PLAN FOR DEVELOPING A CONTAINMENT PERFORMANCE DESIGN OBJECTIVE FOR SAFETY GOALS

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(DRAFT)

I. Introduction

A. Purpose

The purpose of this plan is to develop a containment performance design objective and subsequently determine whether such an objective would be useful and implementable as a quantitative design objective to be added at a later date to the Commission's safety goals. While the containment performance design objective would consider routine operations, present regulations governing containment performance clearly result in very low risk to the public for design basis events. Therefore, the principal thrust of implementation of such a design objective would be for beyond the design basis accidents, i.e., degraded core to severe core damage accidents.

The containment performance design objective would not be intended to be a regulation that would result in explicit, deterministic design requirements which must be met by the designers and by the operating plant. Instead, it is similar to the quantitative design objective for core melt frequency in the Commission's safety goals. Therefore, it would be an objective that one would strive to achieve on a reasonable basis with reasonable confidence. The purpose of the supplementary containment performance design objective is to provide added assurance that the safety goal quantitative design objectives for public risk are achieved. This should provide an additional independent assurance of the public health and safety, conforming to the NRC philosophy of defense-in-depth.

B. Background

The Nuclear Regulatory Commission's Policy Statement in NUREG-0880, Revision 1, presents and discusses qualitative goals and quantitative design objectives for limiting individual and societal risks of nuclear power plant operation. The quantitative design objectives include limits on individual prompt fatalities, cancer fatalities, and the likelihood of a large scale core melt accident.

In the event of a core melt accident the risk to the public depends to a large extent on the performance of the containment system. The proposed safety goals do not provide guidelines for containment performance, nor can such guidelines be determined directly from the safety goals. No objectives currently exist defining the required performance of the containments under severe accident conditions.

One needs to understand containment performance to assess the overall public risk of prompt and latent fatalities. Recent work by the Containment Loads Working Group (CLWG) and the Containment Performance Working Group (CPWG) in support of the Accident Source Term Program Office (ASTPO) activities published in their reports NUREG-1079 and NUREG-1037, respectively, has helped to a great extent in the understanding of these phenomena and the behavior of different containment systems under severe accident conditions, although uncertainties still remain. Based on our current understanding of the design and performance of containments, it is now possible to develop quantitative design objectives for containment performance.

C. Objective

The basic objective of this plan is to develop on a timely basis a containment performance design objective for subsequent trial use and perhaps eventual incorporation into the Commission's Safety Goals Policy Statement. This plan provides for input from organizations other than the NRC. The design objective that is ultimately developed should consider the following attributes:

1. Establishment and maintenance of desired levels of containment performance and defense-in-depth for future designs.

2. Improvement in present designs or operating practices where warranted, considering: uncertainties; the level of accident prevention achieved; the resultant level of public risk and risk reduction potential; and, where the needed minimum level of protection of public health and safety is achieved, the costs of such improvements.

3. The release of more toxic fission products compared to noble gases.

4. The significance of early containment failure compared to late failure, because of the larger quantities of fission products that would be released and the decreased warning time that might be available.

5. Appropriate consideration of engineered safety features to reduce the pressure and temperature loads and the air-borne fission product inventory, as well as the structural strength of the containment under pressure and temperature loads.

6. Flexibility to accommodate uncertainties and to permit reasonable tradeoffs between accident prevention and consequence mitigation.

7. Establishment of a minimum level of containment performance below which tradeoffs with accident prevention would no longer be appropriate.

8. Small scale release of radionuclides from the containment due to severe core damage accidents should be unlikely.

9. Large scale release of radionuclides from the containment due to severe core damage accidents that threatens prompt fatality should be very unlikely.

10. Consideration of accidents that might be initiated from seismic and other external events including initiating events which directly challenge containment integrity.

As a parallel effort, a Containment Performance Design Objective Implementation Guidance will be developed to enable analysts to evaluate specific containment system performance in a standardized and unambiguous manner. The development of such a guide is felt to be crucial to the understanding of how a containment performance design objective would be implemented. A framework for this guide is shown in Table 1.

D. Technical Issues

A rigorous analysis of the performance characteristics of a containment system requires understanding of a number of different important technical issues. First, the magnitude and consequences of a reactor accident depends not only on the accident initiating event but also on the sequence of events that follow. One needs to study these sequences and identify the risk-dominant ones for use in the containment performance analysis. Secondly, one needs to understand the phenomenology of accident progression after core melt, both in-vessel and ex-vessel, the resulting pressure and temperature loads inside the containment structures, and the subsequent response of the containment and estimates of the corresponding source terms. Third, the response of containment systems under severe external events, such as an earthquake, should be investigated. The fourth important issue is a proper understanding of the modes of failure of the different containment types under different loading conditions and/or in different accident sequences. Finally, one needs to be able to combine the accident likelihood and the consequences and consider uncertainties to obtain a perspective on plant risk.

Appendix I contains detailed discussion of these and other related issues.

E. Special Issues

In addition to the above, the plan for developing a Containment Performance Design Objective should include provision for addressing the following special issues.

1. Containment bypass events, in which integrity of the containment structure is insufficient to prevent the fission product release to the environment.

2. Extent of reliance on predictions of accident phenomenology.

3. Whether there should be a different design objective or implementation scheme for present versus future containments.

4. Whether the design objective is intended to improve (if so, to what extent) or just to protect against any erosion of containment quality.

II. Possible Formulation of Containment Performance Design Objective

A. General Philosophy

The development of the containment performance design objective should focus on a simple statement, readily communicable to the public. A simple scalar quantity may, however, not be sufficient for generic or plant-specific implementation: a more complex performance objective may be necessary.

