

ANNEX 1: NEA NUCLEAR INNOVATION 2050

R&D COOPERATIVE PROGRAM PROPOSAL

TITLE/SUBJECT: Enhancing Severe Accident Management in existing and future NPPs

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1. Justification of the selection

1.1 Major expected program outcomes

For the nuclear power to keep a significant level of contribution in the energy mix in the future¹ there must be a renewed and strengthened assurance that nuclear power is safe and reliable which is a prerequisite for wider acceptance by the public and also for strengthened economics and sustainability.

Looking forward three decades from now, it is conceivable that future NPPs may be a mix of LWR and non-LWR varieties. In all likelihood, a number of fuel types will be in use in the future. For any type of NPP and fuel, strengthened safety measures will have to be implemented to avoid the possibility of a severe accident (SA) with large enough radioactivity release in the environment to cause significant adverse effect on public health, ecology and biota and to incur significant and long term societal cost.

The severe accident (SA) at the Fukushima Daiichi nuclear power plant (NPP) in March 2011 was a reminder that despite every effort to prevent severe accidents they can happen and that the need to enhance mitigation strategies remains for existing NPPs but also for future NPPs as long as the possibility of accidents with significant adverse effects, as described above, is not eliminated by design. The present proposal is thus addressing enhancement of severe accident mitigation strategies, including for containment of radioactivity, in existing and future NPPs.

For existing LWRs, the Fukushima Daiichi accident, complementary safety evaluations performed afterwards in NPPs worldwide and the on-going challenging works to prepare the decommissioning of the damaged facilities have highlighted that management of all phases of a SA from the emergency phase up to the long term phase following plant stabilization could still be significantly enhanced. A considerable amount of research work has been completed over past decades [1,2,13,14] but more remains to be done *to consolidate knowledge and tools directly supporting demonstration and implementation of safe and effective severe accident management (SAM) strategies at NPPs.*

Moreover, future research in the field should be adapted in the near future to cope with the rise of new technologies, new fuel concepts, new reactor designs, including non-LWRs to be commensurate with the expected reactors fleet for 2050 and beyond, and with the need of more integrated knowledge management in the field to be informed at best by decades of past research and plant data and information from past accidents such as TMI-2 and Fukushima-Daiichi and to a degree from Chernobyl.

A cooperative international research program involving all stakeholders (regulators and their technical support organizations (TSOs), research organizations, industry) collecting, sharing and producing relevant knowledge should have a significant positive impact on safety of NPPs worldwide with (1) *possible improvements in operating plants* (mostly back-fitting in generation II plants) considering in some countries and for some of them their expected extended operation beyond the initial design lifetime (2) *consolidation of safety demonstration of SAM strategies for more recent and future designs*

¹ The World Nuclear Association's most recent forecast points to a 25 % contribution of nuclear power in the energy mix. This level of contribution can only be achieved if many more NPPs are constructed and operated.

for which design options are mature (generation III/III+, some generation IV plants and small modular reactors (SMR)) and (3) establishment of innovative solutions for future designs with possible applications to existing ones.

In addition, such a program would support maintaining or developing necessary and critical resources, research facilities and capabilities when many countries that had conducted SA research in the past decades, e.g. European countries, USA and the Republic of Korea, are currently struggling to do it, while other countries are building new capacities, e.g. Japan in response to the Fukushima-Daiichi accident or China to support its NPPs deployment.

A cooperative program supporting the *development of more harmonized severe accident management guidelines (SAMG)* ensuring reduction of adverse effects of a SA would certainly strengthen economics and sustainability of NPPs in the future facilitating new and innovative industrial developments for safe SA mitigation. As such, it would also be of interest to decision-makers responsible for outlining energy plans to cope with the climate change, the security and affordability of energy supply.

It would also contribute to wider acceptance by the public and authorities in charge of emergency management producing *knowledge beneficial to emergency preparedness.*

At this stage, it is recognized that the present version of the SA program proposal (established in June 2018) is built focusing primarily on LWR technology and considering a conventional LWR-centric definition of a severe accident, i.e. an accident leading to fuel degradation and potential loss of containment and consequent release of radioactivity. This reflects that in the near term, up to about 2025, consolidation of SAM strategies and supporting knowledge base for LWR NPPs will be the focus while it is proposed to launch at the same time actions to further prioritize research needs for future NPPs designs. It should be emphasized that an enhanced knowledge base for LWR NPPs will certainly inform definition of research needs for other designs. The program focus and scope will have to be adjusted in the near term to address by 2025 more specifically future designs and be more commensurate with the reactor fleet profile foreseen for 2050 and beyond. Program revisions will have to be internationally discussed on a regular basis in close partnership between industry, utilities, manufacturers, regulators and researchers.

It should also be emphasized that the document is not intended to be a roadmap for SA accident research, addressed in documents that have been recently released in Europe, Japan and USA [33-37] after the Fukushima-Daiichi accidents. It is rather intended to highlight areas where maintaining or developing resources, research facilities and capabilities is critical. However, information from these documents is implicitly considered in the present proposal.

1.2 Overall technical scope of the program

Considering a SA with significant fuel degradation in a NPP, in reactor or in SFP, the following “primary” safety targets would have to be met by the operator: avoid any re-criticality event that would negatively affect the plant safety², ensure degraded fuel cooling and retention – either in the reactor vessel (in-vessel fuel retention, IVR) or outside the reactor vessel (ex-vessel fuel retention, EVR) depending on the reactor design -, maintain confinement of radioactivity and, in case of failure, mitigate as far as possible radioactive releases to the environment. These “primary” safety targets are both for phases where the operator aims at reaching a “stabilized” plant state³ after a SA but also for

² Primarily the confinement of radioactivity.

³ Depending on the situation, different “stabilized” plant states may be envisaged (e.g. the confinement may have been damaged at different extents during the accident). It is generally considered that a “stabilized” state

longer term phases where the operator has to maintain this “stabilized” state, up to damaged fuel retrieval from the damaged plant⁴. Further, their final common aim is the limitation of radiological consequences by maintaining as much as possible the confinement of radioactive materials and by reducing as much as possible people exposure - on and offsite - during all accident phases.

Gaining additional knowledge and establishing harmonized methods to better assess risks of losing the confinement of radioactivity for any reactor designs and designing improved measures to efficiently monitor and mitigate these risks, if significant, are in the scope of the research program and of critical interest for designers, operators, regulators and decision-makers.

One should also highlight that additional critical safety issues have been identified in the aftermath of the Fukushima Daiichi accident as deserving further investigations, e.g. management of spent fuel pool (SFP) severe accidents, of accident resulting from major external events and involving more than one NPP unit, safety systems and components performance and reliability under SA conditions - including passive systems and structures and components ensuring confinement -, instrumentation dedicated to SAM, human and organizational factors (HOF), effect of ageing on SAM, ... Some of these issues (SFP severe accidents, SA instrumentation, safety systems and components performance and reliability) will be addressed to some extent later in the program proposal, some (passive safety systems and ageing issues) will be treated in separate NI2050 proposals and are thus not addressed in detail in the present document. Others (external events and multi-units accidents, HOF) have not been considered at the present stage in NI2050 but they may be considered in the future as cross-cutting issues.

It should also be stressed that data and information collected through the decommissioning process of the Fukushima Daiichi damaged NPPs would deepen knowledge of SA phenomenology and help narrowing down remaining research gaps.

Behavior under SA conditions of advanced and new fuels and materials has to be assessed during their development. This is partly addressed in the present proposal but it could as well be included in programs dedicated to the development and licensing of advanced and new fuels and materials which are presented as separate NI2050 documents.

1.3 Main tools, skills, expertise and infrastructures for the program

The program should also sustain and foster the development of (1) unique research infrastructures able to produce data easily transposable to SA modeling at reactor scale – i.e. enabling to address scaling issues, e.g. fuel platforms for degraded fuel cooling and retention studies, radiation chemistry platform for source term (ST) studies (2) SA codes and material databases that can support SA analyses and validation of SAMG with the treatment of realistic complex configurations at proper scale for best-estimate calculations (e.g. for the modeling of degraded fuel retention and cooling in-vessel and ex-vessel in relation to IVR and EVR strategies) (3) knowledge of uncertainties and of safety margins of severe accident strategies (4) instrumentation, methods and analytical tools supporting implementation of SAMG, decision-making during the emergency phase (prognosis/diagnosis) and development of simulators.

is reached when the damaged fuel is cooled and sub-critical and the risk of energetic events and radioactive releases are mitigated.

⁴ These long term phases encompass recovery actions such as site clean-up and decontamination, radiological wastes management and damaged fuel retrieval.

It is clear that the program would enable, and benefit from, a large partnership between regulators, TSOs, research organisms, utilities, and industry and a stronger cooperation between experts from the SAM, emergency management and environment management fields. Multilateral cooperation with a search for harmonization in approaches is key to efficiently progress in the field.

