EVALUATION OF SEVERE ACCIDENT MANAGEMENT GUIDANCE – MAAP5 SIMULATION – INSIGHTS TO ENHANCE TRAINING

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ABSTRACT

In 2012 EPRI updated the Technical Basis Report to reflect lessons learned from the accidents at Fukushima. Both of the BWR and PWR Owner's Groups took the updated TBR in addition with other insight information and updated their respective severe accident management guidelines. For the BWR Severe Accident Guidelines (SAGs), additional input was provided by EPRI as a result of the technical work done in support of the Containment Performance and Release Reduction (CPRR) Rulemaking.

EPRI and the US Department of Energy conducted a gap review for the severe accident knowledge base considering what had been learned from the Fukushima accidents. One outcome was that the SAMG actions should be evaluated using the updated computer models generated during the analysis of the Fukushima accidents. The purpose was not to validate the actions (these were validated by other activities), but to determine the relative importance of the actions and decisions in the SAMGs in light of the new code capabilities and the understanding of severe accident uncertainties.

This project was undertaken to help the industry understand the relative importance of the actions and decisions in the SAMGs. This can be used to help with the implementation of any new SAMGs and the associated training. It can be used to help develop relevant drills for exercising the SAMGs

Using machine learning techniques, the large amount of simulated data was able to be interrogated in order to provide insights on severe accident management. It is acknowledged that there are significant uncertainties in severe accident plant behavior. This project was able to assist the industry in better understanding the importance of uncertainties on the success of actions taken and to focus future training programs.

KEYWORDS
Modular Accident Analysis Program (MAAP)
Severe Accident Management Guidelines (SAMG)

1. INTRODUCTION

Nuclear power plants are designed to withstand challenging transients such as one resulting from an instantaneous, double-ended rupture of a reactor coolant pipe. This underlying design philosophy, combined with the conservatisms used in the design-basis analyses, has produced reactor systems and containments that can accommodate a broad range of accident conditions. This is the case for operating commercial light water reactors (LWRs).

Systems are designed to mitigate excursions in reactivity, removal of decay heat from the fuel, and challenges to the integrity of the containment. Beyond the defense against transients due to malfunctions internal to the nuclear power plant, the design basis includes consideration of events external to the plant that can challenge the capability of various mitigation systems required to return the plant to a safe, stable state. Experience gained over the past decades has shown the following:

- Accident conditions beyond those considered in the design can occur that threaten adequate cooling of the reactor core
- Severe and extremely unlikely initiating events and common mode system failures can occur and lead to a wide-scale challenge of the capability present and mitigate accidents of escalating severity.

Studies of the accident at Three Mile Island Unit 2 (TMI-2) [1] prompted the need for formal guidance to support/manage accident conditions in which the function of reactor core cooling has been challenged. The individual plant examinations performed in response to NRC Generic Letter 88-20 [2], resulted in an assessment of plant capabilities in terms of preventing core damage and maintaining containment integrity. This information was of substantial use in formulating accident management guidance.

The significant effort that followed the accident at TMI-2 led the U.S. nuclear industry, and subsequently the international nuclear industry, to develop severe accident management guidelines (SAMGs) for the nuclear fleet. The availability of SAMGs provides significant defense in depth and ensures that means exist to cope with the occurrence of a severe accident. Because individual plant SAMGs are designed to respond to ongoing challenges to core cooling and containment integrity, they provide the capability to cope with a severe accident, independent of the type of event that led to the occurrence of core damage. SAMGs are intended to deal with reactor coolant system (RCS) and containment conditions that can occur following core damage.

The SAMGs were, and continue to be, developed by the owners' groups for the various Nuclear Steam Supply System (NSSS) types. EPRI's role is to develop the technical basis for what are called the candidate high-level actions (CHLA) that are employed in the SAMGs. The initial technical basis report (TBR), EPRI TR-101869 [3] was issued in 1992.

Following the recent accident at the Fukushima Daiichi nuclear station, following the Great East Japan Earthquake of 2011, the TBR was updated to reflect lessons learned. The accident at the Fukushima Daiichi station illustrated that severe damage could occur to the basic infrastructure required to support various accident mitigation and management functions. It has been recognized since TMI-2 that the management of severe accidents is the most effective when addressing symptoms associated with the potential degradation of fission product retention boundaries. Fukushima Daiichi highlighted the potential for severe initiating events to challenge the implementation of symptom-based accident management measures. The updated TBR, EPRI 1025295 [4], highlights several additional issues including multi-unit and spent fuel pool challenges. The updated list of CHLAs are shown in Table 1-1.

