7ICMSNSE-071

Benchmarking of fast-running software tools used to model releases during nuclear accidents

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Abstract

Fukushima highlighted the importance of effective nuclear accident response. However, its complexity greatly impacted the ability to provide timely and accurate information to national and international stakeholders. Safety recommendations provided by different national and international organizations varied notably. Such differences can partially be attributed to different methods used in the initial assessment of accident progression and the amount of radioactivity release.Therefore, a comparison of methodologies was undertaken by the NEA/CSNIand its highlights are presented here. For this project, the prediction tools used by various emergency response organizations for estimating the source terms and public doses were examined. Those organizations that have a capability to use such toolsresponded to a questionnaire describing each code's capabilities and main algorithms. Then the project's participants analyzed five accident scenarios to predict the source term, dispersion of releases and public doses.

1. Introduction

After the events at Fukushima, the NEA recommended that "CSNI [Committee on the Safety of Nuclear Installations] should analyze the comparison of source term methodologies utilized by countries and determine if or why the dose prediction differed for Fukushima." Therefore, a comparison of methodologies was undertaken by the CSNI; a summary of itis presented in this paper. More specifically, the following information is provided:

- A list of the software tools for assessing the source terms
- Summary of accident scenarios developed to determine the source term and that compare the software tools' capabilities
- A comparison of some of the software tools' results from modelling the accident scenarios

Twenty organizations, representing twelve countries and two international organizations, participated in this benchmarking project. Between them, a total of seventeen software tools were included in this exercise and used to assess five hypothetical accident scenarios.

1.1 Objective

The objective of this CSNI activity was to benchmark software tools used to estimate consequences of accidents at nuclear facilities. This activity and the proposedfollow-up activitiesare expected to promote better understanding of the existing predictive capability currently available in a number of organizations to rapidly assess and recommend protective measures during nuclear emergencies. Several recommendations are provided to direct future efforts in this area.

2. Methodology

The benchmarking activity was carried out in three steps:

1. Identifying software tools to be compared;
2. Selecting accident scenarios for the fast-running software tools to model for a comparison;
3. Having the software tools simulate the accident scenarios and comparing the results;

2.1 Identifying software tools

The first step in the project was to identify software tools to be included in the benchmarking exercise based on their ability to meet the following criteria:

- Calculate the fission product source terms and provide an estimate of core damage state and the condition of the physical barrier
- Predict doses resulting from fission product releases
- Ability to run with small number of input parameters (at the start of a nuclear accident only limited information will be available for use)
- Incorporate additional details as more information becomes available and improve the predicted results
- Versatility in dealing with different reactor technologies
- Speed of calculation
- Accuracy and confidence in the results
- Output the results in a clear, user-friendly and logical manner that can be useful in recommending necessary actions

This was accomplished by requesting the participating organization to bring forward themodelling software that is currently used in their respective organizations for modelling of the fission product releases from nuclear facilities during emergencies. The list of software tools that were involved in this undertaking is presented in Table 1.

Questionnaires forcharacterization of these tools were developed early into the project and distributed to the participants. The information collected allowed getting a better understanding of how the tools work, what they are used for, and what their strengths and weaknesses are.

Country	**Organization**	**Software Tool**
Belgium	Bel V	CURIE V5[1]
Canada	Atomic Energy of Canada Ltd. (AECL)[2]	RASCAL 4.3
	Canadian Nuclear Safety Commission (CNSC)	RASCAL 4.3
		VETA
	Health Canada (HC)	ARGOS
		MLPD
Denmark	Danish Emergency Management Agency (DEMA)	ARGOS
France	Institut de radioprotection et de sûreté nucléaire (IRSN)	MER
		PERSAN
		C^3X
Germany	Areva	MC_Transport
	Gesellschaft fur Anlagen und Reaktorsicherheit (GRS)	ASTRID
		QPRO[2]
	Karlsruhe Instute of Technology (KIT)	RODOS
	Ministerium fur Umwelt, Klima und Energiewirtschaft/University of Stuttgart	ABR
India	Nuclear Power Corporation of India Ltd. (NPCIL)	ACTREL
Italy	Agenzia nazionale per le nuove tecnologie, l'energia e lo sviluppo economico sostenible (ENEA)	IDRA[2]
Korea (Republic of)	Korea Atomic Energy Research Institute (KAERI)	XSOR (SURSOR)[3]
		MACCS2[4]
Poland	National Center for Nuclear Research (NCBJ)	MELCOR 1.8.4
		RODOS
Slovakia	ABmerit	ESTE
	VUJE	RTARC
Sweden	Swedish Radiation Safety Authority (SSM)	RASTEP
United States	Nuclear Regulatory Commission (USNRC)	RASCAL 4.3.1
International	European Commission (EC) – Joint Research Centre (JRC)	MAAP4 4.0.8
	International Atomic Energy Agency (IAEA)	InterRAS[2]

