IMPORTANT CONSIDERATIONS IN SELECTING A SIMULATION END-TIME IN LEVEL 2 PSA DETERMINISTIC ANALYSES

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This paper describes important aspects that should be considered when selecting a simulation end-time for deterministic accident progression analyses that will support a Level 2 probabilistic safety assessment (PSA) for a nuclear power plant. The paper starts by describing why simulation end-time selection is an inevitable part of this type of analysis, what precedents exist for selecting the end-time, and the general considerations that apply to this decision. It goes on to describe how severe accident uncertainty, as well as limitations in the state-of practice of human reliability analysis and PSA technology, elevate the importance of the decision. Next it presents a set of sample MELCOR results to show how simulation end-time affects the cumulative radiological releases to the environment for different types of postulated accidents. Following this, a set of sample source terms and release category frequencies are used to illustrate how simulation end-time can impact Level 2 PSA risk surrogates.

I. INTRODUCTION

Historically, probabilistic safety assessment (PSA), a term used in this paper interchangeably with probabilistic risk assessment (PRA), for nuclear power plants have not explicitly modeled the role of long-term onsite, or offsite, resources in terminating accidents after core damage has occurred. Rather, such analyses typically either truncate the accident (i.e. cease to consider events past a specified time) or assume that all long-term accident sequences will contribute to a particular release category (e.g. late containment failure). Such treatment is becoming more and more difficult to defend in PSAs given:

- the maturity of onsite accident management and emergency preparedness,
- the relative slowness (when compared to historical studies) with which accident progression proceeds for many scenarios,
- the desire to characterize realistic accident outcomes,
- recognition of the potential for additional failures occurring significantly after the initiating event which can complicate accident response, and
- the inclusion (in some studies) of reactor-at-shutdown and spent fuel pool accidents that are often slowly-evolving by nature.

In addition, the 2011 Fukushima Daiichi accident illustrated that severe accidents can have a wide range of realized timelines. For instance, compare a 48-hour analytical truncation time to the description of events on pg. 124-128 of the 2014 National Academies Study[1] for Unit 2 at Fukushima Daiichi, where core damage and resulting release started roughly 72 hours after the initiating earthquake and tsunami. Some thoughts about the importance of considering long-duration scenarios in accident management preparation are captured in a 2014 Nuclear Energy Agency report[2], most notably Sections 4.2.2 and 4.5.

Contemporary examples of the use of truncation times in accident analysis include the US Nuclear Regulatory Commission's State-of-the-art Reactor Consequence Analyses (SOARCA) project documented in NUREG-1935[3] and a recent US NRC consequence analyses for a large earthquake impacting a Mark I spent fuel pool documented in NUREG-2161[4]. The former generally used a truncation time of 48 hours based on qualitative justifications of the capacity of local and regional infrastructure for the two sites studied to flood containment (and an assumption that such flooding would terminate the accident), while the latter used a 72-hour mission time as a compromise between the available offsite resources and the significant hazards that could develop. The use of truncation times like 24, 48, or 72 hours after event initiation are typical. Alternatively, truncation times can be anchored to the onset of fuel damage, as this is typically the time when severe accident management guidance is invoked.

The selection of a truncation time must consider a handful of different factors. These include:

- the modeling uncertainty associated with the severe accident analysis and the associated validity of the computational results at long durations,
- the prognosis for cooling molten debris,
- the availability of accepted human reliability analysis methods for the domains of interest,
- the role and maturity of accident management at the facility being studied,
- the expected impact of truncation time on the deterministic and probabilistic results,
- the facility design and capabilities, and
- the scope of the investigation being conducted.

All but the last two of these (since they are facility- or study-specific) are discussed in the following sections.

II. SEVERE ACCIDENT MODELING UNCERTAINTY

A major consideration in selecting a simulation end-time is the degree to which severe accident modeling uncertainty naturally compounds as the simulation continues, in addition to some specific long-term phenomena that are less well-understood. This paper does not dwell on the former issue, but it is important to acknowledge that due to threshold effects (e.g. a component accumulates damage and degrades or retains capability, a combustion event occurs or is narrowly avoided) small changes in the accident simulation can ultimately lead to large changes in the predicted environmental release. Of course, these points of departure also have the potential to 'wash out' over a long simulation time, so the best that can be generally said is that overall uncertainty increases as the simulation end-time increases.