In early 2015, EPRI and the US Department of Energy (DOE) conducted a gap review [5] for the severe accident knowledge base in light of what had been learned from Fukushima. It included stakeholders such as the BWR Owners Group (BWROG), PWR Owners Group (PWROG), and the US Nuclear Regulatory Commission (NRC). One outcome was that the SAMG actions should be evaluated using the updated computer models generated during the analysis of Fukushima. The purpose was not to validate the actions (these were validated by other activities), but to investigate code predictions for the timing of action management functions, provide input on accident management training and assist in the prioritization of actions.

This project was undertaken to help the industry understand the relative importance of the actions and decisions in the SAMGs. This can be used to help with the implementation of any new SAMGs and the associated training. It can be used to help develop relevant drills for exercising the SAMGs by using the scenarios outlined in this analysis. In addition, there is an objective to help the industry understand the uncertainties inherent in the estimation of severe accident progression and how to deal with that in the implementation of SAMGs.

The following sections provide a description of the MAAP analyses performed, and a summary of the insights gained.

Table 1-1
Candidate High-Level Actions

No.	Candidate High-Level Action
1	Inject into (make up to) reactor pressure vessel/reactor coolant system (RPV/RCS)
2	Depressurize the RPV/RCS
3	Spray within the RPV (BWR)
4	Restart reactor coolant pump (RCP) (PWR)
5	Depressurize steam generators (PWR)
6	Inject into (feed) the steam generators (PWR)
7	Operate isolation condenser (IC) (BWR)
8	Spray into containment
9	Inject into containment
10	Operate fan coolers
11	Operate hydrogen recombiners
12	Operate hydrogen igniters
13	Inert containment with non-condensable gases (BWR)
14	Vent primary containment
15	Spray secondary containment
16	Flood secondary containment
17	Inject into spent fuel pool
18	Spray spent fuel pool
19	Vent/ventilate reactor building/auxiliary building
20	Spray building to scrub releases

2. DESCRIPTION OF THIS INVESTIGATION

Following the events at Fukushima, a comparative study [6] was performed using MAAP5 and MELCOR codes to understand the principal modeling decisions in the two codes and assess their impact on the simulation results. It has been understood that the two codes can be used in this type of assessment to investigate the uncertainties associated with complicated severe accident phenomena. As a logical extension of the Crosswalk investigation [6], EPRI and the NRC, in collaboration with DOE, have decided to sponsor an investigation into the impact that uncertain conditions could play in the execution and response of severe accident guidelines. The first phase of this investigation will focus on a BWR-4/Mark I containment due to the existence of a plant model for both MAAP and MELCOR. The following describes the approach to further our understanding of severe accident management guidelines. At this time, only the MAAP results are being reported and a future update is being planned to integrate the MELCOR results.

2.1 Sequence Identification

Using a representative BWR-4/Mark I plant probabilistic risk assessment (PRA), dominant sequences contributing to the core damage and large release were identified. SAMG actions are typically given limited credit in the plant PRA and this investigation will provide valuable insights as to the impact that these actions could have to mitigate core damage and reduce any potential fission product releases.

Based on significant contribution to core damage and the accidents at Fukushima, an extended loss of offsite power (LOOP) sequence was selected to demonstrate the SAMG actions and decisions, consistent with the recommendations in the EPRI/DOE gap review [5]. Where the PRA is able to identify actions important to the quantification of CDF and LERF, this analysis was able to identify actions important to the timing of CDF and the magnitude of the release.

2.2 SAMG Actions Identified

Once the sequences had been identified, the generic BWROG EPG/SAGs [7,8] were reviewed and a set of candidate mitigation actions established. The terms emergency procedure guidelines (EPG) and severe accident guidelines (SAG) refer to the BWROG products, where SAMG is the industry-wide term used for similar tools. The SAMGs provide multiple options for accident mitigation and combinations of successes and failures are assumed to create a list of specific scenario definitions. Table 2-1 lists the individual steps modeled in the MAAP analyses. The major actions and decision points identified in Table 2-1 were selected to be consistent with the EPG/SAG detailed guidance which are also based on the general CHLAs described in the EPRI TBR.

Based on these modeled actions, a core damage event tree (CDET) and a containment event tree (CET) have been developed to represent the total number of possible end states represented by the MAAP analysis. The core damage event tree represents 37 unique end states and the containment event tree 52 unique end states. Combined, a total of 1924 unique end states are represented using individual MAAP simulations. As provided in the detailed event tree logic, the core damage event tree includes those actions performed to prevent core damage and establish adequate core cooling. For the total loss of AC power event, the major actions involve establishing and maintaining core injection, controlling pressure within the reactor vessel and venting containment to provide for decay heat removal. With the possibility for core damage to occur, the containment event tree is developed to represent actions associated with providing cooling to the core material and establishing decay heat removal through venting of containment.

