NUREG-1335

Individual Plant Examination: Submittal Guidance

Draft Report for Comment

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research
Office of Nuclear Reactor Regulation

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ABSTRACT

Based on a Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants, the performance of a plant examination is required by the licensee of each nuclear power plant. The plant examination looks for vulnerabilities to severe accidents and cost-effective safety improvements that reduce or eliminate the important vulnerabilities. This document delineates the guidance for reporting the results of that plant examination.

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FOREWORD

This draft document contains proposed guidance to utilities for reporting to the Nuclear Regulatory Commission the results of their Individual Plant Examinations (IPEs) of each licensed nuclear power plant for vulnerabilities to severe accidents pursuant to a Generic Letter sent to the licensees, dated November 23, 1988.

Following discussion of this document at a workshop, it will be reissued in final form. Questions and points for clarification are being solicited. Written comments should be directed to the undersigned.

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1. INTRODUCTION AND OBJECTIVES

1.1 Background

On August 8, 1985, the U.S. NRC issued a Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants (50 FR 32138) that introduced the Commission's plan to address severe accident issues for existing commercial nuclear power plants. (The staff in a separate effort is developing recommendations on the treatment of severe accident issues for future LWRs.) Over the past several years, the Commission has developed an approach to implement this plan for existing plants and recently has issued a Generic Letter (Ref. 1) that communicates this plan to all utilities. Each licensed nuclear power plant is required to perform a plant examination that looks for vulnerabilities to severe accidents and cost-effective safety improvements that reduce or eliminate the important vulnerabilities. The specific objectives for these Individual Plant Examinations (IPEs) are for each utility to (1) develop an overall appreciation of severe accident behavior; (2) understand the most likely severe accident sequences that could occur at its plant; (3) gain a more quantitative understanding of the overall probability of core damage and radioactive material releases; and (4) reduce the overall probability of core damage and radioactive material release by appropriate modifications to procedures and hardware that would help prevent or mitigate severe accidents. Upon completion of the examination, the utility will be required to submit a report to NRC describing the results and conclusions of the examination. This submittal will be reviewed and evaluated by the NRC.

This individual plant examination submittal guidance document establishes style and content guidelines for the utility submittals. There are NRC and industry reports that help to put this document into proper perspective and help to give background to many of the specific matters presented herein.

- "Severe Accident Insights Report," NUREG/CR-5132 (Ref. 2). This report describes the conditions and events that nuclear power plant personnel may encounter during the latter stages of a severe core damage accident and what the consequences might be of actions they may take during these latter stages. The report also describes what can be expected of the performance of the key barriers to fission product release (primarily containment systems), what decisions the operating staff may face during the course of a severe accident, and what could result from these decisions based on our current state of knowledge of severe accident phenomena.

- "Assessment of Severe Accident Prevention and Mitigation Features," NUREG/CR-4920, Volumes 1-5 (Ref. 3). This series of reports describes plant features and operator actions that have been found to be important in either preventing or mitigating severe accidents in LWRs with five different types of containments.

- "PRA Procedures Guide," NUREG/CR-2300 (Ref. 4). This report is a guide to the performance of probabilistic risk assessments (PRAs) for nuclear power plants.

- "PRA Review Manual," NUREG/CR-3485 (Ref. 5). This report describes an approach for reviewing a Level 1 type PRA (a PRA that carries the accident analysis up to the point of calculating the probability of core damage or core melt).

- "Probabilistic Safety Analysis Procedures Guide," NUREG/CR-2815 (Ref. 6). This report provides the structure of a probabilistic safety study that is to be performed and indicates which products of the study are valuable for regulatory decisionmaking.

- "Individual Plant Evaluation Methodology for LWRs," IDCOR (Ref. 7). This industry report provides, in a BWR volume and a PWR volume, methodology for plant-specific evaluation of the probability of severe accidents.