Table 1: Participants and tools for the FASTRUN benchmark

[1] While information was provided on CURIE, QPRO, IDRA, and InterRAS, they were not actually used to run the scenarios
[2] In November, 2014, the section of AECL that contributed to this project became Canadian National Laboratories
[3] XSOR is KAERI's fast-running software tool for generic PWRs. SURSOR is designed specifically to model the Surry reactor. In section 2 the capabilities of all XSOR models are discussed. However, KAERI only provided results for the Surry scenario (see section 3) using SURSOR. Therefore, all KAERI results presented in section 4 are from SURSOR
[4] While KAERI used MACCS to determine dose results, no information on the software tool was provided

2.2 Selecting accident scenarios

The second step in the project was to select a set of appropriate hypothetical scenarios for the software tools to model. The scenarios were developed to represent several types of nuclear reactors: PWR, BWR, and CANDU reactors. The amount of data given out initially was limited, so the participants could perform blind simulations.

It is important to note that *the accident scenarios used in this benchmarking exercise are hypothetical. They were deliberately selected to represent relatively extreme cases, regardless of the fact that they would be* ***extremely unlikely****.*

Five different hypothetical accident scenarios were developed and used in the benchmarking modelling, which are as follows:

- An unmitigated, long-term station blackout at Peach Bottom Unit 3, an American BWR
- An unmitigated, long-term station blackout at Surry Unit 1, an American PWR
- A transient resulting in a loss of residual heat removal at Oskarshamn Unit3, a Swedish BWR
- A large break LOCA with failure of safety functions at Golfech Unit 1, a French PWR
- A station blackout with emergency power generators at Point Lepreau, a Canadian PHWR

The scenarios were at some point modeled using detailed, analytical software tools. The Peach Bottom and Surry accident scenarios were modeled with MELCOR by the USNRC, the Oskarshamn scenario was modeled with RASTEP by the SSM, the Golfech scenario was modeled with ASTEC by the IRSN, and the Point Lepreau scenario was modeled with MAAP4-CANDU by AECL.

For each of the scenarios three different datasets were presented. These datasets are meant to represent the different amount of information that would be available to emergency response organizations at different times during the accident progression. The three datasets are as follows:

- 1 hour into the accident scenario where only the location and initiating event are known;
- 6 hours into the accident scenario at which point information on the state of core cooling is available, as well as a slightly more detailed account of the accident scenario;
- 24 hours into the accident scenario at which point the status of containment is known;

Table 2 shows the information provided for the accident scenarios.

Time data is available	**Peach Bottom**	**Surry**	**Oskarshamn**	**Golfech**	**Point Lepreau**
	BWR	PWR	BWR	PWR	CANDU
	3514 MW(th)	2546MW(th)	3900MW(th)	3817MW(th)	2180 MW(th)
1 hour after accident	Reactor shutdown successful	Reactor shutdown successful	Reactor shutdown successful	Reactor shutdown successful	Reactor shutdown successful
6 hours after accident	AC power lost at 0:00 and batteries depleted at 4:00	DC power available until 8:00 (AC power lost at 0:00)	Power available	Power available	Loss of AC power. Emergency generators power ECCS valves for 12 hours
6 hours after accident	Water level reaches top of active fuel at 8.4 hrs	Water level reaches top of active fuel at 14 hrs	Core uncovered – after 12.7 hours Loss of residual heat removal	Water reaches to of active fuel after 10 minutes	ECCS starts at 3.9 hours; ends at 11 hours Water is depleted inside the calandria at 22 hours
1 day after accident	Containment pressure: 690kPa @ 20hr 138kPa @ 24hr	Containment pressure at 24 hours: 350 kPa	Containment pressure at 24 hours: 6.7 bar abs.	Containment pressure at 24 hours: < 5 bar	Containment pressure at 24 hours: 130kPa(a)
1 day after accident	Containment fails at 20 hours	Containment fails at 45.5 hours	Filtered venting starts (9-10 kg/s) at 13.8 hours	Venting starts when pressure reaches 5 bar. Planned at 1.5 days	At 13 hours the airlock seals fail.