Table 2-2
SAG Actions Modeled

Step	Description	Options to be Modeled	Dependency on Previous Action	Comments
Initial Conditions	1. Loss of all onsite and offsite AC power 2. Main Turbine Trip 3. Main Steam Isolation Valves close 4. Scram 5. Loss of all injection except Reactor Core Isolation Cooling (RCIC) 6. Assume Reactor Coolant Pumps (RCP) seal leakage of 36 gpm at normal operating conditions		None	
RCIC Injection	RCIC auto-start on low Reactor Pressure Vessel (RPV) water level Assume suction immediately auto-switched to suppression pool	1. Success 2. Failure	None	Potential future option to allow initial suction from CST Evaluate cases with 4-72 hr. RCIC run time
Defeat RCIC interlocks	Defeat RCIC trip logic for low RPV pressure and high turbine exhaust	1. Success 2. Failure		
RPV Pressure Control 1	At 10 minutes: using 1 Safety Relief Valve (SRV), control pressure in 800-1000 psig range	1. Success 2. Failure		Prevent auto SRV cycling
RPV Pressure Control 2	At 1 hr.: using 1 SRV, control pressure in 400-600 psig range		Only assume success if previous pressure control 1 succeeds	Depress at approx. 80 °F/hr.

Step	Description	Options to be Modeled	Dependency on Previous Action	Comments
RPV Pressure Control 3	At 2 hr.: using 1 SRV, control pressure in 200-400 psig range		Only assume success if previous pressure control 1 succeeds	Hold above 200 psig to maintain RCIC operation
Primary Containment Control 1	Vent containment to maintain adequate core cooling (i.e. RCIC)	1. Success 2. Failure	Only assume success if initial RCIC Injection success	Assume vent pressure of 15 psig. We could consider different pressures in future analysis, but again, focus is on SAG, not EOP.
RPV Level Control 1	RPV water level drops below min steam cooling water level limit – blowdown using SRVs	1. Success 2. Failure	None	Assume level limit at 38” below TAF Also indicates transfer to SAGs
Primary Containment Control 2	Once RCIC is lost, the need to keep the vent open no longer applies and the vent would be closed	1. Success 2. Failure	Only applies if vent was initially opened to maintain RCIC	
Primary Containment Control 3	Suppression pool temperature and RPV pressure exceed Heat Capacity Temperature Limit (HCTL) – blowdown using SRVs	1. Success 2. Failure	None	To simplify, assume suppression pool temperature > 202 °F as threshold.
SAG entry – branch determination		1. SAG branch 5 2. SAG branch 4	None	Upon entry into SAG 1, likely decision is for leg 5, “Core debris cannot be retained in the RPV” due to loss of all injection. However, given that level is just below TAF, an optimistic view might be that leg 4 applies, “Can core debris be retained in the RPV”.
SAG -1	Vent primary containment via the wetwell to control suppression chamber pressure below Pressure Suppression Pressure (PSP)		Only if SAG entry to branch 5 from above	Assume PSP = 30 psig

Step	Description	Options to be Modeled	Dependency on Previous Action	Comments
SAG -1	Vent primary containment via the wetwell to control suppression chamber pressure below Primary Containment Pressure Limit (PCPL)	1. Vent at low pressure. 2. Vent at PCPL	Only if SAG entry to branch 4 from above, or if successful determination of vessel breach as described in the next block.	Assume PCPL = 60 psig
SAG - 1	Vessel Breach determination	1. Success 2. Failure	Based on code calculation	This decision point is to either successfully acknowledge vessel breach or not. If not, we will remain in Leg 5 of the SAG 1
SAG -1	Initiate severe accident water addition (SAWA)	1. Success at 1 hour after onset of core damage 2. Success at time of vessel breach 3. Success at 1 hour following vessel breach 4. Failure		Assume injection directly to the RPV Success path 1 expected to result in in-vessel retention
SAG - 1	Severe Accident Water Management (SAWM)		Only if SAWA has been a success	Maintain level below level requiring closure of wetwell vent by reducing SAWA flow to 100GPM or less.

2.3 Accident Analysis

With the specific scenario definitions developed in Section 2.2, MAAP analyses were carried out using MAAP 5.05 beta [11] release. The event tree logic described in Section 2 was used to generate MAAP5 input files representing over 1900 unique end states. These end states represent individual accident scenarios that are analyzed in order to investigate success and failure of a variety of human actions and system operations that occur prior to and after the onset of core damage.

2.4 Results Interpretation

A considerable amount of data is generated from the execution of over 1900 simulations. Section 3 provides summaries and conclusions from the analysis. The results from the MAAP analyses were interrogated as to how the specific SAMG actions influence the accident progression. There were two primary figures-of-merit (FOM) identified to allow interpretation of the results. First is the impact on the fission product release magnitude. CsI is selected to represent the fission product release and embodies the potential health consequences associated with the event. The other FOM is related to time available for staff members to perform the actions identified in the EPG/SAGs. Additional time available before the onset of core damage can be crucial to the successful implementation of key operator actions and offsite response.

