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Evaluation of Severe Accident Environmental Conditions Taking Accident Management Strategy into Account for Equipment Survivability Assessments

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ABSTRACT

This paper presents a methodology utilizing accident management strategy in order to determine accident environmental conditions in equipment survivability assessments. In case that there is well-established accident management strategy for specific nuclear power plant, an application of this tool can provide a technical rationale on equipment survivability assessment so that plant-specific and time-dependent accident environmental conditions could be practically and realistically defined in accordance with the equipment and instrumentation required for accident management strategy or action appropriately taken. For this work, three different tools are introduced; probabilistic safety assessment (PSA) outcomes, major accident management strategy actions, and accident environmental stages (AESs). In order to quantitatively investigate an applicability of accident management strategy to equipment survivability, the accident simulation for a most likely scenario in Korean Standard Nuclear Power Plants (KSNPs) is performed with MAAP4 code. The accident management guidance (AMG) actions such as the reactor control system (RCS) depressurization, water injection into the RCS, the containment pressure and temperature control, and hydrogen concentration control in containment are applied. The effects of these AMG actions on the accident environmental conditions are investigated by comparing with those from previous normal accident simulation, especially focused on equipment survivability assessment. As a result, the AMG-involved case shows the higher accident consequences along the accident environmental stages.

1. Introduction

Although safety-related equipment, both electrical and mechanical, must perform its safety function during design bases events, the equipment necessary for mitigating the severe accident consequences is required to provide a reasonable level of confidence that it will function in severe accident environment for the time span for which it is needed. This requirement is commonly referred to as “Equipment Survivability” and is fundamentally different from “Equipment Qualification (EQ)” which is common terminology used for the level of assurance provided for equipment necessary for design basis accidents. This implies that the equipment survivability assessment may be performed with a practical engineering approach accompanying best-estimated accident environments. Of course, the accident environmental conditions should be determined as harsh or conservative as possible, in compliance with regulatory requirements such as 10CFR50.34(f) (US NRC, 1984) and SECY-93-087 (US NRC, 1993).

In the advanced nuclear power plants where equipment survivability assessment has been performed, the general approach of the following three steps was used. First, the equipment and instrument to be assessed are selected. Second, the accident environmental conditions to which these equipments are exposed, are defined with accident simulation computer codes such as MAAP4 (R.E.Henry et al., 1990) and MELCOR. Third, considering the time span for which each of the equipments is needed, its survivability is determined by comparing its technical specification with accident conditions. Whereas technical specification of the equipments is predetermined in its initial design although it is possible to change or modify it, the prediction of accident environmental conditions has, in essence, a large uncertainty due to complexity of severe accident progression. Therefore, whether an equipment survives under severe accident environment or not, relies on how reasonably accident environmental conditions can be predicted. From this reason, this study proposes an alternative in predicting the accident environmental conditions.

The review of previous studies (ABB-CE, 1995; KHNP, 2002; US NRC, 1999) on the determination of an accident environmental condition, however, indicates that it was determined with representative or likely accident scenarios only. Even given an accident scenario, the characteristics of severe accident mitigation features such as hydrogen igniter or cavity flooding system, was only considered as their initial operability status, i.e. on and off. The operating timings and adverse effects of the equipments used for accident management strategy were not considered during the accident progression. As a result, previous studies did not seem to provide any practical rationale that accident environmental conditions determined in their studies could envelop those from a series or spectrum of accident scenarios for an individual plant. Of course, it is not economical to perfectly examine a spectrum of all possible accident scenarios as for equipment qualification, since it is very time-consuming work.

In case that there is well-established accident management strategy for a specific nuclear power plant, an application of this tool can provide two advantages. First, any countermeasure’s response to mitigate the accident consequence can be reflected as practical as possible. Second, the adverse effects from which a strategy taken to perform a particular safety function can induce overdue environmental loading than those found in usual accident simulation, can be considered. Using a well-established accident management strategy can provide a technical rationale on equipment survivability assessment so that plant-specific and time-dependent accident environmental conditions could be clearly defined in accordance with the equipment and instrumentation required for accident management strategy or action appropriately taken. For this work, three different tools are introduced; probabilistic safety assessment (PSA) outcomes, major accident management strategy actions, and accident environmental stages (AESs). These tools are expected to be consistent with practical engineering approach for equipment survivability assessment. In order to quantitatively verify an applicability of accident management strategy on equipment survivability, as an example, the accident simulation for most likely scenario in Korean Standard Nuclear Power Plants (KSNPs) is performed with MAAP4 code (R.E.Henry et al., 1990). MAAP4 calculation results for an accident scenario selected will quantify the effects of AMG action on the accident environmental conditions, and outcomes of present work are discussed by comparing them with those from usual accident simulation for equipment survivability assessments.