- "Staff Evaluation of the IDCOR IPEM for PWRs;" "Staff Evaluation of the IDCOR IPEM for BWRs" (Ref. 8). These two reports describe the enhancements to the front-end of the Individual Plant Examination Methodology that the staff considers necessary before the front-end IPEM should be used for an IPE.

1.2 Purpose

The purpose of this document is to provide style and content guidelines for the utility submittals. The reasons for having these guidelines are to provide sufficient submittal content for an effective review and to provide a format that allows for an efficient submittal review and facilitates comparing the many submittals. This document should be used by the utilities as they perform their IPEs and prepare their submittal reports.

1.3 Scope

The scope of this report is consistent with the IPE program as outlined in the Generic Letter (Ref. 1). This report presents submittal guidance for the Individual Plant Examination Methodology (IPEM) and the PRA method of performing an IPE. These are the first two of the three options discussed in the Generic Letter. (The third option, that of choosing some other method (unspecified), will be treated on a case-by-case basis as necessary.) It should also be noted that the IPE program stops with the radionuclide release characterization, the endpoint for a so-called Level 2 PRA. The IPE should carry through evaluation of the behavior of the containment and radionuclide releases to enable utility personnel to understand these phenomena and to provide a basis for the development of a severe accident management capability. Finally, this document makes no substantive distinction between the two IPE options, namely, the IPEM by IDCOR (Ref. 7) and PRAs, in the submittal guidelines. All limitations of the IPEM and enhancements to the front-end IPEM for use in the IPE program are delineated in the staff evaluation reports (Ref. 8). Therefore, they are not repeated in this document.

1.4 Goals for This Report

The goal for this report is to provide a uniform mechanism for allowing the NRC staff to draw conclusions regarding the implementation of the Severe Accident Policy Statement for existing plants.

- The NRC staff will draw a conclusion about the acceptability of the IPE for a given submittal. Specifically, as stated in the Generic Letter, "The NRC will evaluate licensee IPE submittals to obtain reasonable assurance that the licensee has adequately analyzed the plant design and operations to discover instances of particular vulnerability to core melt or unusually poor containment performance given a core melt accident. Further, the NRC will assess whether the conclusions the licensee draws from the IPE regarding changes to the plant systems, components, or accident management procedures are adequate. The consideration will include both quantitative measures and nonquantitative judgment." A positive staff conclusion would be that there is reasonable assurance that the IPEM or the PRA represents the plant, its operation, and its safety strength and vulnerabilities so that the utility is on firm ground in making improvements and/or implementing an effective accident management program.

- The NRC staff will want to conclude how much the utility has integrated the PRA/IPEM methods and applications into the operation and daily activities of the facility. The basis for the request in the Generic Letter (Ref. 1) for involvement of utility staff in the IPE review is the belief that the maximum benefit from the performance of an IPE would be realized if the utility's staff were involved in all aspects of the examination and that involvement would facilitate integration of the knowledge gained from the examination into operating procedures and training programs.

2. SUBMITTAL GUIDELINES: STYLE AND CONTENT

This section provides the content and style guidelines for the utility submittals. The major parts of this section are the front-end (Section 2.1), the back-end (Section 2.2), unique safety features and plant improvements (Section 2.3), and the utility team (Section 2.4). The utilities are requested to submit their IPE reports using the Standard Table of Contents given in Table 2.1. This will facilitate review by the NRC and allow for intercomparisons among various submittals. The content of the elements of this Table of Contents is discussed in Sections 2.1, 2.2, 2.3, and 2.4 below.

The level of detail needed in the documentation should be sufficient to enable NRC to understand and review the validity of all input data and calculation models used; to assess the sensitivity of the results to all key aspects of the analysis; and to audit any calculation. It is not necessary to submit all the documentation needed for such an NRC review, but its existence should be cited and it should be available in easily usable form. The guideline for adequate retained documentation is that an independent expert analyst should be able to reproduce any portion of the results of calculations in a straightforward, unambiguous manner. To the extent possible, the retained documentation should be organized along the lines identified in the areas of review.

