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A Joint Report on PSA for New and Advanced Reactors

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NUCLEAR ENERGY AGENCY

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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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NUCLEAR ENERGY AGENCY

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The mission of the NEA is:

– to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as

– to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information.

The NEA Data Bank provides nuclear data and computer program services for participating countries. In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

THE COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

“The Committee on the Safety of Nuclear Installations (CSNI) shall be responsible for the activities of the Agency that support maintaining and advancing the scientific and technical knowledge base of the safety of nuclear installations, with the aim of implementing the NEA Strategic Plan for 2011-2016 and the Joint CSNI/CNRA Strategic Plan and Mandates for 2011-2016 in its field of competence.

The Committee shall constitute a forum for the exchange of technical information and for collaboration between organisations, which can contribute, from their respective backgrounds in research, development and engineering, to its activities. It shall have regard to the exchange of information between member countries and safety R&D programmes of various sizes in order to keep all member countries involved in and abreast of developments in technical safety matters.

The Committee shall review the state of knowledge on important topics of nuclear safety science and techniques and of safety assessments, and ensure that operating experience is appropriately accounted for in its activities. It shall initiate and conduct programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach consensus on technical issues of common interest. It shall promote the co-ordination of work in different member countries that serve to maintain and enhance competence in nuclear safety matters, including the establishment of joint undertakings, and shall assist in the feedback of the results to participating organisations. The Committee shall ensure that valuable end-products of the technical reviews and analyses are produced and available to members in a timely manner.

The Committee shall focus primarily on the safety aspects of existing power reactors, other nuclear installations and the construction of new power reactors; it shall also consider the safety implications of scientific and technical developments of future reactor designs.

The Committee shall organise its own activities. Furthermore, it shall examine any other matters referred to it by the Steering Committee. It may sponsor specialist meetings and technical working groups to further its objectives. In implementing its programme the Committee shall establish co-operative mechanisms with the Committee on Nuclear Regulatory Activities in order to work with that Committee on matters of common interest, avoiding unnecessary duplications.

The Committee shall also co-operate with the Committee on Radiation Protection and Public Health, the Radioactive Waste Management Committee, the Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle and the Nuclear Science Committee on matters of common interest.”

EXECUTIVE SUMMARY

Probabilistic Safety Assessment/Probabilistic Risk Assessment (PSA/PRA) for new and advanced reactors is recognized as an important approach to achieve improved safety for the future nuclear power plants. For the purpose of this work, new reactors are defined as having a stable general design which is typically within five to ten years of commencing power operations. Advanced reactors are reactors which are generally in the earlier conceptual or preliminary design stage.

The application of PSA to these reactors encounters concurrent challenges, which are slightly different for new and advanced reactors due to their development phases. The technical challenges of the PSA for new reactors, which are in the last phases of design and commissioning stage, include a lack of design detail, a lack of empirical data, and the possibility of failure scenarios that differ in character from those treated in PSAs for current reactors. These challenges can affect a variety of decisions (e.g., plant safety level assessment, defense-in-depth assessment and balanced risk concept). The technical challenges of the PSA for advanced reactors, which are in research stage or in the early phases of conceptual design, in addition to the above-mentioned aspects, also include the potential need to address very different systems and phenomenology.

The ability of current PSA technology to support design decisions for such reactors, and the potential value of advanced methods have not been internationally assessed in recent times. In order to address the above issues, the OECD/NEA Committee for the Safety of Nuclear Installations' (CSNI) Working Group on Risk Assessment (WGRisk) conducted two coordinated tasks: "PSA for Advanced Reactors" and "PSA in the Frame of Design and Commissioning of New NPPs". The objectives of the two tasks are:

- <u>PSA for Advanced Reactors:</u>
    - Characterize the ability of current PSA technology to address key questions regarding the development, acceptance, and licensing of advanced reactor designs;
    - Characterize the potential value of advanced PSA methods and tools for application to advanced reactors;
    - Develop recommendations to CSNI for any needed developments regarding PSA for advanced reactors.
- <u>PSA in the frame of Design and Commissioning of New NPPs:</u>
    - Identify and characterize current practices regarding the role of PSA in the frame of design, construction and commissioning of new nuclear power plants in the member states;
    - Identify key technical issues regarding the PSA for new reactors, current approaches for dealing with these issues and associated lessons learned, as well as issues requiring further work;
    - Develop recommendations regarding the use of PSA by different actors in the frame of new nuclear power plant projects: appropriate PSA scope and level of details, pertinent PSA applications and decision-making process;
    - Identify future international cooperative work on the identified issues.

In order to support the objectives of these two tasks, two task-specific questionnaires were developed by the task core groups and answered by the participating countries and organisations. (13 responses were received for the advanced reactor questionnaire, and 16 responses were received for the new reactor questionnaire.) Additionally, a joint workshop entitled "OECD/NEA Workshop on PSA for new and advanced reactors" was held at the OECD Conference Centre during June 20-24, 2011.

The questionnaires were developed and answered before Fukushima accident and the common workshop was held shortly after. Some discussions during the workshop referred to this event, especially regarding the external events PSA aspects, but the Fukushima issues are not specifically treated in this report. However, based on the current understanding, none of this report's conclusions or recommendations are contradicted by the worldwide ongoing post-Fukushima analyses.

The results of the two tasks show that, despite recognized challenges, there is a general consensus on the usefulness of the PSA in ensuring improved safety of new and advanced reactors. The two tasks characterized current relevant PSA practices and identified further issues to be addressed in order to improve the representativeness of the PSA for new and advanced reactors, in a context of increased usage of integrated decision making approaches. Several recommendations were formulated, mainly referring to the enhancement of the international collaborations in key areas related to development and application of the PSA for new and advanced reactors.

Regarding advanced reactors, PSA is recognized as one of the key approaches to justify safety-critical aspects in the conceptual and preliminary design stage and to address new operation concepts. However, there still remain some issues related to PSA in order to better reflect the specific design features. Most of these issues identified by survey responses and workshop participants are well recognized. Also, most of them are relevant to all reactors in the design stage (not just advanced reactors). However, there naturally are greater difficulties in addressing these issues when the reactor is in the conceptual design stage (and detailed design information has not yet been developed). It can be noted also that there is an increased emphasis on the notion of "PSA quality" as dictated by PSA standards and associated guidance documents. Other infrastructure challenges raised include the lack of "peers" (with experience in PSAs for specific advanced reactor designs) to perform PSA peer reviews and the need for regulators to understand the advanced reactor PSA models. A further regulatory challenge for the PSA for advanced reactors concerns the PSA framework, including the definition of appropriate risk metrics and the scope of PSA. Although the task was not designed to achieve consensus, there was considerable agreement as to the relevance of the issues identified. The task provided a list of topics which can be pursued by WGRISK or others. The topics would have to be prioritized based on organizational as well as technical considerations, as this was not done by the task.

For new reactors, PSA is playing a major and increasing role, in the frame of design, construction and licensing, as a complement to traditional deterministic methods. PSA is used by the industry at all stages of the design for a wide variety of applications, including demonstration of safety level, supporting "balanced design", balancing between accident prevention and mitigation features of the design, identification of design vulnerabilities and improvements, comparison with the risk of existing plants, and establishment of requirements for systems/sub-systems. Regulatory agencies are using PSAs mainly to support auditing applicant analyses and to identify risk-significant areas for safety reviews. The level of formalisation of the development and the role of PSA for new reactors is different in different countries, but generally for new reactors, the role of PSA in the regulation is more important and more systematized as compared to that for existing reactors. Some regulatory agencies have developed requirements for PSAs and their applications for new reactors. All countries recognize that the scope of PSA for new reactors should cover, generally, a wide spectrum of initiating events, including the internal and external hazards, in all reactor states and all radioactive sources (reactor and spent fuel pool). However, although the development status for internal initiating events and hazards (typically the internal fire and the internal flooding) is similar for different new reactors projects, the status of the development of the external hazards PSA varies, mainly due to differences in the project development status (site unknown, expected importance of various external hazards, etc). Also, the spent fuel pool is usually not included in the PSA, especially during the initial phases of the design. A Level 2 PSA is generally available or requested by the safety authorities for each new reactor project. A Level 3 PSA (with some exceptions) is not developed or requested by the safety authorities for new reactors. However, despite data, modelling and code limitations, Level 3 PSA was identified by project participants as a necessary support for some new reactor applications (e.g. definition of the emergency zones).

Most related activities in progress today do not seem to be aimed at a specific reactor type and consist of an extension of the conventional PSA framework to new and advanced reactors.

The main challenges to using the PSA for new reactors are related to intrinsic difficulties to ensure the representativeness and the quality of the model for a reactor in a preoperational phase. It appears that the exchange of lessons-learned on ongoing new reactor project PSAs may greatly contribute to developing improved best-practices and guidelines for the development and usage of the PSA by all actors involved in the frame of new reactor projects (industry and safety authority), mainly during the different design stages. Participants recognize the need to better integrate PSA into the design and safety review process; interaction between the PSA and design teams is important and needs to be strengthened. The challenge of appropriately using PSA results as the PSA is still evolving (to match the increasing detail of the design) is especially interesting and is worth follow-on discussion.

Regarding the insurance of PSA quality, independent verification (e.g., by peer review) is a typical method used for existing reactors. For new designs, this can be a challenge because of a lack of peers, the limited scope of most design-stage PSAs (mainly at the beginning of design), and challenges in ensuring strong interaction between design and PSA teams as the design evolves. Typically, vendors have a generic design phase PSA or a reference plant PSA, but in the construction license phase it is hard for reviewers to get a site-specific, full scope PSA (covering Level 1 and 2 PSAs, all IEs, and all operational modes). The regulator may be in a position to make decisions using a PSA which does not exactly reflect the future as-built as-operated plant and whose scope may be, temporarily, not exhaustive. The need for decision makers to consider uncertainties associated with analyses of plants in the design stage is well recognized. It is further recognized that such uncertainties could be much larger than those for operating plant analyses. Suggestions include: providing decision makers with additional information from additional analyses (e.g., margin demonstration analyses, bounding calculations and sensitivity studies), performing focused research on selected topics, development of appropriate safety goals/targets, development of appropriate PSA standards and development of appropriate peer review approaches.

The technical challenges of the PSA for advanced reactors, in addition to the above-mentioned aspects, also include the potential need to address very different systems and phenomenology, the potential unavailability of important reliability and experimental data, the potential unavailability of knowledge on new key phenomena, and the potential unavailability of accident analysis models

Both new and advanced reactors can have some new/unique features such as: evolutionary components, severe accident features, modern human-machine interfaces, and high redundancy of plant systems (increasing the potential importance of inter-systems common-cause failures – CCF), which may need improved PSA modelling. Previously accepted modelling simplifications may no longer be appropriate for the new reactor PSAs, as it is recognized that new applications (e.g., identification of licensing basis events; structures, systems and components classification) put more burdens on the PSA, for example to treat “all” engineering features. As an example, the applicability of PSA for addressing the safety-security interface was identified as an interesting topic that can raise challenges in the development and use of PSA.

Despite these modelling and applications challenges, the PSA community appears to be comfortable with the existing fundamental PSA technology (i.e. event tree/fault tree formulation). Some work is ongoing to explore the use of the non-ET/FT methods/tools as a means to more explicitly tie advanced reactors-specific phenomenological modelling into the PSA framework. Work is also being done to develop severe accident models/tools to support the risk assessment of advanced non-LWR reactors. Generally speaking, however, research on advanced methods generally does not appear to be a priority for the advanced reactor PSA task participants. As a related topic, a number of countries and organizations recognize the need to address the results of deterministic and probabilistic models in integrated decision making. Although not an issue unique to advanced reactors, it may be affected by the advanced reactor context, including the lack of detailed design information, imprecise understanding of potential accident phenomenology, and a lack of operational experience.

For both new and advanced reactors, in addition to the above, there are a number of well-recognized challenges (e.g., passive system reliability estimation, treatment of digital I&C, human reliability analysis–HRA, and external events analysis). These issues are not unique to new or advanced reactors, but their relative contributions become more important as designs eliminate recognized weaknesses and overall risk estimates decrease. External events, in particular, have gained significant attention in the light of the Fukushima accident. As discussed at the workshop, external events analyses for existing as well as new and advanced reactors need to better account for beyond design basis external events, combinations of external events, induced internal hazards and impacts on the whole site (reactors and fuel pools). The PSA methodology may need some enhancements in order to realistically model long term accident sequences, including accident management and emergency response (before and after core damage). Of particular relevance to this report, workshop participants noted that the modelling of external events in PSAs for new reactors and advanced reactors is generally not exhaustive (often because the PSAs are for generic designs), and this can pose challenges in risk-informed decision making.

In order to deal with the above-mentioned aspects, regulatory and industry organizations in some countries are supporting the development of standards to apply PSA, mainly for the new reactors. Regulatory PSA models are developed in some countries and are used for a confirmatory check of an applicant's model. Regarding the advanced reactors, the participants expressed a need for the development of specific PSA guidance and standards and recognized the ongoing development of a PSA standard for non-light water reactors under the auspices of the American Society for Mechanical Engineers (ASME).

In general, the task participants expressed the need for better advice, based mainly on the international exchange of lessons learned, on how to use PSA during the different design and preoperational phases of the new and advanced reactors.

As the two tasks concentrated mostly on the differences between existing and new/advanced reactors and on general aspects for new/advanced designs and less on PSA applications specifics for different stages of new/advanced reactor development (conceptual design, detailed design, construction, commissioning), participants noted that a future survey on these design-stage specific aspects could be useful. As some generic design assessments are underway and others can be expected in the future, such a survey (which could compare practices and results) could be valuable. This kind of activity can be part of a broader international, collaborative work.

As already mentioned, many of the new or advanced reactors PSA issues identified in the two tasks are not new and are not specific for these types of reactors, but as it is expected that future plants will have a substantial improvement in safety level, the role of PSA, as a systematic tool for identifying key risk contributors and potential opportunities for improvement, becomes more important. As a consequence, the already recognized PSA key uncertainties areas become more critical in the frame of decision making. In this context, it is recommended to:

- promote the international exchange of lessons learned in using PSA for new reactors during the preoperational phases. The challenge of appropriately using PSA results as the PSA is still evolving is especially interesting. It would also be interesting to evaluate the practice of making assumptions on absent detailed design information. The identification of failure mechanisms and scenarios, for situations involving new design features or entirely new designs, is also an important issue. It has to be noted that several international working groups (MDEP, EPR Family Group, WANO, WENRA, CANDU Senior Regulator Groups, etc.) are ongoing on this subject. However, as these working groups separate, in general, the regulatory perspective from the industry perspective and are sometimes dedicated to one type of new reactor, the promotion of a larger collaboration framework may be of benefit;
- promote the information exchange with other ongoing international advanced reactors activities (e.g., GIF-RSWG and IAEA CRP on passive system reliability);

- promote international collaboration on the PSA methods which need to be enhanced in order to model the unique features of new reactors (digital/software based I&C, HRA, modern human-machine interface, CCF of high redundancy systems and intersystem CCF, and modelling of passive systems). As developments in the above-mentioned areas are ongoing in many countries, it is recommended that, in the frame of future WGRISK reports on the use and development of PSA, particular attention should be paid to the status of development of the methodologies and the obtained results;
- promote international exchange of lessons learned regarding the PSA for advanced reactors and on specific aspects of interest to member countries (e.g., definition of risk metrics, modelling of specific phenomena for different advanced reactor types, assessment of potential severe accidents at the pre-conceptual design phase, assessment of the applicability of such frameworks/methods as the Technology Neutral Framework-TNF and the GIF-RSWG's Integrated Safety Assessment Methodology-ISAM, and development of PSA standards);
- promote international collaboration on the development of external hazards PSA as applied to new and advanced reactor designs;
- develop, as a future activity, additional guidance on the application of PSA for the design of new reactors that provide recommended practices for such matters as identifying potential failure scenarios and for determining the appropriate level of analysis. Such guidance could use the higher-level ASME PSA standard for non-light water reactors currently being developed as a starting point.

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1. INTRODUCTION

Probabilistic Safety Assessment (PSA) for new and advanced reactors is recognized as an important approach to achieve improved safety and performances of new nuclear power plant, comparing to the existing plants[1]. However, the development and the application of PSA to these reactors are subject of concurrent challenges, which are slightly different for new or advanced reactors due to their development phases.

The technical challenges of the PSA for new reactors, which are today in the last phases of design and commissioning stage, include, mainly, the lack of design detail, the lack of direct empirical data, and the need to treat some accident scenarios that may be different in character from those treated in the PSA for current reactors. These challenges can affect the development and use of PSA for decision-making support, in the frame of conception, design and construction of the next generations of NPPs.