2. Challenges and opportunities

As briefly exposed in chapter 1, a cooperative program to consolidate SAM could cover a wide range of topics of safety interest. The NI2050 program could first cover selected topics for which (1) it is important to maintain unique and endangered competences and infrastructures and to help building resources and funding for the performance of well-targeted large multilateral research programs (2) effective results in terms of technological developments supporting SAM strategies and providing significant safety improvements – with potential back-fitting developments in generation II and III plants - can be expected in the next 5 to 10 years (3) results supporting the licensing of new fuel concepts (e.g. accident tolerant fuels (ATF)) and the development of future designs (e.g., generation IV and SMR) can be expected in the next 10 to 20 years (4) innovation can be pushed for future industrial developments to be deployed by the 2040's.

In addition, it is also proposed to cover in the program transverse topics dealing with SA knowledge collection, review, sustainability, sharing and integration in SA analysis tools and methodologies as they are critical for SAM consolidation for existing designs and for proposing innovative SAM developments for future designs. They are also critical for designing well-targeted research in the field taking advantage of the largest possible existing background and of more harmonized approaches.

Challenges and opportunities for four selected “high priority” topics are thus further presented in this chapter. These topics are:

- Topic 1: Consolidation of degraded fuel management strategies (retention and cooling) for existing reactors and development of innovative ones for future reactors (LWRs and non-LWRs). This topic includes reinforcement of elimination of energetic events that could challenge the confinement of radioactivity. Behavior of new fuels in SA conditions, in particular where chemical reactions involving new clad and new fuel materials can affect SAMG (e.g. IVR strategies) is addressed in this topic,
- Topic 2: Reinforcement of confinement preservation measures, radioactive releases mitigation means and consolidation of Source Term (ST) evaluations for existing and future reactors. New fuels specificities in relation to ST should be addressed in this topic,
- Topic 3: Reinforcement of SA instrumentation in NPP including harmonization in qualification, behavior during a SA of NPP existing instrumentation and development of innovative SA instrumentation for NPP,
- Topic 4: Consolidation of SA knowledge base, methods and tools and related uncertainties to better support SA analyses (including PSA), SAMG validation and implementation and decision-making in emergency phases.

These high priority topics should also include aspects related to management of severe accident in SFP⁵ and of long term management⁶ of SA.

The overall chart of the proposed program for enhancing SAM in NPPs is provided in Figure 1. Each topic is further described in the following paragraphs.

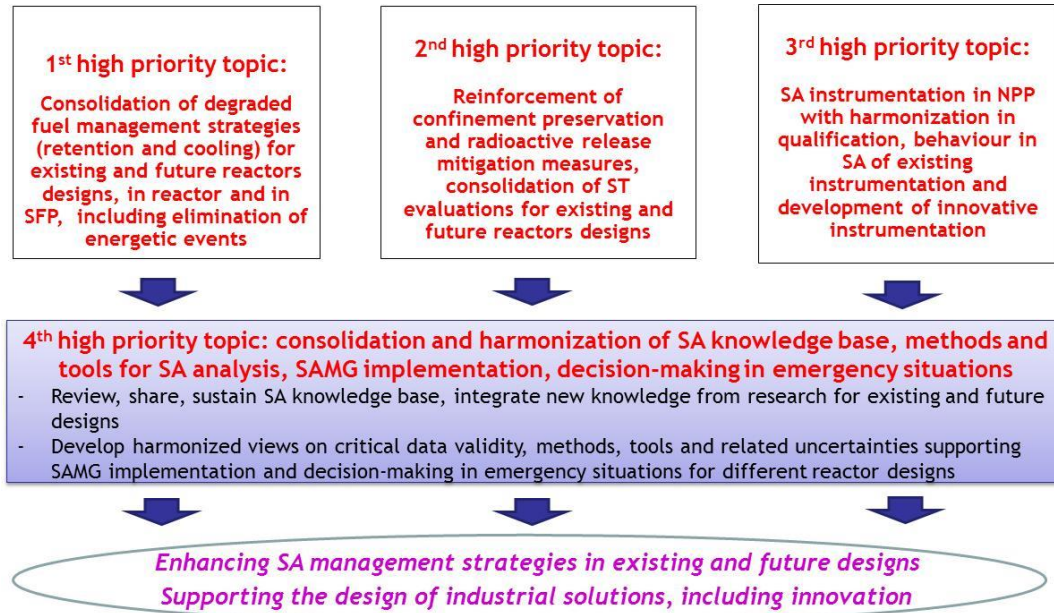


Figure 1 : Chart of first selection of topics for the NI2050 program for consolidation of SAM strategies in existing NPPs and to support the design of innovative industrial solutions for future designs

1st high priority topic: consolidation of degraded fuel management strategies (retention and cooling)

This topic is considered a high priority one since successful degraded fuel retention and cooling strategies would largely reduce the risk of radioactive releases to the environment in a SA. Designs with safe IVR or/and EVR strategies would prevent uncontrolled liquid or gas radioactive releases through a breached confinement. Depending on reactor design and type of fuel, significantly different strategies may apply.

⁵ The relative importance of SFP severe accidents is still being debated and strongly depends on SFP configuration (e.g., in or outside the reactor containment). However, considering all possible initiating events from safety as well as security perspectives, some SFP could possibly lead to large radiological consequences on-site and off-site, if the accident cannot be mitigated. One can refer to recent OECD/NEA reports on the subject: Status Report on Spent Fuel Pools under Loss-of-Coolant Accident Conditions, Final Report, NEA/CSNI/R(2015)2, 2015; Phenomena Identification and Ranking Table (PIRT) on Spent Fuel Pool under Loss-of-cooling and Loss-of-coolant Accident Conditions, NEA/CSNI/R(2017)18, 2017.

⁶ As shown by the past major accidents experiences, managing a damaged plant on the long term is challenging since (1) knowledge of systems reliability in maintaining critical safety function on the long term is limited (2) knowledge and analytical tools are not developed to address risks due to long term processes (e.g. corrosion and radiolysis reactions, long-term radiation-induced damages, fuel leaching and exposition to humidity, effect of radiobiological processes) (3) methods and tools for risks mapping and ranking, tracking evolution of the plan damaged state upon recovery, are not developed and validated. One can refer to the upcoming OECD/NEA report on the Long Term Management of a Severe Accident in a NPP to be released in 2018.

For generation II and III reactors, designs with safe IVMR (In-Vessel Melt Retention) strategies would be more advantageous as degraded fuel would be safely contained in the reactor pressure vessel (RPV) during a SA - thereby eliminating risks of containment failure by steam explosion during corium-water interaction after RPV failure and by molten corium concrete interaction (MCCI) in the pit. Reactors with safety relying on IVMR should be designed with sufficient margins to prevent RPV failure during all phases of a SA and possibly during a prolonged period of time.

With the current knowledge on in-vessel degradation, particularly on the late phases when corium and debris accumulate in the lower RPV plenum, the safety of IVMR can be effectively demonstrated for some low power reactors with properly designed external circulating coolant (e.g. VVER 440 reactors where IVMR was back-fitted) but safety margins related to IVMR strategies for higher power reactors (1000 MWe and more) using external circulating coolant should be further assessed to increase the confidence in IVMR strategies implementation [3].

In many existing generation II reactors (BWRs and PWRs), partial containment flooding is implemented as a SAM measure to flood at least partly the external side of the RPV wall and evacuate heat from it. However, the existing knowledge is not sufficient to demonstrate that this measure is effective in preserving in all cases the RPV integrity. It should be added that generation II reactors were not specifically designed to optimize heat evacuation by external cooling (water access to the RPV wall is sometimes hindered by structures and/or equipment surrounding the RPV) and, particularly in BWRs, RPV penetrations represent fragilities that can fail when attacked by corium. To our knowledge, back-fitting for IVMR is not envisaged at present in existing reactors with the notable exception of VVER-1000 reactors.

The consolidation of IVMR strategies for generation II and III reactors still faces challenges for which R&D is on-going or foreseen. Additional knowledge is required on [2,3]:

- Cooling of realistic complex in-vessel degraded core configurations after loss of fuel rods integrity and the associated H₂ production (e.g. significant scatter was seen in major SA codes results in the OECD/NEA/CSNI BSAF exercise),
- Stratified corium configurations and their transient evolutions in the RPV lower head as they drive the thermal loads on the RPV wall,
- RPV resilience - considering ablation by corium and fragilities in some designs (e.g., penetrations) - and efficiency of cooling countermeasures (from inside and outside RPV) on the long term,
- Possible design innovations for internal cooling, external cooling and in-vessel retention.

In addition, research should address how IVR strategies may be affected by the use of advanced cladding and fuel, including for accident tolerant fuel (ATF).

For future reactors, design maturity is variable, as e.g. between generation IV SFR, VHTR, GFR, SCWR, LFR and MSR, with technological choices still to be consolidated. For some designs, research may be necessary on specific phenomena that could challenge IVR: progression of in-core fuel degradation, fuel-coolant interactions and chemical interactions addressing clad, fuel and in-core structures specificities.

EVMR (Ex-vessel melt retention) to retain and cool corium and debris in the containment after RPV failure is currently applied as a SAM strategy in a large number of reactors, but with different approaches [4]. In generation II reactors, operators generally rely on water already present in the reactor pit prior to vessel failure to cool debris and corium in water. It must be shown in that case that the plant safety cannot be challenged by steam explosion. Back-fitting is envisaged in generation II operating French reactors to ensure corium spreading on the containment base-mat without water (in the reactor pit and in an adjacent room) and cooling by top water flooding after spreading at the

beginning of MCCI⁷. Such an approach, if shown to be safe, would largely reduce the risk of containment failure by steam explosion at RPV failure and by MCCI on the longer term. Some generation III and III+ reactors are designed with core-catchers where the approach is similar to that described earlier for French operating reactors, i.e. with corium spreading and cooling by top water flooding in the core catcher with, in addition, bottom cooling by water circulation implemented in core catchers.