A complete severe accident assessment requires analysis of external events. Previous guidance documents have discussed procedures for performing such analyses (NUREG/CR-2300 (Ref. 4) and NUREG/CR-2815 (Ref. 6)), and several full-scope PRAs and NRC's reviews of them have addressed external events. There is a technical basis for analyzing whether a given plant has significant vulnerabilities with respect to a given external initiator. Although IPE submittals are not presently required to address external events, the staff encourages early consideration of certain aspects of external events in the IPE process. Section 2.5 provides a discussion of future external event analysis.

2.1 Front-End Submittal: Probability of Severe Accidents

The content and style of the front-end portion of the IPE submittal is addressed for the following key areas:

1. General Methodology
2. Information Assembly
3. Accident Sequence Delineation
4. System Analysis
5. Quantification Process
6. Front-End Results and Screening Process

Reporting guidelines for each of these key areas are detailed in Sections 2.1.1 through 2.1.6.

Table 2.1 Standard Table of Contents for utility submittal.

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Table 2.1 (Continued).

	Corresponding Section in This Report
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7. Summary and Conclusions (including proposed resolution of USIs and GSIs)	

2.1.1 General Methodology

Reporting guidelines include a concise description of major tasks of the methodology employed and how these tasks interact with each other to generate the list of plant vulnerabilities. This includes such major tasks as event tree modeling, systems analysis, dependency treatment, quantification process, and vulnerability identification and treatment.

2.1.2 Information Assembly

Reporting guidelines include:

1. A general description of the plant and its layout.

2. A concise description of the containment building and its layout.

3. A list of PRA studies and/or IPEs of this plant or other similar plants that the IPE team has reviewed along with a concise summary of insights derived from these reviews.

4. A concise description of plant documentation used such as the Final Safety Analysis Report (FSAR); system descriptions, procedures, and licensee event reports (LERs); and a concise discussion of the process used to confirm that these documents represent the as-built, as-operated plant.

5. A description of the walkthrough activity of the IPE team, including scope and team makeup.

2.1.3 Accident Sequence Delineation

Reporting guidelines include:

1. A list of all generic and plant-specific initiating events and groups of events considered (including internal flooding), their frequencies, and the rationale for the grouping used. Additionally, list the minimum success criteria for front-line systems that mitigate each initiating event or group of events, the bases for those criteria (e.g., expert judgment, realistic calculation, FSAR), and the consistency of the criteria with the as-built, as-operated plant.

2. All event trees (functional and/or systemic) developed or adapted from a reference plant for the initiating events or groups of initiating events, including a concise discussion of the assumptions and event heading dependencies considered.

3. If separate event trees are developed to support special event analysis (e.g., ATWS, station blackout, PWR reactor coolant pump seal loss-of-coolant accidents (LOCAs), interfacing LOCA, internal flooding), include the same information as in item 2 above.

4. The support system event trees, including modifications if they have been adapted from the IDCOR reference plant or other applicable PRAs. A concise description of each support system state and its effect on each front-line system.

5. An explanation of the method of grouping accident sequences into various "bins," "categories," or "plant damage states," including the unique bins considered and their physical meaning in terms of controlling factors such as initiating events, time of core melt, and performance of containment safety features.

6. A table summarizing the bins associated with the accident sequences that lead to core melt.

2.1.4 System Analysis

Reporting guidelines include:

1. A description and a simplified diagram of front-line and support systems considered in the IPE; appropriate line diagrams of electrical systems.

2. The fault tree diagrams, including a discussion of the method used to develop and evaluate the fault tree.

3. The dependency matrix for all support systems and front-line systems (or functions) considered, including all interdependencies among the systems. This also includes dependencies caused by systems that are shared among multi-unit plants.

4. Differences between the subject plant and the reference plant if the dependency matrix is adapted from a reference plant.

2.1.5 Quantification Process

Reporting guidelines include:

1. Types of common cause failures considered in the analysis (both in the event tree sequences and in the system analysis), including the quantification process employed and sources of common cause failure data used. Include a list of component groups subjected to common cause failure analysis.