The technical challenges of the PSA for more advanced reactors, which are in research stage or in the early phases of conceptual design, additionally include the potential need to address very different systems design and phenomenology. The ability of current PSA technology to support design decisions for such reactors, and the potential value of advanced methods have not been internationally assessed in recent times.

In order to address the above issues, the WGRisk conducted two coordinated tasks: “PSA for Advanced Reactors,” which was led by KAERI (Korea) and “PSA in the frame of Design and Commissioning of New NPPs," which was led by IRSN (France). Both tasks involved a survey based on questionnaire and one common workshop, which allowed identifying and discussing the main topics related to the PSA for new and advanced reactors.

The common workshop on “PSA for new and advanced reactors” was organized at OECD headquarter in June 2011. As given in Table 1.1, fifty experts from 13 countries and one international organization (IAEA) participated in the workshop, and 35 technical papers and 2 WGRisk task activities were presented during the 3 days workshop (20-22 June 2011). Various topics of interest to PSA practitioners were discussed, including regulatory aspects, risk informed methods, technical aspects for the new reactors design PSA, technical aspects for advanced reactors PSA, hazards PSA (internal and external), severe accident/source term/Level 2 PSA, and consequence analysis/Level 3 PSA.

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1 The task participants have different views as to what constitutes an advanced reactor design and whether the distinction between such designs and other, more near-term designs makes a difference for PSA and associated RIDM. WGRISK (or, more broadly, the PSA community) is in no position to develop a consensus definition of “advanced reactors.” For the purpose of this report, “new reactors” are reactors with a stable general design and which are typically within five to ten years of commencing power operations. ”Advanced reactors” are reactors which are generally in the earlier conceptual or preliminary design stage.

Table 1.1 The WGRisk Joint PSA Workshop: National contribution

National contributions (35 papers from 12 countries & IAEA)

	France	USA	Korea	China	Japan	Germany	Others	Total
Papers	8	9	4	3	2	2	7	35

(*) Others (1 paper per country) : Belgium, Finland, Italy, Russia, UK, India, IAEA

Category	France	USA	Korea	China	Japan	Germany	Others	Total
New	3	4		3			3	13
Advanced	4	3	3		2	1	1	14
Common	1	2	1			1	3	8

(*) **New:** Gen-III/III+ (EPR/AP1000/ABWR…); **Advanced:** Gen-IV (HTGR/VHTR/FBR/SMR…)

Category	France	USA	Korea	China	Japan	Germany	Others	Total
Level 1	7	1	2	2	1	1	2	16
Level 2	1	1			1	1	1	5
Common		7	2	1			4	14

The workshop proceedings [1] include two main sections, dedicated to PSA for advanced reactors and to PSA for new reactors, respectively. The common conclusion section presents the main conclusions and results of the workshop presentations and discussions. As the workshop objective was to offer a discussion framework as input for the ongoing tasks, no recommendations were formulated.

The questionnaires were developed and answered before Fukushima accident and the common workshop was held shortly after. Some discussions made during the workshop referred to this event, especially regarding the external events PSA aspects, but the Fukushima issues are not specifically treated in this report.

Section 2 of this report presents the objectives and findings of the advanced reactor PSA task. Similarly, Section 3 presents the objectives and findings of the new reactor PSA task. Section 4 presents the common conclusions and recommendations from both tasks.

2. PSA FOR ADVANCED REACTORS

2.1 Background

The PSA is recognized as one of the key approaches to assess the safety critical aspects for the existing plants and for the new designs. However, the ability of the current PSA technology in the frame of the design of advanced reactors and the potential values of advanced methods for it have not been assessed in recent times.

For these types of reactors, the lack of plant-specific operating experience data and procedures at the conceptual design stage may lead to PSA results that do not reflect the future as-built, as-operated plant, consequently causing an impediment to the implementation of risk-informed design and regulatory initiatives. One way to help ensure quality of the PSAs for advanced reactors, and eventually lead to better guidance/standards, would be to share lessons and best practices.

Started in June 2008, the OECD/NEA WGRisk task activity on “PSA for advanced reactors” was conducted in order to (1) characterize the ability of current PSA technology in order to address key questions regarding the development and licensing of advanced reactor designs, (2) characterize the potential value of advanced PSA methods and tools, and (3) develop recommendations to CSNI for any needed development.

Since advanced reactors are still in the earlier conceptual or preliminary design stages, the most important questions regarding the PSA were primarily focused on (1) the evaluation of the new safety concepts for advanced reactors and comparison with the existing plants, (2) the scope and criteria of PSA to characterize risk profiles for these reactors, and (3) the role of PSA in the risk-informed decision making process and in defining safety measures to various design alternatives. Another important aspect of the present task was to clarify the potential PSA issues in the frame of licensing and regulatory of advanced reactors, i.e. relevant modelling approaches, evaluation methods and scope, as well as standards, requirements and guidance for PSA.

A questionnaire had been distributed to the WGRisk member countries during the period of 2009 to 2010, and 13 answers from the 12 countries and 16 organizations including regulatory bodies, technical support organizations, research institutes, and academia (see Table 2.1) were collected until February 2010. In addition to key technical and regulatory issues requiring further works, and potential areas for future international collaboration, a wide spectrum of issues and current practices related to the PSA of advanced reactors were mentioned in the respondents’ answers to the questionnaire.

This chapter summarizes (1) the questionnaire and answers survey on PSA for advanced reactors and (2) the relevant workshop presentations, discussions and findings. The ‘advanced reactors’ mentioned in this report are reactors in the early conceptual or preliminary design stages like near-term deployable PBMR, Gen IV reactors and SMRs.

Table 2.1 Respondents to the Questionnaire

Countries	Organizations
Belgium	Bel V (A Part of the Belgian Regulatory Body)
	TE (Tractebel Engineering, Architect-Engineering Company)
China	INET (Institute of Nuclear and New Energy Technology)
Czech Republic	SUJB (State Office for Nuclear Safety)
Finland	STUK (Radiation and Nuclear Safety Authority of Finland)
	VTT (Technical Research Center of Finland)
France	IRSN (Institut de Radioprotection et de Surete Nucleaire)
	CEA (Commissariat à l'Energie Atomique et aux Energies Alternatives)
Italy	ENEA (Italian National Agency for New Technologies, Energy and Sustainable Economic Development)
Japan	JAEA (Japan Atomic Energy Agency)
	JNES (Japan Nuclear Energy Safety Organization)
Korea	KAERI (Korea Atomic Energy Research Institute)
Slovakia	UJD (Nuclear Regulatory Authority of Slovakia)
Slovenia	SNSA (Slovenian Nuclear Safety Administration)
UK	ONR (Office for Nuclear Regulation)
USA	USNRC (United States Nuclear Regulatory Commission)

2.2 Definition and Matter of Concern

2.2.1 Definition of Advanced Reactors

While a wide spectrum of advanced reactors employing different design and safety features (including SMR types and Gen-IV reactors like SFR, LFR, GFR and VHTR) are currently being proposed and developed worldwide, a clear classification of these reactors into the relevant category for the application of PSA seems not to exist up to now. For example, one SFR system proposal is primarily intended for electricity production and actinide management, and is estimated to be deployable by 2015. One GFR concept is primarily envisaged for electricity production and actinide management, although it may also support hydrogen production, and is estimated to be deployable by 2040. One LFR concept is specifically designed for distributed electricity generation and other energy products, including hydrogen and drinking water, which is estimated to be deployable by 2025. One VHTR design employs advanced concepts for helium-cooled, graphite moderated thermal neutron spectrum reactors with a core outlet temperature higher than 900°C, with a full passive decay heat removal and the electric power conversion unit operating in an indirect Brayton-type cycle (i.e. gas turbine mixture in the secondary circuit).

Comparing with the new/evolutionary reactors situation, a clear classification of advanced reactors is an important point, since there is a wide variety of proposed designs and the relative importance of PSA related issues can be different: types of hazards and scope of the analysis, relevant technical and regulatory issues, best approaches and practices; standards and guidelines, acceptable level of detail for potential applications; R&D activities and international cooperation, etc.

Concerning the R&D activities the questionnaire answers show that, generally, the effort is more important for advanced reactors comparing with the new and evolutionary reactors (Table 2.2).

Table 2.2 Categorization of new and advanced reactors according to the Q/A survey

Categorization	Reactors under Q/As	Respondents to Q/As	R&D and developmental status
Advanced reactors	Gen IV reactors (SFR, LFR, GFR, and VHTR), HTGR, SMRs (iPWR, SMART), etc	China, France, Japan, Korea, USA	Going through major activities
New reactors	New and evolutional LWRs (EPR, APR, AP1000, ABWR, APWR, ESBWR)	Czech Republic, Finland, UK	Planned and minor activities
	New LWRs or Not explicit	Belgium, Italy, Slovenia, Slovakia	No plan or little activities

2.2.2 Matter of Concern and Key Questions

While many PSAs for advanced reactors seem to be made with the conventional PSA methods such as ET/FT approaches, a few countries are considering a development of PSA methods more specialized (e.g., a complex type event tree development in case French GFR and a development of severe accident model by utilizing event tree approach in case of Japanese SFR, etc).

More efforts are currently being focused on identifying and resolving the PSA issues for designer and regulatory bodies (e.g., digital systems, passive systems, human reliability analysis, Level 2 PSA under advanced design features, etc.). While the current PSA methods and tools are in general viewed as being applicable for the advanced reactors, in case of some specific areas, some issues specific to advanced reactors may need additional developments. Moreover, given the task's emphasis to survey practices rather than come to consensus, the PSA related aspects for some advanced reactors of non-LWR types were not assessed to the extent to reach a common agreement. Further efforts could be focused on assessing the ability of the current PSA technologies for the advanced reactors and on the need to develop new PSA methods and tools. The starting point for this is identifying potential issues which can characterize the PSA of advanced reactors and to investigate the current practices and positions. Table 2.3 summarizes the matters of concern on the PSA for advanced reactors, which were addressed by the questionnaire answers.

Table 2.3 Typical issues which should be clarified for the PSA of advanced reactors

Category	Technical items for the PSA of advanced reactors
PSA scope	- Hazard types: internal events, external events (fire, flooding, seismic, etc)
	- Operating modes: full power PSA, low power and shutdown PSA
PSA framework	- The applicability of the current PSA frameworks (e.g., Level 1, 2 and 3 PSA). - The definition of risk metrics such as plant damage states (incl. core damage state). - The appropriate methods, models, and tools for the assessment (including source term release categorization and consequence analyses) - The quality of PSA
Specific technical issues	- Reliability database for advanced reactors (e.g., Initiating events frequencies, SSCs reliability data, CCF parameters, etc) - Reliability of passive systems - Reliability of digital I&C systems - Human reliability under new advanced design interface such as a digital man-machine interface - Containment/confinement performance including severe accident phenomena and countermeasures - Source terms assessment - Treatment of multiple reactor modules

Besides the above issues, there were additional concerns and comments raised from the member countries, mainly focused on regulation aspects, i.e., how to incorporate those issues into the relevant regulation framework. As the regulation framework is currently being shifted to incorporate more and more risk-informed approaches (i.e. employing both deterministic and probabilistic viewpoints), the existing PSA technologies may require further improvement, in order to allow decision-making taking into account the plant risk in an integrated way. Accordingly, the availability of PSA and the needs of additional methods applicable to the advanced reactors may be differentiated according to the regulatory environment and positions of the member countries.

An essential prerequisite for a risk-informed regulation framework approach is the scope and quality of PSA, the methods to complement the deterministic information with risk insights and applicable risk indicators. From the viewpoint of risk-informed applications, the survey identified the following aspects as being important:

- Definition of the scope and quality of the PSA required for the decision-making process for advanced reactors, including its technical acceptability (development of PSA standards);
- Integration of the deterministic principles (e.g., defense-in-depth) with PSA insights in the regulation framework;

- Definition of risk metrics specific to advanced reactors as compared to the existing ones (e.g. similar to CDF/LERF), by taking into account their specificities and including the aggregation of various risks caused by different external and internal hazards;
- Application of risk information in the design stages.

The most important technical aspects highlighted by the survey are the following:

- Further improvement of the existing methods (including modelling and analysis of the physical phenomena):
    - o Applicability of new methods and tools for the advanced reactors PSA (e.g., Non ET/FT methods);
    - o Integration of deterministic and probabilistic analysis aspects in the frame of PSA development;
    - o Containment performance including severe accident phenomena and countermeasures;
    - o Source terms and off-site probabilistic consequence analysis.
- Specific issues related to the PSA for advanced reactors:
    - o Identification and quantification of initiating events including formal classification of events (AOO/DBA/BDBA);
    - o Reliability database for new SSCs and features: reliability of passive systems and reliability of digital I&C systems including the new man machine interfaces;
    - o Human reliability under new design features.

The above issues are summarized in the following section; details are provided in the questionnaire and in the questionnaire answers (Appendices 2 and 3).

2.3 Results of the survey and of the common workshop

Appendix 3 presents the answers collected from twelve countries/sixteen organizations (covering regulatory bodies, technical support organizations, research institutes and academia). The answers provided by countries where advanced reactors are in conceptual or developmental stage refer to the current practices and activities. The answers provided by the countries with little activities or no plan for the advanced reactors refer mainly to the related expectations and concerns. The answers reflect also the different R&D and developmental status for advanced reactors (preliminary or early conceptual design stage) in different countries: countries with major activities, countries with planned but minor activities and countries with little activities or no plan. Nevertheless, most respondents highlighted the role of PSA as an effective tool to assess the safety of advanced reactors and improve their design. In addition, most countries which have a plan to develop advanced reactors anticipate that the PSA scope and the PSA role should be more important in the frame of the development of advanced reactors comparing with the current nuclear plants. Table 2.4 shows typical survey answers regarding the scope of the PSA which should be taken into account for the advanced reactors.

Table 2.4 The respondents' answers on the scope on the PSA for advanced reactors

Country	Reactors being applied	Hazard types	Operating modes	PSA levels covered	Special PSA methods
China	HTGR	Internal hazard & earthquake	All operating modes	Level 1 & 2	No special PSA method
France	SFR & GFR	All hazards			
Japan	SFR	Internal hazard & earthquake			
Korea	SFR, VHTR	All hazard		Level 1 - 3	No special PSA method, except PSR
USA	SFR, HTGR/VHTR, iPWR				No programs for special PSA method
Czech Republic	New & Evolutionary LWRs(Expected)			Level 1 & 2	Not applicable
Finland					
UK				Level 1 -3	
Others	N/A (No answers)				

2.3.1 Ability of Current PSA to Address Key Questions for Advanced Reactors

2.3.1.1 Use of PSA in the Design and Licensing Stages

Regarding the value of the PSA in the frame of advanced reactors, while there were different viewpoints depending on regulation environment of different countries, there is a common agreement on its usefulness as a complement of the deterministic methods. Some answers identified the need for PSA standards / formal guidance / requirements to assure the quality of PSA for licensing of the advanced reactors, which may include some differences comparing with actual requirements and additional reviews. In this context some R&D efforts being made to address specific regulatory issues on advanced reactors were mentioned.

Regarding the use of PSA in the design and licensing stages of advanced reactors,

- While a few countries are limitedly using PSAs in the conceptual or preliminary design stages of advanced reactors, more efforts are currently being focused on identifying and resolving the PSA issues for designer and regulatory bodies;
- While a few questionnaire answers are addressing the potential use of regulatory approaches where risk plays a greater role, these activities have not yet led to actual changes in regulation and many other countries in answer are not developing new approaches for advanced reactors and relevant PSA technologies yet;
- While the questionnaire answers express many technical and regulatory issues which should be resolved for advanced reactor designs, many of them do not seem to think that advanced reactors pose fundamentally different types of challenges and that methodological work for

some specific areas (e.g., digital systems, passive systems, HRA, etc.) is expected to be generally applicable for a wide variety of designs.