2. Analytical Tools

In the present work, the analytical tools for a determination of accident environmental conditions are composed of three well-known severe accident fundamentals. These are briefly discussed below, where KSNP-oriented descriptions are given in order for comprehensive further discussion.

2.1 Probabilistic Safety Assessment

Conventionally, PSA Level 1 considers only those systems having direct potential to contribute to core damage and Level 2 addresses phenomena related to containment performance and accident progression to core damage. Some Level 2 information is defined by the Level 1 event trees because the systems in question perform a dual function and the operability of the system can be assumed to be the same before and after core uncovery. Since it is, however, necessary to ensure that all pertinent questions about containment system operability and reactor state parameters have been defined prior to containment accident progression analysis, Level 1 event trees have to be extended and are called the plant damage state (PDS) event trees. This PDS concept is used in this study since it includes all the information necessary for assessing the availability of all systems important to containment performance and accident mitigation.

The PDS concept provides a rationale of the selection of accident sequence; which sequence is most probable and challengeable to containment integrity. If several accident sequences of higher PDS priority are considered with respect to core damage occurrence and plant vulnerability, the accident environmental conditions derived from these sequences can be accepted as most feasible and representative in its own plant. For one of KSNPs, the five highest PDS sequences cover approximately 70% of accident sequences.

2.2 Accident Management Guidance (AMG)

An AMG provides the information on the timings of the actions to be taken, if it does not fail. Severe accident management encompasses the actions which could be considered in recovering from a severe accident and preventing or mitigating the release of fission products to the environment. Those actions that would be taken were initially clarified by EPRI and were designated as Candidate High Level Actions (CHLAs). In general, Accident Management Program (AMP) for an individual plant are developed by considering the spectrum of the CHLAs for specific plant type as well as the anticipated effects associated with the implementation of the high level actions at various stages of an accident. The effects that each action would create are, to varying degrees, dependent upon the extent of damage to the core, RCS, and the containment. Under this consideration, AMG for KSNPs has being developed and basically seven accident management strategies are developed (KHNP, 2002); Inject into the steam generator, Depressurize the RCS, Inject into the RCS, Inject into the containment, Control the fission product releases into the environment, Control the containment pressure and temperature, and Control hydrogen concentration in containment. For the equipment and instrumentation needed for accident prevention or mitigation, the timing and the condition that these actions would be taken are very important with respect to their functionabilities. These characteristics can be only determined by strategy performance control logic being established for individual plant.

Figure 1 illustrates the control logic chart for strategy performance for one of KSNPs. This chart is the primary tool used by the Technical Support Center (TSC) severe accident management team to identify immediate and severe challenges to containment fission product boundaries and to select the appropriate mitigation strategy guideline for strategies to respond to the challenge. This chart identifies the severe challenges for all possible plant conditions that may occur following a severe accident where the plant conditions were defined based on the severity of the challenge and capability to take actions to control the conditions in time to mitigate the challenge to the containment fission product boundaries. For example, if the containment pressure exceeds 0.115 MPa (133.6 cmH_2O) as seen in Figure 1, a guideline for containment status control is consulted to evaluate the benefits of the various severe accident management actions which may be used to control this condition. This control logic chart is continuously monitored while the overall set of severe accident guidelines is being used.

2.3 Accident Environmental Stages

After the severe accident initiation, the timings and degrees of the accident progression are dependent upon the initiating events being considered and subsequent failure status of the related safety systems. Under this consequence accompanying severe accident-specific phenomena such as rapid cladding oxidation and melt relocation, any equipment and instrumentation coping with them can be exposed into a variety of accident environmental conditions. For equipment survivability assessments, the discrimination of “accident environmental stages” during severe accident progression is very effective for matching the needed equipment and instrumentation for exposing environmental conditions at that time. The purpose of defining AESs is to identify the time span in the severe accident in which specific equipment is required to perform its function and to facilitate the definition of the environment which challenges the equipment survivability.