2. Internal flooding initiators such as overfilling of water tanks, hose and pipe ruptures, and pump seal leaks along with their frequencies and resulting damage to important plant equipment. Include the result of the quantification of the flooding sequences that lead to core damage.

3. Types of human failures considered in the IPE, such as human failures in maintenance and operation and human failure to recover and mitigate accident progression.

4. List of human reliability data and time available for operator recovery actions considered, including the sources of these data. If the human errors are screened, include a list of errors considered and a list of "important errors," as well as the criteria for determining importance.

5. Method used for determining unavailability of plant hardware, including a description of the unavailability consideration for standby and operating equipment and equipment in test and maintenance.

6. List of items for which plant-specific experience is used, including the method of generating failure data from such experiences (e.g., classical or Bayesian method). Include the rationale if plant-specific experience for important items such as auxiliary feedwater (AFW) and emergency core cooling system (ECCS) pumps, initiating events, batteries, feed pumps, electrical buses, breakers, and diesel generators has not been used. (Generally, plants with several years of experience should use plant-specific experience for these types of items.) Also list any generic failure data used for equipment or initiating events.

7. Method by which accident sequences are quantified. If computer programs are used, identify the program and nature of calculations performed by using this program (e.g., cutset generation, sequence quantification, and sensitivity analysis).

2.1.6 Front-End Results and Screening Process

Reporting guidelines include:

1. A description of how the screening criteria in Appendix 2 to the Generic Letter (Ref. 1) are used in the screening process.

2. A list of functional sequences selected using the Generic Letter screening criteria, including a concise discussion of accident progression; specific assumptions; sensitive assumptions and parameters; essential equipment subjected to environmental conditions beyond the design bases and those conditions; and applicable human recovery actions.

3. A list of major contributors to those accident sequences selected using the Generic Letter screening criteria. Major contributions such as those from front-line systems or functions and support states, as well as contributions from unusually poor containment performance, are important for inclusion. Also include an estimate of total core damage frequency.

4. A thorough discussion of the evaluation of the decay heat removal function because the adequacy of the decay heat removal capability at the plant for severe accident situations is to be resolved within this examination program. Plants with marginal feed-and-bleed capability should particularly address the capability of the plant to remove decay heat for loss of all feedwater events.

5. A list of vulnerabilities identified by the review process, a concise discussion of the criteria used by the utility to define vulnerabilities, and the fundamental causes of each vulnerability. Vulnerabilities associated with the decay heat removal function should be specifically highlighted.

6. Identification of sequences that, but for low human error rates in recovery actions, would have been above the screening criteria. Important human recovery actions for which greater credit than 1 error in 10 is claimed should be discussed.

7. If applicable, and a plant-specific PRA was performed, include a discussion of other evaluations regarding the unresolved safety issues (USIs) and generic safety issues (GSIs) that have been assessed, including a discussion of the technical basis for resolutions proposed by the licensee for any USI or GSI.

2.2 Back-End Submittal: Containment Response

The IPE analysts must keep in mind the main objective of performing the back-end study. The primary objective is to provide the utility with a framework for obtaining an understanding of and appreciation for containment failure modes, the impact of phenomena and plant features, and the impact of operator actions. The second objective is to segregate out, over a broad spectrum of credible accidents, specific vulnerabilities associated with containment and containment mitigating systems. By achieving these objectives, an appreciation of procedures, mitigating system performance, and mitigating system resources (e.g., electrical power, water, instrument air) will be achieved. In some accident scenarios, specific vulnerabilities may be reduced or eliminated by enhancing procedures or improving mitigating system performance. In other scenarios, a release of radioactive material may be delayed or reduced allowing for effective evacuation and reduction in offsite risk. For all scenarios, the back-end analysis should focus on these benefits and allow for the evolution of an effective accident management program.