Rather, this part of the paper will highlight several specific phenomena that have significant uncertainty, and affect prediction of environmental release. The first of these is molten-core concrete interaction (MCCI). The incoming melt composition, redistribution of the material, the rate of ablation, the stability of an overlying crust, the associated ability of overlying water to intrude, the gas generation rate, and the overall heat transfer characteristics are all uncertain, despite significant experimental and analytical progress in their understanding. These uncertainties in turn fundamentally affect the ability to predict coolability of ex-vessel melt, and in particular in cases where MCCI has been ongoing for some time prior to introduction of water. For light-water reactor severe accident simulations extending to beyond 48 hours after core damage, these uncertainties can greatly affect the confidence with which one can predict ex-vessel coolability, as well as the relative race between long-term containment over-pressurization versus basemat melt-through.

The rate of long-term containment over-pressure is in turn affected by a different category of uncertainties, those associated with containment failure degradation and long-term performance. For concrete containments, the trend in the US has been to move away from simplified gross failure containment response models, in favor of more realistic tear-before-rupture (or leak-before-rupture) models. In the latter, when containment internal pressure reaches a high level (e.g. 2.5 times the design pressure), a small opening is introduced which slowly grows with increasing pressure until a quasi-equilibrium condition is reached (which generally occurs without a large opening being introduced).

Such modeling creates an interesting effect when viewed in concert with simulation end-time discussions. A gross failure results in a quick depressurization and a puff-like release, followed by a period of lower releases owing to the lack of a driving force to push fission products out of containment. The tear-before-rupture (or leak-before-rupture) model, on the other hand, can result in containment pressure remaining elevated for days after the introduction of the tear, leading to a long sustained blowdown of the containment. This long sustained blowdown can results in comparable (or even larger) cumulative fission product releases as compared to the gross rupture model, but only if the accident is not subsequently mitigated (and the simulation is carried out to its logical conclusion). Thus, this shift in containment response modeling raises the importance of long-term mitigation modeling (discussed later), in addition to the selection of the simulation end-time.

Related to both of the above categories of uncertainty is the general lack of consensus physical models associated with basemat melt-through. For instance, in MELCOR, MCCI is allowed to continue indefinitely if not stopped by the user or cooling (equivalent to there being an infinite thickness of concrete), eventually leading to non-physical cumulative gas generation. Were a limit to be placed on the ablation depth, it is not obvious how to mechanistically model the response of containment to this threshold, particularly for plants with subterranean cavities. This has the effect of ultimately affecting the long-term airborne release, since it affects the containment pressure signature (recall previous statements regarding long sustained pressurization when using a tear-before-rupture model). Meanwhile, typical Level 2 and Level 3 PSA tool sets do not address the transfer of aqueous (or at least non-airborne) releases for the purpose of offsite consequence calculations.

Finally (on this topic), is the issue of fission product speciation phenomena unique to long-running scenarios. As a key example of this, MELCOR models ex-vessel core debris fission products as being in either the oxide phase or the metal phase, and there is no redistribution between these phases. The data reviewed during model development indicated that molybdenum (among other fission products) partitioned almost exclusively into the metal phase. This metal phase dictates the oxygen potential of gas sparging through the melt. Release of fission products from either the oxide or the metal phase of core debris is largely by vaporization. Molybdenum release from the metal phase is as vapors of a variety of species. The oxygen potential of gas within the metal phase is, however, typically too low to drive sufficient formation of molybdenum-oxygen species to produce extensive vaporization of molybdenum. This is especially true when zirconium or chromium are present in the metal phase. However, even with only iron in the metal phase, molybdenum release is very small. In the long term, the metal phase is eventually depleted by oxidation so that zirconium, chromium, and iron are incorporated into the oxide phase of the core debris. The amount of rebar in the concrete and potential addition of metals from other sources (melting of structures) can delay metal phase depletion. Nevertheless, once these other metals are depleted (in long-running calculations), MELCOR predicts large releases of molybdenum from the ex-vessel debris (e.g. in the 24 to 72 hour range after vessel failure), and when paired with elevated containment pressure this can result in large predicted releases of molybdenum from the containment.