The measure of the objective could be radionuclide release to the biosphere and hence independent of risk tradeoffs offered by siting and emergency planning. However, if it were expressed in terms of offsite consequences, it would not be simply the inversion of the current quantitative design objectives of health effects and core melt.

In assessing containment performance, one must consider uncertainty in PRA as well as safety tradeoffs for various systems, subsystems, and components such that less robust or reliable design or operating performance features in parts of the overall defense-in-depth systems are fully compensated for by superior features in other parts of the system.

B. Specific Conceptual Alternatives

A containment performance design objective may take several possible forms, both probabilistic and deterministic. It may be desirable to have probabilistic performance goals, with a set of deterministic requirements for meeting these goals. Several conceptual forms are described below, for information.

1. One form could be the ACRS proposal (NUREG-0739) which defines performance in terms of the likelihood of a large-scale uncontrolled release of radioactive materials (more than 10 percent of iodine inventory and 90 percent of noble gases), given a large-scale fuel melt and an order of magnitude improvement of new plants compared to the existing plants.

2. A second form might be as follows:

   a. The attenuation capability of a containment system for airborne radionuclides that would be expected to deposit as solids if released to the environment, and that have half-lives greater than X days, should normally not be less than 10(+n).

   b. Further, the mean probability that this attenuation capability is met or exceeded, given a core melt accident, should not be less than 10(-m).

3. A third possibility could be the estimated reduction in the radioactive releases over a specified period of time after a large-scale core melt, such as the reduction in total curies (perhaps weighted as to isotope toxicity) released to the environment due to the presence of the containment building compared to no containment.

4. A fourth way might be to establish a goal (for example, 10^{-6} per reactor-year) for a major release from a containment, allowing a split between the prevention and mitigation systems, with the constraint that the conditional probability of early containment failure is at least 10^{-1}. Thus, for this example, (a) if the core melt frequency is greater than 10^{-4} per reactor-year, one must reasonably demonstrate that no credible mechanism exists for an early containment failure or that the risk from such a failure would be low; (b) if the core melt frequency is between 10^{-4} and 10^{-5} per reactor-year, the conditional probability for early containment failure must be less than or equal to 10^{-2} and, (c) for core melt frequencies less than or equal to 10^{-5} per reactor-year, the conditional probability of early containment failure may be as high as 10^{-1}.

III. Plan to Develop Design Objective

A. Organizational Responsibilities

RES has been involved with much of the work in severe accidents and their consequences. In addition, RES has been charged with preparing the plan for containment performance safety goal. Therefore, RES will assume the role of the overall coordinator in developing the design objective and guidance and work closely with NRR, IE, AEOD and

other offices as appropriate. An interoffice working group will be formed to monitor progress and a senior review group should oversee the overall program.

B. Pertinent Information

The program of developing containment performance design objectives will utilize current research results. The following reports or activities will provide input to the present effort:

1. NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms," draft report for comment.

2. NUREG-1037, "Containment Performance Working Group Report," draft report for comment.

3. NUREG-1079, "Estimates of Early Containment Loads from Core Melt Accidents," draft.

4. SARRP Reports and the subsequent draft of NUREG-1150.

5. Past PRAs and their reviews with emphasis on Indian Point/Zion, Limerick, Millstone 3, GESSAR.

6. Seismic Studies completed to date.

7. Safety goal evaluation steering group report.

8. PRA Reference Document (NUREG-1050).

9. Other programs/activities including RES and industry sponsored work as available.

C. Schedule

Containment Performance Design Objective
(Potential future addendum to Safety Goal Policy Statement)

October 1985	Final Draft Program Plan to Office and ACRS
October 9, 1985	Meet with ACRS subcommittee on Safety Philosophy, Technology and Criteria[1] to discuss plan, staff intent and alternate conceptual approaches
November 1985	Final Plan--Commission Information Paper

[1] Messrs. Okrent, Ebersole, Kerr, Michelson, Remick, Ward, Wylie

February 1986	Meet with ACRS subcommittee to discuss progress on developing a Draft Containment Performance Design Objective and Implementation Guidelines
March 1986	Initial Draft Containment Performance Design Objective and Implementation Guidelines for public comment
April 1986	ACRS Subcommittee to discuss draft
May 1986	Workshop
June 1986	ACRS Subcommittee to discuss results of workshop
August 1986	Redraft for Office comments
August 1986	ACRS Subcommittee to discuss draft
September 1986	Commission paper to EDO
October 1986	Draft Policy Statement to Commission with Implementation Guidelines
November 1986	Publish for Public Comment
March 1987	End of Comment Period
June 1987	Final for Office Review
August 1987	Final to EDO
September 1987	Final to Commission

D. Resources

	Organization	PSY	T/A
FY 86	NRR	1.0	75K
	RES	2.0	200K
	AEOD	0.25	-
	IE	0.25	-
FY 87	NRR	1.5	200K
	RES	2.0	200K
	AEOD	0.25	-
	IE	0.25	-

FY 88*	NRR	1.5	75K
	RES	2.0	200K
	AEOD	0.25	-
	IE	0.25	-
FY 89*	NRR	0.5	75K
	RES	1.0	100K
	AEOD	0.25	-
	IE	0.25	-

*Assuming a two year use and final implementation process.

Table 1

Containment Performance Analysis Guide

I. Introduction

- A. Containment Performance Design Objective
- B. Factors Affecting Estimates of Containment Performance
- C. Internal versus External Events
- D. Use of the Guide

II. Accident Sequence Analysis

- A. Accident Sequences to be Considered
- B. Containment Performance Induced Core Melt
- C. Performance of Analysis
    1. Codes
    2. Assumptions
- D. Uncertainty and Sensitivity Analysis

III. Containment Accident Progression Analysis

- A. Ranges of Initial Conditions
- B. Core Melt Progression and Accident Phenomenology--Input to Containment Analysis
- C. Performance of Analysis
- D. Uncertainty and Sensitivity Analyses

IV. Source Term Analysis (As Appropriate for Chosen Design Objective)

- A. ....
- B. ....