The consolidation of EVMR strategies for generation II and generation III reactors still faces challenges for which R&D is on-going or foreseen. Additional knowledge is required on [2, 4-6]:

- Realistic initial corium/debris configurations at start of MCCI (transfers from failed RPV, fragmentation and debris formation in water if present, spreading or local accumulation of debris and corium),
- Steam explosion risk and potential consequences on equipment, structures and possibly the containment,
- Cooling at large scale in cavities representative of actual reactor designs, with metal-rich corium and on the long term,
- Possible design innovation in core-catchers designs, possible use of sacrificial “innovative” materials.

For future designs including generation IV designs, EVR should be addressed in designs where RPV failure cannot be excluded.

On that specific topic, the program should provide opportunities to (1) progress with experiments and models transposable to reactor scale (treating scaling effects) and better supporting SA analysis and validation of IVR and EVR strategies for all reactor designs (2) support, where possible, the development of back-fitting solutions for existing plants and eventually design innovations for implementation of IVR and EVR strategies in future designs.

As part of the consolidation of degraded fuel management strategies, knowledge on risks of energetic events and their mitigation should be increased. In a SA, different energetic events may result from re-criticality, combustible gas production (e.g. H₂ by metal oxidation in and ex-vessel and CO by concrete erosion during MCCI) and combustion, fuel coolant interactions (e.g. steam explosion when corium interacts with water), coolant water interactions (e.g. Na interactions with water) depending on fuel type and reactor designs. These energetic events could affect safety important components, equipment and systems and confinement of radioactivity and could contribute to remobilize radioactive materials.

A concerted international effort to strengthen and harmonize methods and approaches to appreciate risks related to energetic events in NPPs during a SA and efficiency of associated mitigation strategies would be beneficial for improving safety of existing designs and to identify design features that could limit them in future designs. This should in particular address how new fuels could affect these risks.

⁷ It is also envisaged to add sacrificial carbonaceous concrete material to promote gas production during MCCI and corium cooling by water (by corium eruption and water ingress mechanisms).

2nd high priority topic: reinforce confinement preservation, releases mitigation and consolidate source term (ST) evaluations

Relative to confinement preservation, the focus is placed on knowledge development on behavior of critical confinement component and material in SA and on robustness of active heat evacuation systems as design and safety of passive systems (e.g. pressure suppression pools, pressure relief tanks, ice condensers, etc.) are assumed to be addressed in a separate NI2050 proposal.

Knowledge on behaviour in SA of component and materials (e.g. coatings and paints, penetration seals, heat insulation materials, some metallic parts...) of critical importance for confinement – their degradation possibly resulting in confinement failure or in debris that can yield failure of heat evacuation systems - should be increased with two objectives (1) better assess confinement reliability in SA for existing reactors and improve management to preserve it (2) select or develop alternate more robust component and material, including innovative ones, for existing and future reactors.

Robustness of containment heat removal and pressure control systems is key for maintaining a “safe” containment during a SA. After the Fukushima Daiichi accident, back-fitting with ultimate containment heat sinks (fixed and/or mobile) and filtered/hardened containment vents (CV) was decided in many generation II operating plants to reinforce means to control containment pressure during a SA.

More recent designs (Generation III and III+) include more robust ultimate containment heat sinks and it is in general considered for such designs that CV does not provide a significant added value for SAM. Containment pressure control is then ensured by SA dedicated systems recirculating and cooling highly active fluid during prolonged period of times, for months or possibly for years.

The challenge is to design SA robust and efficient heat sinks systems and to demonstrate their robustness for prolonged period of times for SA conditions, assessing in particular clogging and failure risks. Extensive research to increase knowledge to appreciate clogging risks for DBA conditions has been performed [7]. Clogging issues in SA have much less been addressed. Also, experience feedback has shown that heat exchangers internals are prone to corrosion-erosion and dirt accumulation in aggressive environments which can result in leaks [8]. Knowledge on LT reliability of these sensitive components in SA should be increased. Innovative materials both to limit clogging (e.g., limiting debris and precipitates formation during accidents) and failure risks of sensitive components should be explored under the NI2050 program.

As long as confidence in safety of ultimate confinement heat sinks systems has not been fully established, there should still be some interest for the design of robust CV systems with efficient filtration.

Opportunities to share experience among countries on long term reliability of emergency core cooling systems to maintain the confinement during a SA should be fostered in view to define the necessary research in the field including innovative developments.

Relative to radioactive releases mitigation and ST evaluations, innovative and SA robust trapping technologies for radioisotopes which are major contributors to radiological consequences should be further developed as well as knowledge of radioactive releases to the environment during a SA. These investigations are critical for the reduction of radiological consequences during an accident and would support the design of optimized emergency protection measures.

The consolidation of ST evaluations during a SA still faces challenges for which R&D is on-going or foreseen. Additional knowledge is required on [2, 9-12]:

- Performance and reliability of components, systems and structures ensuring the confinement during a SA (assessment of leakages to the environment up to the long term),
- Contributions to releases of “remobilization” of radioactive materials once they have deposited in various parts of the damaged plant (sumps, pools, pipes and walls),
- Filtering and scrubbing (in BWR suppression pools, in liquid filters) of radioactive materials to assess retention and release, in particular when venting – filtered or not - is considered as a SAM measure,
- Chemistry in the environment and its impact on radiological consequences,
- Releases for other important categories of accident, e.g. steam generator tube rupture (SGTR) and SFP accidents,
- Possible innovation in trapping or mitigating systems (inside or outside containment, in SFP) for major contributors to the radiological consequences.

Extensive collaborative R&D is on-going on in-containment ST and filtration/scrubbing/trapping, however hardly addressing “remobilisation” phenomena in a damaged plant – which were shown to significantly contribute to radioactive releases to the environment at Fukushima Daiichi 1F - and chemistry in the environment, both possibly affecting significantly radiological consequences.

The research momentum should be maintained in the future when present major international programs will terminate in 2019 (e.g. NEA/CSNI THAI3, BIP3 and STEM2) only partially closing these issues.

In addition, research should address how source term may be affected by the use of advanced fuels, including accident tolerant fuel (ATF).

On that specific topic, the program should provide opportunities to (1) progress with experiments, models and, in relation to topic 4, methods to consolidate ST evaluations for major types of reactor designs and accidents (2) support back-fitting and innovations for radioactive releases mitigation.

3rd high priority topic: reinforce SA instrumentation in NPP

Large efforts have been undertaken in the past on NPP instrumentation for normal operation and accidents in the Emergency Operating Procedures (EOP) domain but less on SA instrumentation. In the SA field, the focus of present international attention appears to be on consolidation of existing techniques and systems that support implementation of mitigative SAM actions, focusing particularly at improved methods for better assessing their capability in SA.

While this is of importance and should be pursued, less effort seems to be undertaken internationally (1) to identify if major possible improvements that would efficiently support comprehensive management strategies for all phases of a SA in reactor and/or in SFP (including emergency planning and long term management of damaged plants) are possible (2) to identify available, under development or promising new technologies that could provide such major improvements (3) to conduct research that would help better assessing critical instruments (existing or contemplated) performance in SA and promote development of innovative approaches and technics, particularly for future reactor designs.

A concerted international collaborative program in the domain would certainly help to progress in this high interest safety research field with expected improvements in SAM. Actors in the field should also further address some level of harmonization for SA instrumentation qualification at the international level, for instance by the definition of dedicated SA qualification criteria adapted to the specific SA requirements for a given instrumentation.

4th high priority topic: consolidation of SA knowledge base, methods, tools and related uncertainties to support SA analyses (including PSA), SAMG implementation and validation and decision-making in emergency situations

Consolidation of the SA knowledge base, including for SFP-related issues, is considered as a high priority topic with SA knowledge collection, review, sharing and management (including sustainability) being critical transverse activities that should be internationally organized.

This should be done taking full advantage of information and data collected for:

- past SA, particularly the accident at the Fukushima Daiichi Nuclear Power Station with information and data covering all phases of the accident including the decommissioning of the damaged reactors. Even though the Fukushima Daiichi Nuclear Power Station has specific BWR designs, lessons learnt from the accident analysis and studies have brought and will continue to bring useful information and data to further improve existing and future reactor designs,
- past and on-going collaborative SA research programs in reactor and in SFP, including programs addressing future designs.

Further, some uncertainties and open issues remain in SA accident analyses in reactor and in SFP warranting further research, particularly regarding the modeling of accident progression and mitigation at reactor or pool scale with consideration of differences in designs. Reviewing and updating the knowledge base for SA accidents in reactor and in SFP is a fundamental basis for *focusing the research for consolidation of accident analyses and management.*

The task should also involve the development of *harmonized views – between research and safety organizations and industry (operators and designers) - on critical data and information validity* for accident analyses. These activities are important prerequisites for the consolidation and harmonization of accident analysis tools and methodologies addressed in the following paragraphs.