Analytical and experimental investigations of these types of uncertainties continue as part of various international programs, but consensus modeling approaches are not imminent. Also note that while MELCOR is the focus in this paper, other severe accident computer models have analogous limitations, and often these code issues are simply a reflection on the boundaries of our state-of-knowledge.

III. HUMAN RELIABILITY MODELING CHALLENGES

Performing human reliability analysis (HRA) for severe accident scenarios with current HRA methods is challenging, particularly for long-running scenarios. A typical HRA includes identifying critical human actions or human failure events (HFEs) in the PSA scenarios, failure modes of the human actions, and the context factors that challenge the success of the human actions. For severe accident scenarios, identification of these elements introduces uncertainties that are not adequately addressed in existing HRA methods. These uncertainties combine to influence what (operationally) will be done or attempted, and stems broadly from lack of standardization in requirements (e.g., qualifications, team composition, training) for operator response for this context. Some of the contributing uncertainties include:

- Key human actions to be included in HRA – Unlike in emergency operating procedures (EOPs) where key human actions are specified in procedures, key human actions following core damage and in very long accident scenarios can vary with differences in scenario progression and mitigation strategies. Though severe accident management (and other) guidelines do provide structure for this response; novel solutions could introduce new human actions outside PSA models. Assumptions on the ending point of PSA simulation can also affect identification of key human actions. Additionally, changes in the composition and role of the response organization and changes to the decision-making process that can evolve during long-duration scenarios are not typically modeled in traditional PSAs.
- Success criteria for key human actions – HRA is to assess the likelihood that operating personnel can successfully perform the required actions. Uncertainties in PSA simulations propagate to the success criteria of the human actions. Unlike in EOPs where the success criteria for human actions are more clearly understood, severe accident management strategies often cannot explicitly define the success criteria for human actions. This is particularly true when the actions involve decision-making based on incomplete, erroneous, or unreliable information, and this uncertainty directly interacts with uncertainties discussed in the preceding and proceeding sections.
- Context factors that challenge human actions – Context factors in very-long accident scenarios are far beyond those typically modeled in existing HRA methods. As an example, accumulated responsibilities of operational personnel in numerous ongoing activities other than performing key actions, although not explicitly modeled, can impact human reliability of the modeled key actions. Meanwhile, various environmental factors impose uncertainties to human reliability, and staffing level and composition vary during a very-long scenario. Assessment of such context factors may vary with different choices of simulation end-time.
- Time uncertainties – Time available for human actions is the most important factor in determining the feasibility and reliability of human actions. Time availability is determined by the system time allowed for completing the key human action and the time needed by operating personnel to perform the action. Existing HRA methods typically treat the system time available for a human action as a constant value from thermal-hydraulic simulation. While this approach may be adequate for normal and emergency operation, it is less satisfying for post-core damage and long-running scenarios. In these cases, assessment of both system time available and time needed for human actions can vary more widely with different assumptions made in the PSA, and different underlying uncertainties and variations within a particular PSA accident sequence. Essentially, just the time factor alone can dominate the uncertainties in HRA results.

The NRC recently developed the General methodology of the Integrated Human Event Analysis System (IDHEAS-G) to provide a general framework for developing HRA models and analyzing human events. IDHEAS-G is based on the cognitive basis structure for HRA developed by the NRC[5]. IDHEAS-G provides a systematic framework and guidance for defining HRA scope, developing operational narrative of the PSA scenario progression, identifying key human actions in the scenario and context factors that challenge human actions, performing time uncertainty analysis, and quantifying human error probabilities of the key actions. IDHEAS-G uses four macrocognitive functions to represent a task demand on human cognition: detecting information, understanding the situation, making decisions and planning the response, and executing actions. Each macrocognitive function is assumed to be performed in a team and organizational context. The structured guidance in IDHEAS-G allows the analyst to address the uncertainties described above, as well as other HRA issues identified thus far by the PSA community[6,7]. In particular, IDHEAS-G provides a structured process for identifying key human actions in complicated, long-lasting scenarios. It also has explicit guidance on treating time uncertainties in key human actions. While the explicit guidance allows HRA analysts to produce a systematic understanding of human performance under complex, imperfect, somewhat unknown operating conditions, the overall IDHEAS-G framework can assist PSA analysts in determining the analysis scope and assumptions.