V. Consequence Analysis (As Appropriate for Chosen Design Objective)

- A. ....
- B. ....

VI. Resultant Estimates of Containment Performance

- A. Best Estimate Analysis
    1. Internal Events
    2. External Events
- B. Consideration of Uncertainties

APPENDIX I

TECHNICAL CONSIDERATIONS

The containment performance design objective must be compatible with the philosophy of the safety goals and the other quantitative design objectives. The development program described in this plan will focus on the objectives of the goal, i.e., why is a CPO desired or necessary? What should the goal be? However, in order to develop an implementable goal against which plants could be assessed, an understanding of the underlying technology is necessary.

This section provides a brief account of the issues that might have a bearing on the development and implementation of a containment performance design objective. This discussion provides insights on the ways that a containment performance design objective could best be structured to be useful considering these important issues.

A. <u>Accident Sequence Analysis</u>

Accident sequences are important to assessing containment performance in two important ways:

1. The nature and the temporal progression of accidents are important to the characterization of the likely mode and timing of containment failure and to the type and quantity of radioactive release.

2. For certain accidents, containment failure could contribute to the likelihood of core melt.

The objective of the Accident Sequence Evaluation Program (ASEP) is to provide updated LWR accident sequence information on all operating and near-term operating plants to support the proposed Severe Accident Policy Statement, the NRC source term reassessment, and other safety and regulatory issues. Specifically, ASEP is to: (1) catalog the dominant accident sequence information from existing PRAs into a single reference document; (2) provide updating of sequence likelihood estimates for the reference plants; (3) develop event tree and fault tree models for the operating and near-term operating plants; (4) identify the dominant accident sequences and describe their likelihood characteristics of the operating and near-term operating plants; (5) identify the generic accident sequence insights; and (6) organize the LWRs into a plant classification hierarchy.

Also, ASEP provides results for other NRC programs and applications including: (1) source term reassessment; (2) accident sequence precursor (ASP) program; (3) prioritization of research activities; (4) Analysis and Evaluation of Operational Data (AEOD) incident review; (5) real-time diagnosis and prognosis of accidents and lesser incidents by the NRC Emergency Operations Center; (6) identification of research needs for Severe Accident Sequence Analysis (SASA), human factors, and data development; (7) technical specification assessment; and (8) generic safety issues evaluation.

ASEP has conducted research from both the plant specific and the generic perspectives. On the plant specific basis, ASEP catalogs information from Probabilistic Risk Analysis (PRA) into comparable results and updates (rebaselines) the accident sequence likelihood for the reference plants. On the generic basis, ASEP provides accident sequence information for all the operating and near-term operating Light Water Reactors (LWRs). For both levels of analysis, ASEP research consists of compiling and analyzing current PRA findings, special safety studies, operating experience and post Three Mile Island (TMI) fixes. This information is used to update the sequence likelihoods of the reference plants and to provide insights to be considered for the generic analysis. Since the scope of ASEP is large (i.e., there are about 150 operating and near-term operating LWR units that ASEP needs to consider) and the limited time restriction on the SARP milestones, the ASEP generic analysis is divided into the interim and final phases. For the interim phase, 100 LWR units and a selected set of PRA dominant accident sequences are analyzed. For the final phase, the rest of the LWR units and more accident sequences are to be considered.

NUREG/CR-3301 was published in 7/85 that catalogs dominant accident sequence information from twelve existing commercial nuclear power plant PRAs. The catalog included the following information: general plant information; description and frequency of initiating events; functional failure descriptions; dominant accident sequences and associated frequencies; success criteria; importance measure of the dominant factors driving the sequence frequencies; containment failure modes and probabilities by release categories; insights into human errors; comment on recovery; and other pertinent comments. An informal report on accident sequence likelihood reassessment was published in 8/83 that summarizes and delineates the current major ASEP findings and insights regarding LWR dominant accident sequences. It categorizes approximately 120 dominant accident sequences into sequence classes. It lists the accident sequence insights for each sequence class and provides a limited updating of accident sequence frequencies for the WASH-1400 and RSSMAP plants. An information letter on the insights on design factors affecting containment response was published in 8/83. In contains: (1) a summary of the definition of containment types; (2) a discussion and rationale for considering other design features which may affect the containment

type definition; and (3) a summary of design variabilities among containments. An appendix to NUREG-0956 was published in 2/85 that provides updated accident sequence likelihood information for the 19 source term sequences. It incorporates the insights from the 8/83 accident sequence likelihood reassessment report and the current system design and procedural information from discussions with the utilities.

A report on plant survey and plant model development was published in 12/83. It contains the simplified schematics of the major systems and support systems of about 100 LWRs. Surrogate system configurations were formulated by comparing all system (e.g., compare all AFW systems) considering significant differences in the redundancy, diversity, or support system dependencies for each system of interest. Plant models were developed based on similarity of surrogate system design. For each surrogate systems, fault tree models were developed; they were published in an information letter on 3/84. Several methodology reports were also published throughout 1983-1984. These reports describe the ASEP methodologies on fault tree modeling, data evaluation, treatment of uncertainties, quantification, recovery analysis, ATWS analysis and event tree development. An information letter on system dependencies was published in 9/84. It summarizes, for each plant model, the system design configurations, system operations, system success, and system interdependencies.