The other identified high priority topics aiming at proposing industrial solutions to enhance SAM strategies should also benefit from outcomes of this task.

Finally, over the course of time, the knowledge base should integrate information and data generated by research for new fuel concepts and new reactor designs (e.g. for generation IV reactors and SMR).

Consolidation of SA methods and tools and related uncertainties is considered of high priority with the objective to consolidate and harmonize between research organizations, safety authorities and TSO's and industry (designers and operators) methods to properly capitalize knowledge gained through research programs in tools supporting SA analyses (including SA accidents in SFP) and validation of SAMG and in tools supporting emergency preparedness. On this subject, multilateral cooperation and stronger cooperation between SAM, emergency management and environment management experts would certainly be beneficial as it was rather limited in the past.

On that specific topic, the program should provide opportunities to (1) develop harmonized approaches to make SA codes more readily usable for SAMG validation for different designs (treatment of realistic reactor configurations, of scaling issues and uncertainties in assessment of degraded fuel progression and radioactive releases and their mitigation during a SA) (2) develop further, with the support of SA codes, prognosis/diagnosis tools to support decision-making in emergency situations, this includes methods to integrate the proper level of knowledge in fast-running tools and methods for backwards ST evaluations based on data collected during the accident (3) see how to transfer knowledge in simulator tools for SAMG verification.

3. What is done/exist already, who is doing what, what are the means (resources and infrastructures)

1st high priority topic: consolidation of degraded fuel management strategies (retention and cooling)

Large international research efforts have already been conducted to investigate, mostly for UO₂ fuel and Zircaloy cladding, corium and debris behavior and their coolability during a SA. These investigations addressed, for existing NPPs and NPPs under deployment (generation II and III/III+), degraded fuel retention and cooling in the RPV and in the containment after RPV failure - with or without a core catcher [13,14].

Major outcomes of these research programs and remaining knowledge gaps in particular for cooling strategies are reflected in recent documents released by NEA/CSNI (MCCI State Of the Art Report (SOAR) [4] and Ex-Vessel Steam Explosion (EVSE) technical report [6]) and by a document to be released at the end of 2018 by EC (NUGENIA/SARNET CORESOAR report on corium and debris behavior and their coolability in-vessel).

Recent international benchmark exercises performed within NEA/CSNI on TMI-2 like scenarios [15] and on the Fukushima-Daiichi accident (BSAF project) [16] have also evidenced gaps in modeling corium and debris behavior and cooling in-vessel and ex-vessel for SA analyses both for PWR and BWR designs.

Main knowledge gaps for the consolidation of IVR and EVR strategies, as identified in these documents and benchmarks, are listed in chapter 2. They in particular point to the need to acquire additional experimental data for insufficiently studied prototypical corium compositions (e.g. metal-oxide compositions for existing reactors) and for more realistic configurations allowing extrapolation to the reactor scale for different reactor designs.

Industry has developed for industrialization of specific designs (generation III and generation III+ reactors) scaled demonstrative research facilities for critical corium management systems, e.g. external vessel cooling systems for IVR and core catchers for EVR [3,4]. Based on experimentation in these facilities, industrial developments have been made for generation III/generation III+ plants with designs implementing different IVR or EVR approaches. However, in part due to remaining knowledge uncertainties, consolidated and shared approaches to assess safety margins related to implementation of these strategies have to be developed. Back-fitting is also planned or envisaged in some existing generation II plants.

A large joint research program involving researchers, safety analysts, plant operators and industry developers, the EC H2020 IVMR project, is presently on-going to consolidate the knowledge base and methodologies for the safety demonstration of IVR strategies for different reactor designs [3]. The program should end in 2019 and expects to address some of the knowledge gaps for safety analyses of IVR approaches.

Additional knowledge will also be obtained through experiments performed in the EC SAFEST (European experimental corium platform) [17] and ALISA [18] (Chinese and European experimental platforms) programs and through NEA/CSNI programs performed to prepare the decommissioning of the damaged reactors at the Fukushima-Daiichi NPP and dealing with corium and debris studies such as PreADES and TCOFF. All these programs should end mid-2020 and produce valuable knowledge to focus future research in the field.

Project proposals addressing some of the issues identified through the OECD/NEA/CSNI MCCI SOAR are currently being discussed at the international level (OECD/NEA/CSNI and EC H2020 frameworks). Pending upon funding, these projects could start in 2019 with 4 to 5 years duration. Concerning new fuels (including ATF), large research efforts, mostly driven by the industry, have been and are still being conducted to establish their technological readiness as described in corresponding NI2050 templates. However, limited knowledge has been obtained related to their behavior under SA conditions and the applicability of knowledge gathered for largely studied fuel and cladding materials has not been established. Related to material aspects, existing databases and models of major chemical processes should be developed to address new fuels and claddings interactions and their interactions with structure materials during a SA. Related to degraded fuel management, retention and cooling systems reliability and efficiency during a SA have hardly been investigated.

A critical aspect to progress on this topic both for existing and future designs is the maintenance and development of versatile experimental research infrastructures allowing investigations of prototypical compositions and configurations. Existing experimental platforms require further technological developments (some identified and feasible) or new platforms should be developed to address remaining open issues.

Main existing corium research infrastructures with a long experience are based in Europe (facilities used in SAFEST program), in the US (ANL), in Korea (KAERI), in Russia (NITI). Some are under development in Japan (JAEA) within the CLADS and IRID initiatives in preparation of the decommissioning of the damaged reactors at the Fukushima-Daiichi plant and in China (facilities used in the ALISA program). In the future, by the early 2020s, it is planned to develop a new experimental platform at CEA in France, the PLINIUS2 platform, allowing investigations of prototypical compositions and configurations including for generation IV designs (and related corium composition).

Related to research prioritization in the area and identification of key experimental platforms, ranking exercises are being conducted, e.g. in the EC SAFEST project [17,37], in the ETSO Research Group [33], in the NUGENIA/SARNET Severe Accident Research Priority (SARP) Ranking exercise, in the updating of the OECD SESAR document. These exercises are mostly performed under the angle of completing the knowledge base on SA phenomena for generation II and III reactors.

2nd high priority topic: reinforce confinement preservation, releases mitigation and consolidate ST evaluations

Relative to confinement preservation reinforcement, existing knowledge and on-going research specifically addressing SA conditions should be reviewed on two aspects (1) behavior of components and materials critical for containment leak-tightness (sealing materials) and as sources of debris and sediments that will eventually accumulate in cooling waters (2) systems and strategies reliability and robustness for containment heat evacuation and pressure control. Due to the diversity of components, materials and systems, some developments that are proprietary, and the challenge to address SA extreme conditions (e.g., dealing with combined effects of elevated temperatures and doses, oxidant or reducing and corrosive environments), limited concerted international research efforts have been undertaken so far in the area.

On the first aspect, sealing materials have been investigated in containment integrity studies that have essentially been performed nationally (see for instance US efforts summarized in the SAREF report in Appendix A.2.2.1 [2]). Research and knowledge on behaviour of materials that are source of debris and sediments has been reviewed for DBA conditions [7]. It should be highlighted that challenging SA conditions have not been thoroughly addressed up to now.

On the second aspect, operational feedback, knowledge and research on heat evacuation systems robustness and reliability have recently been reviewed by OECD/NEA working groups [7,8]. Again, knowledge is limited for challenging SA conditions, particularly on dirt accumulation, clogging, corrosion-erosion, including on the long term, in sensitive components of the systems.

Many different passive or active safety systems (e.g. pressure suppression pools, relief tanks, condensers, internal and/or external sprays, cooling recirculating loops with heat exchangers, venting lines, ...) and SAMGs aimed at limiting confinement pressure increase during a SA have been developed mostly by industry for existing reactors of different designs, and some innovative ones are being developed for future reactor designs.

Reviewing, sharing and capitalizing knowledge on these systems and related SAMGs, their efficacy and robustness in SA, considering past-accident feedback (including Fukushima-Daiichi), is a prerequisite to design future research and development of innovative systems. Note that necessary future research for passive systems is addressed in a dedicated NI2050 template.

Relative to releases mitigation and ST evaluations, large international research efforts have already been conducted [9,10,13,14,20] (e.g. Phebus FP and International Source Term (CHIP, EPICUR, VERDON) programs conducted in France, programs conducted in NUGENIA/SARNET, EC PASSAM, NEA/CSNI BIP, STEM and THAI programs) and are still on-going (phase 2 of STEM, phase 3 of BIP and THAI, NUGENIA/SARNET IPRESCA) to consolidate the knowledge base for ST evaluations and for characterizing and supporting the development of filtered containment venting systems (FCVS). Major outcomes of these research programs and remaining knowledge gaps are reflected in recent documents released by NEA/CSNI (International Iodine Workshop Summary Report [9], Status Report on FCVS [10], final reports of BIP2 [9], STEM [21], THAI2 projects [20]) and by EC (Synthesis of knowledge gained in the ST area [11], PASSAM Final Report [12]).

The on-going international benchmark exercise performed within NEA/CSNI on the Fukushima-Daiichi accident (BSAF phase 2) have also evidenced gaps in modeling fission products releases into the environment particularly related to “remobilization” processes once they have deposited in surfaces and pools in the damaged reactors.