Regarding the regulatory framework for the advanced reactors, the summary of the survey answers is the following:

- China is preparing a policy statement on the use of PSA for nuclear power plants, but the role of PSA for advanced reactors is actually incompletely considered. It is intended to make a case-by-case application of the PSA approaches to the advanced reactors in the future;
- The French regulators encourage the development of PSA for advanced reactors, as for example the PSAs which has been performed for SFR. It is expected that the future regulation will be based on deterministic principles complemented by PSA insights; Reflections are in progress for defining a regulatory framework for Gen-IV reactors.
- In Japan, PSA play an important role in the ongoing design of the SFRs although the implementation of PSA is not required but strongly recommended for licensees. More specifically, NISA analyses the use of risk information widely in safety regulation. AESJ already published the PSA procedure standard for Level 1 through 3 PSA for power operation, shutdown and seismic. The guidelines for the application of “risk information” are planned in the future;
- Korea currently requires the PSA report for all of the operating reactors since 2000. Recently, new R&D plans are being prepared to develop regulatory requirements and licensing guidance for advanced reactors (such as SFR and VHTR);
- In the USA, the approach for the licensing of advanced reactors will follow the existing regulatory framework for all new reactors. However, an advanced reactor design may require exceptions to certain requirements or additional review to determine how the requirement can be met. The NGNP licensing strategy includes the use of PSA to support risk-informed processes for selection and categorization of licensing basis events and evaluation of defense-in-depth measures. A revision of the NRC risk-informed performance base licensing framework is also intended to be written in a technology neutral way;
- Other countries are concerned with PSA within the current regulation framework, without considering explicitly special issues related to the PSA of advanced reactors yet.

Regarding the development of advanced reactors specific regulatory or licensing guidance, the survey showed that:

- In China, a project has been initiated to develop the set of regulatory codes for the HTGR reactors, but currently focuses on the technical specific issues, such as material, component design, etc., and so on.
- In Japan, corresponding to the expectation by the Nuclear Safety Commission of Japan:
  - o JAEA develops research activities for preparation of regulation guidance of SFRs, including the safety evaluation of FBR fuel, the assessment of the prevention of severe accidents and the evaluation of consequences of the severe accidents;
  - o JNES develops research activities on Monju SFR, including the analyses of the basis of the allowed outage time (AOT) in the “Safety operation guideline” and the preparation of the databases of the Emergency Response Support System.
- In Korea, KINS has performed a continuous research program to apply the risk-informed regulatory approach to the existing reactors, including a licensing guidance to review the technical adequacy of Level 1 and 2 PSAs for evolutionary (APR1400) and advanced reactor

(SMART). In addition, the Level 3 PSA guidance has been proposed based on the recent research experience on a technology-neutral framework for non-LWRs.

- In the USA, NRC has research programs in place to address the need for development of regulatory guidance related to the licensing of advanced reactors.
  - o A specific research and development plan has been written to support the licensing of the NGNP prototype (e.g., VHTR), and additional research programs are planned for other advanced reactor design types;
  - o Subsequent follow-on research tasks on an advanced non-LWR PSA are likely to include development of regulatory guidance for PSA technical acceptability, development of tools, methods, and data to support a PSA technical acceptability review, and development of a scoping-level PSA model to support the ongoing identification, prioritization, and selection of advanced non-LWR research topics.

In some countries where the advanced reactors are in the development stage, efforts of designers and regulatory bodies are currently focused on identifying and addressing advanced reactor-specific PSA issues, mainly by developing relevant R&D programs and plans, as for example:

- In China, the designer proposed a research project regarding the PSA methodology for HTGR within the framework of “Important National Science & Technology Specific Projects” in 2008, to serve the PSA development for full power operation only, including safety goal of HTGR, technical framework for HTGR PSA, initiating events (IEs) analysis, reliability data collection and evaluation, and passive system reliability assessment methodology;
- In Japan, the following R&D programs are under development:
  - o Use of PSA in design & reliability data collection for the SFR components;
  - o Implementation of PSA for advanced reactor (Joyo, Monju);
  - o Development of severe accident evaluation technology (Level 2 PSA) and plant thermal-hydraulic dynamic code for SFR;
  - o Development of FP transports code and its validation with experimental data;
  - o Development of core disruptive analysis code and containment vessel response code;
  - o Preparation of detailed thermal-hydraulic analysis tool for LMFBR;
  - o Research for the level-2 seismic PSA and statistical safety analysis.
- In Korea, core technologies are being developed for evolutionary and GEN-IV reactors, mainly focused on DI&C reliability and HRA, seismic PSA methodology, and passive system reliability. In addition, the followings activities are currently being performed:
  - o SFR: pilot study to verify applicability of the conventional Level 1 PSA method to the SFR PSA;
  - o VHTR: preparation of PSA methods and procedure, reliability of passive safety system, and evaluation of interfacing events risk between nuclear facility and hydrogen production facility.
- The USNRC research projects that are aimed at or may be relevant to advanced reactor PSA include the followings:
  - o Advanced Reactor Research and Development Plan for the NGNP prototype plant;
  - o Advanced Reactor PSA Planning Study to determine PSA technical acceptability for a VHTR licensing review and Research on DI&C System Reliability Models;

  - o Long-term research on the potential benefits of advanced modelling techniques for Level 2 and 3 PSAs, focused on dynamic PSA methods utilizing a phenomenological accident simulator in conjunction with human response modelling.

- The other countries have minor activities for the PSA of advanced reactors.

Additionally, according to the workshop presentations, the current practices for PSAs of advanced reactors being performed among the OECD member countries can be summarized as follows:

- PSAs are being used at the conceptual or preliminary design stages, most analyses use currently available PSA methods including the conventional ET/FT and RMPS approaches;
- The regulatory agencies have expressed explicitly their expectations or requirements to encourage the activities to integrate the use of risk insights more fully into the design and safety review;
- Many efforts are focused on identifying and resolving well-recognized issues in an advanced reactors-specific context, which are major concerns of designers and regulators;
- Nevertheless, there is no consensus and guidance on how to take into account the foregoing issues (including the level of depth of the analyses) and on how to incorporate them into the PSA framework;
- The use of the non-ET/FT methods/tools is being explored as a mean to more explicitly tie phenomenological modelling into the PSA (e.g., RMPS, Dynamic PSA, and DDET);
- RSWG is developing TNF [2] and ISAM [3] for safety assessment of advanced reactors, which synthesizes the different approaches to safety;
- The USA is developing ASME/ANS standards for advanced non-LWR PRA.

In the same line as the current practices, many task participants recognized that further work for PSAs of advanced reactors needs to focus on the following issues:

- Necessary scope for advanced reactor PSA (e.g., importance of treating external events);
- Modelling of severe accidents, especially for non-LWRs;
- Methods and tools to aid in the search for potentially important hazards and scenarios;
- Guidance to help determine whether and when the PSA state of the art is sufficient;
- Risk metrics and goals;
- Risk/safety implications of advanced reactor security features;
- Accommodating a lack of peers when addressing the need for peer review.

Most of the PSA issues identified by survey responses and workshop participants are well recognized. Also, most of the issues are relevant to all reactors in the design stage (not just advanced reactors). However, there naturally are greater difficulties in addressing these issues when the reactor is in the conceptual design stage (and detailed design information has not yet been developed).

It can be noted also that there is an increased emphasis on the notion of "PSA quality" as dictated by PSA standards and associated guidance documents. Other infrastructure challenges raised include the lack of "peers" (with experience in PSAs for specific advanced reactor designs) to perform PSA peer reviews and the need for regulators to understand the advanced reactor PSA models. In addition, there is widespread support (across countries and organizations) for the use of PSA in support of the design and regulation of advanced reactors.

Although the task was not designed to achieve consensus, there was considerable agreement as to the relevance of the issues identified. The task provided a list of topics which can be pursued by WGRISK or others. The topics would have to be prioritized based on organizational as well as technical considerations, as this was not done by the task.

2.3.1.2 PSA Technology issues

Many of the current PSA methods and practices can be applied or extended to advanced reactors. However, some specific issues should be addressed in order to support the effective, confident use of PSAs in the frame of risk-informed decision making (RIDM) for advanced reactor. The mentioned particular concerns are the followings:

- China:
  - Definition of risk metrics for advanced reactor (such as VHTR);
  - Initiating events analysis (IE) and human reliability analysis;
  - External plant hazards analysis (to reduce large uncertainty and over-conservatism).
- France:
  - Essential feature to define advanced reactors;
  - Selection of initiating events, definition and analysis of accident sequences;
  - Reliability of passive safety systems, digital I&C systems, and reliability data for advanced reactors;
  - Severe accidents analysis & consequential analysis;
  - Use of PSA in design and PSA standard;
  - Regulatory aspects on PSA framework for advanced reactors.
- Japan:
  - Component reliability estimation method and relevant data;
  - Non-LWR specific PSA methods (SFRs);
  - Computational tools for severe accident of SFRs.
- Korea:
  - Selection of initiating events;
  - Analysis of accident scenario;
  - Reliability of passive safety systems, digital I&C systems, and reliability data for advanced reactors;
  - External events, Level 2 & severe accidents, and Level 3 & consequence analyses;
  - Consensus on PSA technology among different stakeholders.
- Czech Republic:
  - Treatment of passive features in PSA, e.g. credibility of passive functions, probability of failure of passive components;
  - Treatment of risk contributors such as component failures, common cause failures (CCFs) and human failures, emergency operation manuals (EOMs).

- Finland:
  - Reliability of passive safety systems;
  - Treatment of diverse systems in PRA;
  - Digital I&C systems.
- Italy:
  - Reliability of passive safety systems;
  - Digital I&C systems;
  - Level 2 PSA.
- Slovenia:
  - Practical implementation of PSAs for advanced non-LWR reactors.

Table 2.5 summarizes the above-mentioned issues grouped by the applicable domain: PSA general, PSA modelling, and specific PSA topics.

Table 2.5 Technical issues that each country is of particular concern

	Technical Issues of Particular Concern		# of the Respondents
PSA General	Consensus of PSA technologies among different stakeholders (e.g., Use of PSA in design and PSA standard)		2
	Regulatory aspects on PSA framework for advanced reactors		1
	Practical implementation of PSAs for advanced non-LWR reactors (e.g., SFR-specific PSA methods)		2
	Definition of risk measures for advanced reactor with essential features to define advanced reactors (e.g., VHTR)		2
	External plant hazards analysis to reduce large uncertainty and over-conservatism		2
PSA modelling	Level 1 PSA	Initiating events analysis (IE) (selection and estimation)	3
		Definition and analysis of accident sequences with deterministic safety analysis for developing accidents scenario	2
		Treatment of risk contributors with diverse systems	2
		Modelling of inter-system CCFs and related database	2
	Level 2 PSA & severe accidents (including computational tools for advanced reactors-specific severe accident analysis)		4
	Level 3 PSA & consequence analyses		3
Specific PSA topics	Component reliability estimation method and relevant data		4
	Treatment of passive features in PSA (e.g., credibility of passive functions, probability of failure of passive components)		6
	Digital I&C systems		4
	Human error probability estimation techniques to digital interfaces and relevant emergency operation manuals (EOMs)		3

2.3.1.3 Risk-Informed Regulation Issues

The recent trend of the PSA for the existing reactors is to extend the PSA scope to all levels and plant states (Level 1/2/3 for full power and low power & shutdown) and to all hazards in order to require a realistic quantification by reducing the unnecessary conservatisms. It is expected that PSA for advanced reactors be in the same line with the existing reactors. For the time being, the current PSA technology will be also used in the PSA for advanced reactors without major changes. This reflects the fact that some specific issues on PSA for advanced reactors have not been clearly defined yet. It is also expected that the PSA technologies for advanced reactors will develop according to the development status of advanced reactors. Since it is expected that the risk-informed approaches will be employed during all the design stages of advanced reactors, the relevant issues will be identified and addressed step by step. In this respect, as reflected by the survey, the following concerns or strategies were expressed by the different countries or organisations:

- China:
  - Regulatory framework to allow the use of PSA during the regulation processes;
  - Detailed understanding of regulatory staff to the advanced reactor PSA model.
- France:
  - Clear position and a harmonization of the world-wide regulatory requirements upon PSA for advanced reactors;
  - Combining probabilistic sequences and physical uncertainties (e.g., SM2A project);
  - Technology-Neutral Framework (not easy to implement).
- Japan:
  - The same grade of safety as LWR should be guaranteed if the NPP generates electricity.
- Korea:
  - For the use of PSA in the regulatory domain, appropriate performance goals for the Level 1 and Level 2 PSAs of advanced LWRs;
  - More systematic framework and methodology for the various risk-informed regulations, which will be closely related to the quality and uncertainty in PSAs.
- In the USA, conservative deterministic engineering judgment will be needed to offset PSA uncertainties and unknowns until the following issues is resolved for advanced non-LWR designs:
  - Use of PSA consensus standards and independent peer reviews against the standards for establishing PSA quality;
  - Development of PSA technical elements for advanced reactors to justify the use of the PSA and to support regulatory programs;
  - Development of appropriate risk metrics for advanced reactors to support regulatory programs;
  - Use of a PSA for a plant in the design phase which can be quite different than an operating NPP;
  - Guidance and methods for advanced reactors with designs featuring multiple reactor modules;
  - Development of new concepts of operations and requirements for some features of advanced reactor designs, such as passive systems, advanced human-machine interfaces, and modularized reactor facilities;
  - Use of scenario-specific mechanistic source terms for advanced reactor PSA;
  - Development of acceptable methods to incorporate different containment concepts employed in advanced reactor designs in PSA and to aggregate the risk from different contributing hazard groups.
- Czech Republic: Approach of the regulatory body to the PSA applications including relevant acceptance criteria;

- Finland: The technical issues related to passive safety system reliability, treatment of diverse systems, and digital I&C systems are also regulatory issues;
- Slovenia: The current PSA standards and QA programs being used as a prerequisite for use of PSA is also in the regulatory domain for non-LWR designs.

In summary, the respondents have different views as to what constitutes an advanced reactor design and whether the distinction between such designs and other, more near-term designs makes a difference for PSA and associated RIDM.

2.3.2 Potential Value of Advanced Methods and Tools

2.3.2.1 Needs for Advanced Methods and Tools

Regarding the ability of current PSA to address PSA for advanced reactors,

- Many questionnaire answers take into account a direct extension of the existing PSA methods to advanced reactor systems (e.g., reliability databases, development and application of appropriate PSA models for advanced reactors);
- The questionnaire answers seem to express different viewpoints on scope of hazards, quality, and framework of the PSA to be treated in advanced reactors, more specifically (1) while some respondents express the applicability of Level 1/2/3 for some reactor types, (2) others have no issue; while some respondents expect treatment of all modes and all hazards, others are more limited; and (3) while some respondents think current ET/FT technology is fine, some are in research stage, some are using different methods for certain problems. No disagreements among the task participants seem to arise from the fact that (1) advanced reactors are still in the stage of conceptual and design to make a consensus into the relevant PSA framework and scope; (2) advanced reactors-specific issues and phenomena were not clearly defined yet; and (3) developments of the relevant methods and tools to resolve the foregoing issues are in an early stage.

Form these points of view, most respondents mentioned the need of advanced methods and tools as an effective means to address some of the advanced reactors specific issues, but the specific R&D efforts seem to be currently rather limited:

- China is going to apply a Monte Carlo based approach to an integration process of T/H calculation and reliability evaluation results, particularly for the passive safety system reliability analysis;
- Japan mentioned the followings activities:
    - Collection of reliability data and development of PSA models and parameter for the JSFR system at a level in detail, and examining applicability of risk information to SFRs;
    - Development of MUTRAN code to evaluate the long term behaviour of the materials remaining in the core, plant T/H dynamic codes, FP transport code and its validation test, core disruptive analysis code, and containment vessel response code;
    - Preparation of detailed T/H analysis tool for LMFBR;
    - Research for the Level-2 seismic PSA and statistical safety analysis;
- Italy mentioned the development of methodologies for evaluating the reliability of T/H passive systems and its inclusion in PSA (e.g. event trees).

Associating with the applicability of current PSA technology to advanced reactors and need of advanced PSA technologies to address the design, RIDM and licensing of advanced reactor, the common workshop participants raised the followings questions:

- Does the PSA scope currently applied in PRAs for operating reactors and new reactor designs need to be modified for applicability to advanced reactors?
- Is the current PSA framework sufficient to address technical issues such as data, computer codes, risk metrics, standards and guidelines and quality assurance programs for advanced reactors?
- Do regulatory bodies have the tools and training necessary to review advanced reactor PSAs and application of PSAs?
- Are current risk metrics in guidance and standards appropriate for decision-making with regard to advanced reactors?
- Are there differences in the life-cycle applications of PSA for advanced reactors from those of current PSAs?
- What effort is needed to develop the technical bases for advanced reactor EOPs and SAMG?
- What R&D is necessary to support advanced reactor PSA development in areas such as: computer code tools, verification of tools, uncertainties, modelling new phenomena, generating experimental data, etc.?
- Can the defense-in-depth and safety margins be implemented in advanced reactors with the same approaches that are used for current reactors?
- Might a technology neutral framework on the PSA of advanced reactors be useful?

2.3.2.2 Further Improvements

While PSA technologies have been continually improved in terms of methods, models, analytical tools, and reliability data, some areas still require further improvement and clear guidance to apply to advanced reactors. The survey mentioned improvements listed in Table 2.6.