2.2.1 General Methodology

The general methodology for containment response has been described in Appendix 1 to the Generic Letter (Ref. 1). Additional, potentially important material may be found in the Containment Loads Working Group Report (Ref. 9), the PRA Procedures Guide (Chapter 7 of Ref. 4), and draft NUREG-1150 (Ref. 10) and its supporting documents. On phenomenological matters, the current status of the NRC position versus that of IDCOR is summarized in a series of so-called "issue" papers (Ref. 11). The methods used by utilities for IPEs are expected to be consistent with the staff's positions in those position papers. Regarding the probabilistic treatment of phenomenological uncertainties, some additional material may be found in the Peer Review of Draft NUREG-1150 (Ref. 12).

2.2.2 Specific Guidelines

In order to facilitate and ensure a high-quality review process, each submittal should be organized in major sections as follows (note Table 2.1):

1. Plant Data
2. Plant Models and Methods for Physical Processes
3. Bins and Plant Damage States (Interface with Front-End Assessment)
4. Containment Failure Characterization
5. Containment Event Trees
6. Accident Progression
7. Radionuclide Release Characterization

2.2.2.1 Plant Data

Identify and highlight component, system, and structure data that may be of significance in assessing severe accident progressions. Additional consideration should be given to equipment whose operability is desired during exposure to harsh environments. Describe systems such as fan coolers or sprays that are important to operation during a severe accident. This description should extend to the reactor building or auxiliary building if appropriate. The utility has the option of submitting a concise set of the plant data that is relevant to severe accident phenomenology or an identification of those containment features that are unique to the facility in question relative to the similar plant that was the subject of previous PRAs such as those for NUREG-1150 (Ref. 10). In addition to the appropriate narrative explanations and sketches, this information should be summarized in a convenient tabular form.

The assessment of the "significance" of such unique features will, of course, have to be judgmental and based upon the understanding of severe accident phenomena, and associated containment challenges, developed in the submittal. For example, debris bed coolability depends strongly on such plant features as available spread area within the cavity and water availability in it. Both aspects are highly individualized even among plants of the same type; thus, an accurate but convenient representation of such plant features would be needed.

The process of providing sufficient plant data gets more complicated when considering mechanisms that are incompletely understood. For example, it is agreed that phenomena associated with high-pressure melt ejection depend strongly on the characterization of the vessel's lower head, the sizes of the flow paths within and out of the reactor cavity, and the lower subcompartment geometry, although actual relationships to resulting containment loads are lacking. Similarly, the potential for non-uniform distribution of combustible gases in the containment air space is clearly related to geometry and location, composition, and intensity of release; however, little basis exists for judging which are important features and the extent of their impact on mixing. It is requested, therefore, that accurate but simple representations of containment geometry be made in this section in as complete a fashion as possible so as to cover the needs in the two examples mentioned above and possibly other situations as they might arise in the submittal's treatment of phenomenology. While blueprints are not necessary, drawings that accurately display the location and rough dimensions of components, systems, and structures that are important for accident progression assessment should be included.

2.2.2.2 Plant Models and Methods for Physical Processes

Provide concise documentation of all analytical models, including selection of empirical factors and data inputs, used in charting out the various potential paths of accident progression. Well-known codes and published models, or even widely accepted results on particular aspects of the phenomenology, may be incorporated simply by reference. To the extent that accepted results can be used, the utility can gain the insights about physical processes without the effort of _de novo_ analysis and without extra review by the staff. For example, if the utility chooses to use CORCON for core-concrete interactions, it can do so provided reference is given to the specific modification to CORCON that is used. General assumptions used in the modeling of phenomenology are just as important as the models themselves and therefore should be fully described. Organization should be such that all particular results quoted in subsequent sections can be referred conveniently to respective analytical models of this section. Clearly, fully integrated analytical tools may not be necessary; however, it is important that the composing of overall accident behavior from separate effects analyses be clearly delineated.