Table 2: Example of information provided for the different datasets

The purpose of having three different datasets each with varying amounts of information available is to examine how well different software tools cope with a limited amount of information.

In addition to the limited information on the accident progression, all participants were provided with meteorological data for each accident scenario so that plume dispersion could be modeled. The meteorological data is representative of real weather conditions at and nearby the reactor sites on certain dates.

2.3 Simulating the accident scenarios

The third and most important step in the project was to use the software tools in simulation of the selected scenarios. Three datasets were availablefor the scenarios; each dataset representing the information that would be available to the external (that is, not involved directly in the managing the accident) organizations after certain duration into the accident and as such the amount of information they provided varied; however some organizations only used one or two of those. The participating organizations used their software tools to run these datasets for as many of the scenarios/reactor types as they considered possible.

3. Results

It was observed that the predicted source terms varied significantly from each other and from the baseline source terms calculated by the analytical tools, sometimes by orders of magnitude. Similarly, the doses predicted by the dispersion models also varied significantly. Factors affecting the calculations are discussed in the subsequent subsections.

3.1 Factors affecting source term calculations

All participants were provided with limited information on the accident scenarios, such as the nature of the accident, the state of core cooling, and the nature of the releases. However, most software tools required more input forcing the participants to make assumptions regarding the accident progression. These assumptions caused variations, some significant, in the source term estimates.

3.1.1 Limited information

One fundamental aspect of this benchmarking study was to see how the existing fast-running software tools would model the scenarios when only limited information was available as would be the case early on into a real accident at a nuclear facility. In the end, only four tools were used by the participants to calculate source terms based on the data sets provided for the three accident times (RASCAL, MAAP4, ASTRID, and SURSOR).

It was found that a code's ability to predict consistent results with limited information can vary depending on the accident scenario. For example the CNSC's RASCAL results were all approximately the same for all datasets, while analysis by another participantof the 24 hour dataset predicted radioiodine releases more than an order of magnitude lower than the corresponding analysis of the 1 hour dataset. (In that particular case, this was because for one hour dataset, it was assumed that the reactor core isolation cooling only operated until DC power was exhausted.) However, the code whose results varied by more than an order of magnitude for the Peach Bottom scenario also predicted radioiodine estimates varying by a factor of less than 1.5 when analyzing the different datasets for Surry. Meanwhile the CNSC radioiodine estimates decreasedby more than a factor of three from between analyzing the 1-hour and 24-hour datasets of the Surry scenario, due to additional information made available for the 24-h dataset.

Probably the most important assumption that participants had to make was what the end state of the accident scenario would be. Many participants, such as the CNSC assumed that the scenario would progress to a severe accident and that a major release would occur and therefore modeled the early datasets with containment failure for certain scenarios.

On the other hand, other organizations did not make any assumptions about the end state of the accident based on the information present in the datasets. Therefore, they assumed that containment remained intact when modeling the early datasets. The result of these two different views is source terms that are orders of magnitude apart for the early datasets. This outcome is probably indicative of the projections of the accident end-state in a real emergency while it still develops

3.1.2 Core inventory

Certain tools have specific reactors built into them, typically the reactors present in the country where the software tool is used. For example, RASCAL has all the American plants built into it. But as this work involved reactors located in four different countries, participants often had to adjust the models built into their software tools. Part of this was adjusting the equilibrium core inventory for each scenario. For many software tools, this was done by scaling the built in inventories by reactor thermal power and average fuel burnup.

However, the fact that the noble gas source terms results varied between the different software tools and the results of the analytical tools, indicates that the initial estimates for core inventory were not necessarily accurate. As the fuel melts, all the noble gas fission products are released and noble gases are unaffected by natural and engineered deposition processes. Given that all scenarios presented in this work proceed to core melt, it would be expected that the noble gas source terms would all be quite similar. The Point Lepreau analyses in particular showed divergence between the noble gas source terms, with MC_Transport basing the Point Lepreau initial inventory off of a German BWR and then attempting to scale the BWR inventory based on reactor power. (Difficulties of software tools modeling reactor types they were not designed to model are discussed further in section 3.1.7).