At previous equipment survivability assessments for practical plant applications (ABB-CE, 1995; KHNP, 2002; US NRC, 1999), the plant damage states have been defined, but in a rough or simple way; onset of severe accident, before and after the reactor vessel failure. Thus, the accident strategy taken and the equipment and instrumentation needed for accident prevention or mitigation at each stage are not clearly defined. The definition of AESs being proposed by this study is summarized in Table 1.

AES 1 is defined as the period of time in the accident sequence after the accident initiation and prior to core uncovery. The fuel rods are cooled by the water/steam mixture in the reactor vessel, and thus the accident has not yet progressed beyond the design basis of the plant. AES 2 is defined as the period of time in the accident sequence after core uncovery and prior to the onset of significant core damage as evidenced by the rapid oxidation of the core. This is the transition period from design basis to severe accident environment, of which ending time is an entry into accident management strategy. During this stage, the overall core geometry is intact and the uncovered portion of the core is overheating due to the lack of decay heat removal. AES 3 is defined as the period of time in the accident sequence after rapid oxidation of fuel rod cladding occurs and prior to a relocation of core melt into lower head. During this stage, the heat of exothermic reaction from rapid oxidation accelerates the core degradation, melting and relocation of the core.

Most of severe accident management strategies have to be exercised during this stage since the accident, if it is developed into severe accident realm, can go into this stage faster, and occupies most of a lot of time the accident progresses. Therefore, recovery actions in this stage may create environmental challenges by increasing the rate of hydrogen and steam generation. AES 4 is defined as the period of time in the accident sequence after the relocation of core melt into lower head and prior to the reactor vessel failure. And AES 5 is defined as the period of time in the accident sequence after the reactor vessel failure until the establishment of a controlled stable state.

The quantitative severity of accident environmental conditions generated for a specific accident scenario will be compared via accident environmental stages defined in Table 1.

3. MAAP4 Simulation for a Selected Accident Scenario

MAAP4 calculation (R.E.Henry et al., 1990) is performed for Loss of Feedwater (LOFW) sequence which was chosen because it was one of the most probable scenarios from the KSNP PSA results and is a representative in which most of AMG actions can be involved. The lumped-parametric mechanistic code, MAAP4 is applied to predict pressure, temperature, and radiation dose at various locations within the reactor vessel, RCS, and containment where accident management actions are implemented. This sequence includes a loss of main feedwater, a failure of auxiliary feedwater system startup, and the inability to depressurize the RCS due to malfunction of safety depressurization system valve. Both failures of primary and secondary heat removal capabilities result in core damage and heatup. Especially, the safety injection systems do not work since the RCS can not be depressurized until the reactor vessel failure. During the accident progression, it is assumed that a containment spray system is operating to flood the reactor vessel, but a hydrogen mitigation system does not work in order to predict the plant response against AMG actions.

As a base case, the MAAP4 calculation results are summarized in Table 2, in which any AMG action was not involved. Table 2 provides all the useful information for selecting the timings for AMG action being taken and determining accident environmental conditions. Since heat removal of both sides is not available, this sequence shows fast core melt progression so that the reactor vessel failure occurs at about 1.8 hours. Initially, RCS pressure rises rapidly over pressurizer safety valve (PSV) opening setpoint and is maintained at this level through PSV's periodic chattering. This high RCS pressure decreases abruptly at the reactor vessel failure. As described in previous paragraph, any safety injection systems, i.e. high-pressure, low-pressure and safety injection tank (SIT) could not operate until the reactor vessel failure. The SIT water was injected during about 30 seconds just after the reactor vessel failure. The containment temperature and pressure rise during accident progression since mass and energy releases from RCS continue and the containment spray system does not operate. Also, hydrogen concentration in containment rises. Best-estimate predicts that about 56 percent of the active fuel cladding was oxidized during core melt progression, which indicates the uniform hydrogen concentration corresponds to approximately 6.1 percent by volume.