As indicated previously, a large amount of data is generated as a result of the MAAP analysis. The problem becomes one of determining which top branches in the event trees affect the figure-of-merit the most. With the large amount of output data, several techniques are available to display the results in a meaningful way. One such technique is machine learning. Machine learning is the evolution of pattern recognition and computational learning theory. Machine learning is essentially building mathematical models to achieve an understanding of data. "Learning" means that the mathematical models use tuning parameters to provide for a prediction of the observed data. Once the models have been tuned, they are then used to understand a variety of aspects of the data. Some familiar examples of applications of machine learning include search engines and spam filtering. This project utilized the Random Forest [10] algorithm. This algorithm is well suited for the type of event tree logic employed here and can investigate how well each decision tree predicts an end state figure-of-merit (e.g. CsI release). The initial results can be viewed as to the importance of each top event to the FOM.

3. RESULTS

As stated in Section 2, over 1900 unique simulations were executed using MAAP 5.05 Beta [9]. Each of these simulations represent the success and failure of the actions discussed in Section 2 along with the success and failure of decisions made in the SAGs. There is no attempt to place probabilities on the success and failures. It is assumed that equal likelihood for all branch points identified in the event trees. By doing this, the analysis is able to identify the most important actions and decisions made in the execution of the SAGs. Also included in the simulations are scenarios where the various mitigation systems either operate successfully or are assumed to fail. Therefore, the totality of end states represented include consideration of both equipment operation along with human interactions.

Figure 3-1 shows a typical machine learning output which evaluates the most important actions/events modeled in the core damage event tree as they impact the magnitude of the CsI (i.e. fission product) release overall. Clearly identified from the MAAP simulations is the action to depressurize the reactor pressure vessel prior to core damage. Within the core damage event tree, the top event "rpv_mscwl" represents the action to depressurize the RPV once the water level drops below the minimum steam cooling RPV water level limit. The importance of this action is well understood by the developers of the EPG/SAGs as it provides the means to utilize low pressure injection systems to prevent core damage, thereby eliminating any significant release of radionuclides. Also, a significant impact of this action on fission product release is that, with an open relief valve during core damage, a large fraction of the initial fission product inventory is released into the suppression pool where it is scrubbed and unavailable for release from containment.

Assuming successful depressurization of the RPV, Figure 3-2 shows the next important event from the results of the simulations, again relative to the CsI release from containment. As shown, the next critical event modeled in the core damage event tree is the time period that RCIC is able to inject. Well understood is that if RCIC is able to run for a longer period of time, deployment and use of portable equipment to prevent core damage becomes more likely. As seen in the simulations is that extending the running of RCIC also has an impact on the source term release. The importance of this event to the CsI release fraction is based on the magnitude of the release at the end of the analysis. All calculations were run for 72 hours of accident time except for the cases assuming that RCIC operates for 72 hours. For these runs, the simulation time was extended to 100 hours. Figure 3-3 shows the next important events, assuming the prior events are successful. From these figures, isolation of the containment vent after it is initially opened to extend the use of RCIC is important to the overall CsI release. Closure of this vent also confirms the importance of timely transfer from the EOPs to the SAGs. In addition, opening of the containment vent as a way to provide containment heat removal, is also shown to be important to the reduction of fission product release.

Where the previous discussion mostly focused on core damage prevention actions, Figure 3-4 is included to also confirm the importance of several actions taken within the SAGs. In the top left of the figure is plotted the CsI release fraction from containment as a function of the time of core damage. The upper left figure contains the full dataset of results from all of the simulations. Notice that the results are most densely located with a CsI release fraction less than 0.7 and a large population less than 0.2 and a time of core damage between 1 and 5 hours. The top right figure highlights in red the results for cases with successful depressurization of the RPV prior to core damage. Where the highest population of results are still in the range of 1-5 hour timing, the majority of simulations resulted in CsI release fraction less than 0.2. Likewise, for cases with RPV depressurization and injection into the vessel prior to vessel breach (shown in blue), the majority of simulations show less than 0.1 CsI release and several of the results show extended timeframe for core damage. And finally, the bottom right figure highlights in green only the results for successful RPV depressurization, injection prior to vessel breach and successful venting of containment for heat removal. The results show a slightly further reduction in the CsI release for this combination of actions.

Another method to display the results of the simulations uses the previously discussed machine learning algorithm to show the relative importance of the top events within the containment event tree. These events generally address actions taken after the onset of core damage that are included in the general SAG. Figure 3-5 shows the results of the machine learning algorithm. Note that the most important action relative to the fission product release is venting of the containment. This is well understood as containment venting can prevent failure of the containment and also provides for scrubbing of fission products prior to release from containment. Other important actions identified in Figure 3-5 include implementation of severe accident water addition (SAWA), injection of water into the RPV prior to vessel breach and severe accident water management (SAWM), management of water addition to preserve the function of the wetwell vent.