IV. OTHER ACCIDENT MANAGEMENT ASPECTS

Beyond strictly severe accident simulation-related or HRA-related factors, there are some other aspects of long-running simulations that warrant mentioning. Three of these are discussed here, though others also exist. The first is flammability within containment late in the accident. Sustained MCCI in some containment designs has a proclivity for establishing inerted conditions, but with substantial amounts of hydrogen and carbon monoxide, if prior burns have not sufficiently consumed these combustibles and depleted oxygen. Uncertainties in containment leakage, prior combustions, ignition frequency, and human actions (e.g. venting, restoring containment heat removal) lead to uncertainty in how this situation will be resolved (either via large damaging combustions, smaller less-concerning combustions, or the dissipation of the combustibles safely). The point is that if an accident results in in-vessel and ex-vessel combustible gas production, a necessary step prior to entering a recovery phase that is not typically treated in severe accident simulations is the disposition of these combustibles.

Related to the above is the issue of the transit of these same combustibles, along with radiological material, into surrounding structures. This was obviously a major impact at the Fukushima Daiichi accident, both in terms of radiological habitability and the damaging effect of the reactor building hydrogen combustions. The most difficult aspects of addressing this issue are the uncertainties associated with the transit paths and the coarse characterization of these surrounding structures in typical severe accident models. For the former, the lack of rigorous knowledge about how normal containment leakage is distributed amongst the numerous containment penetrations and the potential pre-existing liner defects, combined with uncertainty in what induced pathways will form at elevated pressures (e.g. dry-well head bolt stretching, penetration seal degradation) make it speculative to predict how leakage will be distributed. This directly translates in to uncertainties in how radiological material and combustible gases will accumulate in sur-rounding structures versus transiting directly to the environment. In addition, the characterization of surrounding structures in terms of geometry, settling areas, other systems (e.g. normal fire suppression systems), and additional leakage paths to other structures or the environment are often coarse (or non-existent). Traditionally this coarse modeling has been justified by making assumptions that “conservatively” assume these structures do not provide as much fission product retention as they might. The Fukushima Daiichi accident underscores that energetic building failure and radiological contamination can not only affect the ultimate environmental release, but can also significantly affect the characterization of the accident management response (i.e. equipment survivability, radiological habitability, direct damage from combustion events, fear generated by combustion events).

Finally, and again related to the above points, is the issue of sustained high temperatures and pressures on equipment within containment. Equipment at nuclear power plants is subjected to rigorous qualification within the design-basis of that equipment. Often during the first couple of days of postulated severe accidents, it is possible to assert that a subset of that equipment will only see design-basis-like conditions during the particular time-period it is needed, while the remainder of the equipment is assumed unavailable because it exceeds its design-basis. The margin in equipment survivability (i.e. how much beyond its design envelope it will continue to reliably perform its intended function) has both aleatory and epistemic uncertainty that is generally not well-characterized (despite continued contributions in this area[8]). This situation is then exacerbated in the long-term wherein containment might experience sustained pressures on the order of 0.6 to 0.9 MPa and sustained temperatures on the order of 200 to 300˚C. As with combustion in the surrounding structures, making “conservative” assumptions on this front can be difficult because equipment response (and particularly instrument response) is not necessarily binary. Degraded equipment performance and erroneous instrument readings can have unintended and difficult-to-anticipate effects on accident behavior and human decisionmaking.

V. EFFECT OF SIMULATION END-TIME ON RADIOLOGICAL RELEASE RESULTS

So given the preceding points, is it hopeless to reliably predict accident consequences during long-running scenarios? This section and the next will argue that it is not, due to some useful features about the important deterministic and probabilistic results. First, a suite of contemporary severe accident simulations (produced using MELCOR) were scrutinized. These simulations covered bypass scenarios, early containment failure scenarios and a range of other scenarios. Three simulation truncation times were employed: 36 hours after the onset of core damage, 60 hours after the onset of core damage, and 7 days after accident initiation.