By analyzing accident progression for base case, the AMG actions to be taken are found to be RCS depressurization, water injection into RCS, and hydrogen concentration control in containment. The actions such as water injection into the containment and the containment pressure and temperature control are excluded since the base case assumed the containment spray system (CSS) was initially operable. According to KSNP severe accident management guidances (SAMGs) (KHNP, 2002), the AMG action can be taken after core exit temperature (CET) exceeds 922 K (1,200 ^{o}F), i.e. 3,614 seconds after accident initiation. From this time to reactor vessel failure of 6,456 seconds, AMG actions such as RCS depressurization and water injection into RCS need to be taken with the highest priority (see Figure 1). Hydrogen concentration control in containment can be performed at any appropriate time when its concentration does not exceed 5 % by volume. In this study, the timings were initially set to be after the reactor vessel failure for conservative estimation of accident environmental conditions.

If an AMG action is successful, its consequence can influence the other actions' trials. Based on this judgment, all the AMG actions considered are programmed to start when the operating conditions are met. Figure 2 shows some of MAAP4 calculation results for the base case, which illustrates the available time band for each AMG action considered in this study.

4. Accident Environmental Conditions for Equipment Survivability Assessments

First of all, the action to depressurize RCS has to be taken at any time from an entry on AMG, no later than the reactor vessel failure. A success of this action can provide a merit that it can make it possible to cooldown RCS by injecting water. Figure 3 shows RCS pressure transient when a Safety Depressurization System (SDS) valve opens just after AMG entry point. The performance criteria of SDS valves was initially decided to accommodate RCS overpressure without core uncovery in which, if one valve opens just after the first PSV lifting, the core uncovery did not happen. Similarly, This opening mode was selected based on the verification calculations that RCS pressure at the reactor vessel decreased below the High Pressure Melt Ejection (HPME) threshold pressure (KHNP, 2002). As seen in Figure 3, the RCS pressure decreases rapidly, and, as a consequence, it is expected that the containment pressure and temperature increase (See the next Figures 6 and 7). This may impose additional loads on the containment and threaten the survivability of needed equipment and instrumentation for subsequent AMG actions.

Once the RCS has been depressurized successfully, the water injection into the reactor vessel is accomplished automatically via a safety injection system, if its operational configuration is successful. In this study, a low-pressure safety injection system is assumed to operate. The timing was 3,848 seconds after accident initiation. Moreover, the passive SIT water began to be injected at 5,782 seconds. Figure 4 shows CET response, which indirectly indicates that the violent interaction occurs between degraded core with the injected water. Repeating competent steam generation and water ingression, the reactor goes to a coolable state. Generally, the successful in-vessel water injection, if its timing is not too late, can prevent the reactor vessel failure since the cooling capacity of the safety injection system is adequate for cooling even the damaged and meltdown reactor core. Owing to this action, therefore, the reactor vessel failure has never occurred during the accident simulation. On the other hand, the core support plate was failed at 10,290 seconds, much longer than that in not AMG-involved case.

As specified in previous section, the hydrogen igniters actuate before the hydrogen concentrations reach 5 vol% at anywhere in containment. In this study, this concentration was conservatively assumed to be 4.5 vol%. Under this circumstance, the calculation showed that the first hydrogen burning occurred at annular compartment where the hydrogen generated in RCS releases via Reactor Drain Tank. The timing was 5,106 seconds, which is in a process of violent metal-water interactions due to large steam generation. Figure 5 compares the hydrogen concentrations in upper compartment between the base case and AMG-involved case. The AMG-involved case shows the earlier hydrogen release, which confirms that the former AMG actions have created the better oxidation environment in the RCS. Figure 5 illustrates the hydrogen control capability of the KSNPs definitely.

The effects of AMG actions are reflected on the ongoing and following accident environmental stages. Basically, implementation of AMG actions makes the timings of AESs changed; ending time in AES 3 was changed from 5,228 sec to 10,290 sec, and ending time in AES 4 was extended over 24 hours. They appear to be distinct with respect to containment responses since RCS transients are accommodated on the containment. Figures 6 and 7 showed the containment temperature and pressure resulted from three AMG actions taken. All the actions induce the higher containment temperature and pressure, which can lead to more harsh containment environmental conditions.