Table 2.6 PSA areas for which further improvement is required to apply advanced reactors

<table>
<tr><th colspan="2">Technical issues</th><th>Ongoing activity and future plan</th></tr>
<tr><td rowspan="3">Advanced techniques for accidents modelling and quantification</td><td>Applicability of new methods for PSA tools (Non ET/FT methods)</td><td>・ Development of PSA models and parameter for the JSFR system at a level of detail (Japan)</td></tr>
<tr><td>Combination of deterministic analysis and probabilistic analysis</td><td>・ Means to more explicitly tie the phenomenological modelling and uncertainty assessment into the PSA, e.g., RMPS, SM2A method, and DDETs.</td></tr>
<tr><td>Containment performance including severe accident phenomena and their</td><td>・ Development of MUTRAN code to use the severe accident analysis for JSFR (Japan).
・ Research for the Level 2 seismic PSA and statistical safety analysis (Japan).</td></tr>
</table>

	countermeasures	
Specific issues on advanced reactors PSA	Identification and quantification of initiating events including events classification (AOO/DBA/BDBA)	• NRC is considering how a PSA for a plant in the design phase should be used for regulatory applications such as the selection of licensing basis events (LBEs) or safety classification of SSCs.
	Reliability database for new SSCs and features	• Many countries are considering this item as a high priority issue for advanced reactors, but practical activities are currently limited for it.
	Passive systems by new methods including the aforementioned combined approaches	• Many countries are considering the reliability analysis of passive systems employed for advanced reactors as a high priority issue; a few specific approaches are under discussion.
	Digital I&C systems including new emergency operation under these new circumstance	• Many countries are considering this item as a high priority issue for advanced reactors. Developments are in progress in this area.
	Human reliability under new advanced design features	• Many countries are considering this item as a high priority issue for advanced reactors. Developments are in progress in this area.
Application of risk information into the design		• Most countries agree to the value of risk-informed decision making approach in improving design of advanced reactors. The future developments are intended.

Regarding the potential use of advanced PSA methods and tools for advanced reactors:

- While some questionnaire answers address development of new methods-related work for challenging topics (e.g., DI&C, HRA, Level 2 under advanced design features), they do not seem to be aimed at specific reactor types;
- A few questionnaire answers express the use of the non-ET/FT methods/tools as a mean to more explicitly tie phenomenological modelling into the PSA (e.g., RMPS (Reliability Methods for Passive Safety systems), Dynamic PSA, DDETs (Discrete Dynamic Event Trees)). While those methods have a potential value for their application to advanced reactors, the tools implementing these methods are under development and there is no clear guidance on how to incorporate into the PSA framework for advanced reactors yet;
- A few questionnaire answers also address the need of advanced reactor-specific severe accident (SA) analysis models, definition of source terms risk, and their countermeasures which could be critical in determining the risk of advanced reactors. While they could be involved in the realm of PSA, however, there seems not to reach any consensus on how to take into account those issues for some advanced reactors and on the level of depth to take into account;
- A number of countries are performing research and development to address widely recognized PSA challenges (e.g., passive systems reliability, digital I&C systems reliability), as well as phenomenological issues that will needed to be addressed in advanced reactor PSAs.

In general, the respondents have not identified a need for advanced PSA methods (e.g., dynamic PSA methods) to address the risk associated with advanced reactor designs. Many organizations are performing design-support PSAs using conventional methods and tools. However, some countries see potential benefits associated with such methods and are pursuing related, limited-scope research and development activities.

As a related topic, a number of countries and organizations recognize the need to address the results of deterministic and probabilistic models in integrated decision making. Although not an issue unique to advanced reactors, it may be affected by the advanced reactor context, including the lack of detailed design information, imprecise understanding of potential accident phenomenology, and a lack of operational experience.

2.3.2 Potential Areas for International Collaboration

A few international activities on PSA for advanced reactors are in progress:

- The IAEA has an ongoing Coordinated Research Program (CRP) on passive systems reliability;
- The CNRA has a Working Group on the Regulation of New Reactors (WGRNR);
- The Generation IV International Forum (GIF) conducts an assistance group as the RSWG for the risk-informed safety evaluation of Gen-IV reactors.

The international collaboration is expected to play an important role for the improvement of PSA technology for the advanced reactors and refining the research and development roadmaps or programs of each country. Table 2.7 shows a summary of the questionnaire and answer survey on the aspect.

Table 2.7 Technical items for more clarification and further improvement

Technical Items	Current similar collaboration
Definition of probabilistic criteria for RIDM and/or PSA applications (which may replace the traditional CDF/LERF criteria)	YES
Development of a methodology to take into account severe accidents as early as possible during the pre-conceptual design phase	YES
Regulatory aspect of PSA, such as regulatory requirements (criteria, scope, QA requirements etc)	YES
Development of standard or guidance on technical acceptability for advanced non LWR PSAs	YES
Advanced Non-LWR PSA Planning Study (including modelling techniques) (Level 1/2/3)	YES
Collection of possible non-LWR specific initiating events or its guidance for selecting;	YES
Sharing of reliability DBs for components and initiators specific to different reactor concepts	NO

Digital I&C model and data under digital environment	YES
HRA under digital environment including for shutdown PSA and external events PSA	NO
Seismic PSA (to reduce uncertainty)	NO
Aircraft crash risk analysis	YES
Reliability issues with severe accident analysis (e.g., different core catcher designs also relevant to evolutionary reactors)	YES
Reliability of passive safety systems	NO
MDEP discussions focusing on advanced reactor designs may share experience and understand technical issues encountered in the licensing process of other regulators	NO

The common workshop participants made a tentative consensus for the need of international collaboration in the following research areas:

- Guidance to determine technical acceptability of PSA for advanced reactors (including the ASME/ANS non-LWR standard) and provide its implementation process;
- Advanced reactors-specific accident phenomena and passive safety system reliability, reliability databases, human and digital system reliabilities, accident sequences and event classifications for PSA modelling, and adequacy of current phenomenological models to support the accident sequence analysis;
- Assessment of the possibility of potential severe accidents in the pre-conceptual design phase of advanced reactors.

3. PSA FOR NEW REACTORS

3.1 Background

The main objective of the task on "PSA in the frame of design and commissioning of New NPPs" was to identify and characterize current practices regarding the role of probabilistic safety assessment (PSA) in frame of design, construction and commissioning of new nuclear power plants in the member states:

- to identify key technical issues,
  - current approaches for dealing with these issues and associated lessons learned,
  - issues requiring further work
- to develop recommendations regarding the use of PSA by different actors in the frame of new nuclear power plant projects:
  - appropriate PSA scope and level of details,
  - pertinent PSA applications and decision-making process.
- to identify future international cooperative work on the identified issues.

The scope of the task covered the technical issues related to PSAs for nuclear plants in the final design, construction, testing or commissioning phases. Plants in these phases have a stable general design configuration and are typically within five to ten years of commencing power operations. The task covered the PSA development and uses during the final design, construction, testing and commissioning pre-operational phases of a new power plant project, addressing applications performed by all actors involved in new NPP design and commissioning, referring to all PSA levels (1, 2, and 3) and to all PSA scopes (reactor, spent fuel pool, internal events, hazards, etc.).

Designs of more advanced reactors are generally in the earlier conceptual or preliminary design stage: they were treated by the specific WGRisk task on "PSA for advanced reactors" which is presented in the chapter 0 of this report.

The performance of this WGRisk task was justified by the fact that, although previous WGRisk tasks have addressed potentially relevant aspects (probabilistic risk criteria, external events, digital instrumentation and control, and low power and shutdown PSA), most of these tasks were focused on currently operating reactors. Today, new power plant projects are ongoing or expected in many countries. In all these countries, the new power plant project actors (safety authorities, technical support organisations, designers, constructors, and operators) are using (or plan on using for intended new plant projects) probabilistic methods in order to achieve improved nuclear power plant safety and performances comparing to the existing plants. The practices may be however different in different organisations and countries. The objectives of using the PSA, the guidance level, the PSA level and scope, the PSA applications and the approaches and methods to integrate the PSA in the decision-making process may be also different. PSAs for reactors in various stages of the design and commissioning process can face challenges (including a lack of design detail, a lack of empirical data, and the possibility of failure scenarios that differ in character from those treated in current PSAs) for which new standards requirements and guidance may be needed. In most member countries, PSA standards requirements and guidance are based on experience with operating reactors. One way to help ensure quality in the PSAs for new builds, and eventually lead to better guidance/standards, would be to share lessons and best practices. For example, many risk-informed regulatory programs (such as risk-informed in-service inspection, technical specification completion times and surveillance frequencies, and graded quality assurance) are based on the accumulation of plant-specific performance data gathered within a largely deterministic programmatic framework. Within a risk-informed decision making context, this baseline data serves as a performance benchmark to ensure that the implementation of risk-informed programs do not degrade plant safety. Additionally, the detailed plant procedures needed to assess human performance may not be available. The lack of plant-specific operating experience data and operations procedures at the design stage may lead to PSA results that do not reflect the future as-built, as-operated plant. This may lead to an impediment to the implementation of risk-informed regulatory initiatives.

In order to complete the above objectives, a questionnaire was distributed to nuclear power project actors (safety authorities, technical support organisations, designers, constructors, and operators).

As the WGRisk task on "PSA in the frame of design and commissioning of New NPPs" addressed certain common aspects with the task on "PSA for advanced reactors", a joint workshop on lessons and best practices, involving PSA experts behind past design-support studies and PSA experts involved today in the field, was organized at OECD headquarter in June 2011.

The task performance was led by IRSN (France).The coordination was performed by the NEA Secretariat. The project core group included IRSN (France), NUBIKI (Hungary), STUK (Finland) and NRC (USA). Other WGRisk members participated through their response to the questionnaire and by participating to the joint workshop.

The following paragraphs were developed based on questionnaire answers and on the joint workshop discussions and conclusions.

3.2 Use and development of PSA for new reactors

In the frame of design, construction and licensing of new reactors, the PSA plays an essential role as complement of traditional deterministic methods. All nuclear industry actors, as well as the safety authorities, recognize the value of using the PSA to develop new improved reactors, by taking advantage of this systematic and comprehensive analysis method.

New nuclear power plant (NPP) projects are under development or intended in many countries. As for example, new reactors are under construction in Finland, France and China. In other countries new designs are under assessment by the safety authorities in order to prepare the future applications for new NPPs, as for example in USA and UK. Finally, in other countries new projects are intended, the preparatory analysis being under way, like, for example, in Switzerland, Italy and Hungary. New reactor types are in general "evolutionary" comparing with the existing NPPs, the basic technology being the same (PWR, BWR,..) improved with new advanced features, intending mainly to enhance the safety and the availability of new NPPs, as for example: EPR, AP1000, ABWR, and ESBWR.

The survey showed that, for the nuclear industry, the PSA results was one of the main reasons of choosing a given design as a candidate for building new plants, since the design PSA provides a preview of the plant risk profile which can be compared versus the new project specific regulatory requirements.

3.2.1 Role of PSA

The PSA is a licensing requirement for new reactors in many countries, like for example:

- In India, the submission of Level-1 PSA for internal events at full power before the first criticality is a mandatory requirement for new NPPs;
- In Switzerland, based on the Nuclear Energy Act and its accompanying ordinance, full-scope, plant-specific Level 1 and Level 2 PSA for all relevant operational modes are required. A Level 3 PSA is not required in Switzerland;
- In USA, For DC and COL applications submitted according to Title 10, Part 52 of the U.S. Code of Federal Regulations (10 CFR Part 52), the applicant is required to submit a description of the design-specific (for DC) or plant-specific (for COL) probabilistic risk assessment (i.e., the PSA) and its results. The applicant is required to develop a Level 1 and a Level 2 PSA. The PSA must cover those initiating events and modes for which NRC-endorsed consensus standards on PSA exist one year prior to the scheduled date for initial loading of fuel;
- In Hungary, the PSA is mandatory according to Hungarian regulatory requirements laid down in the Nuclear Safety Codes. Level 1 and 2 PSAs are required for a nuclear power plant covering all plant operational states, modes and initiating events (internal and external). PSA quality is addressed in a regulatory guide on PSA. No explicit requirements exist on PSA applications;
- In UK, as part of the site licensing process ONR requires that a Site Specific Pre-Construction Safety Report (PCSR) submitted prior to legal permission being given for the start of major safety related construction activities. This PCSR should include a full scope Level 1, Level 2 and Level 3 PSA;
- In Finland, the Living PRA is formally integrated in the regulatory process of NPPs already in the early design phase as a part of the licensing documentation. PRA and its applications are to run throughout the construction and operation phases of the plant service time. A plant specific, design phase level 1 and 2 PRA is required as a prerequisite for issuing a positive statement for an application of the construction license for a new NPP design and a complete level 1 and 2 PRA for issuing a positive statement for an application of an operating license. The plant specific Level 1 and 2 PRAs include internal initiators, fires, flooding, harsh weather conditions and seismic events for full power operation mode and for low power and shutdown model.

In other countries, the PSA is not a mandatory requirement, but is considered as an essential complement of deterministic analyses, like for example:

- In France, as indicated in the French PSA Fundamental Safety Rule (2002-01), the safety of French nuclear reactors is based essentially on a deterministic approach. PSA supplements the conventional deterministic analyses. For the new generations of reactors, PSA has however a formal role as a supplemental tool in safety assessment, as stated in the "Technical Guidelines For The Design And Construction Of The Next Generation Of Nuclear Power Plants With Pressurized Water Reactors";
- In Japan, the PSA is not a mandatory regulatory requirement, but is considered to be important information to regulate NPP safety. In some area, especially in inspection of NPPs, risk-informed regulation is put in place;
- In Slovenia, the regulation has some general requirements for the use of PSA in design and operation of NPPs.

Finally, in other countries, for the time been, the role of the PSA is not formalized in the regulation, as for example, in Italy and in Czech Republic.

The role of PSA in the regulation for new reactors is diverse in different countries, as for example:

- In Switzerland, the Nuclear Energy Ordinance enacts a number of PSA applications. Furthermore, the ordinance authorizes the Inspectorate to issue two PSA guidelines: Guideline ENSI-A05 "PSA quality and scope" and Guideline ENSI-A06 "PSA applications." Although both guidelines are in general written for existing and new reactors, they are currently focusing on existing reactors. In particular ENSI-A06 may be changed in the future in order to reflect the more stringent requirements for new reactors. The PSA applications are:
    - Evaluation of the safety level and the identification of potential plant-specific vulnerabilities. Corresponding evaluation criteria are given in the regulatory guideline ENSI-A06. This evaluation is performed within the framework of plant-specific licensing actions and/or the periodic safety review. The balance among the risk contributions from initiating event categories, accident sequences, components and human actions shall be evaluated. If accident sequences belonging to any of the initiating event category, and components or human actions are found by PSA to have a remarkably high contribution, measures to reduce the risk shall be identified and – to the extent appropriate – implemented. The regulatory guideline ENSI-A06 provides criteria for the evaluation of the balance of the risk contribution of the various initiating event categories. According to the ordinance on "Hazard Assumptions and Evaluation of Protection Measures against Accidents in Nuclear Installations" the plant shall be designed against natural hazards such as earthquakes, flooding and extreme weather conditions. In particular, sufficient protection against natural hazards with a frequency greater than or equal to 1E-4 per year shall be demonstrated. The corresponding hazard curves are taken from the PSA;
    - The impact of a plant modification on the risk shall be assessed. This applies to all PSA-relevant structural or system-related plant modifications as well as to changes of the technical specification involving PSA-relevant components. Criteria are given in the regulatory guideline ENSI-A06;
    - In defining the allowed outage times, it shall be ensured that components shown to be significant to safety from the PSA point of view (ENSI-A06) are considered in the technical specifications (completeness), and assigned to correspondingly short allowed outage time categories (balance). Based on the risk measures CDF and LERF, a review of the completeness and the balance of the allowed outage times shall be carried out in the course of the periodic safety review;
    - At the beginning of every year, the licensees submit ENSI a probabilistic evaluation of the operational experience of the previous year. In this study initiating events as well as component unavailability due to planned or unplanned maintenance or tests are considered. The study involves among other things the determination of the probabilistic safety indicators (maximum annual risk peak and incremental cumulative core damage probability) and the risk contribution of the online maintenance;
    - PSA is one element in the integrated decision-making. Therefore, PSA is also used to classify reportable events (provided the event affects a PSA-relevant structure, system, component or operator action). Since ENSI classifies all reportable events by the INES-Scale, regulatory guideline ENSI-A06 provides a relationship between the cumulative conditional risk of an event and the INES-Scale;
    - Insights from the plant-specific Level 2 studies are used as part of the technical basis of the Severe Accident Management Guidance (SAMG) in order to provide information on the possible accident progressions and plant states. Furthermore, Level 2 PSAs are also used for the preparation of emergency exercises dealing with severe accidents;
    - The Nuclear Energy Ordinance also enacts the requirement of a PSA Level 1 and Level 2 for the construction permit and the operation permit of a new plant to show the safety

level, the balance of risk contributors, and the balance and completeness of the technical specifications (only for operation permit).

