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Phenomenological uncertainty analysis of containment building pressure load caused by severe accident sequences

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ABSTRACT

This paper illustrates an application of a severe accident analysis code, MAAP, to the uncertainty evaluation of early containment failure scenarios employed in the containment event tree (CET) model of a reference plant. An uncertainty analysis of containment pressure behavior during severe accidents has been performed for an optimum assessment of an early containment failure model. The present application is mainly focused on determining an estimate of the containment building pressure load caused by severe accident sequences of a nuclear power plant. Key modeling parameters and phenomenological models employed for the present uncertainty analysis are closely related to the in-vessel hydrogen generation, direct containment heating, and gas combustion. The basic approach of this methodology is to (1) develop severe accident scenarios for which containment pressure loads should be performed based on a level 2 PSA, (2) identify severe accident phenomena relevant to an early containment failure, (3) identify the MAAP input parameters, sensitivity coefficients, and modeling options that describe or influence the early containment failure phenomena, (4) prescribe the likelihood descriptions of the potential range of these parameters, and (5) evaluate the code predictions using a number of random combinations of parameter inputs sampled from the likelihood distributions.

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1. Introduction

A level 2 probabilistic safety analysis (PSA) is used to assess the performance of the containment in mitigating severe accidents. The analysis includes an evaluation of the accident progression in the containment; an estimation of the timing, location, and mode of containment failure; and an estimation of the source term characteristics. A typical containment performance analyses have made use of a containment event tree (CET) modeling approach, to model the containment responses by depicting the various phenomenological processes, containment conditions, and containment failure modes that can occur during severe accidents. A CET predicts the accident sequence progression from the core melt to radionuclide release into the environment. The CET is constructed in sufficient detail to address the important phenomena that significantly affect the containment integrity and radiological source term. The Accident Progression Event Tree (APET) in the Surry plant of NUREG-1150 (USNRC, 1990), which is equivalent to the CET, uses seventy-one questions to describe all the phenomena and operator actions during the severe accident progression within the containment for a Surry plant. However, the CET should not be so detailed as to be inscrutable, and the number of top questions in the CET can be reduced by introducing a supporting tree for each top event in the CET. The details for each top event can be considered in the supporting tree. Fewer than twenty top events are sufficient to describe the accident progression inside the containment with the use of the decomposition event tree (DET). Detailed phenomena or operator actions for the top events in the CET are treated in the DET. The ultimate strength of the plant-specific containment for the static load inside the containment is evaluated in this step. Furthermore, the information of the pressure loads for a given accident sequence are needed to estimate the likelihood of containment failure.

A level 2 PSA of OPR-1000, which is the reference plant of this analysis, have made use of a CET modeling approach, where a general approach in the quantification of a small event tree is to use DET to allow a more detailed treatment of the top event. A quantification of the physical phenomena in the DET is achieved based on results obtained by validated code calculations or expert judgments. The phenomenological modeling in the event tree still entails a high level of uncertainties. Such uncertainty exists because of our incomplete understanding of reactor systems and severe accident phenomena.

This paper illustrates an application of a severe accident analysis code, MAAP (Fauske and Associates, 2005), to the uncertainty

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evaluation of an early containment failure DET, which is one of the CET top events in the reference plant of this study. An uncertainty analysis of a containment pressure behavior during severe accidents has been performed for the optimum assessment of an early containment failure model. The MAAP code is a system level computer code capable of performing integral analyses of potential severe accident progressions in nuclear power plants, whose main purpose is to support a level 2 probabilistic safety assessment or severe accident management strategy developments. The code quantitatively predicts the evolution of a severe accident starting from full power conditions given a set of system faults and initiating events through events such as core melt, reactor vessel failure, and containment failure. A key element tied to using a code like MAAP is an uncertainty analysis (Roberts and Sanders, 2013). The code employs lots of user-options for supporting a sensitivity and uncertainty analysis. The present application is mainly focused on determining an estimate of the containment building pressure load caused by severe accident sequences. Key modeling parameters and phenomenological models employed for the present uncertainty analysis are closely related to an in-vessel hydrogen generation, direct containment heating, and gas combustion.

2. Analysis methodology

The basic approach of this methodology is to (1) develop severe accident scenarios for which the containment pressure loads should be performed based on a level 2 PSA, (2) identify severe accident phenomena relevant to an early containment failure, (3) identify the MAAP input parameters, sensitivity coefficients, and modeling options that describe or influence the early containment failure phenomena, (4) prescribe likelihood descriptions of the potential range of these parameters, and (5) evaluate the code predictions using a number of random combinations of parameter inputs sampled from the likelihood distributions. This method of characterizing uncertainty in the reactor accident progression is similar to the method used by Gauntt (2005), where the MELCOR code was used. To limit the number of "realizations" (code calculations) needed to characterize the full range of uncertainty, the Monte Carlo Sampling method is used to sample the input parameter distributions.