Table 3 summarizes accident environmental conditions at various locations for an AMG-involved accident scenario in which those from the base case are also provided. The data shown in Table 3 are maximum values in a given AES. All the AMG actions considered in this study were taken during AES 3 and could terminate the accident during AES 4. As easily seen in Table 3, the AMG-involved case shows the higher accident consequences in the reactor and containment along AES 3. In AES 4, because the corium was confined in the RCS, the harsher environmental condition appears to be limited on somewhere the effects of AMG actions appear.

5. Conclusion

In case that there is a well-established accident management strategy for a specific NPP, a method utilizing this tool in determining severe accident environmental conditions for equipment survivability assessments was proposed. For quantitative verification of this method, MAAP4 calculations were performed for a representative accident scenario with respect to severe accident management. As a result, compared with a base case which corresponds to a usual simulation for equipment survivability assessment, an AMG-involved case produced more harsh environmental conditions. If accident environmental conditions via this method are determined and enveloped for a spectrum of different accident scenarios, plant-specific and time-dependent accident environmental conditions could be defined practically and realistically in consistency with the equipment and instrumentation required for accident management strategy or action appropriately taken.

Acknowledgement

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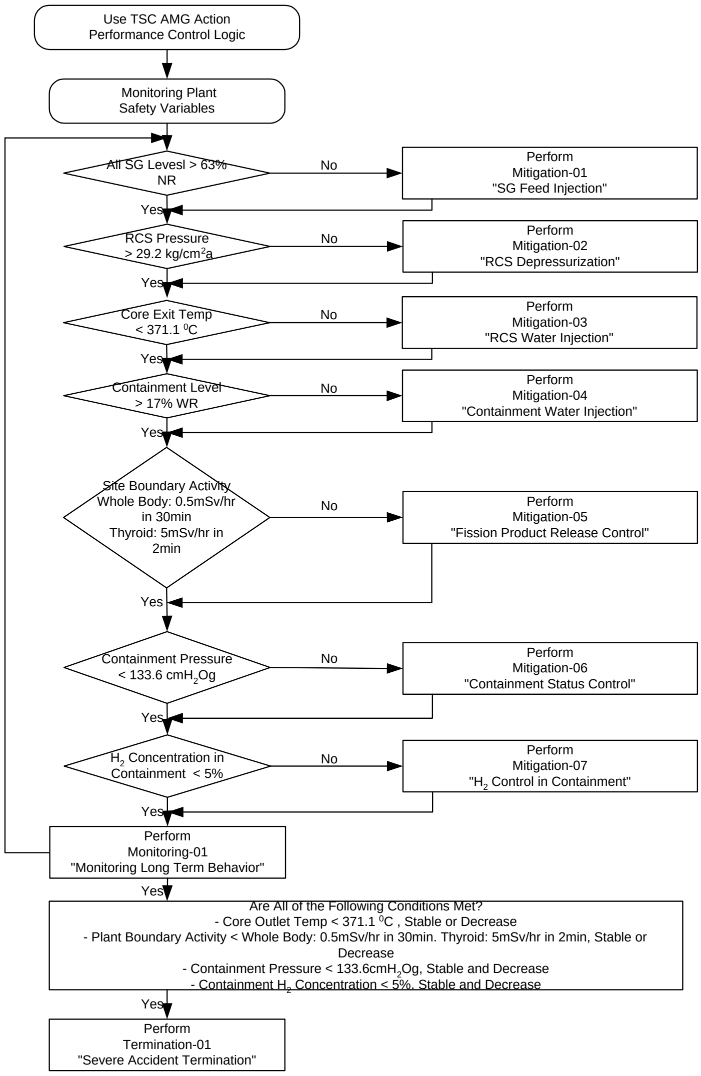
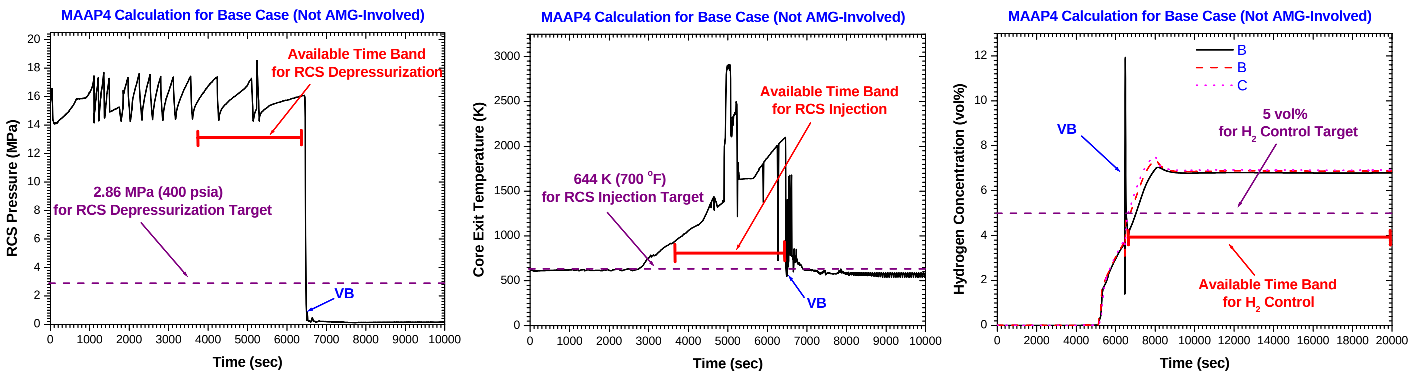