It should be understood that these are severe accident simulations, so by their nature they represent outcomes from combinations of events that have a very low combined likelihood of occurrence. The simulations are typical of PSA-oriented scenarios for US pressurized water reactors with a large, dry containment, and employ typical MELCOR modeling assumptions. Any given accident scenario has a range of potential outcomes based on modeling assumptions and boundary conditions, and so results from a given simulation may exceed a central tendency in the range of results for a given scenario (so as to envelope some portion of that range). Also of importance is that the simulation boundary conditions did not include mitigation actions much beyond vessel breach (i.e. long-term plant mitigation and stabilization activities were not modeled). Therefore, the simulations did not include either positive (e.g., scrubbing due to late water addition) or negative (e.g., inadvertent triggering of a late combustion event by de-inerting) impacts of such actions. The full set of simulations, of which only a subset are presented here, include cases with successful and unsuccessful mitigation actions through the time of vessel breach.

There are two fundamental trends of interest that were observed in the simulations scrutinized. The first is that bypass and early containment failure scenarios lead to the highest radiological releases, regardless of truncation time. This can be readily seen by comparing Figure 1 to Figure 2, and has historically been the case in severe accident simulation. The second is that bypass and early containment failure radiological releases for volatile chemical classes are insensitive to truncation time. This can be seen in Figure 1, was found to be true for chemical classes other than just cesium, and again is generally consistent with the conclusion of past severe accident analyses. The key exception to both of these trends for the simulations scrutinized was molybdenum, whose releases were (i) sensitive to truncation time (see Figure 3) and (ii) tended to be more similar for the later timeframes regardless of the scenario type. What this suggests is that to a first-order, the consequences that would be predicted from such environmental release estimates are not sensitive to truncation time, on the basis that iodine and cesium are traditionally the best indicators of early and latent effects (respectively), and the scenarios resulting in the largest environmental releases are the ones least sensitive to the selected simulation end-time. This gives confidence that the insights that these environmental releases and offsite consequences are ultimately used to generate are reasonable and reliable, so long as they don't rely on a high degree of precision in the environmental release estimates.

VI. EFFECT OF SIMULATION END-TIME ON PROBABILISTIC RESULTS

To further the discussion on simulation end-time effects, an illustrative example is provided showing how simulation end-time might impact common Level 2 PSA risk surrogates. Risk surrogates are used when full Level 3 PSA results are not available, and while such surrogates are often well-defined qualitatively, there is significant variability in the US with regard to what quantitative definition is used to operationalize the qualitative definitions. For this reason, the evaluation here includes a set of different definitions for two of these risk surrogates. To do this, a sample release category profile is provided in Table 1, which is loosely based on typical internal events Level 2 PSA results for US pressurized water reactors.

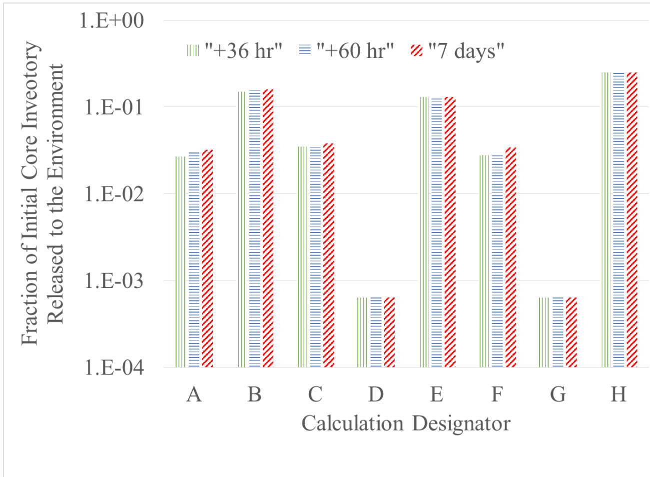

Fig. 1: Sample Cumulative Cesium Releases for Bypass and Early Containment Failure Scenarios

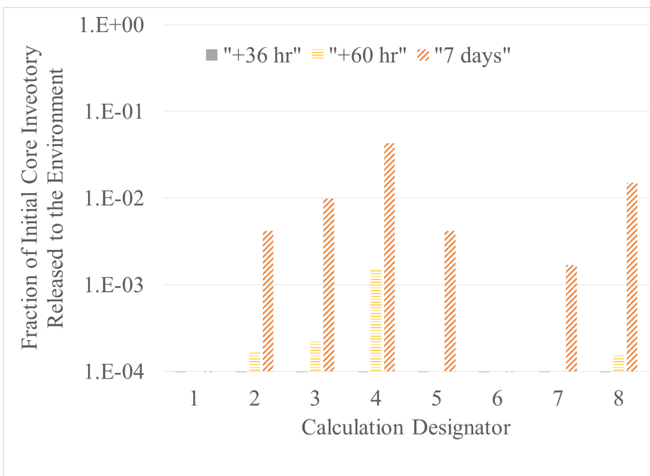

Fig. 2: Sample Cumulative Cesium Releases for Late and No Containment Failure Scenarios