3.1.3 Release pathway

The effect ofthe release path on the source term magnitude is most clearly seen in the BWR scenarios, particularly Oskarshamn. This is because, in BWRs, the participants had to assume whether or not the release traveled through the suppression pool which would scrub some of the fission products from the release.

A limitation of RASCAL is that the release can only go through either the wetwell or the drywell; users can’t change the release path partway through an accident. For the Oskarshamn scenario the CNSC assumed that the release was through the wetwell and when analyzing the 6 hour dataset, the USNRC made the same assumption. The CNSC’s results for the 6 hour dataset, which assumed a wetwell release only, are about an order of magnitude lower than the baseline RASTEP predictions made by a Swedish organization. In analyzing the 24 hour dataset, though, the USNRC assumed that the release was through the drywell with an external reduction factor applied to account for the initial hours of the release prior to the drywell melt through. Other software tools are not limited to one of the drywell or wetwell. ESTE allowed for the release to start off passing through the wetwell, before going through the drywell later in the accident scenario. With MC_Transport the release went through the drywell; however, it was assumed that the drywell was flooded, so some material would be removed prior to the release.

3.1.4 Filtered venting

The Oskarshamn and Golfech scenarios both ended in filtered venting. However, not all the software tools were able to model the specific filter systems installed at those reactors (a multi-venturi scrubber system at Oskarshamn and a high efficiency sand filter at Golfech). Some software tools only modeled basic filters. For example, while RASCAL does include a default external filter model, it does not model the specific filter designs used at Oskarshamn and Golfech. Because of this, when the USNRC modeled Oskarshamn and Golfech, they exported the source terms, applied a reduction factor, then imported the source terms back into RASCAL. The CNSC did not do this and instead relied on the default filter models built into RASCAL. As a result of this, the RASCAL calculations predicted higher non-noble gas source terms than ASTEC calculations, which used a more mechanistic filter model.

For the software tools that do have filtered venting models, it appears the filters lead to under predicting the source term. Many of the fast-runningsoftware tools predicted lower releases than RASTEP and ASTEC did for Oskarshamn and Golfech respectively for most aerosol fission products (e.g. Cs). This discrepancy is typically within an order of magnitude.

3.1.5 Defining containment failure

Participants tended to model containment failure one of two ways: either by drastically increasing the assumed leak rate (as was used by ESTE), or by specifying pressure in containment and assuming a hole of a certain size in the containment structure (such as is typically done in RASCAL). The CNSC used both approaches (using an assumed leak rate for the 1 and 6 hour datasets and pressure measurements for the 24 hour datasets).The CNSC found that defining containment failure based on pressure and an assumed hole size led to lower releases and aerosol source term that were closer to the analytical results.

3.1.6 Radioiodine speciation

It was found that several software tools predict much greater iodine releases than the ASTEC analysis calculated for Golfech, whereas most of these software tools under predicted the amount of caesium released compared to the ASTEC calculation. These software tools assumethe mainchemical form of the radioiodine released into containment during an accidentis either organic iodine or elemental iodine.Organic iodine is not affected by filters, while the filter efficiency for removing elemental iodinewas assumed to be 50%. These tools all calculated iodine releases on the order of 10^5TBq. However, the baseline results by a mechanistic code (ASTEC) indicate that the amount of organic and elemental iodine is significantly lower than estimated by the fast-running tools, in the order of 10^2TBq. A similar effect can be seen in one of tool's modeling of the Oskarshamn scenario. The tool estimated radioiodine releases on the order of 10^4TBq, while the Swedish software tool RASTEP, designed specifically for Oskarshamn, estimated releases of just under 10^3TBq.

Software tools that assumed an aerosolized particulate was the dominant form of iodine in containment did not necessarily predict more accurate iodine source terms. However, they would over-predict or under-predict both the iodine and caesium releases as compared to the base case.