Fig. 1 KSNP's AMG action performance control diagram



(a) RCS Pressure (b) Core Exit Temperature (c) H_2 Concentration

Fig. 2 MAAP4 simulation for a selected accident scenario (Base Case)

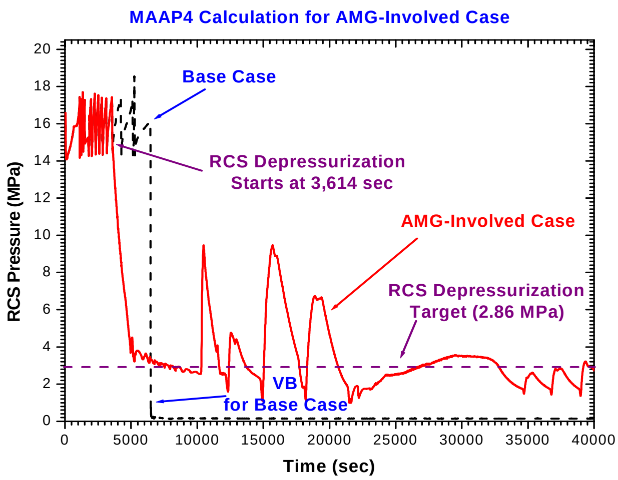


Fig. 3 RCS pressure response for RCS depressurization

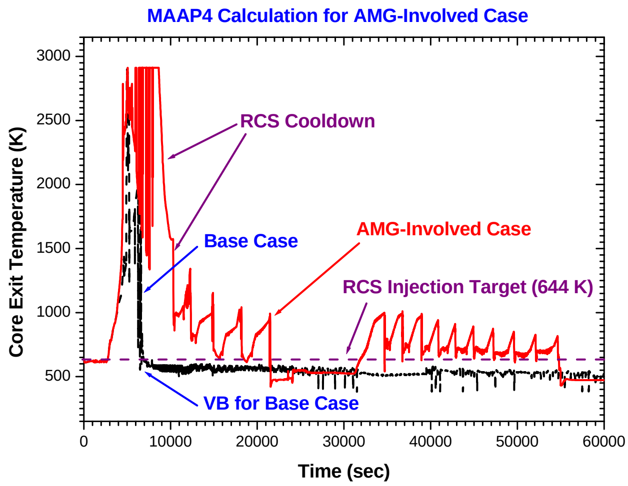


Fig. 4 Core exit temperature response for RCS injection

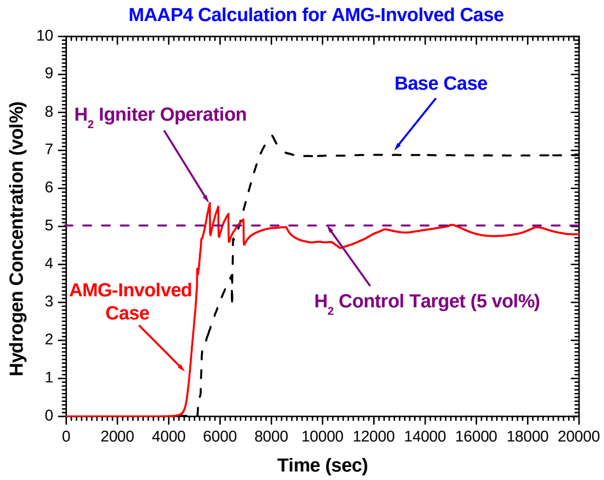
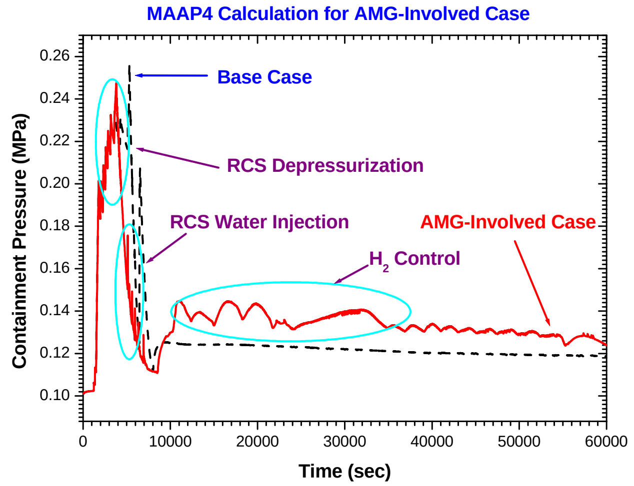


Fig. 5 Hydrogen concentration response resulted from actuation of hydrogen igniters



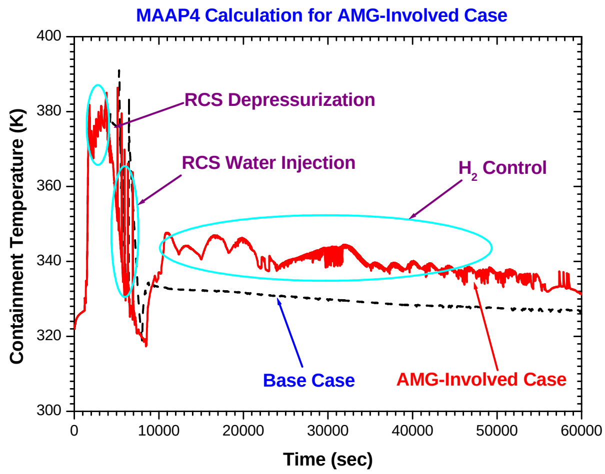


Fig. 6 Cont. pressure response against AMG actions

Fig. 7 Cont. temperature response for AMG actions

Table 1 Definition of accident environmental stages for equipment survivability assessments

AES	Beginning Time	Ending Time	Remarks
1	Accident initiation	Safe stable state or core uncovery	Bounded by design basis EQ environment
2	Core uncovery	Safe stable state or significant core damage	All the AMG actions are considered, but Bounded by design basis EQ environment
3	Significant core damage	Safe stable state or corium relocation into lower head	All the AMG actions are considered