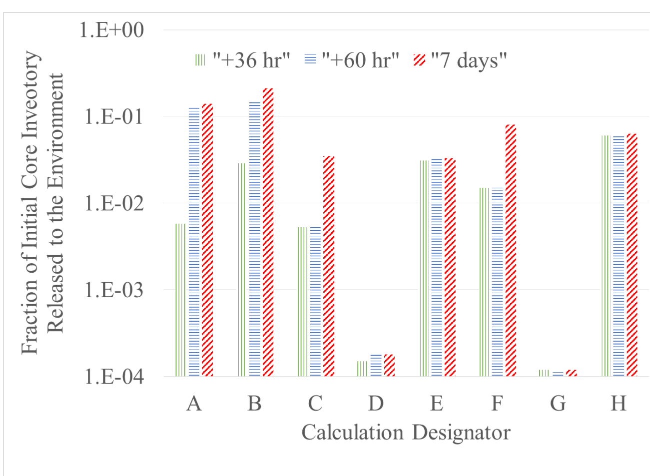

Fig. 3: Sample Cumulative Molybdenum Releases for Bypass and Early Containment Failure Scenarios

TABLE 1: Illustrative Release Category Profile

Release category	Contribution to total release frequency
Interfacing system loss-of-coolant accident (LOCA) with failure of the auxiliary building	2%
Other interfacing system LOCAs	2%
Steam generator tube rupture (SGTR) without scrubbing	2%
Other SGTRs	2%
Containment isolation failure	2%
Containment failure during in-vessel core degradation	1%
Hydrogen-induced containment failure 1 to 24 hours after vessel breach	5%
Long-term containment over-pressure failure	20%
Long-term containment over-pressure failure with scrubbing	20%
Basemat melt-through	10%
Intact containment	34%

The following definitions are used for evaluating the associated risk surrogates. They follow the cited source to the extent practicable:

Large early release frequency (LERF):

- NUREG/CR-6595[9]: In this study, a set of calculations investigates source terms which would lead to an early fatality within 1 mile of the plant and it was estimated that for early releases (within four hours of accident initiation, an iodine or tellurium release fraction of around 2.5% was sufficient to cause a fatality; as applied here, the 4-hour warning time is taken to be the time that the radiological release exceeds 1% iodine minus the time that General Emergency conditions are met.
- Electric Power Research Institute (EPRI) PSA Applications Guide[10]: An unscrubbed containment failure pathway of sufficient size to release the contents of containment within one hour, which occurs before or within four hours of vessel breach, or an unscrubbed containment bypass (e.g. SGTR) pathway occurring with core damage.
- Potential new definition: Release categories are defined to contribute to LERF if their representative source term has a cumulative release greater than $1.4{\cdot}10^{17}$ Bq of I-131 simultaneous with a warning time (based on the time that iodine release exceeds 1% minus the time General Emergency conditions are met) less than 3.5 hours (for early fatalities) or 20 hours (for early injuries).

Large release frequency (LRF):

- NUREG/CR-6094[11]: The authors performed several calculations in order to identify the characteristics of a release that would result in an early fatality; a no-evacuation case with approximately $1{\cdot}10^{13}$ Bq of noble gases and $4{\cdot}10^{11}$ Bq of iodine released had the potential for an early fatality. [Note that this publication pre-dated the common use in the US of LERF.]
- EPRI Utility Requirement Document[12]: The cumulative frequency of all sequences with a dose greater than 0.25 Sv whole body at a distance of 1/2 mile from the reactor assuming exposure to the plume for the first 24 hours after core damage.
- Swiss Federal Nuclear Safety Inspectorate[13]: "…the expected number of events per calendar year with a release of more than $2{\cdot}10^{14}$ Bq of Cs-137 per calendar year."
- Finnish Radiation and Nuclear Safety Authority[14]: "…the mean value of the frequency of a release of radioactive substances from the plant during an accident involving a Cs-137 release into the atmosphere in excess of 100 TBq."
- Potential new definition: The summation of the frequency of all release categories that include containment bypass or containment failure, excluding those where fission product scrubbing (or other mechanisms) result in a source term comparable to, or smaller than, the remainder of the (intact containment) source terms; a cutoff of $8{\cdot}10^{13}$ Bq of Cs-137 is proposed for the threshold.