Current ASEP effort is concentrated on the plant specific analysis of six reference plants (Surry, Peach Bottom, Sequoyah, Grand Gulf, Zion, LaSalle). The analysis will utilize information from the generic ASEP results as well as results from other programs (RMIEP, Review of Zion PRA, etc.) and will be completed by mid-1986. The research tasks include the following: plant visit, event tree and fault tree modeling, quantification of sequences, uncertainty and sensitivity analysis, and assessment of the generic information base. The result will be used to assist NRC in resolving severe accident issues with IDCOR as well as for the preparation of NUREG-1150.

Since in ASEP, plant specific insights will be used in a generic manner, ASEP results are likely to have large uncertainties. For a better understanding and possible quantification of these uncertainties, the program includes tasks for detailed uncertainty and sensitivity analyses.

B. Containment Performance Analysis

1. Containment Loads and Response Analysis

Estimates of loads (temperature and pressure) experienced by containment structures and the assessment of the responses of specific containments under such loads have been the subject of several studies in the past. But a systematic evaluation of loads and responses for a wide range of containment types under severe

accident conditions has been undertaken only recently by NRC Accident Source Term Program Office (ASTPO). This effort resulted in the formulation of two expert groups consisting of NRC staff members, consultants, staff of National Laboratories and representatives from the nuclear industry, called the Containment Loads Working Group (CLWG) and the Containment Performance Working Group (CPWG).

The overall approach used by CLWG and CPWG in their assessment of containment loads and responses was based on a "standard problems" methodology. The NRC management team formed during this study selected specific reactors to represent each of the six primary containment designs used in the U.S. and defined the respective containment challenge mechanisms as follows:

Standard Problem Number	Containment Type	Reactor Name	Challenge Mechanism
SP-1	Large Dry	Zion	Steam Spike
SP-2	Subatmospheric	Surry	Concrete Attack
SP-3	Ice Condenser	Sequoyah	Hydrogen Burn
SP-4	Mark I	Browns Ferry	Concrete/Liner Attack
SP-5	Mark II	Limerick	Concrete Attack
SP-6	Mark III	Grand Gulf	Diffusion Flames

The in-vessel aspects considered parametrically in these studies are (1) quantity, composition and temperature of corium released at times of vessel failure, (2) quantity and timing of hydrogen release, (3) vessel breach size, and (4) primary system and containment pressures.

Details of CLWG and CPWG work and their findings are published respectively in NUREG-1079 (draft) and NUREG-1037.

a. Containment Loads Working Group

The primary objective of the CLWG was to develop an updated evaluation of containment loads (temperature and pressure history) and associated challenges to containment integrity during severe accidents. The standard problem methodology allowed relatively unambiguous comparisons of the results by the different participants. Efforts were made to investigate the differences and to arrive at some consensus conclusions. During the CLWG proceedings, the phenomenon of "direct heating" drew substantial interest among the experts. Direct heating is relevant to molten corium dispersal to the containment atmosphere following the release at high primary system pressure. The standard problem SP-A was added to the previous six to consider this phenomenon.

Results of the CLWG studies are summarized in Table A-1 (reproduced from NUREG-1079). The results have uncertainties which may, in some cases, be large. A better understanding of the physical phenomena such as direct heating, hydrogen generation and combustion, long-term temperature evolution of the melt, etc. would help in reducing these uncertainties. It should be noted that these results are for specific plant designs and for selected scenarios as defined in the standard problems. Extrapolation of these results to other plant designs would involve additional uncertainties.

b. Containment Performance Working Group

The CPWG was established to develop models for containment leakage. These models quantify leakage areas as functions of containment pressure and temperature loading for specific accident sequences for the six selected containment types. The leakage models were incorporated into the existing computer codes for assessing containment loads as functions of time, corresponding leakage, and the mode and timing of containment failure. The approach of the study was to perform detailed reviews of containment penetration designs and an analytical treatment of penetration performance to estimate leak area as a function of predicted pressure and temperature conditions.

A comprehensive leakage model should consider leakage as a function of pressure, temperature, as well as other parameters such as aging and radiation. The CPWG report presented leak areas a a function of pressure only, with the assumption that leakage will occur only if seal degradation has occurred due to high temperature, aging and other effects.

Results of the CPWG effort is summarized in Table A-2 (reproduced from NUREG-1037). The results are, to a large extent, based on engineering judgment and, thus, are subject to large uncertainties. Several research programs are currently under way that should provide additional bases to supplement and modify the engineering judgment as appropriate.

2. <u>Source Term Analysis</u>

The source term analytical procedure that has been developed for the NRC is discussed in NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms." In contrast to the assumptions and methods presently used in regulatory practices, that is, those from TID-14844 and the Reactor Safety Study, the new source term analytical procedures has been found to yield results that are strongly dependent on plant characteristics and on the accident sequence definition. That is, where releases of radioactive materials within the containment or to

the environment were previously considered to be simple and generic, they are now perceived to be complex and specific, making development of useful generic source terms difficult.

The source term analytical procedure depends not only on the accident sequence definition--the timing and mode of failure of plant equipment, plant procedures, operator actions and automatic actuation of plant equipment--but also on the assumed timing and mode of containment failure. This latter factor has been found to be the largest single factor affecting source terms for most accident sequences. The larger source terms have been calculated for sequences with early containment failure and high flow rates of exiting gases and aerosols. On the other hand, a delay in the time of containment failure for several hours allows time for natural processes to reduce the airborne fission product concentration significantly.