- In France, for the new generations of reactors, PSA is used as a supplemental tool in safety assessment during the design phase. The contributions of these assessments include the following:
    - Help for the design of safety systems, particularly in terms of redundancy and diversification;
    - Verification of a balanced conception of reactor safety related to the absence of scenarios having a predominant contribution to the frequency of core damage;
    - Estimation of the deviations with respect to the safety requirements applied to operating reactors;
    - Comparison of the level of safety of the future reactor with that of operating reactors or of other reactors under development;
    - Help with the definition of operating conditions related to multiple failures;
    - Preliminary assessment of the safety improvement resulting from the planned measures in the case of a severe accident;
    - Help in the demonstration that the sequences leading to large (and/or early) releases are practically eliminated.
- In the USA, the uses of the PSA during reactor licensing are discussed in NRC's Regulatory Guide 1.206. These uses include:
    - Identify risk-informed safety insights based on systematic evaluations of the risk associated with the design, construction, and operation of the plant;
    - Demonstrate how the risk associated with the design compares against the Commission's goals of less than $1x10^{-4}$/yr for core damage frequency, less than $1x10^{-6}$/yr for large release frequency, and that the conditional containment failure probability be less than approximately 0.1 for the composite of all core damage sequences assessed in the PRA;
    - Demonstrate whether the plant design, including the impact of site-specific characteristics, represents a reduction in risk compared to existing operating plants;
    - The PSA results and insights are used to support other regulatory programs as follows:
        - Support the process used to demonstrate whether the Regulatory Treatment of Non-Safety Systems (RTNSS) is sufficient and, if appropriate, identify the SSCs included in RTNSS;
        - Support the regulatory oversight processes for operating reactors, e.g., the Mitigating Systems Performance Index (MSPI) and the significance determination process (SDP), and programs that are associated with plant operations, e.g., technical specifications (TS), reliability assurance, human factors, and Maintenance Rule (10 CFR 50.65) implementation;
        - Identify and support the development of specifications and performance objectives for the plant design, construction, inspection, and operation, such as Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC); the Reliability Assurance Program (RAP); TS; and COL action items and interface requirements.
- In UK, the PSA should be used to help in showing that the design satisfies *As Low As Reasonable Practicable* (ALARP) requirement. For the pre-licensing GDA process the

Generic PCSR should also include a full scope Level 1 and level 2 PSA. A Level 3 PSA is also required but as details are site specific, a high level outline analysis is acceptable. Where numerical targets are given in the SAPs, ONR will seek sufficient information for it to judge that the target is likely to be achieved and the overall risk is ALARP. Further guidance on ONR's (HSE) expectations relevant to PSA can be found in the SAPs (Safety Assessment Principles for Nuclear Facilities, 2006 Edition and Revision 1, January 2008) and in the PSA TAG (2 Probabilistic Safety Analysis, Technical Assessment Guide. T/AST/030, Issue 3, Health and HSE Nuclear Directorate, February 2009). GDA is following a "Claims – Arguments – Evidence" structure:

- Step 2 was "claims" and for PSA these were interpreted as approach, outline scope, criteria and output of the PSA;
- Step 3 was "arguments" which were broadly interpreted as being the methods, techniques and detailed scope (Step 3 Probabilistic Safety Analysis Assessment of the Westinghouse AP1000. HSE-ND Assessment Report AR 09/017, November 2009 and Step 3 Probabilistic Safety Analysis Assessment of the EDF and AREVA UK EPR. HSE-ND Assessment Report AR 09/027, November 2009);
- Step 4 has concentrated on the "evidence" and for PSA this is the detailed implementation of the methods and techniques, and the data and parameters used to quantify the PSA.

Step 4 GDA reports were published in ONR website in December 2011. As well as the detailed review of all the technical areas of the PSA, during GDA Step 4 a Risk Gap Analysis (RGA) has been undertaken. The RGA was designed to meet the following objectives:

- Support to GDA conclusion whether the EPR or AP1000 are reactors that can be built and operated safely in the UK;
- Evaluation of the importance of the findings of the GDA review in the various PSA technical areas;
- Evaluation of the overall gap between the plant design risk claimed by RP and risk contributors that may have been underestimated or omitted.

- In Slovenia, the regulation has some general requirements for the use of PSA in design and operation of NPPs, like:
  - The NPP must be designed so as to assure that the total CDF is less than $1x10^{-6}$/yr and the LERF is less than $1x10^{-6}$/yr; If the CDF is less than $1x10^{-5}$/yr, but greater than $1x10^{-6}$/yr or if the LERF is less than $1x10^{-6}$/yr, but greater than $1x10^{-7}$/yr, investor or operator shall substantiate that further risk reduction is not possible or reasonable;
  - The licensee must establish the PSA analysis as a part of the Final Safety Analysis Report;
  - The PSA analysis must be used in decision making process, for plant modifications, periodic safety review, online maintenance, development and verification of programs, event analysis.
- In Finland, as mentioned, the Living PRA is formally integrated in the regulatory process of NPPs already in the early design phase as a part of the licensing documentation. STUK will review the PRA and make an assessment of the acceptability of the design phase and construction phase PRAs prior to giving a statement about the construction license and operating license applications, respectively.

3.2.2 Risk informed applications

Both industry and safety authorities use the PSA for a variety of applications during all stages of the new NPP projects. In general, full scope level 1 and level 2 (internal events, internal and external events hazards) for power and shutdown states is developed and used in the frame of new reactors projects. These PSA are more or less complete as they develop with the design. The internal events PSA is in general more developed, while the other internal and external hazards are sometimes treated with other screening methods. The PSA is in general available for the reactor. The spent fuel pool is not always included. The site specific aspects are not always taken into account, since for many projects the exact site is not known. These aspects are treated in more detail in chapters 0.

The main PSA uses during the new reactor projects design are:

- safety demonstration, including the demonstration of meeting probabilistic safety goals;
- support of the choice of design options;
- well-balanced safety concept;
- defence in depth assessment /multiple failures conditions definition;
- appreciation of the improved safety level compared to existing plants;
- confirmation of the protections from external and internal hazards;
- cost-benefit /ALARP;
- support of the severe accidents analysis.

In general, the development of risk-informed applications is not required by safety authority (only few countries have legislation in this respect). Nevertheless, many risk informed applications are performed or are intended by all new reactor project actors, as for example:

- technical specifications definitions;
- safety classification of SSCs;
- in-service testing (IST) and in-service inspection (ISI) definitions;
- online preventive maintenance definition;
- emergency operation procedures development;
- development of operator training program and of simulator training;
- development of accident management countermeasures;
- Reliability Assurance Program definitions;
- risk-informed security analysis.

In general, it is recognized that the design PSA has important uncertainties. The decision making process has to consider this aspect and to provide means to take into account the uncertainties while interpreting and using the PSA results. In general, the data uncertainties are analyzed quantitatively by using the uncertainties propagation. The impact of epistemic uncertainties is evaluated in general by performing sensitivity analyses. The PSA limitations and assumption should be always indicated and discussed.

STUK mentioned that, because the safety systems are not yet designed in all details, the design phase PRA includes uncertainties related to the systems configuration. There may be also uncertainties involved in the data used if the plant unit in question is a prototype. In the design phase PRA, operating experiences collected from similar plants or corresponding applications shall be used. As to the PRA of an operating plant, the plant specific data and if necessary, combined with data received from other similar plants or corresponding applications, shall be used and, in the absence of such data, general data shall be used. The feasibility and uncertainty of the data shall be justified. Provided that no adequate design, site and reliability data are available for the design phase PRA or if some safety related systems are constructed using a technology such that there are no well established methods available for computing the system reliability estimate, expert judgment, experiences and information from corresponding applications and corresponding sites can be used. In that case the estimation procedure must be justified.

NRC has developed guidance on the treatment of uncertainties associated with PSA in NUREG-1855. In addition, as part of the licensing reviews for new designs, NRC may request additional calculations and studies to be performed in order to assure that the uncertainties associated with novel aspects of the designs are adequately addressed. These may include, for example, studies demonstrating the appropriate use of conservatism in developing PSA success criteria, the use of bounding parameters for PSA supporting calculations and sensitivity studies, and testing programs to validate calculations.

Generally, regarding the overall roles and applications which PSA played during the design and review of the new reactors, the regulatory agencies have expressed explicitly their expectations or requirements to encourage the activities to integrate the use of risk insights more fully into the design and safety review. PSA has been used and are also highly recommended to be used from the very early stage of the design phase. The nature of PSA can provide a frame for the synthesis of the different kinds of available knowledge at the design stage. The continuous iteration and interaction among PSA team and design teams are recognized to be of very important necessity to the success of risk-informed processes. Participants recognize the need to better integrate PSA into the design and safety review process; interaction between the PSA and design teams is important and needs to be strengthened.

Epistemic uncertainties due to lack of design information, unknown phenomena, plant-specific hazards, data, etc., may be larger than that from existing reactors, and will impose a significant challenge to the decision making. The PSA quality and technical adequacy are critically important when the PSA is to be used in reactor design and licensing. Because new reactors are likely to use innovative technologies, with which we have relatively little experience, this need is greatly magnified relative to PSA applications for current plants. Appropriately focused research in selected areas, careful thinking about both qualitative and quantitative safety goals, well conceived PSA standards, and independent peer reviews are important to allow PSA to fulfil its potential in supporting design and licensing. Establishing a regulatory PSA model for confirmatory check of the technical adequacy during the design certification phase is a practice in some countries (like in France for example). Since technical adequacy of a design phase PSA is an essential part of risk-informed decision making, the regulatory PSA model which is independent from the applicant's model may be one of the possible choices that some countries may endorse.

Member countries have diverse requirements for PSA in support of licensing, ranging from no requirements to "full scope Level 3." Some require PSA at specific stages in the licensing process; at least one country (Finland) requires a Living PSA commencing with design and continuing through operation.

The members make broad use of PSA. At a high level, it appears that the members' ranges of uses are similar. Commonly cited example uses include:

- demonstration of achievement of a safety target (e.g., specific numerical criteria, balanced design, defense-in-depth, achievement of improvement over operating plants and ALARP);
- assessment of risk impact of proposed modifications;
- support development of design/operations (e.g., EOPs, training programs, technical specifications, inspection and classification and treatment of SSCs).

Although it is probably implicit in many country responses (e.g., through the demonstration of meeting safety targets), one country (UK) indicates that the PSA provides direct support to the overall decision as to whether a new reactor design can be built and operated safely.

The need for decision makers to consider uncertainties associated with analyses of plants in the design stage is well recognized. It is further recognized that such uncertainties could be much larger than those for operating plant analyses. Suggestions include:

- providing decision makers with additional information from additional analyses (e.g., margin demonstration analyses, bounding calculations, sensitivity studies);
- performing focused research on selected topics;
- development of appropriate safety goals/targets;
- development of appropriate PSA standards; and
- development of appropriate peer review approaches.

The challenge of appropriately using PSA results as the PSA is still evolving (to match the increasing detail of the design) is especially interesting and is worth follow-on discussion (e.g., to see if there's anything that can be suggested beyond "do the best you can") It would also be interesting to evaluate the practice of making assumptions absent detailed design information (e.g., whether it's more effective to include an analysis based on such assumptions or to perform a conditional analysis that excludes affected contributors).

3.2.3 Guidelines and international cooperation

In general, the development and the use of PSA in the frame of new reactor projects are based on international available guidelines. The most used are the: IAEA Safety Series No. 50-P-4, IAEA-TECDOC-1144, ASME RA-S 2008 PRA Standard, U.S. NRC Regulatory Guide 1.200, PRA Procedures Guide NUREG/CR-2300, NRC-RES Fire PRA Methodology for Nuclear Power Facilities, NUREG/CR-6850, ANS External-Events PRA Methodology, ANSI/ANS-58.21-2007, WENRA requirements and EUR requirements. These guidelines are completed, in many cases, with national references, like for example:

- China: specific PSA guide HAF.J008, Standard formats and Contents of Nuclear Power Plant Probabilistic Safety Assessment Report;
- India: AERB/NPP&RR/SM/O-1;
- UK : "UK Health and Safety Executive (HSE) - Safety Assessment Principles for Nuclear Facilities," 2006 Edition Revision 1, January 2008 and "UK Health and Safety Executive (HSE) - Technical Assessment Guide, Probabilistic Safety Analysis," T/AST/030 issue 03, February 2009;
- Switzerland: ENSI-A05 (quality and scope of the PSA) and ENSI-A06 (PSA-Applications);
- France: French PSA fundamental safety rule (2002-01), ASAMPSA2 guideline will be used in the future for L2 PSA;
- Japan: Guidelines for PSA quality for NPP (prepared by JNES and NISA), Level 1 and 2 PSA standards (prepared by Atomic Energy Society of Japan);
- Czech Republic: Guideline for Level 1 PSA BN-JB-1.6;
- Hungary: Regulatory Guide 3.11. on PSA of the Hungarian Atomic Energy Authority;
- Finland: Regulatory Guide YVL 2.8 "Probabilistic safety analysis in safety management of nuclear power plants. "

No specific guidelines for new reactors PSA currently exist, as the existing guidelines are generally judged to be fully applicable. However, as the existing guidelines typically were not developed for pre-operational reactors. ASME is now developing a PSA standard for new LWRs that will be applicable to pre-operational reactors.

Beside the OECD/NEA WGRisk, several other international working groups are dealing with the development of new reactors:

- MDEP (EPR, sometimes AP1000);
- EPR Family Group;
- WANO;
- CANDU Senior Regulator Groups.ft

The performed survey revealed the following most sensitive fields for the development and use of PSA in the frame of new reactor projects:

- reliability analysis of digital I&C and software reliability;
- assessment of internal and external hazards;
- risk-informed application for new plants and use of PSA throughout the reactor design cycle;
- role of PSA in licensing new nuclear power plants;
- reliability analysis of passive systems;
- probabilistic safety criterion;
- Common Cause Failures;
- HRA approach for NPP with advanced computerized HMI;
- long term analyses;
- dynamic PSA;
- assessment of site risk for multi-reactor sites;
- uncertainty quantification methods and communication of uncertainty results;
- assessment of PSA limitations;
- design features credited in the PSA for the prevention and mitigation of severe accidents.

These subjects are potential candidates for future methodological developments and/or international cooperation.

Regarding research and development, participants identified a number of potential topic areas. Most of these topics are similar to those identified previously (in the report) for advanced reactors. As in the case of the advanced reactor list, additional work will be needed to clarify the specific issue(s) embedded in the topic description and the associated R&D work needed.

3.2.4 Technical aspects

3.2.4.1 Internal events, Level 1 PSA

<u>*Initiating events*</u>

The initiating events list for new reactors is mostly developed based on similar existing reactors PSA and on generic references like IAEA-TECDOC-719, NUREG/CR 3862, NUREG/CR 6928 and NUREG-1829. This list is then complemented with some specific analysis to take into account the specificities of the new designs, like, system analyses, FMEA and Master Logic Diagram. It appears that the resulting initiating events list contains, in general, few new initiating events specific to unique or novel plant design feature. The new initiation events are associated mainly with spuriously automatic actions.

PSA supporting studies

In most cases, specific support studies (best estimate) are developed for the PSA for new reactors. These may include, for example, system engineering analyses and thermal hydraulic analyses for determining success criteria for mitigating systems. The need for specific studies development is identified in accordance with the PSA standards and guidance. For example, in UK, specific studies for PSA have been developed for both new reactors (EPR and AP1000), which were included in the GDA.

The Safety Report analysis and the design basis reports are another source of information for the development of the new reactors PSA, as well as the PSA reports of similar reactor PSAs.

Passive systems modelling

Many new reactor designs use passive safety systems. Due to the specificities of passive systems that utilize natural circulation (small driving force, large uncertainties in their performance, lack of data, etc.), there is a need for the development and demonstration of consistent methodologies and approaches for evaluating their reliability. In order to get confidence in the achieved results, it is necessary to reduce the level of uncertainty pertaining to the passive system behaviour, and in particular the phenomenological uncertainty. The determination of the dependencies among the relevant parameters adopted to analyse the system reliability is also essential. The study of the dynamical aspects of the system performance, because the inherent dynamic behaviour of the system should to be characterized, is another important aspect.