The results of applying these definitions to the previously-defined release category profile (using actual MELCOR-generated source terms for the release categories) are shown in Table 2, this time considering two different truncation times rather than three. The following observations can be made:

- LERF was not sensitive to simulation end-time, nor was it particularly sensitive to the definition selected;
- LRF was only sensitive to simulation end-time for the proposed new definition (i..e, the last entry in Table 2), and was somewhat sensitive to the definition selected;
- CCFP is extremely sensitive to the simulation end-time.

These observations support the general concepts that one might expect, namely that (i) early injury/fatality risk surrogates are not sensitive to simulation end-time (because the bulk of the early release has occurred prior to a reasonable simulation end-time), (ii) measures of longer-term impacts are only somewhat sensitive to simulation end-time (because bypass scenarios and early/intermediate containment failures tend to drive these impacts) and the use of broadly-defined surrogates tends to mask actual scenario-to-scenario variability, and (iii) containment failure probability is very sensitive to simulation end-time (because a long-term containment failure is just as penalizing as an early failure). Along with reinforcing these concepts, this illustration also suggests that selection of the risk surrogate definition can be as or more important as the selection of the simulation end-time (for LERF and LRF, but not CCFP). A key unresolved question is whether advances in the state-of-practice of severe accident simulation (by reduction of the types of uncertainties described earlier in this paper) will have an effect on these observations.

TABLE 2: Illustrative Risk Surrogate Results

	Risk surrogate: Truncated at 48 hours after core damage / Truncated 7 days after event initiation		
	LERF	LRF	CCFP
NUREG/CR-6595	4% / 4%	-	16% / 66%
NUREG/CR-6094	-	10% / 10%	
EPRI PSA Applications Guide	6% / 6%	-	
EPRI URD	-	14% / 14%	
ENSI	-	16% / 16%	
STUK	-	16% / 16%	
Potential new definition	5% / 5% (injuries)	12% / 16%	
	4% / 4% (fatalities)		

VII. THOUGHTS ON THE IMPACT OF EVOLVING PLANT CAPABILITIES AND RISK ASSESSMENT TOOLS

Like many other countries, the events at the Fukushima Daiichi site in 2011 have prompted additional regulatory requirements in the US. These requirements include the addition of capabilities that have the potential to extend the timeline of some typical PSA sequences. Such capabilities, if they are to be included in quantitative risk assessments, will necessitate the need for furthering the state-of-practice in modeling such long-running scenarios, including the local actions and offsite support that are intrinsic to these capabilities. The development of better analytical competencies for this purpose can help to reduce the types of uncertainties described earlier in this paper.

While traditional risk assessment approaches (namely event tree / fault tree PSA) are likely to remain the central focus of nuclear power plant PSAs, there is also an opportunity to utilize other types of tools to address the types of analytical limitations described herein. In particular, discrete dynamic event tree (a.k.a., simulation-based PSA, integrated deterministic probabilistic safety analysis) offers an opportunity to directly couple the exploration of baseline scenarios and uncertainty affected by limitations in phenomenological modeling, system response, and human performance. These tools can complement traditional approaches when addressing uncertainties in long-running simulations.

VIII. CONCLUSIONS

The observations in this paper are offered in the context of PSA technology, they are not intended as views or guidance with respect to regulatory requirements associated with any of the topics covered herein.

This paper asserts that there are many important considerations when choosing a simulation end-time for deterministic analyses that will support a PSA, and that long-running simulations challenge the state-of-the-art in nuclear power plant accident modeling, in terms of phenomenological, systems, and human performance modeling. Nevertheless, the paper has also illustrated that only a subset of the deterministic and probabilistic results are inherently sensitive to the simulation end-time selection, and thereby, the selection of a simulation end-time can be made in concert with the specific set of goals that the analysis seeks to achieve. The evolution of modeling techniques (and associated reduction of uncertainty) has the potential to affect this conclusion, and so it is important that the observations herein be viewed within the context that they are offered.

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