The analytical procedure is complex and involves several scientific disciplines, including those of reactor physics, thermal hydraulics, materials science, chemistry, and aerosol physics. To the extent that these disciplines and the models to account for them in the source term analytical procedure also are important to the evaluation of other phases of the problem, for instance, the sequence definition or the containment response matrix, the source term evaluation cannot be divorced from the evaluations of the other phases. For instance, the thermal hydraulic analyses that are used to evaluate the containment loads and the several containment behaviors that may result from those loads are important in the determination of retention of fission products within the reactor coolant system because the flow rates determine the aerosol residence time and therefore the natural removal.

Although the new analytical procedure represents a major advance in technology compared to the methods of the Reactor Safety Study, remaining areas of uncertainty have been identified and they persist in some of the same areas where major advances have already been made. During the many reviews of the analytical procedure, other organizations recommended areas for improvement and for continued research. From these reviews, it is clear that the weak areas in the source term technology have been consistently identified by all major participants in the review process. NUREG-0956 discusses the major areas of uncertainty and the research programs in place to address them.

The results of source term calculations for the different accident sequences have large uncertainties. The Quantitative Uncertainty Estimation for Source Term (QUEST) study, which was

completed in parallel with the development of the analytical procedure, provided the first attempt to evaluate the sensitivity of source term results to "reasonable" changes in the input parameters and the models. The QUEST study was limited and generalizations are not justified, but it was concluded for the sequences studied that if the mode and timing of containment failure is considered to be fixed, the uncertainties in the release fractions would likely show a span of two decades.

The major areas of uncertainty identified in NUREG-0956 and their importance to development of a containment performance design objective are as follows:

1. Natural circulation in the reactor vessel could have an important effect on reactor vessel pressure at time of failure.

2. Core melt progression and hydrogen generation will affect at least: core-coolant interaction, core-concrete interactions, containment pressure and temperature loads, time of containment failure, and fission product release.

3. In-vessel fission product release from the fuel and aerosol generation will have an important effect on the fission product release.

4. Retention and revaporization of fission products in the reactor coolant system will have an important effect on the fission product release.

5. Fission product release and aerosol generation from the core-concrete interaction will have an important effect on the fission products available at time of containment failure.

6. Scrubbing efficiency of suppression pools and ice compartments will have an important effect on the amount of fission products removed from the containment atmosphere.

7. Containment pressure loads will have an important effect on whether and when containment fails.

8. Containment failure modes will have an important effect on the fission product release.

3. <u>Seismic Risk Analysis</u>

The design objective for containment performance includes consideration of accidents that might be initiated from seismic events. However, such accidents and their consequences have

major uncertainties because (1) the recurrence frequency of seismic events far above the design basis accident are highly uncertain, and (2) the likelihood, mode, and timing of potential containment failure and core melt, given a magnitude of ground motion sufficiently above the plant design basis to threaten the plant, are highly uncertain. The major sources of seismic uncertainties are discussed in the section.

a. Seismic Hazard

In recent years the Nuclear Regulatory Commission has made increasing use of probabilistic estimates of seismic hazard and seismic risk at nuclear power plant sites in the United States. Seismic hazard means the frequency of occurrence of earthquake ground motion, as compared to the integrated effects of ground motion damage and consequences, which is usually referred to as seismic risk. The most recent use of probabilistic methodology in the earthquake problem has been in the incorporation of seismic initiators into Probabilistic Risk Assessment (PRA). For the regulator, all PRAs, including seismic PRAs offer an attractive and powerful tool for rational decisionmaking. It provides a systematic and comprehensive way of looking at the behavior of nuclear power plants under various and complex scenarios and a resulting understanding of the consequence of the occurrence of events well beyond the design basis. Problems arise, however, since the very nature of PRA requires it to make a formal assessment of the uncertainty involved. This is often very difficult.

In the seismic hazard area there are several fundamental shortcomings which inhibit our ability to make "reliable" estimates. First and foremost is the inadequacy of the existing historical and instrumental seismic record (two or three hundred years in the United States). PRAs try to utilize this record to draw inferences on earthquakes that appear to have mean return periods from two to four orders of magnitude or more beyond the record. This extrapolation of numerical estimates is speculative, particularly since we lack a fundamental understanding of the causative mechanism of earthquakes in the eastern U.S. Secondly, estimation of ground motion from an earthquake of known size also poses a problem, particularly in the eastern U.S. where few records of strong ground motion from earthquakes actually exist. Attempts to deal with these problems lead to the observation that most of the calculated uncertainty in the seismic event PRAs is related to uncertainty in the seismic hazard.

Methodologies are being used that formally take into account "expert judgment" as to the range of models and the appropriate parameters that need to be used. The extent to which, and consistency with which, this major source of uncertainty is dealt with in non-seismic initiators appears to vary greatly. As a result, there exists a significant potential for systematic bias that cannot be simply accounted for. These biases would place into question comparisons between seismic and internal initiators. However when making comparisons where such biases may be common to the entities being compared, the resulting errors tend to be minimized. In other words, a seismic PRA may not give the regulator a sufficiently reliable picture of the absolute seismic contribution to core melt frequency or risk, but it will give a more robust estimate of which among possible sequences are the most and least worrisome.

In addition, reliance upon simple point estimates such as means or medians to characterize actual risk is unwarranted. There has been an extensive effort to define the uncertainty. The wide bands of uncertainty, often several orders of magnitude, associated with estimates of seismic hazards can be thought of as representing a large part, but not all, of the actual uncertainties. They may be used to gain insight as to the range of the actual risk associated with seismic initiating events. That does not mean to imply that higher risk estimates (e.g., 95th percentile) are more appropriate than the median, mean or lower (5th percentile) estimates. For many aseismic areas of the eastern U.S., we certainly cannot exclude from the range of reasonable possibilities the judgment that there essentially is no risk to the public resulting from earthquake-induced damage at a seismically-engineered nuclear power plant during its operating life.