Main knowledge gaps for the consolidation of ST evaluations are listed in chapter 2. They in particular point to the need to acquire additional knowledge (experimental data and modeling) on 4 main aspects: FP release from fuel in view of developing predictive models for any type of fuel (including new fuel types) and accident transients (including SGTR and SFP in existing designs), remobilization of FP once deposited in a damaged plant in view of developing models able to assess releases at short and mid-term, filtering/scrubbing/trapping of FP in view of assessing existing release mitigation systems efficiency and developing new innovative systems and chemistry of FP in the environment to assess its impact on radiological consequences and support the design of optimized protection measures.

Industry on its side has developed for industrialization FCVS [10]. Following the Fukushima Daiichi accident, it has been decided to install FCVS systems in a large number of generation II NPPs worldwide not yet equipped with these systems or to improve already implemented ones (e.g. for robustness and filtration efficiency). However, reliability and efficiency of these FCVS are still being internationally debated due to uncertainties on actual operating conditions in a SA and lack of knowledge on dose effects on retention on the LT. Generation III/III+ reactors are generally designed with SA robust Emergency Containment Cooling System (ECCS) but there too the reliability of such systems for SA conditions may be questioned as earlier discussed.

Large joint research programs involving research organizations, safety authorities, TSOs, plant operators and industry developers, are presently on-going to consolidate the knowledge base for ST evaluations (BIP3, STEM2, THAI3, IPRESCA). These programs should end by 2019 but only partially filling the identified knowledge gaps for ST evaluations. It is important to keep the momentum after that to preserve competences and experimental infrastructures to tackle remaining major knowledge gaps but also to support the development of innovative solutions for FP releases reduction.

A ST workshop will be organized early 2019 under the OECD/CSNI/WGAMA auspices to support the identification of future research priorities and necessary resources and infrastructures in the area.

3rd high priority topic: reinforce SA instrumentation in NPP

Many initiatives on SA instrumentation were launched following the accidents at Fukushima Daiichi NPPs [2].

In the US, DOE, BWROG, PWROG and USNRC collaborated on the instrumentation survivability evaluation (based on SOARCA for pilot plants and MELCOR simulations) that was concluded in 2015 [22]. This evaluation covered the following main aspects:

- identification of critical parameters for SA management and mitigation,
- required instrumentation to provide these parameters,
- characterization of environment for instruments in SA,
- identification of gaps in instrumentation capability.

In Japan, the “SA instrumentation and monitoring systems to prevent accident escalation program (SA-Keisou Japanese program)” was undertaken involving utilities, vendors and manufacturers. An evaluation was conducted covering the following aspects [23]:

- identification of critical parameters for SA management and mitigation,
- required instrumentation for “safe state” monitoring over the long term,
- identification of R&D actions to improve SA instrumentation.

Recommendations from this program were issued in 2015 which resulted in launching a SA instrumentation program.

Initiatives were also conducted by international organizations. IAEA issued guidance on “Post-Accident and SA Monitoring Systems” in 2015 [24] and more recently in 2017 a TECDOC on “Assessment of Equipment Capability to Perform Reliably under Severe Accident Conditions” [25]. These covered aspects such as equipment capability assessment under severe accidents, equipment use in the scope of SA mitigation and monitoring, estimation of environmental parameters for instruments, demonstration of reliable performance for severe accident (qualification).

NEA/CSNI issued in 2016 the SAREF report [2] where SA instrumentation is quoted as a high safety research interest. The document underlined that performance of critical instruments for SAM is not well understood in part because qualification testing does not include assessment of margins to failure and also because environmental conditions of instruments during SA are difficult to capture. Feedback from Fukushima-Daiichi accidents provided, up to now, limited information on real causes for instruments failure.

The issue of SA instrumentation is considered as high-priority by the Technical Area 2 of NUGENIA/SARNET in the R&D roadmap “NUGENIA Global Vision” published in April 2015.

Specific research programs on instrumentation for severe accidents conditions have been launched in many countries, with:

- in Japan, development of SA instrumentation (RPV and PCV water levels, pressures, temperatures, H₂ concentration, neutron detection for reactor state diagnosis),
- in France, ANR DECA-PF (fission products measurements) and DISCOMS (corium monitoring) led by CEA, MITHYGENE (H₂ measurement) led by IRSN,
- in UK and Germany, research on thermocouples and ex-core neutron analysis,
- in Japan (NRA) and France (IRSN), survivability of aged cables,
- in US (EPRI TG) and Japan, development of guidance for development of SA instrumentation and monitoring systems.

There are important differences in the methods of instrumentation qualification in SA conditions between different countries (source terms, transfer of FP, qualification process, qualification checks). Harmonization between different countries would be beneficial for SA instrumentation improvement.

It should be emphasized that the program on this topic should benefit from technological developments both from the nuclear and the non-nuclear fields.

4th high priority topic: consolidation of SA knowledge base, methods, tools and related uncertainties to support SA analyses, SAMG implementation and validation and decision-making in emergency situations

Following the TMI-2 and Chernobyl accidents, a large number of SA research projects have been completed, including in particular NEA/CSNI and EC (in last years within NUGENIA/SARNET) international joint undertakings, in order to accumulate information and knowledge of severe accident phenomenology and characterize severe accident scenarios [13,14].

The Fukushima Daiichi accident triggered intensive discussions on existing knowledge gaps on severe accidents as well as priority areas for narrowing safety research gaps taking advantage of data and information collected from the Fukushima Daiichi accident analysis and long term management processes. Additional data and information is expected from the decommissioning of the damaged plants. The NEA/CSNI's Senior Expert Group on Safety Research Opportunities Post-Fukushima (SAREF) issued a report that identified a number of priority research topics to improve nuclear safety and proposed to initiate near-term projects as a preparatory phase for the decommissioning of the damaged plants [2]. These projects as well as other on-going projects within NEA and EC (NUGENIA/SARNET) should contribute filling some of the identified knowledge gaps.

Even though the existing SA knowledge base is quite extensive and still expanding, researchers, safety analysts, plant operators and designers are currently facing two difficulties (1) despite initiatives to develop SA databases [26,27], there is no agreed, shared and managed validated core knowledge base supporting SA analysis and SAMG validation, (2) the knowledge base is sometimes not sufficient to demonstrate the safety of some SAM strategies often due to the necessity to further address scaling and uncertainties with consideration of different designs (existing or future).

The first difficulty could be overcome by joint efforts within the nuclear community to produce a common core SA knowledge base supporting SA analysis and SAMG validation.

The second difficulty could be tackled by additional joint research efforts on major mitigation strategies as described for the 1st and 2nd high priority topics and efforts to harmonize tools and methods supporting SAMG implementation as discussed below.

Important SA knowledge integration tools are SA codes such as e.g. ASTEC developed in Europe (latest version within the EC CESAM project), MAAP, MELCOR in the US and SAMPSON in Japan. These codes have been developed for SA analyses in existing PWR, PHWR and BWR designs and can be used for

more recent designs (Generation III/III+) with developments for specific safety systems (e.g. core catchers in generation III plants). They are used by researchers, safety analysts and industry.

It should be noted that specific versions of some SA codes (e.g. ASTEC and MELCOR) have been developed for future reactors (generation IV and SMR) – ASTEC for generation IV within the recently concluded EC JASMIN project. These developments should be completed in the future.

Recent international benchmarks on large scale tests (e.g. on Phebus FPT3 within NUGENIA/SARNET [28]) and accidents (e.g. on “TMI-2 like” scenario [15] and on Fukushima-Daiichi accident [16] (BSAF project phase 1) within NEA/CSNI) have been completed. They have evidenced codes limitations particularly in the treatment of advanced core degradation in-vessel (once fuel rods lose their integrity). Following BSAF phase 1, more detailed benchmark exercises, called “crosswalk”, are pursued between major SA codes in different frames [29,30]. These “crosswalk” exercises have helped identifying major differences in modeling approaches focusing on in-vessel degradation. These differences have major effects on the prediction of in-vessel degradation progression and the related hydrogen production.

On SA codes, the program should foster joint actions involving research organizations, safety authorities, TSOs and industry to develop approaches to make SA codes more readily usable for SAMG validation for different designs – including for future designs - and develop more harmonized approaches and methods for SA analysis (modeling, treatment of uncertainties and scaling issues, including for degraded fuel retention and cooling and ST). Promoting concerted approaches in modeling SA phenomenology is critical in reaching some level of harmonization in SA analysis.

Fundamental knowledge for SA analysis is also integrated in nuclear material databases (e.g. TAF-ID, NUCLEA, MEPHISTA, etc) supporting SA analyses. Joint initiatives for harmonized development and validation of such tools should be continued in the future following the concluded NEA/NSC TAF-ID and on-going TCOFF projects.

SA knowledge should also be integrated in fast-running tools that provide information supporting decision-making in emergency situations. Following the recent NEA/CSNI FASTRUN fast-running tools benchmark exercise (completed in 2015 [31]) that evidenced limitations in these codes with large differences in ST and radiological consequences predictions for selected NPP accidents, a project was launched under the EC H2020 frame, the FASTNET project (2015-2019) [32], to work on methods for the development of tools for prognosis and diagnosis during an emergency situation for selected accident scenarios for various reactor designs. This includes methods to integrate the proper level of SA knowledge in fast running tools and methods for ST and radiological consequences evaluations based on data – which may be scarce – collected during the accident on-site and off-site.