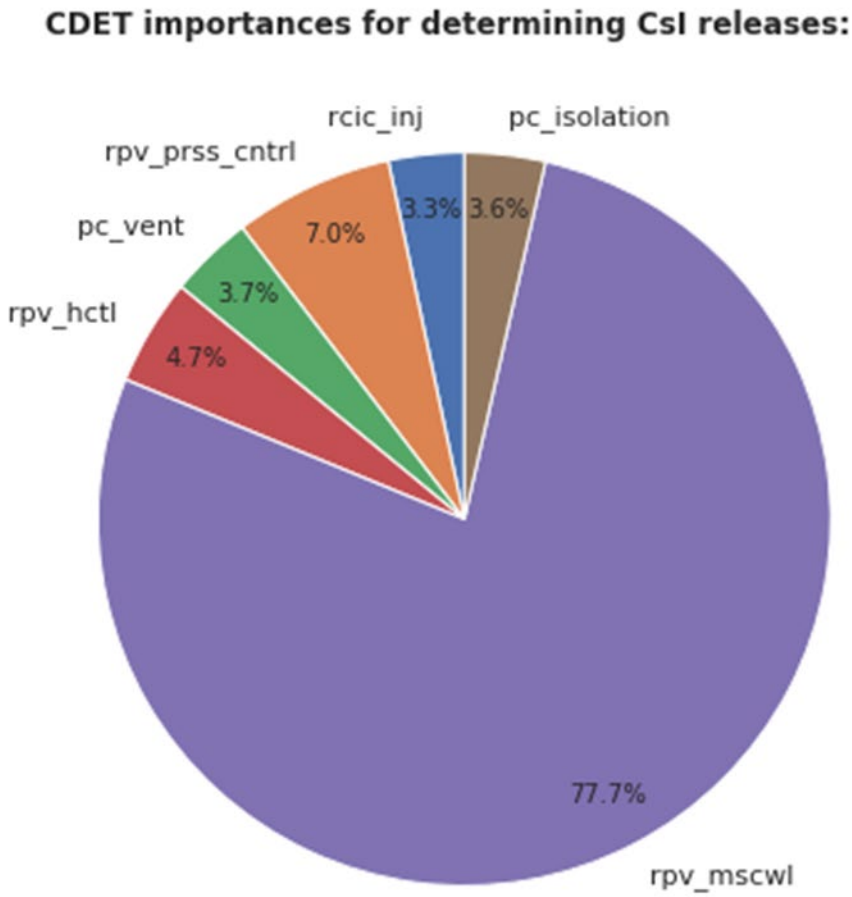

Figure 3-1
Importance of Core Damage Events

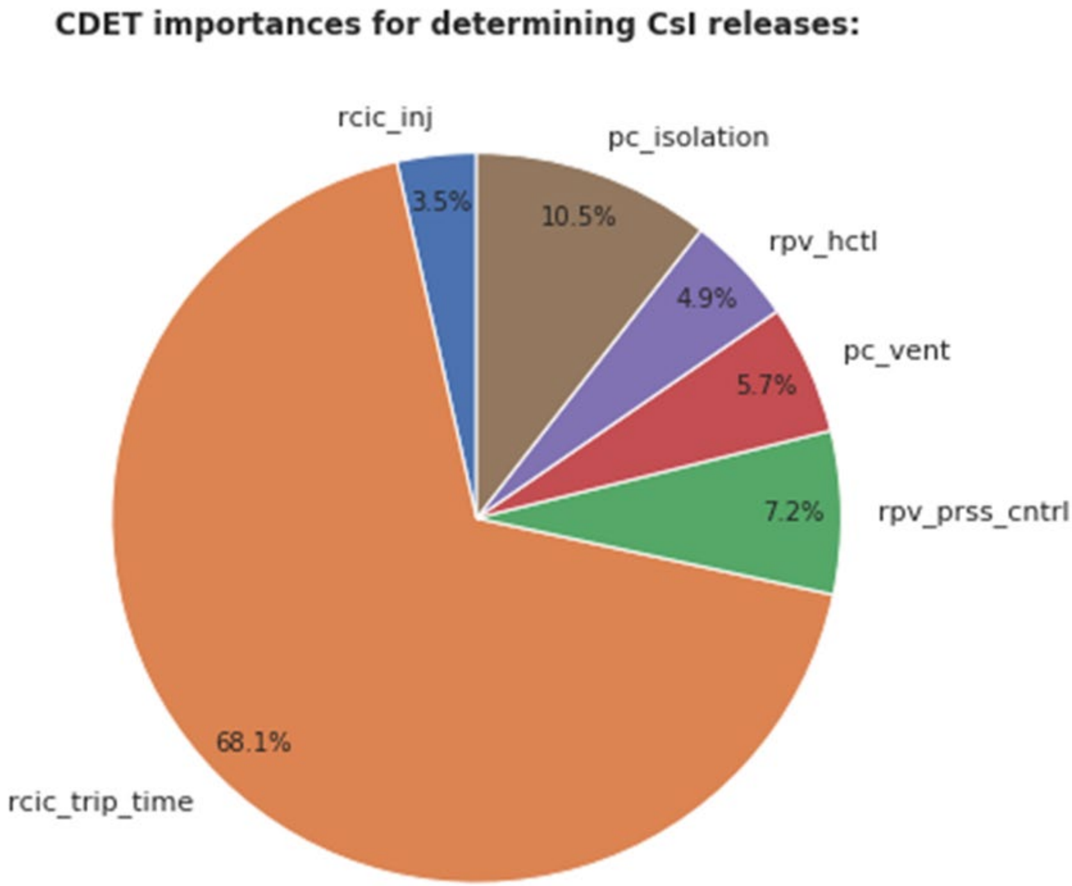

Figure 3-2
Successful RPV Depressurization

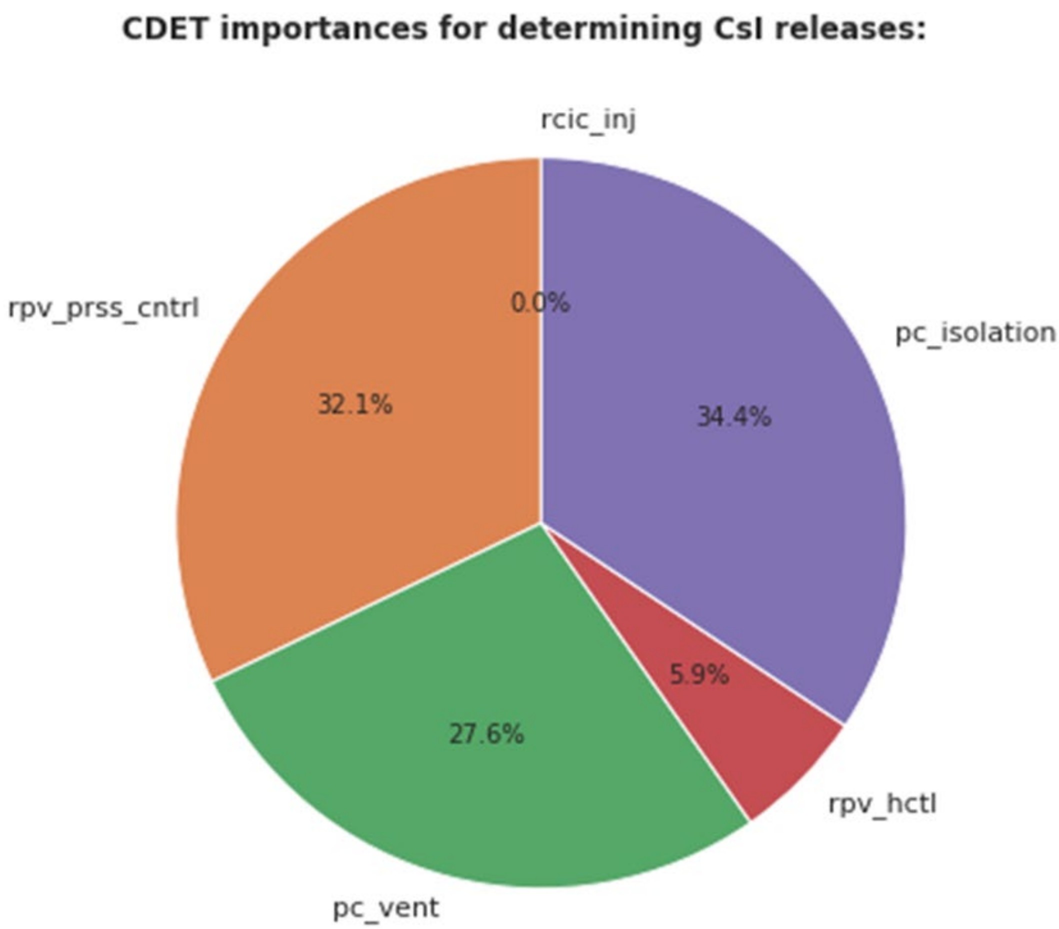

Figure 3-3
Successful RPV Depressurization and Extended RCIC Operation

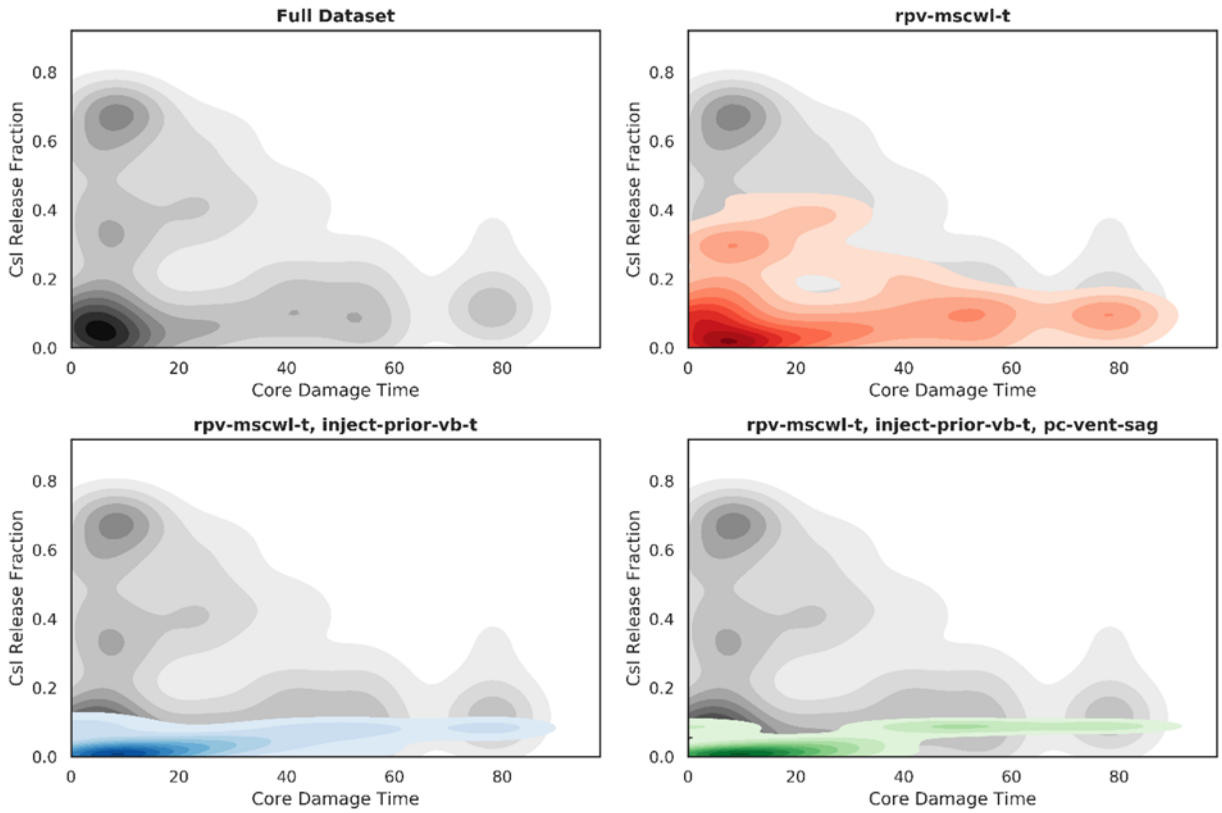

Figure 3-4
Summary for CsI Release and Accident Timing

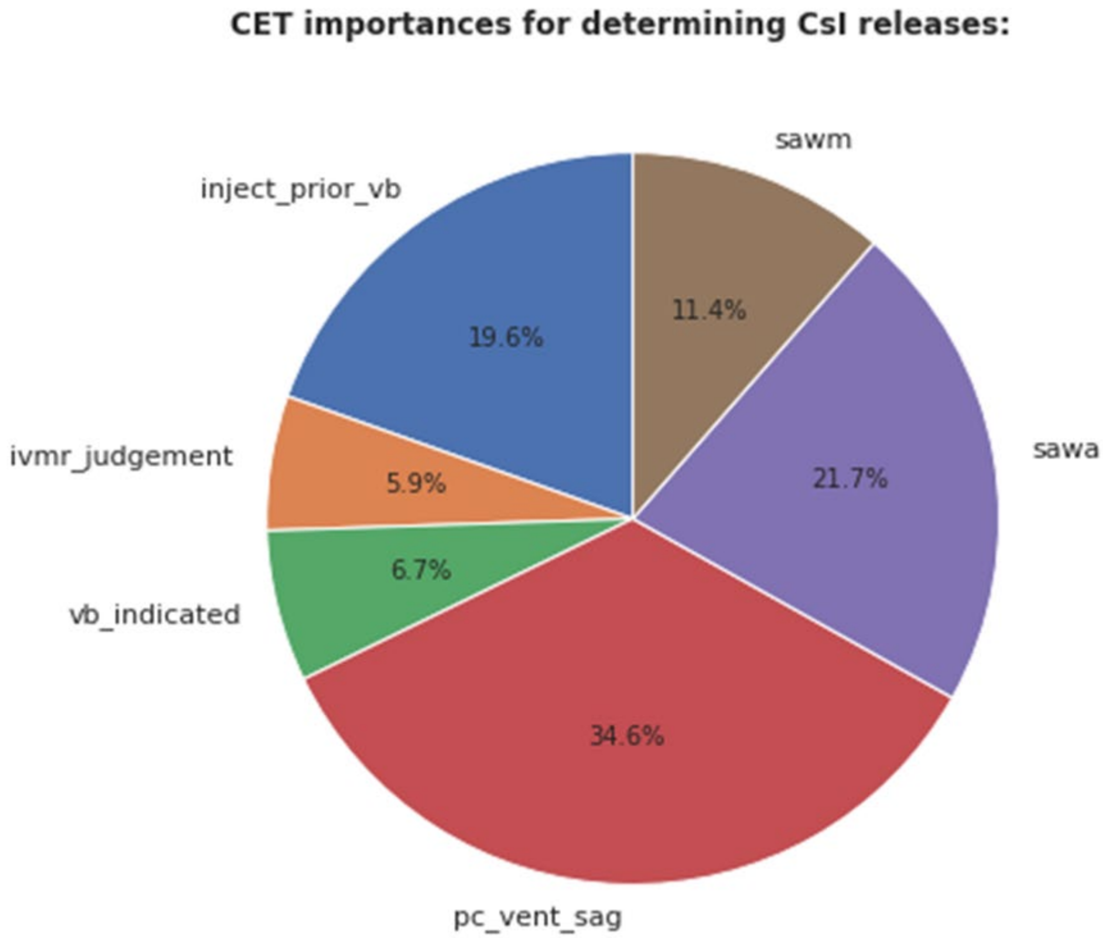

Figure 3-5
CET Top Actions

4. CONCLUSION

Using machine learning techniques, a large amount of simulation data can be interrogated in order to provide insights on severe accident management. It is acknowledged that there are significant uncertainties in severe accident plant behavior, however, this project was able to assist in better understanding the importance of those uncertainties on the success of actions taken and to focus future training programs.