The most vexing problem associated with probabilistic estimates of seismic hazard lies within their highly quantitative nature. The ability to generate vast quantities of numerical results such as can be done utilizing probabilistic estimates is often assumed to be a measure of greater credibility and accuracy than simple judgmental estimates. Often simple qualitative judgments as to input parameters are the root causes of significant differences in the quantitative results. This potential for misinterpretation of probabilistic estimates of seismic hazard (and risk) appears to be integrally bound up into the most important and positive aspect of these estimates, that is the ability to provide a quantitative framework for the integration of diverse elements and their associated

uncertainties. This apparent contradiction can be alleviated, however, by the recognition that all probabilistic estimates of seismic hazard and risk need also to be evaluated in the light of physical and engineering insight.

b. Containment Failure Modes Under Seismic Loads

Containment structures designed in accordance with Section III of the ASME Boiler and Pressure Vessel Code can be expected to perform adequately even if subjected to seismic loads far in excess of their design basis. The fundamental reason for this expectation is that the structures were proportioned and detailed so that they would respond elastically when subjected to the combined effects of a loss of coolant accident and the safe shutdown earthquake. Although different parts of a containment are most affected by seismic loads and internal pressures, the net outcome of including the effects of both loadings is to significantly increase the capacity of the containment to withstand either loading, considered separately. In addition, the design criteria specified essentially elastic response. Since significant yielding must occur before the onset of failure, that aspect of design practice adds to the margin above design levels that can be expected in any containment.

Estimates of structural capacities for a number of containment shells have, indeed, indicated large expected margins in earthquake resistance. Two studies are especially worth noting. One (NUREG/CR-4334) summarized estimates of failure levels for concrete containments made in connection with seismic PRAs. The other (NUREG/CR-3122) focussed on providing estimates of the resistance of steel containment shells that could be used in probabilistic evaluations. The plants studied were all located in the eastern U.S. (i.e., east of the Rocky Mountains) had safe shutdown earthquakes ranging from 0.10g to 0.25g. The concrete containment study estimated, for the failure modes considered, median capacities ranging from 2.5g to 9.7g. The steel containment study provided mean estimates of capacity ranging from 1.45g to 16.25g. Considerable uncertainty is associated with these estimates. For example, in the concrete containment study, an attempt was made to include the effects of both randomness and uncertainty and make estimates of capacity that were felt to have a high confidence of low probability of failure (HCPLF). The net result of reflecting the uncertainties in this way was a reduction of the estimated capacities by a factor of about three--with HCPLF estimates of shell capacity ranging from 0.8g to 2.7g. Notwithstanding the large uncertainties, it

is still apparent that structural failure of the containment shell is not a significant potential failure mode under earthquake loading.

It is more likely that, for soil sites, any containment failure would be associated with a foundation failure, leading to failure of piping. This potential failure mode has not been studied extensively. But, in one case where it has been considered, it has been found to be dominant. In the previously cited review of seismic PRA studies (NUREG/CR-4334), the median estimates for failure of a containment structure and for soil failure beneath its foundation slab were 2.9g and 0.9g, respectively. The corresponding HCPLF estimates were 0.8g and 0.3g, respectively.

The fundamental question is: Will containment performance in a severe accident be significantly degraded if the accident was initiated by a large earthquake? A soil failure would amplify motions at piping penetrations and could create a leak path. Another possible failure mode is local damage at operable penetrations which might reduce pressure retaining capability.

Research on containment failure modes under seismic loads will be carried out during FY86-88 with the intention of assessing the ability of analytical methods to predict behavior near failure. Necessary experiments will be identified and planned during FY 86 and carried out in FY 87 and 88. It is likely that the outcome of this research will, in the context of PRAs, improve confidence in mean or median estimates of failure level. However, the number of experiments will be too small to be of any real help in reducing uncertainties.

c. Integrated Risk Insights

A number of seismic PRAs have been published in the open literature to date. Although these studies vary in scope, methodology, input information, plant type, etc., they provide valuable engineering insights on both generic and plant-specific levels, that can be useful for assessment of containment performance. Some of the major insights are discussed here.

In general, seismically initiated accidents have been estimated to be significant risk contributors. Large uncertainties exist in the numerical results of seismic PRAs, the main contributors to which are due to the lack of adequate knowledge of site-specific seismic hazard. It should be noted, however, as discussed before, that comparison between risks from seismic and non-seismic initiators may be subject to methodological bias.

The frequency of a major release that would be permissible under the containment performance safety goal is expected to be of the order of 10^{-6}/year or less. The recurrence frequency of a seismic event several times greater than SSE at some sites may be higher that this (of the order of 10^{-5}/year). However, the occurrence of such an event does not necessarily result in core melt. Past PRAs indicate that plant systems and structures have substantial margins to failure given such seismic occurrences.

Seismic PRAs include numerous simplifying assumptions. Although some of these assumptions may be unconservative (e.g., treatment of relay chatter), a number of these are obviously conservative. An example is the assumption that when a structure fails, all equipment dependent on or underneath that structure also fails.

Seismically induced accidents leading to core melt almost always involve loss of offsite power. The emergency diesel generators are usually strong against seismic effects. But their support systems and anchorages for other mechanical and electrical equipment often are the weak points. Components, in general, are very robust against seismic effects. Large concrete containment structures also have high seismic capacity.

Current PRAs do not consider effects of design and construction errors, or the effects of increased stress during a seismic event on human performance. Contribution of risk due to these factors to the overall plant risk may be significant.