3.1.7 Knowledge of different reactor designs and ability to model them

All the software tools, regardless of their capabilities, have to be run by human operators. The effect of the users assumptions based on limited information was previously mentioned. However, even if the user has a significant amount of information regarding how the accident started and how it is progressing, a lack of familiarity with a reactor design could easily lead to problems modeling the situation. Different organizations would have expertise on certain reactor types, often based on which reactors are present in their countries. For example, the Areva office participating in this exercise is based in Germany, whose nuclear fleet include PWRs and BWRs, but no CANDU reactors. As a result, Areva was not familiar with the CANDU design and had to make several assumptions in order to model the Point Lepreau scenario. Areva based the MC_Transport analysis of Point Lepreau off of a BWR design. However, later discussions indicated that a PWR model would have been more appropriate due to similarities in containment. The resulting discrepancies in the noble gas and strontium source terms could be due to this lack of familiarity.Conversely, the CNSC was able to model all LWR accident scenarios using RASCAL. However, the CNSC's estimates on volatile releases were an order of magnitude or more greater than theresults given by detailed mechanistic codes with proper plant models, and often greater than the other predictions, while noble gas source terms were estimated to be less than the analytical results. These discrepancies could be caused by CNSC staff's lack of familiarity with accident progression with LWR designs as Canada only has CANDU reactors. Work is underway at the CNSC to improve on this.

The purpose of including the Point Lepreau scenario in this benchmarking exercise was to examine how well the existing tools handled a reactor design that is less known in many countries and it appears that few software tools can be quickly adjusted to model an accident at a reactor of less common design. It is worth remembering that in the event of a nuclear emergency, national organizations involved in response to nuclear emergencies will be looked to for an estimate of what material may be released and the dose consequences, regardless of their familiarity with the type of the reactor experiencing the accident.

In the end only, three tools attempted to model the Point Lepreau scenario and only two tools, ESTE and MC_Transport, were used to model all five accident scenarios.Two more software tools, RASCAL and PERSAN, were used to model all but the Point Lepreau scenario, although the CNSC did use RASCAL to model dispersion and dose for the Point Lepreau scenario.

3.2 Factors affecting dose calculations

3.2.1 Source term

For the Oskarshamn, Golfech, and Point Lepreau scenarios there was no standard source term, although RASCAL and ARGOS did use the same source term for the Point Lepreau scenario. Because the different software tools considered different amounts of different radionuclides, it is not surprising that the resulting doses are different. The CNSC and the USNRC used the same software tool and the same weather data but different source terms. As a result the behavior of the doses over distance was similar, but as the CNSC's predicted a lower source term, the CNSC's doses were consistently lower than those of the USNRC.

Even for the Peach Bottom and Surry scenarios, which have a common source term,different software tools may handle it differently. The common source terms for the two SOARCA scenarios provided to all participants consisted of 66 different radionuclides. It could be that certain tools used for dispersion do not consider certain radionuclides and/or use different meteorological sources. For example, ARGOS's Lagrangian dispersion model can only handle 20 radioisotopes when combined with the Canadian Meteorological Centre's weather data model (its Gaussian puff model can handle more). More complex models require more computational power in analyzing the dispersion of each isotope, which could force them to limit how many are considered. With different tools considering different isotopes, it would be expected that the doses are different.

3.2.2 Dispersion models

It was mentioned in the previous section that different dispersion models may be able to handle different amounts of radioisotopes due to varying complexity. The different degrees of complexity lead to the different models handling the dispersion quite differently. In comparing the maximum doses over distance, ABR and ESTE calculated a much more gradual decrease than the other software tools. The dispersion model in ABR is a Lagrangian particle model and ESTE used its Lagrangian particle dispersion model to analyze the scenarios. Other software tools (RODOS, ACTREL, and MACCS) used a simpler Gaussian model when estimating doses for the scenario and the doses that they predicted were noted to drop off much more quickly as one moved away from the reactor. Also, ABR uses radiation transport calculations that consider 30 energy groups when determining cloudshine doses. As a result, this often yields higher doses than simpler methods.

3.2.3 Wind speed

The meteorological data provided for Surry calculations included very low wind speeds with variable wind directions. It is expected that the emergency response codes would exhibit significant differences for this case because they likely handle this condition differently. RASCAL contains a low wind speed correction algorithm in its atmospheric transport and dispersion models that account for this meteorological condition. If other codes do not include this type of low wind speed correction in their diffusion coefficients, their resulting dispersion will vary from the RASCAL results.