This project identified the three most important actions for operators to focus upon in a severe accident for the representative BWR. These are depressurization of the RPV to prevent high pressure melt ejection, injection of water into the containment, and controlled venting of the containment vessel to prevent catastrophic failure. These actions sould be emphasized in severe accident training.

This technique can be applied to other reactor and containment types.

NOMENCLATURE

Abbreviation	Description
rcic_inj	RCIC injection
rcic_trip_time	Time that RCIC injection is lost
rpv_prss_cntl	Initial Reactor Pressure Vessel (RPV) pressure control to maintain RCIC operation
pc_vent	Initial containment venting to maintain RCIC operation
rpv_hctl	Depressurization of RPV based on elevated pool temperature (heat Capacity Temperature Limit)
rpv_mscwl	Depressurization of RPV based on water level below top of active core (Minimum Steal Cooling Water Level Limit)
pc_isolation	Isolation of containment vent that was initially opened to support RCIC operation
inject_prior_vb	Injection of water into RPV after core damage, but before vessel breach
ivmr_judgement	Operator determines if core material can be maintained in RPV
vb_indicated	Operator determines that vessel has breached
pc_vent_sag	Operation of containment vent per SAG instructions
sawa	Severe accident water addition following core damage
sawm	Severe accident water management to preserve wetwell vent function

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REFERENCES

1. NSAC-1, “Analysis of Three Mile Island—Unit 2 Accident.” Nuclear Safety Analysis Center. United States, 1980.
2. NRC Generic Letter 88-20, “Individual Plant Examination for Severe Accident Vulnerabilities.” U.S. Nuclear Regulatory Commission, Washington, DC. November 23, 1988.
3. *Severe Accident Management Guidance Technical Basis Report, Volume 1: Candidate High-Level Actions and Their Effects*. EPRI, Palo Alto, CA: 1992. TR-101869.
4. *Severe Accident Management Guidance Technical Basis Report, Volume 1: Candidate High-Level Actions and Their Effects*. EPRI, Palo Alto, CA: 2012. 1025295.
5. ANL/NE-15/4, “Reactor Safety Gap Evaluation of Accident Tolerant Components and Severe Accident Analysis”, Argonne National Laboratory, March 31, 2015

6. *Modular Accident Analysis Program (MAAP)-MELCOR Crosswalk: Phase 1 Study*. EPRI, Palo Alto, CA: 2014. 3002004449.
7. EPG/SAG Revision 3, BWROG Emergency Procedures Committee, BWROG, February 2013
8. EPG/SAG Revision 4, BWROG Emergency Procedures Committee, BROG, 2018
9. Modular Accident Analysis Program (MAAP), version 5.05 Beta, EPRI, January 2017, 3002009328.
10. Murphy, K. P. (2012). *Machine Learning, A Probabilistic Perspective*, Cambridge, MA: The MIT Press