To quantify the uncertainties addressed in the MAAP code, a computer program, MOSAIQUE (Lim and Han, 2009), has been applied, which was recently developed by the Korea Atomic Energy Research Institute. The program consists of fully-automated software to quantify the uncertainties addressed in the thermal hydraulic analysis models or codes. The MOSAIQUE employs a methodology of sampling-based uncertainty analysis using thermal hydraulic or severe accident analysis codes (Ahn et al., 2002; Helton and Davis, 2002; Crécy and Bazin, 2007). The Korean standardized nuclear power plant, an OPR-1000, has been selected as a reference plant for this analysis.

2.1. Development of DET scenarios for the early containment failure

An early containment failure is defined as a failure of the containment shortly before, at, or soon after a reactor vessel failure. An early containment failure can potentially result from a combination of the energetic processes and events that may occur at a reactor vessel breach. To evaluate the early containment failure, the total pressure inside the containment should be calculated. In addition to the base pressure and reactor coolant system (RCS) blow-down pressure, the rapid pressurization owing to rapid steam generation in the cavity, direct containment heating (DCH), hydrogen burn, and containment spray system (CSS) operability have been considered. Eventually, the probability of a containment failure and its failure mode will be calculated using the containment fragility curve, which is out of scope of this paper.

The first top heading of an early containment failure DET is the operability of the CSS. If containment heat removal is available by the CSS, the amount of steam in the containment will be decreased. Hence, the base pressure is low but the possibility of hydrogen combustion will be increased. The second concern of the DET is a cavity condition. There is a water flow path from the containment sump level to the reactor cavity in the reference plant. This path allows the cavity to be flooded if the inventory of the refueling water tank is injected into the containment. The third event is the amount of hydrogen produced in-vessel. Two discretized regimes have been selected to represent the uncertainty in the magnitude of in-vessel hydrogen production. The sequences of flooded cavity without the CSS operation are the cases in which the high pressure safety injection (HPSI) or the low pressure safety injection (LPSI) system is working or recovered, where only a high amount of hydrogen generation has been assigned owing to the long-term in-vessel melt progression. The fourth top heading is a fraction of the mass involved in DCH. The fraction of core debris mass that participates in a DCH event is one of the most important parameters impacting the peak pressure associated with a vessel failure. Two discretized levels have been selected to represent the uncertainty in the amount of core debris which fully participates in a DCH event at vessel failure. The fifth top heading asks whether hydrogen burn occurred before or at the reactor vessel failure. The extent of hydrogen combustion at vessel failure is another important parameter impacting the peak containment pressure associated with a vessel failure. Two branches of 'global burn' and 'local burn' are considered. It is believed that all sequences with a high DCH fraction result in sufficient hydrogen combustion (global burn) at vessel failure. However, there is only local burn in the sequence of high DCH fraction with wet cavity and without the CSS operation (LPHPHHN in Table 1) owing to a high amount of steam generation at vessel failure.

In addition to the above severe accident phenomena, an occurrence of a significant ex-vessel steam explosion can be an important factor for the containment failure when the cavity is in a flooded condition, which was not considered in this analysis because it was assumed to fail the containment integrity.

Fourteen scenarios were developed as DET scenarios of an early containment failure. Developed DET scenarios are shown in Fig. 1 and Table 1: six scenarios are cavity flooded cases by CSS operation, two scenarios are flooded cavity by high pressure safety injection (HPSI) without recirculation, and the other six scenarios are dry cavity cases. The loss of offsite power (LOOP) accident sequences and the station blackout (SBO) accident sequences are applied to simulate the wet (flooded) cavity and dry cavity cases, respectively.

2.2. Selection of MAAP modeling parameter and sampling

In the severe accident analysis there were uncertainties in the physical phenomena. There were also uncertainties in the MAAP phenomenological models. Users had control over the uncertainties through the so-called 'model parameters' of the MAAP program. They were either used as an input to a given physical model or to select between different physical models. This feature of the code architecture was included specifically to facilitate sensitivity or uncertainty in the analysis. In this study, input variables assigned as the model parameters to affect the pressure load of containment building during severe accidents were identified, and their uncertainty was characterized using a user specified distribution. These parameters were selected based on MAAP input parameter files.

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