Joint initiatives should be pursued – involving nuclear safety, emergency management and environment communities - for the development of integrated approaches and tools able to provide diagnosis/prognosis for emergency management on-site and off-site, extended to management of environmental consequences.

4. What can be done to improve/accelerate through cooperation, plan of actions

Research for systems and strategies for mitigating degraded fuel progression and radioactive releases in SA for existing and future reactor designs can only be done effectively involving all major stakeholders (industry (utilities and manufacturers), research organizations, safety authorities and TSOs) and through international cooperation using as far as feasible already developed knowledge,

skills and existing/foreseen infrastructures since it is costly and requires complex technological developments for breakthrough in terms of systems and strategies development (treatment of scaling issues and actual reactor, SFP and mitigation systems configurations).

Outlines of actions plans are provided in the following sub-sections for each identified high priority topic. They consider what has been done and is currently being done in each area (as described in previous chapters), address identified main remaining issues for existing reactor designs and innovations for future designs. They also identify prerequisite actions that should be conducted in the short term with the support of international organizations (OECD/NEA, EC and IAEA) to establish more detailed future actions plans, prioritizing further research needs, implementing as far possible needs for the reactor fleet profile foreseen for 2050 and beyond and defining necessary means (resources and infrastructures) in each area.

As earlier stated, proposed actions plans are presently built focusing primarily on LWR technology but they should be adjusted in the future in close partnership with all stakeholders to further address by 2025 future designs. This should be supported by conclusions from prerequisite short term actions.

An organizational scheme should be implemented to periodically revise the program scope and related actions plans considering progress in technology readiness levels of new reactors and new fuels.

The effective deployment of resources and infrastructures to conduct the actions plans will require significant funding from major stakeholders. As establishing funding for SA investigations is particularly challenging, NI2050 could undertake building the bases with decision-makers for facilitating future concrete involvement of major stakeholders.

Finally, sharing between complimentary research projects founded under different auspices (e.g. OECD/NEA, EC and IAEA) should be sought.

1st high priority topic: consolidation of degraded fuel management systems and strategies

One critical aspect in the area is the remaining need to perform demonstrative experiments with sufficiently large masses of representative fuel, clads and structural material mixes. Only a limited number of existing infrastructures currently offer such possibilities as discussed in chapter 3. Such facilities should be maintained and developed with international support through cooperative experimental programs, pending development of new facilities in other countries (e.g., in France with the CEA PLINIUS2 platform project) by 2020-2025.

A second critical aspect is the necessity to perform experiments for innovative technological developments at sufficiently large scale for industrial demonstration. Since different systems and strategies can be envisaged, versatile experimental platforms will have to be implemented. Research will then require a strong involvement and support of industry.

A third critical aspect is the need to address new fuels, cladding and structural materials, their interactions as well as their interactions with fuel coolant in a SA. While important research has been conducted and is still being conducted to establish technological readiness of these new materials (refer to the corresponding NI2050 templates) for existing and future reactors, potential issues for SAM have not been systematically addressed.

A fourth critical aspect is the importance of gathering and implementing knowledge on actual fuel debris from the Fukushima-Daiichi accident.

A first tentative overall actions plan to address these critical aspects is provided in figure 2.

A first major action that should be undertaken at the international level and involve main stakeholders is the preservation, adaptation or development of versatile experimental platforms for investigations related to the design of systems and strategies for degraded fuel management in SA. This action should be conducted in the short-term for complementary investigations for generation II and III reactors and in the mid-term (by 2025-2030) for investigations for the future generation of reactor.

As a prerequisite to the development of future platforms, an action should be conducted in the very short term to identify and prioritize research issues and define necessary resources and infrastructures in the area. It should in particular consider (1) inputs of concluded relevant international research projects, conclusions of recent SOARs and conclusions of on-going research priority ranking exercises in the area (2) knowledge of mitigation systems and strategies and their rationale both for existing and future designs (3) knowledge of existing research infrastructures and their capacities. Such an action could be conducted under the auspices of an international organization, possibly within existing working groups (e.g. the CSNI/WGAMA of the OECD/NEA). It could be initiated through a dedicated workshop followed by review meetings. It should propose a well-prioritized research approach, necessary technological and infrastructures development to progress significantly in mitigation systems and strategies. It should be highlighted that current initiatives are already partly addressing this pre-requisite action, e.g., the EC SAFEST roadmap [37], the NUGENIA/SARNET SA Research Prioritization (SARP) exercise, the ETSON Research Group strategic orientations, the OECD/SESAR document revision. *These initiatives are however conducted more under the angle of SA phenomena investigations than that of development of mitigation systems and strategies and are scarcely addressing future designs and innovation.*

In parallel to the action related to research infrastructures development, experimental research targeted at consolidation of IVR and EVR strategies for generation II and III reactors up to 2025 (e.g. with the ROSAU project, if funded) and, after 2025, at the development and demonstration of systems and strategies for future designs would be conducted.

Another important action that should be undertaken at the international level and involve major stakeholders is to develop, where necessary, and capitalize knowledge on new fuels behavior in SA, including ATF. As a prerequisite, existing knowledge of fundamental physico-chemical data and chemical reactions driving degradation of new fuels, cladding and structural materials, their interactions with fuel coolant in a SA should be reviewed to identify critical aspects that may affect significantly SAMG. Main related gaps and needed experimentation or theoretical development to fill them should be identified and prioritized. The focus should initially be on fuels that are technologically ready and that are planned to be used in the near future. Such an action could be organized under the auspices of an international organization within existing working groups (e.g. groups of the CSNI and NSC of the OECD/NEA dealing with new fuels behaviour, nuclear material databases and SA). The action could consist in a knowledge review and a PIRT exercise.

Another major action concerns knowledge acquisition from the Fukushima-Daiichi accident. Short-term projects have already been launched under the CSNI and NSC auspices (e.g. PreADES and TCOFF) and, based on their results, long term research projects shall be launched by 2021, particularly to obtain data on real degraded fuel samples. These projects should a priori be conducted under existing committees of the OECD/NEA (CSNI and NSC).

As part of topic 4, actions to integrate the data and information generated through these actions in the SA knowledge database and to consolidate harmonized SAMG evaluation methods should be conducted.

Further prototypical experimental investigations and well-focused Fukushima Daiichi forensic investigations should be supported and conducted jointly - industry (operators and designers) being, in addition to research organizations, safety authorities and TSOs, major actors - to build a common knowledge base and common methodologies to harmonize assessment of IVR and EVR systems and strategies for different reactor designs. This would certainly accelerate industrial development, licensing and implementation of back-fitted solutions in existing reactors and innovative solutions in future ones.

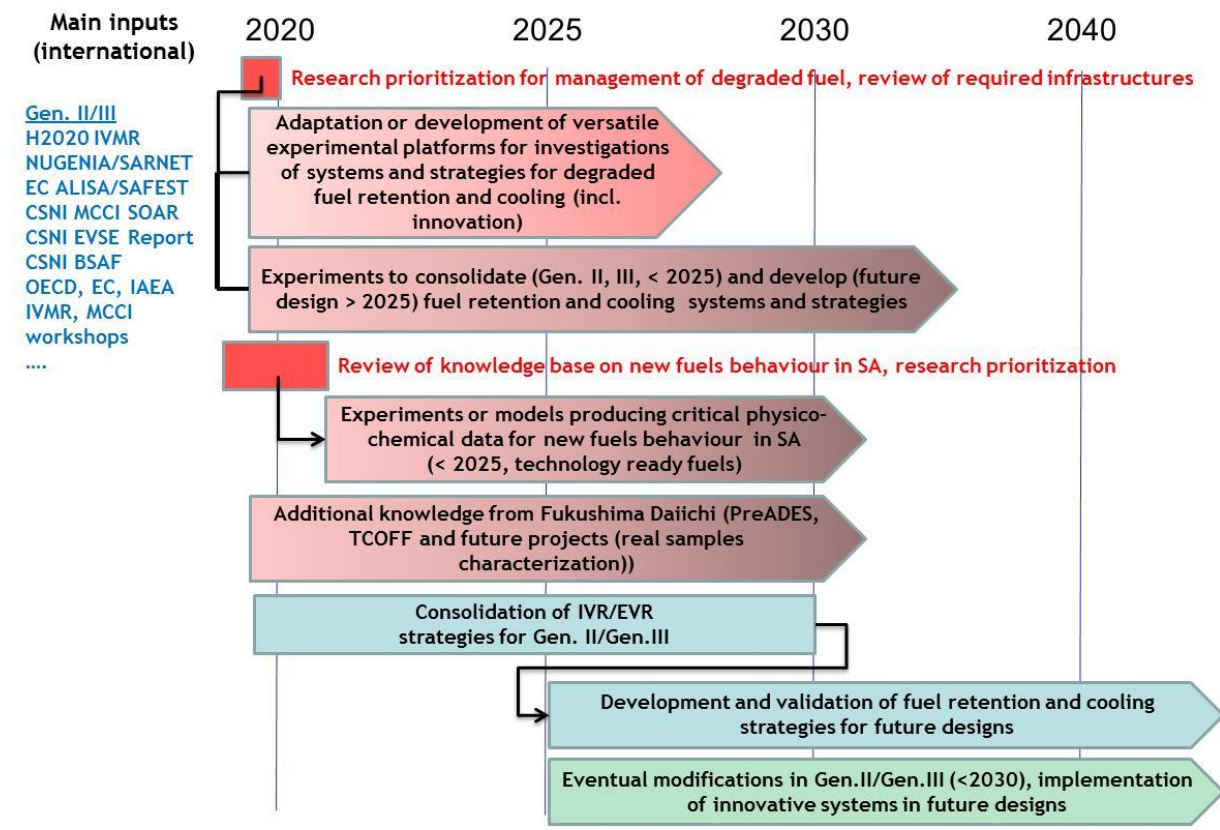


Figure 2 : Overall plan of actions to address future research related to degraded fuel retention and cooling systems and strategies in a SA. Pre-requisite actions to be launched at short term are indicated in red (see text for more details).