C. Containment Failure Modes

There are inherent capabilities in all containment types to accommodate to varying degrees severe accidents beyond the design bases. These capabilities to mitigate the consequences of core melt accidents should be thought of in terms of release characteristics of the radioactive fission products; in particular, the composition, the mode, the quantity, and the timing of releases. Ultimately, it is the release of radioactivity that defines the performance of a given containment and not the strength or energy absorption capabilities of containment systems themselves. In the analyses to date, containment strength as well as the containment features designed to absorb energy and reduce airborne radioactivity (ice in ice condenser plants, water in the suppression pool of Boiling Water Reactors, sprays in large dry containments) and design characteristics unique to specific containments (e.g., penetration seal materials) are the key factors in an integrated risk assessment of containment failure modes and release characteristics. The assessment of these features, common

to all reactors within a containment type classification, must be combined with plant-specific design features which may have equal importance in determining containment performance. A list of containment failure modes is provided in Table A-3. Important examples of plant-specific features are:

- o capability to isolate the containment on demand;
- o capability of containment to maintain pressure without gross leakage prior to structural failure (e.g., capability of penetration seals to hold up under the severe accidents conditions of a core melt);
- o capability of containment cooling to remain functional under accident conditions;
- o potential for containment bypass and release of the radiological source directly to the environment (e.g., by steam generator tube failures).

The above examples point to an integral and comprehensive approach to containment analysis that considers plant-specific as well as generic features of containments.

With this general approach in mind, mechanistic core melt accident evaluations can determine the containment loading and containment failure characteristics for existing facilities, facilities under design, and facilities with mitigation features in place. The analytical approach should be as realistic as possible and should include, where appropriate, dynamic loadings from hydrogen and carbon monoxide detonations, quasi-static loadings from hydrogen (and other combustibles) burning, quasi-static loadings from steam and noncondensibles overpressurization, associated temperature loadings, basemat penetration by core melt materials, and effects of aerosols as well as the loadings on engineered safety features (ESF) and containment performance.

Just as important as determining the containment performance is an evaluation of the radioactive source terms and the quantities of fission products released to the atmosphere upon loss of containment integrity. Our present analytical processes for calculating source terms is described in NUREG-0956; but there are significant uncertainties in the source term analysis.

While both development of a containment performance objective and assessment of conformance with that objective will take account of new source-term information and rely on those aspects of new source-term information which are well understood, the development of the containment performance objective will not place undue reliance on new source-term information.

D. Consequence and Risk Perspective

Containment leakage rates are controlled under 10 CFR Part 100 (siting) and 10 CFR Part 50 (design criteria). Under these criteria, offsite radiological damages in the event of a contained core melt accident would be relatively small. The accident at Three Mile Island in 1979 is the sole and outstanding example of this. However, should the containment leak substantially beyond its design basis, either before or after a core melt, driving forces inside the containment would allow sizeable radiological releases to occur, up to and including releases of sufficient magnitude to cause injuries and fatalities offsite. Well below such release magnitudes (source terms), considerable offsite property damage and latent (long term) cancer fatalities could be induced. For perspective, containment design basis leak rates average about 0.1 percent per day, whereas leakage rates above about 10 percent per day would be required for a core melt accident to induce significant physical damage to the environment.

Any radiological release to the environment above technical specifications would be of major regulatory concern, of course; but, in the context of beyond the design basis containment threats, three potential offsite consequences are of import when considering a separate containment performance goal: (i) early fatality risk and injuries, (ii) cancer fatality risk, and (iii) total man-rem and offsite property damage due to a potential accident sequence. Of these, the first pertains to the area close to the site and the latter two pertain to a much broader area.

Both cancer fatality risk and total man-rem depend strongly on accident probability and release fraction (source term). However, early injuries and fatality risk depends on these parameters plus the timing of a major release--a delay of even a few hours not only decrease the source terms but would provide the public with time to accomplish effective protective action (e.g., evacuation) in the higher risk areas nearby.

Given a core melt safety goal and a set of public health safety goals, certain containment performance design objectives should be amenable to derivation. The former leads to the latter only via the containment. Thus, a containment performance design objective is implicit in the draft NRC safety goals. Such a containment performance design objective could that relate to physical attributes which, with a given probability, would contain or delay a major release due to a core melt; or it could relate to the probability of incurring early injuries or fatalities.

Table A-1

SUMMARY OF TABLES OF RESULTS

SP-1 (Zion)

Containment Capability:	150 psia*
Upper Bound Spike:	107 psia
Early Failure Physically Unreasonable	
Best Estimate Pressure Rise: (Including heat sinks)	10 psi/hr
Best Estimate Failure Time: (Unlimited water in cavity)	16 hrs

SP-2 (Surry)

Containment Capability:	135 psia*
Upper Bound Spike:	107 psia
Early Failure Physically Unreasonable	
Best Estimate Failure Time: (Dry cavity)	Several Days

SP-3 (Sequoyah)

Containment Capability:	132 psia, 330°F*
Upper Bound Loading:	132 psia in 40 min.
Lower Bound Loading:	132 psia in 2 hours
Thermal Loads:	500° - 700°F
Early Failure Quite Likely	

SP-4 (Browns Ferry)

Containment Capability:	132 psia, 330°F*
Upper Bound Loading:	132 psia in 40 min.
Lower Bound Loading:	132 psia in 2 hours
Thermal Loads:	500° - 700°F
Early Failure Quite Likely	

Table A-1 (Continued)

SP-5 (Limerick)

Containment Capability:	155 psia, 330°F*
Upper Bound Loadings:	145 psia in 2-3 hours
Lower Bound Loadings:	100 psia in 3 hours
Thermal Loads:	550° - 700°F

Early Failure Rather Unlikely
(Upper bound too conservative)

SP-6 (Grand Gulf)

Containment Capability:	75 psia*
Upper Bound Loadings:	30 psia
Wall Fluxes:	10^3 - 10^4 BTU/hr ft^2
Penetration Seal Temperatures:	345°F*

Pressurization Failure Physically Unreasonable
Seal Failure Unlikely

SP-A (Based on SP-1 Results)

Containment Capability:	150 psia*
Upper Bound Loads:	176 psia
Thermal Loads:	1340°F

Early Failure Quite Likely
(100% core dispersal with
100% clad oxidation - No
early depressurization
- Unobstructed flow)

*From NUREG-1037, Containment Performance Working Group Report, Draft dated May 1985.