4	Corium relocation into lower head	Safe stable state or reactor vessel failure	Some AMG actions are considered
5	Reactor vessel failure	Safe stable state or containment failure	Only AMG actions for containment integrity are considered

Table 2 Accident event summary for a selected accident scenario

Event	Timing	Event	Timing
First Lift of PSVs	1,109 sec	Main Coolant Pump Off	1,748 sec
Core Uncovery	2,698 sec	Core Exit Temperature (CET) Exceeds 644 K (700 ^{o}F)	2,784 sec
Core Exit Temperature (CET) Exceeds 922 K (1,200 ^{o}F)	3,614 sec	Corium Relocation into Lower Head	5,228 sec
CSS Actuation Signal	5,257 sec	Reactor Vessel Failure	6,456 sec
Start of SIT Injection	6,476 sec	Depletion of SIT Water	6,507 sec

Table 3 Accident environmental conditions for a selected accident scenario

Parameter		AMG not involved (base case)		AMG involved	
		AES 3	AES 4	AES 3	AES 4
RCS pressure (MPa)		17.39	18.55	14.76	9.47
RCS temperature (K)		1,199	1,272	1,384	1,198
Containment pressure (MPa)	Upper compartment	0.234	0.257	0.248	0.145
	Annular compartment	0.234	0.257	0.248	0.145
Containment Temperature (K)	Upper compartment	382	391	386	369
	Annular compartment	389	397	1,004	430