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Simplified Approach for Estimating Large Early Release Frequency*

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Abstract

The U.S. Nuclear Regulatory Commission (NRC) Policy Statement related to Probabilistic Risk Analysis (PRA) encourages greater use of PRA techniques to improve safety decisionmaking and enhance regulatory efficiency. One activity in response to this policy statement is the use of PRA in support of decisions related to modifying a plant's current licensing basis (CLB). Risk metrics such as core damage frequency (CDF) and Large Early Release Frequency (LERF) are recommended for use in making risk-informed regulatory decisions and also for establishing acceptance guidelines. This paper describes a simplified approach for estimating LERF, and changes in LERF resulting from changes to a plant's CLB.

1 Introduction

The August 1995 U.S. Nuclear Regulatory Commission's (NRC's) Policy Statement [1] related to Probabilistic Risk Analysis (PRA) encourages greater use of PRA techniques to improve safety decision making and enhance regulatory efficiency. One activity in response to this policy statement is the use of PRA in support of decisions related to modifying a plant's current licensing basis (CLB). Draft Regulatory Guide DG-1061 [2] includes staff guidance for using risk information from plant specific PRA study findings and insights, as well as risk metrics such as core damage frequency (CDF) and Large Early Release Frequency (LERF; defined in Reference [2]) in making risk-informed regulatory decisions. In those cases where only the Level 1 PRA analysis is available, Appendix B of Draft DG-1061 provides a simplified approach to estimate LERF, and change in LERF resulting from changes to a plant's CLB.

It was decided to test the guidance in Appendix B of Draft DG-1061 by performing several case studies. One objective of the case studies is to use information from the Level 1 PRAs documented in Individual Plant Examinations (IPEs) together with the simplified approach to estimate LERF for several different plants. These estimates were in turn compared against LERF estimates obtained directly from the Level 2 results reported in the IPEs for the same plants. The purpose of these comparisons was to identify causes for discrepancies, especially in cases where guidance leads to underestimation or significant over estimation of LERF. It was also decided to extend the scope of Draft DG-1061 guidance to include external events and modes of operation

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other than full power. Appendix B of Draft DG-1061 was therefore expanded and published as a draft NUREG/CR Report [3]. The approach described in Reference [3] includes lessons learned from the case studies and expands the scope of the guidance. Public comments received on Draft DG-1061 that are pertinent to the simplified approach have also been incorporated into Reference [3].

2 Objective

This paper summarizes the approach described in detail in Reference [3] for estimating the frequencies of large and early releases of radioactivity from a nuclear power plant. The objective is to develop a relatively simple approach that can be interfaced with a Level 1 PRA with a minimum of additional work. The approach uses simplified containment event trees (CETs) to process information obtained from a Level 1 PRA into an estimate of LERF.

3 Approach

The interface between the accident sequence information obtained from a Level 1 PRA and the simplified CETs is crucial and depends, to some extent, upon the system model used. When a large fault tree approach is used, different cutsets of the same accident sequence may have a different impact on the progression of the accident. Therefore, the information associated with each cutset has to be used along with the sequence definition to respond to questions in the simplified event trees. After the dominant cutsets of a sequence are considered, it can be determined if all cutsets of the sequence have the same response to the questions. If not, it is possible to develop basic event based rules that can be applied to the remaining cutsets to determine the response to the questions. If a large event tree approach is used, the sequence definition should provide sufficient detail to quantify the questions in the simplified event tree. However, with a large number of sequences to be considered, it may also be necessary to determine rules based on the sequence logic to apply to the sequences. Once the interface between a Level 1 PRA and the simplified approach has been established, the accident sequence information can be processed through the containment event trees (CETs). However, not all of the information necessary to quantify the event trees is available from a Level 1 PRA. Reference [3] indicates, in detail, what information can be expected from a Level 1 PRA versus what will have to be generated as part of this approach. Five simplified CETs, have been developed to process the Level 1 PRA results. The following five types of plants are represented by these CETs:

- Pressurized water reactors (PWRs) with large volume and subatmospheric containments
- PWRs with an ice condenser containment
- Boiling water reactors (BWRs) with a Mark I containment
- BWRs with a Mark II containment
- BWRs with a Mark III containment

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Each accident sequence is allocated to a risk category based on the status of the plant. The intent is that the split fractions for most of the questions in the trees will be determined from plant-specific accident sequences and plant characteristics. The CETs include recommended split fractions only for questions related to the likelihood of early containment failure.