Table A-2. Summary of Containment Performance

Plant/type	Containment performance findings	Impact of leakage on offsite consequences
Zion/PWR large dry	Without leakage the containment would reach its capability pressure between 12 hours to several days after the start of the accident depending on conditions in the reactor cavity. Leakage before the capability pressure would not exceed 5 $in.^2$ Such leakage would be sufficient to delay or possibly prevent the containment from reaching its capability pressure. Significant leakage could occur at 8 hours after the accident, but only in the case of a flooded cavity.	The impact of leakage on offsite consequences would be low for sequences in which containment integrity is threatened by long-term pressure/temperature buildup. This is because significant leakage is predicted only if the reactor cavity is flooded. A flooded cavity implies a coolable debris bed and hence limited core/concrete interactions, which in turn results in minimum fission product release via this mechanism. Hence, as most of the fission products are released to the containment during invessel degradation of the reactor core for this sequence, aerosol agglomeration and settling will reduce the fission products suspended in the containment atmosphere before the beginning of significant leakage. Other mechanisms that result in early containment failure, such as hydrogen burns, isolation failure, etc., have potentially a high impact on offsite consequences but would be unaffected by leakage induced by the severe accident conditions considered in this report.
Surry/PWR subatmospheric	Without leakage the containment would not approach its capability pressure until several days after the start of the accident. Leakage before the capability pressure would not exceed 0.4 $in.^2$ Such leakage would have little effect on containment performance.	
Sequoyah/PWR ice condenser	Without leakage the containment would reach its capability pressure after about 8 hours following the start of the accident assuming no early failure via hydrogen burns. Leakage before the capability pressure would not exceed 0.3 $in.^2$. Such leakage would have little effect on containment performance.	

Table A-2. Summary of Containment Performance (Continued)

Plant/type	Containment performance findings	Impact of leakage on offsite consequences
Peach Bottom/ BWR Mark I	Without leakage the containment would reach its capability pressure between 4 and 5½ hours after the start of the accident. A leakage of approximately 10 in.² to 12 in.² would be sufficient to prevent the containment from reaching its capability pressure. Leakage before the capability pressure could be as high as 35 in.²	The leakage in Mark I and II containments can be large and has both positive and negative effects. A negative effect is the potential for significant release of the fission products earlier than would have been predicted based on threshold models. This earlier release when coupled with the potential for greater pool bypass (less fission product scrubbing) could result in increased offsite consequences. However, a positive effect of leakage vs. gross failure is that it could result in significant aerosol agglomeration and settling in the reactor aerosol agglomeration and settling in the reactor building. In addition, the standby gas treatment system could be utilized to scrub the fission products under these circumstances. The above uncertainties associated with leakage vs. gross failure could be eliminated by using current operating procedures which call for wet-well venting.
Limerick/ BWR Mark II	Without leakage the containment could reach its capability pressure as early as 5 hours but not later than 1 day following the start of the accident. A leakage of approximately 5 in.² would be sufficient to prevent the containment from reaching its capability pressure. Leakage before the capability pressure could be as high as 42 in.²	
Grand Gulf/ BWR Mark III	Without leakage the containment would reach its capability pressure after about 13 hours following the start of the accident, assuming no early failure via hydrogen burns. Drywell leakage equivalent to 9 in.² at a pressure of 4 psid was determined on the basis of observed leakage from an integrated leak rate test. Drywell leakage sufficiently large to result in pool bypass could occur 5 to 6 hours after the start of the accident.	The impact of drywell leakage is that it provides a mechanism for fission products to bypass the suppression pool. However, significant pool bypass is expected to be late in the accident after most of the fission products have already been scrubbed by the pool. Hence, the impact on the offsite consequences of loss-of-drywell integrity late in the accident sequence is not expected to be significant. The leakage estimate at 4 psid is important because of the potential for suppression pool bypass early in the accident sequence and should therefore be carefully assessed in future studies. However, the leakage paths at this pressure would be tortuous, with the potential for significant retention of aerosol fission products.

Table A-3. Containment Failure Modes

	LARGE DRY/ SUBATMOSPHERIC	ICE CONDENSER	MARK I/ MARK II	MARK III
STEAM EXPLOSIONS	--- very unlikely to be sufficiently energetic to fail containment - - -			
Failure to isolate	- - - Design dependent - - - Subatmospheric less likely than large dry		Unlikely. Intermediate time frame seal failure may be a problem	Design dependent
Hydrogen burn/ detonation	Unlikely. Intermediate time-frame failure may be a problem in some containments	Dominant failure mode	Unlikely	Dominant failure mode. Wetwell failure more probable than drywell failure and less risk significance.
Overpressurization	Early failure unlikely. Dominant late failure mode but lesser risk significance.	Late failure	Dominant intermediate time-frame failure mode significant to risk.	Late failure mode of lesser risk significance.
Basemat penetration	- - - Possible to unlikely, of lesser risk significance - - -			
Containment Bypass	- - - Design dependent, very risk significant - - -			