2.2.2.3 Bins and Plant Damage States

As in standard methodology, the coupling of the front-end analysis to the back-end is through the binning of the multitude of front-end sequences into a few groups of damage states with similar back-end characteristics. It is important that the bins be justified on the basis of such factors as timing of important events or operation (or non-operation) of key features. Also, the state of the various systems and components, as deduced from the detailed front-end analysis, should be accurately translated into the back-end plant damage states considered. The impact of severe accident phenomena on the operability of such systems and components must be reflected where appropriate.

Accordingly, this section, in a manner consistent with the binning guidelines of Section 2.1.3 (items 5 and 6), should concisely cover or reference the methodology and results of binning, as well as the actual procedures employed--particularly front-end/back-end team interactions. Further, all front-to-back-end sequence interfaces (i.e., reactor coolant system and containment thermal-hydraulic conditions, containment mitigation system availability, support system availability, human factor assumptions) need to be concisely documented and the adopted binning needs to be justified. Care should be taken to have different bins for sequences that will progress under different assumptions regarding physical phenomena, for example, high reactor coolant system pressure versus low reactor coolant system pressure or different timing--slowly developing or fast.

Recent studies, such as NUREG-1150 (Ref. 10), have stressed the importance of mission times, inventory control (of such resources as instrument air or battery power), and dual usage (e.g., when the condensate storage tank supplies water for both vessel injection and containment sprays, early injection may deplete the water so that it is not available for sprays). Therefore, for the screened sequences, it is important that the impact of mission times, inventory control, and dual usage be discussed with respect to the progression of the accidents, the estimated frequencies, and the binning process.

2.2.2.4 Containment Failure Characterization

This section should provide the results of structural calculations or comparisons with structural calculations for other plants of similar design performed to assess containment strength and the magnitude of various loads necessary to fail containment, e.g., static pressure, localized heat loads, and localized dynamic pressures. A sample list of potential containment failure modes and mechanisms is provided in Table 2.2; these have been considered in the final version of Reference 10. Other failure mechanisms may be appropriate for specific designs. Some of the modes in Table 2.2 are more important for some containment designs than for others. If the analysts choose to incorporate results obtained previously for other containments, it is important to provide a concise rationale of their applicability. The vulnerability of containment

Table 2.2 Potential containment failure modes and mechanisms.

Direct bypass
Failure to isolate
Vapor explosions
Missile generation
Quasi-static pressure rise
Overpressurization
Steam
Noncondensible gases
Combustion processes (hydrogen, carbon monoxide, methane)
Blast
Quasi-static pressure rise
Core-concrete interaction
Basemat penetration
Structural failure and tearout of penetrations
Blowdown forces
Vessel thrust force
Meltthrough
Direct contact of containment shell with fuel debris
Thermal attack of containment penetrations

penetrations to thermal attack is discussed in Reference 13. The licensee submittal should include an assessment of the penetration elastomer seal materials and their response to prolonged high temperatures. Particular attention should be paid to seals in areas where standing hydrogen flames are likely.

In each case, potential failure locations should be identified together with respective failure sizes.

Finally, an assessment of failure size and location should be made for any other structures within which radionuclide retention will be considered (e.g., the reactor building in BWRs).

2.2.2.5 Containment Event Trees (CETs)

It is important to note that this section is closely coupled to the following section (2.2.2.6), "Accident Progression and CET Quantification." Not only does Section 2.2.2.6 quantify the split-fractions for the CETs; but, depending on the results of the accident progression analysis, it could dictate the structure of the CETs themselves.

All functional accident sequences (represented now by plant damage states or bins) that meet the Generic Letter screening criteria should be represented by CETs according to standard practice. Helpful guides and standard practice concerning the structure and methods of analysis of CETs can be found in a number of Level 2 PRAs such as those for Oconee (Ref. 14) and Seabrook (Ref. 15).