2nd high priority topic: reinforce confinement reliability, releases mitigation and consolidate ST evaluations

The first critical issue that research related to radioactive release mitigation in a SA should address is the reinforcement of confinement reliability.

This should go through increasing knowledge on behavior in SA (combined effects of prototypical temperatures, irradiations and chemical conditions) of critical components and materials contributing to the confinement and which can negatively affect, by their degradation, both the confinement leak-tightness and the functioning of containment residual heat evacuation systems - including on the long term - and developing and qualifying more robust ones. This concerns in particular sealing and heat insulation materials, coatings and paints and some metallic parts subject to extensive corrosion in SA.

This should also go through increasing knowledge on reliability and robustness under SA conditions of active and passive containment residual heat evacuation and pressure control systems and developing

and qualifying more robust ones. Issues that should be addressed are potential clogging of these systems and failures of sensitive components such as heat exchangers by corrosion –erosion reactions (other issues may be identified in a more detailed review process).

A tentative overall actions plan to address both aspects is provided in Figure 3.

As a prerequisite, an action should be conducted on the short term to review the knowledge base, on-going research and existing infrastructures and prioritize necessary future research in the area. This should be done considering extensive research work already conducted for DBA conditions (see e.g. the OECD/NEA ECCS reliability SOAR [7]), feedback from past events (e.g. OECD/NEA ICDE project [8]) and past accidents (including Fukushima-Daiichi) analyses. Such an action could be organized under the auspices of an international organization within existing working groups (e.g. groups of the CSNI of the OECD/NEA dealing with ECCS reliability, involved in ICDE projects and in SA). The action could consist in a specialized workshop to identify necessary resources and infrastructures to progress in the area.

Following this prerequisite action, actions that should be undertaken at the international level and involve main stakeholders are the adaptation or development of versatile experimental platforms for investigations related to (1) critical components and materials behavior in SA (2) containment heat evacuation systems reliability in SA and the performance of related experiments.

These actions should be conducted in the mid-term (< 2025) for investigations for components, materials and systems of generation II and III reactors and in the longer term (by 2025-2030) for investigations for future generations of reactors including innovative developments.

In the ST research field, it is critical to further share and review recently obtained knowledge (through international ST research programs and Fukushima-Daiichi NPPs analyses) since most on-going major international research programs in the area will end by 2019. This should be done through the OECD/NEA CSNI ST workshop scheduled early 2019 with the objectives to review the knowledge base, identify major remaining knowledge gaps and prioritize future research needs in the area. Established infrastructures and competences (in particular through the NEA THAI3, BIP3, STEM2 and EC PASSAM and NUGENIA/SARNET IPRESA projects) should be used to address identified remaining gaps. This research should be done in close connection with consolidation of calculation tools and methods for best-estimate ST calculations and related uncertainties and would largely benefit from a close cooperation between the various involved communities using ST research data (SA, environment, emergency management).

As evidenced by Fukushima Daiichi accident analyses, it is required to generate additional data to model FP “remobilization” phenomena in a damaged plant through experimental research. These data are necessary to develop capacities to assess releases in the days following fuel damage that can significantly contribute to radiological consequences. This was scarcely addressed in past research programs.

Innovative research to reinforce FP release mitigation, including for specific types of accidents that could result in unacceptable consequences if unmitigated (e.g. SA in some SFPs, some SGTRs), should be further developed. This could go through fundamental research investigating further specific trapping media that would be robust and efficient under SA conditions and industrial developments of collecting and trapping systems. This could involve experiments at sufficiently large scale for industrial demonstration. This research would benefit from a strong support from the industry.

Also, integration of the data and information from the Fukushima-Daiichi accident should be pursued.

A tentative overall actions plan to address future ST research is provided in Figure 3.

Conclusions of the 2019 CSNI ST workshop should be used to further identify research priorities and necessary resources and infrastructures and develop a more detailed action plan in the domain.

An action should consist in investigations related to improving knowledge to consolidate ST evaluations for all reactor designs. As part of this action, remobilization phenomena would be investigated for generation II and III reactors till 2025 to establish their potential contribution to radiological consequences. The action should also address further ST for future designs and new fuels, addressing their specificities. Here too, the focus should initially be on fuels that are technologically ready and that are planned to be used in the near future.

Another important research action should investigate the development of innovative FP collecting and trapping systems using recent knowledge gathered on specific chemical trapping media for isotopes that can be major contributors to radiological consequences in a SA. For core melt accident, priority would be given to development of in-containment systems to avoid contamination transfer outside containment.

A last identified action concerns knowledge acquisition from the Fukushima-Daiichi accident. Short-term projects have been or will soon be launched under the OECD/NEA CSNI auspices (e.g. ARC-F, TCOFF) and, based on their results, long term research projects shall be launched by 2021-2022 particularly to progress in FP behavior analyses. As for fuel-related projects, these projects should a priori be conducted under existing OECD/NEA committees (CSNI and NSC).

As part of topic 4, actions to integrate the data and information generated through these actions in the SA knowledge database and to consolidate harmonized SAMG evaluation methods should be conducted.

Further prototypical experimental investigations and well-focused Fukushima Daiichi forensic investigations should be supported and conducted jointly - industry (operators and designers) being, in addition to research organizations, safety authorities and TSOs, major actors - to build a common knowledge base and common methodologies to harmonize assessment of containment preservation and release mitigation systems and strategies for different reactor designs. This would certainly accelerate industrial development, licensing and implementation of back-fitted solutions in existing reactors and innovative solutions in future ones.

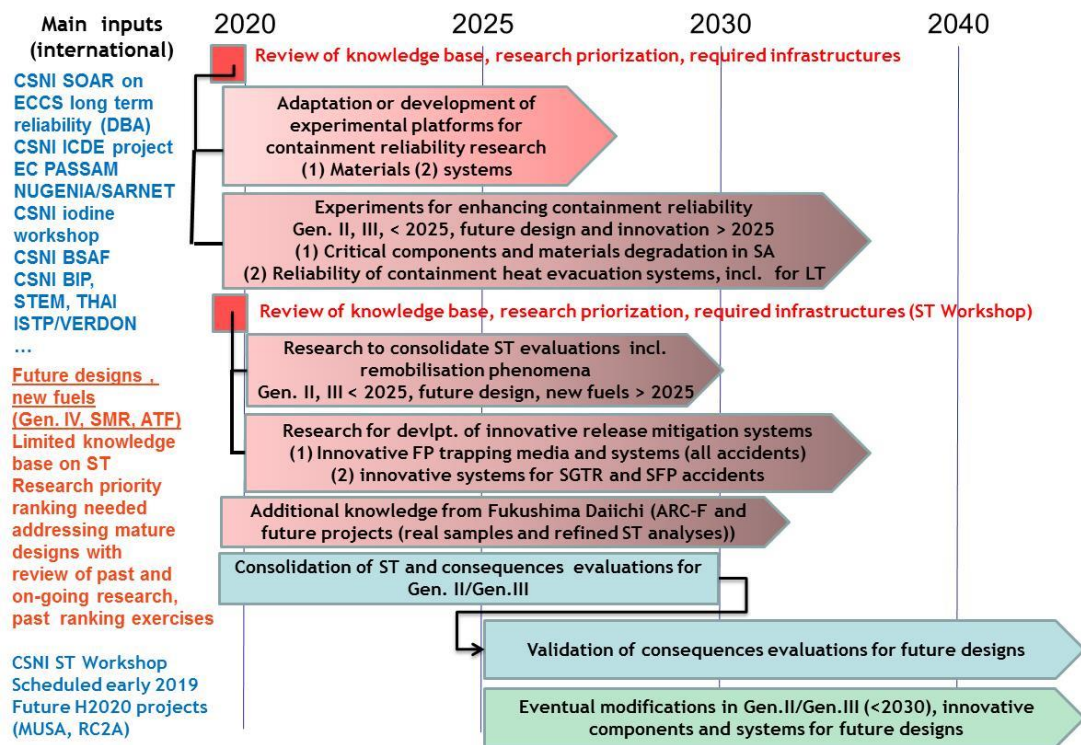


Figure 3 : Overall plan of actions to address future research related to containment reliability, radioactive release mitigation and ST evaluations in a SA. Pre-requisite actions to be launched at short term are indicated in red (see text for more details).