4 Scope and Limitations

The simplified CETs presented in Reference [3] use information provided in a Level 1 PRA to estimate LERF. The trees are structured to interface with Level 1 PRAs for accidents initiated during full power operation and other modes of operation. Accidents initiated by events internal to the plant and external events (such as seismic) can also be processed through the event trees.

The simplified CETs are based on the results of severe accident research performed over the last several years. The CETs were constructed to capture the most important characteristic of severe accident progression that influence the potential for early containment failure or bypass. This focus on estimating early loss of containment integrity allows significant simplification of the CETs but also means that later modes of containment failure are generally not estimated. The exception is those accident sequences that are initiated by external events which can impede evacuation of the population. Under these circumstances, it is possible that late containment failures could result in early health effects.

As noted above, most of the questions will be determined from information provided in the Level 1 PRA supplemented by additional analysis and information. The CETs include split fractions only for questions dealing with the likelihood of containment failure. These split fractions are intended to encompass the likelihood of containment failure for most plants in each of the five containment types. Consequently, the split fractions are somewhat bounding in nature and consequently should only be used as a first step scoping study to determine the change in LERF caused by the change in the CLB. If the change in LERF is negligible (defined in Reference [2] in terms of changes to CDF and LERF), then a more accurate estimate of LERF is not warranted. However, if the change in LERF is not negligible, then the proximity of LERF to the decision criteria established in Reference [2] has to be determined. If the estimated LERF is significantly below (about an order of magnitude or more) the acceptance guideline then (depending on the magnitude of the change in LERF) expenditure of additional resources to obtain a detailed Level 2 model and a more accurate estimate of LERF may not be warranted. However, if the LERF estimated from this simplified approach is close to or larger than the acceptance guideline, further analysis may be necessary to obtain a more accurate LERF for the purpose of risk-informed decisionmaking.

5 Simplified Event Trees for PWRS

Figure 1 presents a simplified CET for PWRs with large volume or subatmospheric containments that allows allocation of accident sequences to one of two categories (i.e., large early release or no large early release). A similar CET has been developed for PWRs with ice condenser containments and is presented in Reference [3]. A CET for late containment failure is also given in Reference [3].

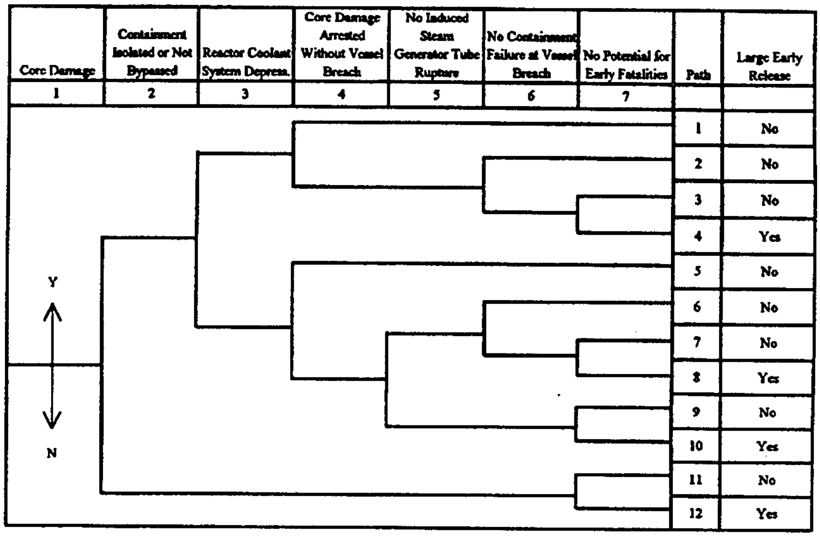

Figure 1 PWR Large Dry Containments

Each accident sequence in a Level 1 PRA would be allocated to one of the categories in Figure 1 based on the plant status as defined by the various accident sequences. The first question deals with the status of containment integrity at the start of the accident (i.e., bypassed or not isolated). The pressure in the reactor coolant system has been found to strongly influence the progression of a severe accident (e.g., core damage arrest prior to vessel breach (Question 4), induced steam generator tube rupture (Question 5), direct containment heating at vessel breach (Question 6), etc.); hence, Question 3 is included in the CET. Question 6 addresses the likelihood of containment failure at vessel breach. The split fractions recommended in Reference [3] for this question reflect a reasonable estimate of the likelihood of early containment failure for large-volume containments given a high-or-low pressure core meltdown accident. Question 7 deals with the potential for a large release to cause early fatalities. The potential for early fatalities depends on the magnitude and timing of the radionuclide release. The magnitude of the release is important because there is a dose threshold below which early fatalities will not occur. The timing of release is important because of radionuclide decay and because of its influence on evacuation of the close-in population around a nuclear power plant.