2.2.2.6 Accident Progression and CET Quantification

The submittal should present all significant containment loads referenced to events or sequences of the CETs. Significant loads are those with potential to challenge containment integrity. In this interpretation, the containment boundary should be taken to include any interface with a more or less direct access to the outside (e.g., primary to secondary pressure boundary, drywell shell in Mark I).

The presentation should be systematic, i.e., damage state by damage state, and each predicted load should be adequately supported by reference to either:

1. A particular model presented in Section 2.2.2.2 or
2. A previously published (i.e., referenceable) analysis.

In the latter case, applicability would be established through comparison of geometry and thermal-hydraulic conditions. Appendix 1 to the IPE Generic Letter (Ref. 1) provides guidance for assessing containment loads. NRC-sponsored calculations of containment loads that take into account certain phenomenological and containment loading issues can be found in Reference 10. In any event, selected pressure and temperature histories for representative CETs should be displayed graphically for the containment compartments and other building compartments of interest.

On the basis of the above and any additional pertinent analyses, this section continues with quantification of the CETs. In the quantification of the CET, human intervention would be based on existing emergency operating procedures and assessed against standards for human performance. If existing emergency operating procedures (EOPs) are used in controlling or ameliorating the outcome of the accident, the submittal should state that these EOPs are operational and that the requisite amount of training has been completed. Documentation should be provided for the availability and survivability of systems and components with potentially significant impact on the CET or the radionuclide release. The equipment environment should be assessed with the same temperature, pressure, humidity, and radiation environment predicted as part of the accident progression analysis. The utility is to pay particular attention to equipment vulnerability and survivability. If containment sprays, for example, are operating to remove heat and washout radionuclides, the utility should assess the capability of the system to perform its function for the allotted time under the expected environmental conditions. Time is an important consideration, especially for accident sequences that do not fail the containment early in the sequence. Additional detail may be required to justify time and component reliability during such harsh environmental conditions. Reference 16 provides additional information and insights into potential risk-significant equipment qualification issues.

A description should be given of information used in determining the conditional probability that the containment is not isolated, given a core melt accident, including capability, testing, trip signals, overrides, diagnostics, and, of course, experience. (This is the so-called "beta failure mode" for containments as used in PRAs.) In addition to the conditional probability, a description of the size and characterization of the isolation failure should be included.

A description should be given of the assessment of accident sequences that bypass the containment (interfacing system LOCA). Reference 3 discusses the plant features that have been found to be important.

Finally, this section should make clear the methods employed for handling uncertainties in this quantification. The staff recognizes that there are significant unresolved phenomenological uncertainties associated with the quantification of containment event trees. The purpose for conducting an uncertainty analysis is to avoid the masking of potential vulnerabilities due to technically unsupportable assumptions regarding the likelihood of certain phenomena. The uncertainty analysis may be either quantitative or qualitative. The submittal should describe the analysis in sufficient detail to gain the staff confidence that phenomenological and other uncertainties have been properly accounted for in the identification of candidate plant improvements.

2.2.2.7 Radionuclide Release Characterization

Quantification of the CETs will produce estimates of the probability and mode of containment failure for the various plant damage states identified. By combining the frequencies of the plant damage states with the probabilities of the various failure modes, the frequencies of containment failure or bypass can be determined. If a functional sequence is found to have a core damage frequency and containment performance that exceed the screening criteria of the Generic Letter, the magnitude of the radionuclide release should be estimated.

This may be done by selection of source terms for similar sequences that have been identified for an appropriately similar plant or by code calculation. References 17 and 18 contain calculations that provide source term information.

Whatever approach is used, concise documentation should be provided. If a code is used, it should be referenced and the input assumptions provided. If a large number of source term calculations are combined into a set of release categories, the rationale for the process should be provided. If sequences are binned prior to calculating a single source term for a representative sequence in the bin, the rationale for the binning process should be provided.