However, it seems that, in the available design PSAs, the passive systems models considers only the failure of the systems components (pipe break, spuriously actuation of valves, etc.), the "failure" of the phenomena (natural circulation for example) not being generally taken into account.

This aspect may need some improvements by modelling, for example, the scenario dependent situations which can lead to a combination of conditions for which the passive system function can not be performed, or by using parametric models. The modelling of passive systems in the PSA raises also the question of the impact on other PSA aspects. For example, the functioning of the passive systems for long term accident scenarios should be carefully analyzed. Another important issue is the treatment of the physical and thermal hydraulic data uncertainties as well as of the uncertainties in the behaviour of the passive systems.

The existing thermal hydraulic codes may not be completely applicable for the analysis of the passive systems behaviour for PSA supporting studies. In fact the main two concerns with thermal hydraulic codes are:

a) whether the codes are still within their domain of applicability when the input parameters are varied over their potential ranges or not, and

b) whether the analyses take into account the possibility of degraded conditions (e.g., following a seismic event) or not.

The international activities performed up to now on the passive systems reliability have not explicitly treated the modelling of the passive systems in the PSA. Furthermore, this survey did not identify any analyses indicating the relative importance of dealing with this issue as opposed to other sources of uncertainties (e.g. digital I&C reliability.) The passive systems reliability and modelling in the PSA is considered an open issue which may need more efforts.

Reliability and CCF data

In general, reliability data and common cause failure parameters for the components considered in the PSA for new reactors are taken from the same sources as for the PSA for existing reactors. The preferred source is data quantified by using the available operating experience, completed with data from international available sources. The most used international sources are:

- NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," Idaho National Laboratory, February 2007;
- Component Reliability Data for Use in Probabilistic Safety Assessment, IAEA-TECDOC-478;
- Institute of electrical and electronic engineers (IEEE) Std. 500 "Guide to the Collection And Presentation of Electrical, Electronic, Sensing Component, And Mechanical Equipment Reliability Data For Nuclear power Generating Stations," 1984;
- T-Book, Reliability Data of Components in Nordic Nuclear Power Plants, 6th edition, 2005;
- NUREG/CR-5750 Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 – 1995;
- NUREG/CR-5497, "Common Cause Failure Parameter Estimations," U.S. Nuclear Regulatory Commission, Washington, DC, November 1998;
- VGB PowerTech e.V. Zentrale Zuverlässigkeitsund Ereignisdatenbank (ZEDB), Zuverlässigkeitskenngröß en für Kernkraftwerkskomponent en, Dezember 2007, ISSN 1439-7498;
- EIReDA 1998, European Industry Reliability Data Bank, Joint Research Centre of the European Commission, 1998;
- EGG-SSRE-8875, Generic component failure data base for light water and liquid sodium reactor PRAs, February 1990;
- European Utility Requirements, Volume 2: Generic Requirements, Chapter 17 PSA Methodology, November 1995;
- NUREG/CR-4780, "Procedures for Treating Common Cause Failures in Safety and Reliability Studies," U.S. Nuclear Regulatory Commission, Washington, DC, January 1988;
- NUREG/CR-5485, "Guidelines on Modelling Common Cause Failures in Probabilistic Risk Assessment," U.S. Nuclear Regulatory Commission, Washington, DC, November 1998.

For evolutionary components and components without available data (limited experience components), in general, generic data for similar components are considered with supporting reliability evaluations, supplier's information and expert judgment.

Modelling of new or evolutionary design features

In general, in the PSA for new reactors, the new /evolutionary design features contribute to the decrease of the core damage frequency. Some new initiating events were identified, mainly related to inadvertent actuation of the new automatic actions.

It appears that the modelling of the impact of new /evolutionary design features is not complete in most of the available PSA for new reactors, mainly due to lack of information regarding the design and due to lack of supporting studies. Some additional calculations and studies may be necessary in order to assure that new and evolutionary design features are adequately addressed. These may include, for example, studies demonstrating the appropriate use of conservatism in developing PSA success criteria, the use of bounding parameters for PSA supporting calculations and sensitivity studies, and testing programs to validate calculations. For example, in UK AP1000 PSA, In-Vessel Retention (IVR) mitigates the consequences of some of the low pressure sequences and therefore decreases the large release frequency. Reactor Pressure Vessel (RPV) depressurization and new passive features as Passive Core Cooling System or Passive Containment Cooling System have a major role to reduce the risk associated with Internal Events and Hazards. There are potential new initiating events associated with Reactor Pressure Vessel (RPV) depressurization or some features of the Passive Core Cooling System due to spurious actuation of some automatic actions or other failure modes. The same, in UK EPR PSA, the Corium stabilization system ("core catcher") mitigates the consequences of some of the low pressure sequences. Also the EPR includes an extension of the primary system depressurization to cover severe accident depressurization. Improvements on the redundancy and/or diversity of systems as feedwater systems, cooling systems, blackout diesel generators, etc. would contribute to reduce the risks. No potential new initiating events have been identified at this stage. However, the FMEAs supporting IE derivation may need to be completed when further design detail will be available and potential initiating events due to actuation of the new automatic actions and others should then be investigated.

While some types of simplifications in the PSA models may have been acceptable in the past for older reactors with higher core damage frequencies, they may not be justified for newer reactors, in particular to support use of PSA for decision making in design and operational matters:

- Application of human error probability estimation techniques to digital interfaces;
- Failure of passive components and structures now more important in advanced reactor designs. It is essential that passive components are adequately addressed in the PSA;
- Modelling of inter-system CCFs.

In general, the need for guidance for searching for failure mechanisms and scenarios (for situations involving new design features or entirely new designs, where operational experience is lacking) was highlighted by task participants.

Modelling of Technical Specifications and preventive maintenance

In general, because Tech Specs and preventive maintenance procedures are not available in the design phase, assumptions are made on preventive maintenance and on corrective maintenance durations. These assumptions are based mainly on the anticipated Tech Specs and on the industry experience. This approach is generally accepted for a design stage PSA. Nevertheless, if the preventive maintenance is foreseen during power operation, detailed maintenance information, mainly related to the configuration management, may be requested in order to ensure that the maintenance configuration risk is properly addressed in the PSA.

In parallel with assessment of the PSA, some safety authority, as for example NRC, evaluate the design stage Tech Specs to confirm that they will preserve the validity of the plant design by ensuring the plant will be operated with the required design conditions, and with operable equipment that is essential to prevent accidents and to mitigate the consequences of accidents. In some instances, detailed design information, equipment selection, allowable values, or other information are needed to establish the information to be included in the Tech Specs. These plant-specific values must be provided when a combined license application is submitted for a specific plant.

Human Reliability Analysis (HRA)

In general, the HRA methodologies for the PSA for new reactors are the same as for the existing reactors PSA. The methodologies are mainly based on generation 1 standards and guidance, like THERP (Technique for Human Error Rate Prediction) and ASEP (Accident Sequence Evaluation Program) described in NUREG/CR-1278 and NUREG/CR-4772, respectively. These methodologies are used to quantify the pre and post-accidental HRA. During the design phase detailed accident procedures and the use of a simulator to support HRA quantification typically are not available. The HRA is based on general information on the operator strategies for different accident scenarios. Additionally, it is expected that the HRA qualitative and quantitative analyses are used to help develop the detailed accident procedures and simulator training scenarios. However, it can be noted the lack of general agreement among participants on the use of plant simulators in HRA (e.g., whether stress is representative or even measurable).

In the future the employment of generation 2 HRA methods is foreseen, as well as the extensive using of the simulators. The availability of detailed information regarding the accident procedures and the severe accident management is seen as an essential aspect, by all new reactor projects actors, for the finalized PSA HRA.

Modelling of I&C

Digital I&C systems is the current design solution for the new reactors. Due to the many unique attributes of these systems, challenges exist in PSA modelling. The main issues are firstly the ability of the models to identify dependences due to I&C, in particular dependencies between an initiating event (due to a spurious signal) and failures of safety functions (in principle the fault tree modelling is a potential solution for this question). The second issue is the problem of data, which are still very difficult to find, especially for software and CCFs.

The digital I&C is not really specific issue to new plants, but due to the improved general safety level the role of I&C is increasing and becomes a potentially dominant issue.

Although there is no real methodology consensus for the digital I&C modelling and quantification, some tentative approaches are developed and integrated in PSAs. Several international working groups are currently dealing with this issue and a specific WGRisk task is also under development.

Common cause failure

The CCF modelling is the same as for the existing PSA. However, questions are raised regarding the applicability and treatment for long-running scenarios (presumably considering recovery), especially if the external events are considered in the PSA. Another important aspect is related to the investigation and modelling of the intersystem CCF, especially for high redundant systems.

3.2.4.2 Level 1 PSA, hazards

External hazards

The status of the development of the external hazards PSA is different for different new reactor projects. This is due to differences in the project development status, mainly if the site is known or not, and to the expected impact of the various external hazards on the future plant safety, which is country and site specific. In general, today only few hazards PSA are available for new reactors, mainly screening analyses being performed completed with other simplified methods allowing to approximating the contribution of the hazards to overall future plant risk. The future possible hazards evolution, induced by climate changing for example, are generally not explicitly taken into account in the analysis (however, the climate changing is sometimes considered in the hazard assessment and external hazard analyses typically to involve bounding assessments that are meant to demonstrate the design margin for these hazards). The combinations of hazards, as well the induced internal hazards, seem to have not been systematically taken into account in the performed assessments.

The status of the development of external hazards PSA for different new reactors projects or countries, as revealed by the survey is the following:

- AREVA:
  - OL3: A detailed external events screening analysis for all kind of potential external events was performed. The screening analysis is based on the SKI report "Guidance for External Events Analysis." Single and Multiple External Events are considered. External events which were not screened out were analyzed as initiating events in the PSA model;
  - TSN: A detailed external events screening analysis for all kind of potential external events was performed. The screening analysis is based on the SKI report "Guidance for External Events Analysis." Single and Multiple External Events are considered. External events which were not screened out were analyzed as initiating events in the PSA model. Seismic risk will be analyzed with a simplified approach;
  - US: A detailed external events screening analysis for all potential external events were performed, based on NUREG 0800.

- EDF:
  - FA3: A simplified PSA based seismic margins assessment has been used. It will be updated in the future with a more detailed seismic PSA. The extreme weather hazards have been screened out by simplified assessment or justification. Loss of ultimate heat sink is combined with loss of external power, for a specific long term PSA analysis.
  - UK: To be included in the PSA scope, an external hazard must be able to impact on plant structures, systems or components and degrade one or more plant safety functions, challenging plant safety systems that act to maintain or bring the plant to a safe state. The process adopted for the probabilistic analysis of external hazards involves the following steps:
    - a screening analysis of the initial external hazard list (which is as exhaustive as possible)
    - a probabilistic analysis of the 'screened in' external hazards

    The seismic margin of the UK EPR is assessed by a PSA-based SMA, following a methodology developed by the US NRC.
- ENEL: The seismic PSA should be used to evaluate the seismic events. Seismic hazard is the most important event that could affect the plant then other hazards are not considered in the preliminary phase.
- Switzerland (ENSI):
  - seismic: The Swiss licensees carried out a large-scale project "PEGASOS" – a German acronym for "Probabilistic Assessment of Seismic Hazards for Swiss Nuclear Power Plant Sites" – in response to a requirement that came out of the Inspectorate's PSA review process. In order to achieve a thorough quantification of the uncertainty of seismic-hazard estimates, the licensees conducted an extensive elicitation process involving technical experts, scientific institutions and engineering organisations from Europe and the USA. The project was conducted in full compliance with the Senior Seismic Hazard Analysis Committee (SSHAC) Level 4 methodology. In 2008, Swiss licensees initiated a follow-up project, the "PEGASOS Refinement Project" (PRP). The project takes advantage of the most recent findings in earth sciences and new geological and geophysical investigations at Swiss NPP sites. A particular objective is to reduce the uncertainty range of the PEGASOS results. In 2009 the scope of the PRP was extended to include the sites for the proposed new Swiss NPPs. The Inspectorate is following the study closely through a system of continuous peer reviews similar to that for PEGASOS. In the PRP full compliance with the SSHAC Level 4 methodology is maintained. The design of the new NPPs and their seismic PSA must be based on the results of this project;
  - extreme weather hazards: Guideline ENSI-A05 defines which external events have to be modelled in the PSA. This includes high winds and tornadoes. In addition, ENSI-A05 includes a list of hazards, which shall be considered (drought, high summer temperatures, ice cover lightning, low winter temperatures and snow (drift) …);
  - other external hazards: Guideline ENSI-A05 defines which external events have to be modelled in the PSA. This includes aircraft crash and external flooding. Additionally, ENSI-A05 includes a list of hazards, which have to be analysed. Methods and criteria for screening are also defined. For external flooding, in the applications for the general license (early site permissions), a two dimensional model was used to quantify extreme river flow rates and the corresponding water levels.
- France (IRSN):

  - seismic: any method can be used, if the application is capable to indicate, with a high degree of confidence, that the seismic contribution to core damage frequency is acceptable (low);
  - extreme weather hazards: WENRA statements are fully applicable for new reactors (“Additionally, external hazards such as severe weather conditions and seismic events shall be addressed in the PSA so that the overall risk of a plant is assessed realistically - This means that these two hazards shall be included in the PSA, except if a justification is provided for not including them, based on site-specific arguments on these hazards or on sufficient conservative coverage through deterministic analyses in the design, so that their omission from the PSA does not weaken the overall risk assessment of the plant.”);
  - other external hazards: A limited study for a long term common mode failure of the external grid and ultimate heat sink was done by EDF for FA3.
- MHI:
  - seismic: PRA based seismic margin method is used;
  - extreme weather hazards: Extreme weather hazards are site specific. The bounding design parameter requirements for meteorology are considered in the design certification application. The site specific meteorology hazards will be assessed in accordance with the requirements for screening and conservative analysis of other external hazards of ASME/ ANS RA-Sa-2009;
  - other external hazards: The other external hazards are site specific. The bounding design parameter requirements for other external hazards are considered in the design certification application. The nearby industrial, transportation and military facilities, hydrologic engineering, and geology, seismology and geotechnical engineering will be assessed in accordance with the requirements for screening and conservative analysis of other external hazards of ASME/ANS RA-Sa-2009.
- USA (NRC):
  - seismic: Seismic hazards are typically addressed by a seismic margins assessment during the design phase. A full seismic PSA is required prior to initial fuel load if a standard for seismic PRA has been endorsed by the NRC one year prior to the date of initial fuel load;
  - extreme weather hazards: High wind hazards are considered. Hazard frequencies are typically based on historical data for a representative site region. Frequently, PSAs that support design certification will treat the hazard in a bounding fashion so that it may apply to all applicants for a COL referencing that design certification;
  - other external hazards: Other hazards that may be considered include external flooding, transportation accidents, and nearby facility accident. Additional hazards may be considered if warranted by site specific considerations.
- Hungary (NUBIKI):
  - seismic: Existing regulatory requirements call for a seismic PSA. Indications are that short-term future modifications of the Nuclear Safety Codes will allow seismic margin assessment instead of a detailed seismic PSA;
  - extreme weather hazards and other external hazards: Safety regulation requires a PSA for all natural and man-made hazards that are relevant for the site. So the actual hazards to be considered must be selected on a plant specific basis. Bidding requirements will include a listing of hazards to be considered in PSA modelling details are not available yet.

- UK (ONR):
  - UK AP1000 PSA:
    - a Seismic Margins Analysis (SMA) has been submitted in GDA to address seismic risk;
    - a set of hazards listed up front was looked at, including, high winds, tornadoes, external floods and transportation and nearby facility accidents and have been screened out from the PSA submitted for GDA;
    - site-specific work on this area will be needed.
  - UK EPR PSA:
    - a Seismic Margins Analysis (SMA) has been submitted in GDA to address seismic risk;
    - for the GDA PSA, the majority of external events are screened out on deterministic or on probabilistic grounds. Frazil ice, solid or fluid impurities released into the water from a ship (e.g., oil spill) and the effect of organic material on the water intake water are included in the loss of ultimate heat sink (LUHS) initiating event and included in the Riskspectrum PSA model;
    - site-specific work on this area will be needed.
- Slovenia (SNSA): There are no detail requirements regarding external hazards. For existing plant:
  - for seismic events a detailed PSHA is prepared. The results are incorporated in the internal events PSA model;
  - extreme weather hazards are included in the other external events (together with external flooding, external fires, and aircraft crash hazard, etc). For the extreme weather most of the hazards are included in other initiating event frequencies (e.g. lightning is included in loss of off-site power and external fire, intense precipitation included in the external flooding, high temperatures, low river water included into drought, etc). For extreme winds (which also include hurricanes and tornadoes) a bounding study was performed;
  - external events considered are: aircraft impact, avalanche, biological events, coastal erosion, drought, external flooding, extreme winds, tornadoes and hurricanes, external fires. Fog, frost, hail, hazardous materials, high tide, high river stage, high summer temperatures, ice cover, industrial and military facility accidents, internal fires, landslide, lightning, low river water level, low winter temperatures, meteorite, pipeline accident, intense precipitation, release of chemical in on-site storage, river diversion, sandstorm, seiche (sloshing of water in an enclosed body of water, which can result in tsunami-type waves), snow, seismic activity, storm surge, transportation accidents, tsunami, toxic gas, turbine-generated missile, volcanic activity, waves;
  - none of the external events, except seismic activity, internal fires and floods, are modelled in the PSA model, but are added to the final PSA results based on performed PSA analysis.
- Finland (STUK): The Finnish Regulatory Guide YVL 2.8 requires Seismic PRA to be performed. For the OL3 see answer form AREVA.
- UNISTAR:
  - seismic: PRA-based seismic margins assessment;

- extreme weather hazards: Analysis of High winds and tornadoes for screening them out;
- other external hazards: Analysis of Airplane crash and industrial transportation for screening it out.