3.2.4 Terrain

The terrain near a reactor affects how much of the radioactive material gets deposited on the ground. Rougher terrain (e.g. urban areas and forests) would result in more of the released material depositing on the surfaces, increasing groundshine in that region, but depleting the plume so the cloudshine and inhalation doses would be reduced further downwind.How the terrain is defined would not only affect the dose values but the shape of the plumes as well. This could be the reason why the shapes of the plumes for the Golfech scenario vary. When calculating the doses, ABR set the release point at the Gravelines plant instead of Golfech. Also, the CNSC used the Seabrook reactor as a surrogate for Golfech, which is located in the U.S. state of New Hampshire. Both the Gravelines and Seabrook sites have water to the northeast. With minimal surface roughness, the plume would easily spread out in that direction and little material would be deposited.The actual Golfech site though, is located in the Midi-Pyrenees region in southwest France and to its northeast are the mountains of the Massif Central, which would be far more difficult for the plume to travel over than water. ESTE analyzed the release coming from the Golfech site, and with the mountains blocking the flow path to the northeast, ESTE may have determined that the plume deflected to the southeast of the reactor.

3.2.5 Release timing

For the Oskarshamn scenario different assumptions were made on the duration of the release to the environment: some participants (e.g. CNSC) assumed venting lasted for one hour while others (e.g. USNRC, ABmerit) assumed it lasted for ten hours. As wind shifts over ten hours, the area over which the plume blows is increased.Codes that ran one hour release predicted a narrow plume confined to the northeast of the reactor.Codes that assumed a ten hour release, though, showed a broad plume ranging from the northeast to the southeast of the reactor.

The release timing also affects the magnitude of the doses. With everything being released in one hour, there’s less time for the wind to shift and disperse the material. Therefore, with greater concentrations of the radionuclides in one location, there will be greater maximum doses.

3.2.6 Dose conversion factors

Even if two software tools were to model the dispersion the exact same, the calculated doses could still be different depending on what dose conversion factors the software tool used. Dose conversion factors (DCFs) are estimates of how much dose a receptor would receive per unit of activity of a specific radionuclide via a specific exposure pathway. Different organizations have different DCFs. For example, the International Council of Radiation Protection, the U.S. Environmental Protection Agency, and the Canadian Standards Association have all produced guidance reports detailing DCFs for different radionuclides. These guidance reports are not necessarily identical and therefore codes that use different DCF would predict different doses.

Also related to the DCF’s is the assumed inhalation rate for receptors. ABR uses an increased rate of inhalation to calculate thyroid doses. This assumption, along with the DCFs ABR uses (which are mandated by German law) likely contributes to the fact that ABR’s predicted doses were almost always greater than the other participants. There are also different dose conversion factors for different age groups as children are more susceptible to radiation. Different software tools may have assumed different receptors. Codes that used children as the receptor would have predicted higher doses than tools that used adults as the receptor.

4. Conclusion

The benchmarking showed that the software tools available will likely provide results differing to a certain degree when attempting to calculate source terms based on limited information, despite assessing the same scenarios. Some reasons why the source terms differ were identified and discussed. Important factors involve both the software models, and the assumptions made by the participants when inputting data. The assumptions made on the basis of limited information were shown to have a major effect on the predicted source terms. Some participants, after being given the accident scenarios, assumed that a release was inevitable and defined such a release in the code input parameters. Other participants assumed that unless a release was specifically stated in the dataset (it was only explicitly stated in the 24 hour datasets for all scenarios) no release occurred besides leakage. The effect of these differences in assumptions can clearly be seen as source terms produced by two different organizations using the same software tool were more than an order of magnitude different for certain scenarios.

Other significant factors affecting source term prediction include:

- Initial core inventory
- Definition of the pathway to the environment.
- Code capability to model certain systems.
- Assumptions related to the containment failure.
- Modeling of chemical species of radioiodine.
- Knowledge of and ability to model different reactor designs

As for doses, several factors affecting the results were identified

- Different source terms
- Different dispersion models
- How weather data is handled by the software tools
- How the terrain was defined
- The timing and duration of releases
- Different dose conversion factors

One apparent option for the future work could be a more in-depth comparative study of the software tools to further understand the causes for the different source term. In a similar vein,an in-depth comparison of software tools used to predicts doses would also be a reasonable follow-up activity.A more fundamental activity to consider would be to take some of the strategies of quickly analyzing accident scenarios, as presented in this benchmarking project, and apply them to build an enhanced system to rapidly diagnose an accident scenario and predict consequences. As well, a future forum for exchange of best practices and perhaps, hands-on training for the users of fast-running tools for prediction of accident consequences could be considered.

Regardless whether the above activities are undertaken or not, the outcomes of this project should be useful to the practitioners in the field to better understand the existing capabilities for rapid assessment of the accident source term as well as their limitations and the need for future improvements.