6 Simplified Event Trees for BWRs

Figure 2 provides a simplified CET which allows allocation of accident sequences to one of two consequence categories for BWRs with Mark I containments. Similar CETs have been developed for BWR Mark II and Mark III containments (refer to Reference [3]). A CET for late containment failures is also presented in Reference [3].

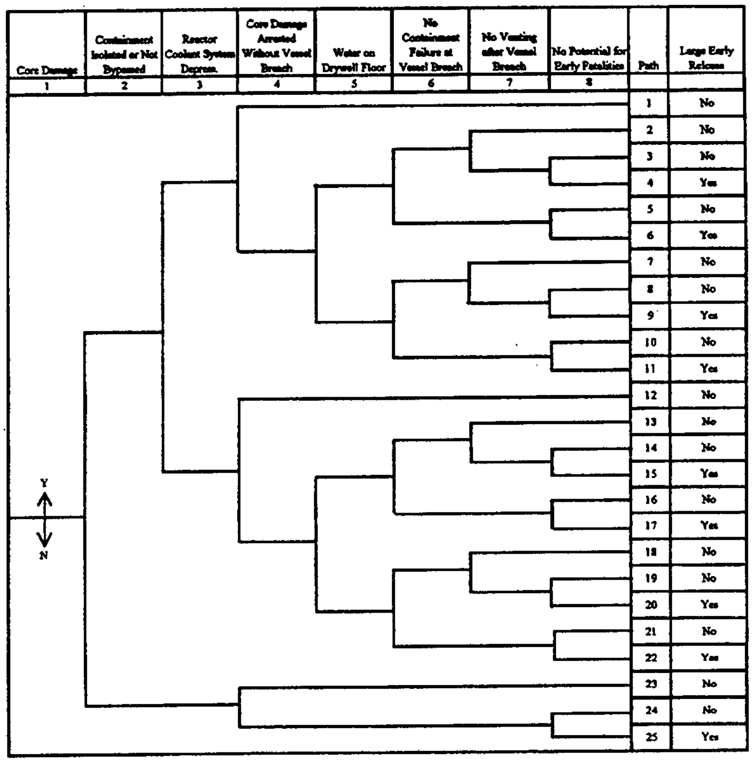

Figure 2 BWR Mark I Containments

The structure of the BWR event tree is similar to the PWR CET. The initial questions again deal with containment integrity, RCS depressurization, and core damage arrest. However, Question 5 (water on the drywell floor) was added to the CET because liner meltthrough by molten core debris has been found to be an important failure mode for Mark I containments and the presence of water can significantly influence the likelihood of failure by this mechanism. Containment venting is also important for Mark I containment, hence, the addition of Question 7 in Figure 2. Finally, Question 8 deals with the potential for early fatalities. The response to this question for BWRs is complicated by the premise that all early releases that are scrubbed by the suppression pool are sufficiently low that by themselves they will not result in individual early fatality risk. Hence, if an early failure occurs with the functionality of the suppression pool intact, it is assumed that the early scrubbed releases will not pose an early fatality threat to the population within one mile of the plant boundary, and that this population will evacuate before substantial core concrete interaction releases or late iodine releases from pools are of a magnitude to cause individual early fatality risk.

7 Summary

This paper has described a simplified approach designed to supplement Level 1 PRAs submitted in support of risk-informed decisionmaking. The intent is to use accident sequence information provided in the Level 1 PRA to estimate the frequencies of LERF. The advantage of this approach is that it allows LERF to be calculated very quickly, though approximately, without the need for performing a detailed Level 2 PRA. The intent is to use this approach to initially estimate LERF. If the estimated LERF is significantly below the acceptance guideline or the change in LERF is negligible, then further analysis may not be necessary. Finally, the approach presented in this paper and described in more detail in Reference [3] is under review and therefore may be revised in the final NUREG/CR Report.

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