In most of the mentioned studies, the site specific aspects are addressed in general by bounding assumptions, since for many projects the site was not known.

Internal hazards

Typically the internal fire and the internal flooding are addressed in detail for new reactors, and in many cases PSAs are being developed.

Internal fire hazards for new designs are typically assessed using the Fire Induced Vulnerability Evaluation (FIVE) methodology developed by EPRI. The methodology for fire PSA is based mainly on the NUREG/CR-6850 – "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities - EPRI TR-1011989" and national standards (like for example in Switzerland ENSI-A05 - "Probabilistic Safety Analysis (PSA): Quality and Scope" which describe the acceptable methods for all internal hazards PSA).

The internal flooding analysis typically includes a systematic approach to identify potential internal flooding sources and their impacts on the plant.

Because the as-built configuration cannot be assessed until construction is complete, the studies are in general based on assumptions regarding the plant layout and they may be updated when walk-downs will be performed after the plant is built.

The other internal hazards (explosions, high energy component breaks, heavy load drops, etc.) are seldom addressed in the PSA for new reactors (for example FA3 and EPR-UK projects), mainly due to lack of detailed information (however, design margins are provided to cope with these risks). This will be reconsidered when detailed information will be available.

3.2.4.3 Level 2 PSA

The Level 2 PSA is available for each new reactor project, allowing estimation of the source term occurrence frequencies of the new design. The Level 2 PSA is generally integrated with the level 1 PSA, allowing the treatment of dependencies among the PSA model, for the systems and for the human factor. The internal fire and the internal flooding which are included in the Level 1 are in general addressed equally in the Level 2 PSA. The external hazards are rarely included in the Level 2 PSA, as they are rarely included in the Level 1 PSA. It appears also that the spent fuel pool is not always included in the Level 2 PSA.

It has to be noted that during the design in some cases a Level 1+ PSA, which may include assessment of containment phenomena and severe accident mitigation measures, was developed at the beginning of the design (like EPR for example). This type of PSA model was demonstrated to be a valuable tool to identify design improvements.

The Level 2 phenomenology modelling is in general supported by specific supporting studies, which are performed with severe accident codes. The most used codes are MAAP 4, MELCOR, ASTEC, complemented with specific phenomena codes, like GOTHIC (hydrogen burn), TEXAS-V (steam explosion), LS-DYNA (containment structure), FLOW-3D (molten core spreading behaviour), MC3D (DCH), TONUS (hydrogen distribution).

In general, not many experimental studies were performed for the new reactor designs. An example of experiments sponsored by NRC for the AP1000 reactor design is described in the report NUREG-1826 "APEX-AP1000 Confirmatory Testing to Support AP1000 Design Certification."

New phenomena may be considered if they are deemed appropriate. For example for US-APWR, the rocket mode reactor vessel failure is newly added as part of the high pressure melt ejection in the model. The US-APWR eliminates the ICIS penetrations at lower plenum so that this new energetic phenomenon is additionally considered.

The new or advanced severe accident features, which are provided by all the new designs, are in general taken into account in the Level 2 PSA. These features are for example, for EPR: core melt stabilization system, severe accident RCS depressurization, hydrogen recombiners and severe accident containment heat removal system; for AP1000: In Vessel Retention (IVR), hydrogen igniters and Passive Containment Cooling (PCS). The modelling of the advanced severe accident features shows, in general, an improved safety comparing with existing reactors.

Severe accident codes were improved in order to be capable to model new complex phenomena, like for example the MAAP5 was enhanced with new models and improvements to address complex phenomena important for IVR. The key parameters affecting the IVR success are the metal layer emissivity and thickness of the top metal layer, which depends on the amount of steel in the oxidic pool and in the heavy metal layer. Equally, the MAAP5 was improved to simulate transients at shutdown conditions (including low reactor inventory states, open reactor and open containment states, refuelling operations, and spent fuel pool accidents, as well as the long term non-severe accident transients).

In general the Level 2 PSAs will be updated as the designs will evolve, in order to include the detailed design information, and all Level 1 PSAs addressed internal and external hazards. The other radioactive sources (e.g., spent fuel pool) will be also included in the future Level 2 PSA.

3.2.4.3 Level 3 PSA

Several Level 3 evaluations were performed in the frame of new reactors projects, like for example:

- AREVA performed a Level 3 PRA, using the MACCS2 code, for each site where a COLA application for the US EPR™ is proposed, even though a Level 3 PRA was not a part of the Design Certification application. This level 3 PRA uses the results directly from the Level 2 PRA, combined with plant specific information to provide estimates of early fatalities, early and latent cancers, population doses, and economic impact;
- For FA3, EDF used the source term results to calculate the effective dose to a hypothetical individual that remains at a fixed location 500 m downwind of the reactor for a period of seven days after an accident. This was performed with the computer codes COSAQUE, CORRA and ASTRAL;
- MHI performed a Level 3 PSA using the MACCS2 code. The fission product source terms for the release categories (six release categories were defined for the US-APWR Level 2 PSA) were evaluated by using the MAAP 4.0.6 code. These release category conditions include the fission product release fractions, release height, release energy, and release duration;
- For UK AP1000 PSA:
    - Limited scope Level 3 PSA analysis which uses the Level 2 PSA results, frequency and Source Term for each RC each with source terms characterized by environmental release rates and timing, together with isotopic content, as input. The Level 3 PRA code used is MACCS2 (MELCOR Accident Consequence Code System);
    - The Level 3 as the Level 2 PSA does not cover all sources of radioactivity (only the reactor core is included; fuel ponds, fuel handling facilities, waste storage tanks, etc, are not included). Low consequence sequences (Level 1 non-core damage sequences) are not included in the scope meaning that their radiological risk contribution is not taken into account;

    - As in the Level 2 PSA, fires, floods and external hazards are not included.
- For UK EPR PSA:
    - The consequences analysis considers at-power conditions, shutdown plant states, internal fire and flood, external hazards as treated in Level 1 and accidents occurring in the Spent Fuel Pool.
- In the frame of other projects, the use of Level 3 PSA is not foreseen in some countries, since is not required by the regulation, for example in Finland, China, Switzerland and Slovenia.

In general, the level 3 assessments considers only the internal events and only the reactor, with some exceptions where the full range of initiators which are considered in the Level 1 PSA are propagated up to Level 3 (APWR in US, EPR in UK, EPR in US).

The site specific aspects are treated in general by specific information, like for example:

- For EPR US, plant specific information was used for the COLA sites: population numbers and distribution, year round meteorological data, Agricultural information – crops, growing season, location of cultivation, economic data – farmland and non-farmland property value. To assess human health impacts, the analysis determined the expected number of early fatalities, expected number of latent cancer fatalities, and collective whole body dose from a severe accident to the year 2050 population within a 50-mile radius of the plant. Economic costs were also determined, including the costs associated with short-term relocation of people, decontamination of property and equipment, and interdiction of food supplies;
- For EPR FA3 bounding assumptions were used (e.g. standard weather condition "DF2");
- For APWR US (MJI) in the analysis for the standard design, the meteorological data of the Surry site have been used as the "typical", which is accessible in the MACCS2 code sample input file attached to the MACCS2 code. The population data of the Surry site in the MACCS2 code sample input file has been adjusted to be representative of about 80% of the U.S. nuclear plant sites in NUREG/CR-2239 "Technical Guidance for Siting Criteria Development";
- For UK AP1000 PSA:
    - These analyses consider 'generic' assumptions about weather and are conducted to estimate the whole-body dose and acute red bone marrow dose, both at the site boundary (0.5 miles). The population whole-body dose out to 80.5 kilometres and the downwind, centerline, ground-level thyroid dose at the site boundary (0.5 miles) are also calculated for information.
    - The analysis carried out is biased towards US methodology and regulatory requirements. Consequently, inferences have to be made when comparing the assessment results with SAP targets as direct comparisons are not possible. Compliance is claimed with Numerical Targets 5 to 9 of the SAPs (corresponding to risk from accidents) on this basis without additional analysis.
- For UK EPR PSA:
    - Consequences are calculated in terms of long term doses for comparison with Target 8. Results do not include the specific calculation of early or late health effects based on organ doses. Based only on these doses an estimate of risk of death is made for comparison with Target 7 and a screening method is adopted to identify releases that are likely to result in significant off-site consequences for comparison with Target 9.

The emergency actions are considered in general by bounding assumptions, like for example:

- For the EPR US the study considers the actions dictated by the Emergency Operating Procedures and Severe Accident Management. Guidelines are included in the Level 1 and Level 2 PRA. The Level 3 PRA includes the sheltering and/or evacuation of populations affected by the plume(s), depending on the severity of the release;
- For FA3 the consequence analysis doesn't credit any emergency actions;
- For APWR US (MHI) no emergency actions are considered in Level 3 PRA for the design certification application. In the analysis for the standard design, Level 3 PRA is calculated for the severe accident mitigation design alternative (SAMDA) calculation. The application of off-site emergency response plan and the evacuation are not assumed, and therefore the population dose risk is conservatively evaluated;
- For UK AP1000 PSA:
    - o The site boundary study doesn't take into account emergency actions.
    - o A site specific L3 PSA is expected in due course.
- For UK EPR PSA:
    - o A site specific L3 PSA is expected in due course.

For the finalized Level 3 assessments, the regulatory require in general that all relevant site specific historical and data must be taken into account with state-of-the-art consequence assessment codes. The emergency actions need to be considered for a credible consequence analysis and sensitivity studies. This assumes the availability of developed emergency plans.

4. CONCLUSIONS

There is a general consensus concerning the usefulness of the PSA in ensuring improved safety of new and advanced reactors. The two tasks allowed identifying the current related PSA practices, as well as further issues to be addressed in order to improve the representativeness of the PSA for new and advanced reactors, in a context of increased usage of integrated decision making approaches. Several recommendations were formulated, mainly referring to the enhancement of the international collaborations in key areas related to development and application of the PSA for new and advanced reactors.

Most of the conclusions are similar for the two types or reactors, e.g. new/evolutionary reactors and advanced reactors. However, some of the highlighted issues are specific, due to different development stage (finalized/stable design for the new reactors and basically conceptual design for the advanced reactors).

Regarding the advanced reactors, the PSA is being recognized as one of the key approaches to justify safety-critical aspects in the conceptual and preliminary design stage and to address new operation concepts. However, there still remain some issues related to PSA in order to better reflect the advanced reactors specific features. Most of the identified PSA issues are well recognized. Also, most of the issues are relevant to all reactors in the design stage (not just advanced reactors). However, there naturally are greater difficulties in addressing these issues when the reactor is in the conceptual design stage (and detailed design information has not yet been developed). It can be noted also that there is an increased emphasis on the notion of "PSA quality" as dictated by PSA standards and associated guidance documents. Other infrastructure challenges raised include the lack of "peers" (with experience in PSAs for specific advanced reactor designs) to perform PSA peer reviews and the need for regulators to understand the advanced reactor PSA models. Although the task was not designed to achieve consensus, there was considerable agreement as to the relevance of the issues identified. The task provided a list of topics which can be pursued by WGRISK or others. The topics would have to be prioritized based on organizational as well as technical considerations, as this was not done by the task.

For the new reactors, the PSA plays a major and increasing role, in the frame of design, construction and licensing, as complement of traditional deterministic methods. PSA is used by the industry at all stages of the design for a wide variety of applications, including demonstration of safety level, supporting "balanced design", balancing between accident prevention and mitigation features of the design, identification of design vulnerabilities and improvements, comparison with the risk of existing plants, and establishment of requirements for systems/sub-systems. Regulatory agencies are using PSAs mainly to support auditing applicant analyses and to identify risk-significant areas for safety reviews. The level of formalisation of the development and the role of PSA for new reactors is different in different countries, but generally for new reactors, the role of PSA in the regulation is more important and more systematized comparing with the existing reactors. In support of licensing member countries have diverse requirements for PSA, ranging from no requirements to "full scope Level 3." Some require PSA at specific stages in the licensing process; at least one country requires a Living PSA commencing with design and continuing through operation.

Some regulatory agencies developed requirements for PSAs and applications for new reactors. All countries recognized that the scope of PSA for new reactors should cover, generally, a wide spectrum of initiating events, including the internal and external hazards, in all reactor states and all radioactive sources (reactor and spent fuel pool). However, if the development status for internal initiating events and hazards (typically the internal fire and the internal flooding) is similar for different new reactors projects, the status of the development of the external hazards PSA is various, mainly due to differences in the project development status (site unknown, expected importance of various external hazards, etc). Also, the spent fuel pool is usually not included in the PSA, especially during the initial phases of the design. The Level 2 PSA is generally available or requested by the safety authorities for each new reactor project. The Level 3 is not, with some exception, developed or requested by the safety authorities for the new reactors. However, despite data, modelling and code limitations, Level 3 PSA was identified as a necessary support for some new reactor applications (e.g. definition of the emergency zones). Additionally, compared to the existing and new reactors, another challenge for the PSA for advanced reactors is the PSA framework, including risk metrics and the scope of PSA.

Most related activities which are today in progress do not seem to be aimed for a specific reactor types and consist in an extension of the conventional PSA framework to the new and advanced reactors.

The main challenges to use the PSA for new reactors are related to intrinsic difficulties to ensure the representativeness and the quality of the model for a reactor which is in a preoperational phase. It appears that the exchange of lessons learned on ongoing new reactor project PSAs may greatly contribute to develop improved best-practices and guidelines for the development and using of the PSA by all actors involved in the frame of new reactor projects (industry and safety authority), mainly during the different design stages. Participants recognize the need to better integrate PSA into the design and safety review process; interaction between the PSA and design teams is important and needs to be strengthened. The challenge of appropriately using PSA results as the PSA is still evolving (to match the increasing detail of the design) is especially interesting and is worth follow-on discussion.

Regarding the enhancement of the PSA quality, the independent verification is one of the identified ways. It is also recognized that is more difficult to ensure PSA quality for the newer designs because of a lack of peers, limited scope PSA (mainly at the beginning of design), and challenges in ensuring strong interaction between design and PSA teams as the design evolves. Typically, vendors have some generic design phase PSA or a reference plant PSA, but in the construction license phase it is hard to get a site-specific, full scope (covering Level 1 and 2 PSAs, all IEs, and all operational modes). Regulator is often in a position to make the decision on a PSA which does not fulfil all requirements. The need for decision makers to consider uncertainties associated with analyses of plants in the design stage is well recognized. It is further recognized that such uncertainties could be much larger than those for operating plant analyses. Suggestions include: providing decision makers with additional information from additional analyses (e.g., margin demonstration analyses, bounding calculations, sensitivity studies), performing focused research on selected topics, development of appropriate safety goals/targets, development of appropriate PSA standards and development of appropriate peer review approaches.

The technical challenges of the PSA for more advanced reactors, which are in research stage or in the early phases of conceptual design, in addition to the above-mentioned aspects, also include the potential need to address very different systems and phenomenology, unavailability of reliability and experimental data, unavailability of knowledge on new phenomena, unavailability of accident analysis models, etc..

Generally, the PSA community is comfortable with existing PSA technology (i.e. event tree/fault tree formulation). However, in the frame of advanced reactors, the use of the non-ET/FT methods/tools is being explored as a mean to more explicitly tie advanced reactors-specific phenomenological modelling into the PSA framework, as well as some severe accident models/tools to support the risk assessment of advanced non-LWR reactors. In general, the need for advanced PSA methods (e.g., dynamic PSA methods) to address the risk associated with advanced reactor designs was not identified as a priority issue. Many organizations are performing design-support PSAs using conventional methods and tools. However, some countries see potential benefits associated with such methods and are pursuing related, limited-scope research and development activities. In conclusion, the research on advanced methods seems not to be a priority for the tasks participants. As a related topic, a number of countries and organizations recognize the need to address the results of deterministic and probabilistic models in integrated decision making. Although not an issue unique to advanced reactors, it may be affected by the advanced reactor context, including the lack of detailed design information, imprecise understanding of potential accident phenomenology, and a lack of operational experience.