The staff encourages assessment of accident management issues concurrent with performing the IPE since the results of the IPE will be a major source of information for use by the utility in developing its accident management program. For instance, the inventory of radionuclides residing in areas to which personnel may need access (e.g., reactor building, auxiliary building) may be accounted for and presented in summary fashion, including the chemical form, decay chain characteristics, and estimates of the final locations of all radionuclides. This assessment is not meant for calculating a risk profile, but rather for providing the technical bases for the utility to judge the effect of the radiological source term on the systems and personnel that should operate in that area.

The section should conclude with the ranking of sequences on the basis of both their conditional and total (i.e., including front-end results) probabilities of radionuclide release. These results should be compared with the criterion for radioactive material release provided in Appendix 2 of the Generic Letter (Ref. 1).

2.3 Submittal on Unique Safety Features and Potential Plant Improvements

On the basis of the understanding developed through the IPE, the utility should develop and document in this section a list of any safety features that are believed to be unique to the facility. Among the family of unique features would be those features that resulted in significantly lowering what are considered to be high-frequency core melt sequences or accident progressions in contemporary PRAs for similar plants.

The utility should document any strategies to further prevent or mitigate the detrimental effects of severe accidents that were developed as part of the IPE process and for which credit has been taken in the analysis. For the vulnerabilities from the functional sequences, identify potential improvements, if any, including equipment changes as well as changes in maintenance, operating and emergency procedures, surveillance, and training programs that have already been implemented or have been selected for implementation. Include a discussion of the anticipated benefits in terms of the vulnerabilities addressed; downside considerations should also be addressed.

For those potential improvements that would only be under consideration because of the unresolved generic phenomenological issues in the NRC Containment Performance Improvements Program (for example, an improvement that would only be justified if direct containment heating caused early containment failure), the staff has made clear in the Generic Letter that the industry will not be placed in a position of having to implement improvements before all containment performance decisions have been made.

Thus those improvements may remain potential candidates only. However, consistent with the IPE Generic Letter, the submittal should "...develop strategies to minimize the challenges and the consequences such severe accident phenomena may pose to the containment integrity and to recognize the role of mitigation systems while awaiting their generic resolution."

Describe the rationale used to select options for implementation. Provide, in tabular form, which options have been scheduled for implementation and the respective timing of implementation. If all the alternatives have been dropped from further consideration because of the high cost, discuss how less expensive alternatives were sought.

2.4 IPE Utility Team and Internal Review

The basis for the request in the Generic Letter (Ref. 1) for involvement of utility staff in the IPE review is the belief that the maximum benefit from the performance of an IPE would be realized if the utility's staff were involved in all aspects of the examination; that involvement would facilitate integration of the knowledge gained from the examination into operating procedures and training programs. Thus the submittal should describe utility staff participation and the extent to which the utility staff was involved in all aspects of the IPE program.

The Generic Letter requests that each utility conduct "...an independent in-house review to ensure the accuracy of the documentation packages and to validate both the IPE process and its results." The staff requests that this review team be in-house, that is, made up, to the extent practical, of utility personnel not directly involved in conducting the IPE. The submittal should contain, as a minimum, a description of the internal review performed, the results of the review team's evaluation, and a list of the review team members and their backgrounds.

The maximum benefit to the utility would occur if the combination of persons involved in the original analysis and in-house review, taken as a group, provide both a cadre of utility personnel to facilitate the continued use of the results and the expertise in the methods to ensure that the techniques have been correctly applied.

2.5 Consideration of External Events

The IPE Generic Letter (Ref. 1) states that examination of external events will proceed separately and on a later schedule from that of the internal events. Because of this, no reporting for external event analysis is required at this time. However, it is prudent for the utilities to properly retain documents and plant-specific data relevant to external events such that they can be readily retrieved for future external event analyses. This minimizes the need for a second performance of similar tasks and allows maximum utilization of the internal event analysis, models, and data.

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Based on the Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, the performance of a plant examination is required by the licensee of each nuclear power plant. The plant examination looks for severe accident vulnerabilities and cost-effective safety improvements that would reduce or eliminate any discovered vulnerability. This document delineates the guidance for reporting the results of a plant examination.

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