3rd high priority topic: reinforce SA instrumentation in NPP

The following pre-requisite actions would help defining a well-targeted research program to better assess critical NPP instruments performance in SA and promote development of innovative approaches and technics:

- Review of critical information for SAM, emergency planning, long term management (up to decommissioning after an accident) with required criteria (range, accuracy, response and mission time) - based on approaches developed or under development for NPPs worldwide - that would efficiently support comprehensive management strategies (both for SA in reactor and in SFP). With consideration of existing instruments, identification of major potential improvements (additional supportive critical information and relevant measuring technics) for future designs and eventually for existing reactors,
- Review of already implemented instruments (including DBA instruments), their expected performance in SA (level of qualification, reading validity, margin to failure) and their possible use in SAM, emergency planning and long term management approaches for NPPs. Identify additional needed research, if any, to better assess their performance,
- Identify available technologies and/or technologies under development and/or innovative technologies that would provide these major improvements, considering possible implementations, protections and expected environmental conditions and covering all accident phases,
- Assess Technology Readiness Levels for considered available technologies with respect to SAM procedures distinguishing their implementation in existing NPPs and in advanced ongoing nuclear systems.

These pre-requisite actions should be conducted involving industry (including from the non-nuclear field for innovative developments), research organizations, safety authorities and TSOs. They could also guide the development of more harmonized instrumentation qualification requirements bearing in mind that dedicated SA qualification criteria should not be too generic and be adapted to the specific SA requirements for a given instrumentation.

Knowledge from Fukushima-Daiichi accident related to SA instrumentation behavior should also be reviewed as well as outcomes of post-Fukushima international initiatives (IAEA guidance) and US, Japan, Europe research initiatives.

A workshop will be organized in 2019 under the CSNI/WGAMA auspices and the SILENCE network (SWINTH-2019) mostly addressing technological development for SA instrumentation including for SA experiments. Conclusions of this WS could guide the development of a research actions plan in the domain. However, this action should be supplemented by another action addressing more specifically instrumentation for SAMG and emergency management and reviewing knowledge from Fukushima-Daiichi accident related to SA instrumentation.

Another important action that should be undertaken at the international level and involve major stakeholders is the development of versatile experimental platforms for investigations related to instrumentation behaviour under prototypical SA conditions and for innovative developments. Research would be targeted at development of SA instrumentation plans for future designs, eventually implementing some improvements in existing ones.

A tentative overall actions plan to address the SA instrumentation in NPP topic is provided in Figure 4.

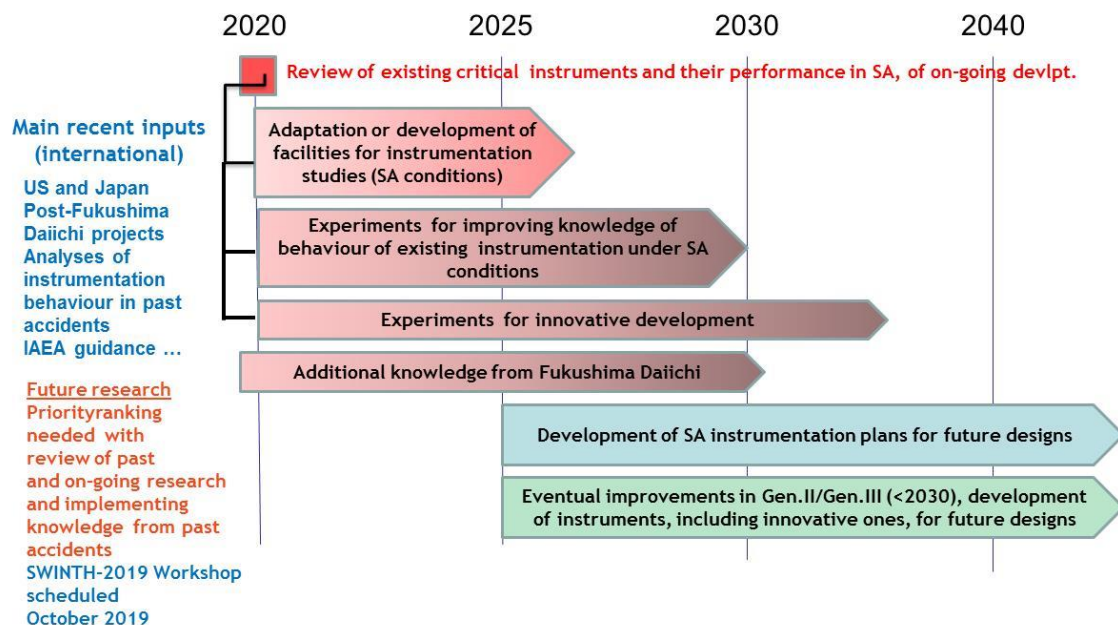


Figure 4 : Overall plan of actions to address future research related to SA instrumentation. Pre-requisite actions to be launched at short term are indicated in red (see text for more details).

4th high priority topic: consolidation of SA knowledge base, methods, tools and related uncertainties to support SA analyses, SAMG implementation and validation and decision-making in emergency situations

For this topic, two major “continuous” actions are described.

One first important action that should be undertaken at the international level is to organize for management (reviewing, sharing, ensuring sustainability) of the SA knowledge base, integrating new knowledge from research for existing and future designs. It could be conducted at the international level under the auspices of existing working groups (e.g. the CSNI/WGAMA). A prerequisite would be to define expectations from the management tool: objectives, structure, critical data selection and presentation. Convergence between international organizations, in particular EC and OECD, should be searched for such a development. This action could be launched in the near term and should in the future ensure continuous integration of relevant new knowledge.

Sharing, reviewing and integrating SA knowledge from past major accidents including Fukushima Daiichi accident, from projects related to its analyses and decommissioning and from recently concluded and on-going major research programs related to corium/debris progression and cooling in-vessel and ex-vessel, ST and mitigation, SA instrumentation, SA tools and methods should be done as much as possible in cooperation. This fundamental part of the NI2050 SA program would benefit from exchanges and iterations between the scientific, the industry and the safety communities with consolidation of a commonly shared knowledge base. This would also serve to identify remaining important knowledge gaps for safety and to initiate or redirect cooperative research projects to fill these gaps.

Relative to Fukushima-Daiichi accident, the NEA/CSNI Senior Expert Group on Safety Research Opportunities Post-Fukushima (SAREF) on-going process with on-going research projects (PreADES, ARC-F, TCOFF) should foster sharing, reviewing and integrating Fukushima Daiichi NPPs knowledge. Conclusions from on-going projects expected by 2020-2021 will serve to define data and samples collection to improve knowledge both for decommissioning and safety. Future projects will be initiated, in 3 to 5 years, for actual samples collection taking into account realistic feasibilities in term of costs and dose exposure. It is important to pursue and develop this initiative in the future as part of SA knowledge base consolidation.

Besides SAREF initiative, SA research is reactor and in SFP is presently on-going at the international level mostly through NEA/CSNI projects or actions and through the EC H2020 and NUGENIA/SARNET initiatives. It is important to pursue and develop some networking, ensuring as much as feasible open exchanges and complementarity between projects funded under different auspices. An important practical aspect that should be organized is data compilation, sustainability and sharing (possibly through databases development). This should also be made with consideration of works performed to assess the validity of SA knowledge base (data, models) for SAMG validation and of technical bases for emergency response.

Such an action would be an important prerequisite for harmonization in severe accident analyses thus contributing to accelerate licensing of industrial developments for SA management in existing and future designs.

A second important action is to develop more harmonized methods and tools supporting SAMG implementation and decision-making in emergency situations for different reactor designs.

A substantial effort is necessary to develop methods to address scaling issues and uncertainties in SA analysis including for ST evaluations. A project proposal is under elaboration for the next H2020 call

(MUSA project) which should establish between 2019 and 2023 the bases for tools and methods for uncertainties analyses related to SAMG assessment. These methods and tools would be generalized to all SAMGs by 2025 for existing designs and could be extended to future designs after that.

Concerning SA codes, pursuing actions launched after the BSAF-1 benchmark (called “crosswalks” exercises) related to the consolidation of key physico-chemical models in major SA codes (ASTEC, MAAP, MELCOR, SAMPSON) and of generic methods for best-estimate calculations at reactor scale for different reactor designs should be fostered. Cooperation between main SA code developers and users would be most effective to progress and achieve effective results in SA codes models improvement.

International benchmark exercises for SA codes for practical applications remain of special importance for SA codes validation. Such exercises should be done periodically.

Specific necessary developments in SA codes for new reactor designs (e.g. generation III, generation IV, SMR), new systems and components should be as much as possible shared between interested partners for effective implementation in codes.

Prognosis/diagnosis tools (including fast running tools and backward ST evaluations) for practical application in emergency situations should be consolidated by a close cooperation between SAM and emergency management communities.

Finally, methods for consequence and economic analyses (e.g. level 3 PSA) have for the time being mostly been developed nationally with limited sharing and no harmonized approach. It could be of importance for public acceptance, economics and sustainability of nuclear power plants to develop harmonized methods for SA consequence and economic analyses and share their applications. Conclusions of such studies should also be of importance for decision-makers.

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