For both new and advanced reactors, there are a number of well-recognized challenges (e.g., lack of detailed design information and operating experience data, passive system reliability estimation, digital I&C, HRA, phenomenological analysis models and tools, and risk metrics). These issues are not unique to new or advanced reactors, but are more important as overall risk estimates decrease. Moreover, the new and advanced reactors can have some new/unique features like: new evolutionary components, new severe accident features, evolutionary human-machine interface, high redundancy of plant systems (treatment of inter-systems CCF), which may need more “specialised” PSA modelling techniques. Previously accepted modelling simplifications may not be anymore appropriate for the new reactor PSAs, as it’s recognized that new applications (e.g., identification of LBE and SSC classification) put more burdens on the PSA, for example to treat “all” engineering features. As for example, the applicability of PSA for addressing the safety-security interface is an interesting topic which can raise other challenges for the development and using of the PSA. Given new/unique design features and the associated lack of operational experience, the development of improved methods and tools to support analysts’ searches for potential failure scenarios could be useful.

Notwithstanding the PSA community’s general comfort with existing methods and tools, there remains interest in the question as to under what conditions the current state of the art may not be well suited for new designs.

Another challenge is due to the fact that the modelling of external events in the PSA for new reactors and advanced reactors is generally not exhaustive and then it is very difficult to be used to support the decision making related to the plant protection against this type of hazards. This aspect is not specific to new or advanced reactors; the improvements in this PSA field can contribute, from the design stage, to ensure a high level of plant safety by enhancing the using of PSA in the decision-making process. In fact, the PSA should better take into account the beyond design basis external events, the combinations of external events, the induced internal hazards and the impact on the whole site (reactors and spent fuel pools). The PSA methodology may need some enhancements in order to realistically model the long term accident sequences as well as the role of the crisis organisation and exceptional means (before and after core damage).

In order to deal with the above-mentioned aspects, regulatory and industry organizations in some countries are supporting the development of standards to apply PSA, mainly for the new reactors. Regulatory PSA models are developed in some countries and are used for a confirmatory check of an applicant’s model. Regarding the advanced reactors, the participants expressed need of development of specific PSA guidance and standards.

In general, the task participants expressed the need for better advice, based mainly on the international exchange of lessons learned, on how to use PSA during the different design and preoperational phases of the new and advanced reactors. The main fields of interest were identified by the questionnaire answers (see chapters 2.3.2.2 and 3.2.3 of this report) and discussed during the common workshop [1].

Periodic survey on further activities among member countries will be helpful in finding and clarifying further issues related to the PSA of new and advanced reactors;

Regarding high-priority technical issues related to the PSA of new and advanced reactors, pilot study and international collaboration will provide more insights into the underlying issues. Such kind of technical issues and common areas of interest among member countries were raised and discussed in the both tasks.

As the two task concentrated mostly on the difference between existing/new reactors and on the specific aspects for new designs and less on PSA applications specifics for different stages of new reactors (conceptual design, detailed design, construction, commissioning), a future survey on these aspects can be useful. Moreover, as GDA (generic design assessments) are undergoing and will be also done in the future in some countries (like USA and UK) where the survey and the comparing of practices and results can be highly valuable.

Finally, it should be noted that the questionnaires were developed and answered before Fukushima accident and the common workshop was held shortly after. Some discussion during the workshop referred to this event, especially regarding the external events PSA aspects, but the Fukushima issues are not specifically treated in this report. However, based on the current understanding, none of this report's conclusions or recommendations are contradicted by the worldwide ongoing post-Fukushima analyses.

5. RECOMMENDATIONS

As already mentioned most of the new or advanced reactors PSA issues are not new and are not specific for these types of reactors, but as is it expected that the future plants have a substantial improvement of the safety level, the role of PSA, as a systematic risk analysis, becomes more important. As a consequence, the already recognized PSA key uncertainties areas become more and more critical in the frame of decision making. It has to be mentioned that some of the mentioned subjects are already treated by other national and international working groups.

The recommendations of the two tasks are the followings:

- To promote the international exchange of lessons learned in using the PSA for new reactors during the preoperational phases by all actors involved in new reactor projects.

  The challenge of appropriately using PSA results as the PSA is still evolving is especially interesting. It would also be interesting to evaluate the practice of making assumptions on absent detailed design information. The identification of failure mechanisms and scenarios, for situations involving new design features or entirely new designs, is also an important issue.

  These aspects are also covered by several international working groups (MDEP, EPR Family Group, WANO, WENRA, CANDU Senior Regulator Groups, etc.). However, as these working groups treat, in general, separately the regulatory perspective from the industry perspective and are sometimes dedicated to one type new reactor, the promotion of a larger collaboration framework may be of great benefice;

- To promote the information exchange with ongoing international advanced reactors activities (e.g., GIF-RSWG and IAEA CRP on passive system reliability);

- To promote the international collaboration on the PSA methods which need to be enhanced in order to model the unique features of new reactors:

  - digital/software based I&C,
  - HRA, taking into account modern human machine interface,
  - CCF of high redundancy systems and intersystem CCF,
  - modelling of passive systems.

  Some of these fields are covered by some other working groups, as the importance of these issues is widely recognized. However the main objective of these working groups is not always the PSA development. The international collaboration may lead to increase the confidence in the PSA models and results.

  As in many countries developments in the mentioned areas (i.e. digital/software based I&C, HRA taking into account modern human machine interface, CCF of high redundancy systems and intersystem CCF, modelling of passive systems) are ongoing is it recommended that, in the frame of yearly countries report on use of development of PSA, a special section be dedicated for the description of the updated status of development of the methodologies and of the obtained results.

- To promote international cooperation in the frame of the PSA for advanced reactors, mainly regarding the following specific aspects:
    - definition of risk metrics,
    - modelling of specific phenomena for different advanced reactor types,
    - assessment of potential severe accidents at the pre-conceptual design phase, especially for non-LWR types of reactor,
    - assessment of the applicability of TNF and ISAM,
    - development of PSA standards.
- To promote the international collaboration on the development of external hazards PSA;

  This aspect is not specific to new reactors, but due to the important safety improvements relating mainly to internal events, external hazards contribution can become dominant. This subject was already treated by several WGRisk tasks. However, the recent events showed that the PSA scope should be widely increased in order to extensively include the beyond design basis external events, the combinations of external events, the induced internal hazards as well as the impact on the whole site (reactors and spent fuel pools). This extension of the PSA scope may need some enhancements in order to realistically model the long term accident sequences as well as the role of the crisis organisation and exceptional means (before and after core damage). The international collaboration on this field may be highly beneficial;
- To promote information exchange with ongoing international advanced reactors activities (e.g., GIF-RSWG and IAEA CRP on passive system reliability);
- Develop, as a future activity, additional guidance on the application of PSA for the design of new reactors that provide recommended practices for such matters as identifying potential failure scenarios and for determining the appropriate level of analysis. Such guidance could use the higher-level ASME PSA standard for non-light water reactors currently being developed as a starting point.

REFERENCES

[1] Proceedings of OECD/NEA Workshop on PSA for New and Advanced Reactors, Organized by OECD/NEA WGRisk, OECD Conference Centre, Paris, France, June 20-24, 2011, NEA/CSNI/R(2012)2

[2] M. Drouin, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing," NUREG-1860, US Nuclear Regulatory Commission, December 2007.

[3] RSWG, "An Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems," GIF/RSWG/2010/002/Rev 1, June 2011.

APPENDIX 1: GLOSSARY

ABWR	Advanced Boiling Water Reactor
ADAMS	Agency-wide Document Access and Management System (US NRC website)
AESJ	Atomic Energy Society in Japan
ALARP	As Low As Reasonably Practicable
AOO	Anticipated Operational Occurrences
AOT	Allowed Outage Time
AP1000	Advanced PWR 1000 MWe (U.S. Type)
APR	Advanced Power Reactor (Korean Type)
APWR	Advanced Pressurized Water Reactor
ASEP	Accident Sequence Evaluation Program
ASME/ANS	American Standard for Mechanical Engineering/American Nuclear Society
ASTEC	Accident Source Term Evaluation Code
ASTRAL	Code to calculate the Effective Dose
ATWS	Anticipated Transients Without Scram
BDBA (BDBE)	Beyond Design Basis Accidents (Events)
CATHARE	Code for Analysis of Thermal-hydraulics during an Accident of Reactor and safety Evaluation
CCF	Common Cause Failure
CDF	Core Damage Frequency
CEA	French Alternative Energies and Atomic Energy Commission.
CFD	Computational Fluid Dynamics
CNRA	NEA Committee on Nuclear Regulatory Activities
COLA	Construction and Operation License Application
CORRA	Code to calculate the Effective Dose
COSAQUE	Code to calculate the Effective Dose
CRP	Coordinated Research Program
CSNI	NEA Committee on the Safety of Nuclear Installations

DB	Database
DBA (DBE)	Design Basis Accidents (Events)
DC COL	Design Certificate Combine Operating License
DDET	Discrete Dynamic Event Tree
DI&C	Digital Instruments and Control (System)
ENEA	Italian National Agency for New Technologies, Energy and Sustainable Economic Development
EOP/EOM	Emergency Operation Procedure
EPR	European Power Reactor
ESBWR	Economic Simplified Boiling-Water Reactor
ET/FT	Event Tree/Fault Tree
FANC	Federal Agency for Nuclear Control in Belgium
FBR	Fast Breeder Reactor
FIVE	Fire Induced Vulnerability Evaluation
FLICA	Computer Code for 3D and 2-Phase Flow T/H Analysis
FLOW-3D	Computational Fluid Dynamic Simulation Code
FMEA	Failure Modes and Effect Analysis
FP	Fission Products
GDA	General Design Assessment (UK)
Gen-III	Generation III
Gen-IV	Generation IV
GFR	Gas-Cooled Fast Reactor
GIF	Generation IV International Forum
GOTHIC	Generation of Thermal Hydraulic Information of Containments
HMI	Human-Machine Interface
HRA	Human Reliability Assessment (Analysis)
HTGR	High Temperature Gas Cooled Reactor
IAEA	International Atomic Energy Agency
IE	Initiating Event
IEEE	Institute of Electrical and Electronic Engineers
INET	Institute of Nuclear and New Energy Technology
iPWR	Integral Pressurized Water Reactor (U.S. Type)
IRSN	Institute for Radiological Protection and Nuclear Safety (France)
ISAM	Integrated Safety Assessment Methodology

ITAAC	Inspections, Tests, Analyses and Acceptance Criteria
IVR	In-Vessel Retention (nuclear power plant)
JAEA	Japan Atomic Energy Agency
JNES	Japan Nuclear Energy Safety Organization
JSFR	Japanese SFR
KAERI	Korea Atomic Energy Research Institute
KINS	Korea Institute of Nuclear Safety
LBE	Licensing Basis Event
LERF	Large Early Release Frequency
LFR	Liquid Metal Fast Reactor
LMFBR	Liquid Metal Fast Breeder Reactor
LOFT4AP	Computer Code for the Analysis of Non-LOCA and ATWS (Anticipated Transient Without Scram)
LOFTRAN	Westinghouse Code for the TH Analysis of Nuclear Steam Supply System
LS-DYNA	ANSYS Computer Code to Predict the Failure of Concrete Structures
LUHS	Loss of Ultimate Heat Sink
LWR	Light Water Reactor
MAAP4	Modular Accident Analysis Program version 4
MACCS2	MELCOR Accident Consequence Code System version 2
MANTA	Code for T/H Analysis of the Primary and Secondary Side Transients
MC3D	Monte-Carlo 3D Radiative Transfer Code
MDEP	(NEA) Multinational Design Evaluation Program
MELCOR	Methods for Estimation of Leakages and Consequences of Releases
MLD	Master Logic Diagram
MOSAIQUE	Module for SAmpling Input and QUantification Estimator
MSPI	Mitigating Systems Performance Indicators
MUTRAN	Severe Accident Analysis Code for JSFR (developed by JAEA)
NEA	Nuclear Energy Agency
NGNP	New Generation Nuclear Project: a USA development project
NISA	Nuclear and Industrial Safety Agency (Japan)
NPP(s)	Nuclear Power Plant(s)
OECD	Organisation for Economic Co-operation and Development
ONR	Office for Nuclear Regulation
PBMR	Pebble Bed Modular Reactors

PCS	Passive Containment Cooling System
PCSR	Pre-Construction Safety Report
PRP	PEGASOS Refinement Project
PSA/PRA	Probabilistic Safety (Risk) Assessment (Analysis)
PWR/BWR	Pressurized Water Reactor/Boiling Water Reactor
Q/A	Questionnaire/Answer
QA	Quality Assurance
R&D	Research and Development
RAP	Reliability Assurance Program
RCS	Reactor Coolant System
RGA	Risk Gap Analysis
RIDM	Risk Informed Decision Making
RMPS	Reliability Methods for Passive Safety systems
RPV	Reactor Pressure Vessel
RSWG	Risk and Safety Working Group
RTNSS	Regulatory Treatment of Non-Safety Systems
SA	Severe Accident
SAMDA	Severe Accident Mitigation Design Alternative
SAMG	Severe Accidents Mitigation Guidance
SDP	Significance Determination Process
SFR	Sodium Cooled Fast Reactor
SM2A	Safety Margin Analysis Phase 2 (International Collaboration Project)
SMA	Safety Margin Analysis (International Collaboration Project)
SMART	System- integrated Modular Advanced ReacTor (Korean Type)
SMR	Small Side Modular Reactors
SNSA	Slovenian Nuclear Safety Administration
SSC	Systems, Structure and Components
SSHAC	Senior Seismic Hazard Analysis Committee
STUK	Radiation and Nuclear Safety Authority of Finland
SUJB	State Office for Nuclear Safety (Czech Republic)
T/H	Thermal Hydraulic
TEXAS-V	A Steam Explosion Analysis Code ver. 5
THERP	Technique for Human Error Rate Prediction
TNF	Technology-Neutral Framework

TONUS	CFD Code for Hydrogen Risk Analysis
TS	Technical Specifications
TRAC	Transient Reactor Analysis Code
UJD	Nuclear Regulatory Authority of Slovakia
USNRC	United States Nuclear Regulatory Commission
VHTR	Very High Temperature Reactor
VTT	Technical Research Center of Finland
WANO	World Association of Nuclear Operators
WCOBRA	Code for T/H Analysis of Rod Bundle Nuclear Fuels and Cores
WEC	Westinghouse Electric Company
WENRA	Western European Nuclear Regulator's Association
WGRisk	Working Group on Risk Assessment
WGRNR	Working Group on the Regulation of New Reactors

APPENDIX 2: QUESTIONNAIRE ON PSA FOR ADVANCED REACTORS

1 Introduction

1.1 Background

In June 2008, a WGRISK-proposed task on advanced reactor PSA was approved by CSNI. (The CSNI Activity Proposal Sheet – CAPS – for the task is available on the WGRISK website.) The approved task involves two sub-tasks: 1) a survey of participating countries regarding the state of PSA technology for advanced reactors, and 2) an international workshop for detailed, follow-up discussions related to the topic. This questionnaire addresses the need of the first sub-task.

1.2 Objectives of the CAPS

The objectives of the "CAPS on PSA for advanced reactors" are (1) to characterize the ability of current PSA technology to address key questions regarding the development and licensing of advanced reactor designs, (2) to characterize the potential value of advanced PSA methods and tools and (3) to develop recommendations to CSNI for any needed developments.

1.3 Objectives of this questionnaire

The purpose of the questionnaire is to elicit the respondent's viewpoints on a number of topics relevant to the objectives of the CAPS. The answers to the questionnaire, possibly combined with the results of follow-up discussions with respondents, will provide the basis for an initial draft report presenting the international status and insights of PSA technologies for advanced reactors. (This report will be finalized following an April, 2010 workshop.)

1.4 Definition of advanced reactors

Based on discussions at a task group meeting held on 27 March, 2009, there is a range of opinions as to the definition of "advanced reactors." Three possible groups are:

(1) LWR with advanced safety features (e.g., ABWR, APWR, AP1000, EPR, ESBWR, etc)

(2) Modular types New LWR (e.g., NUSCALE, SMART, etc)

(3) Non-LWR types (e.g., Liquid Cooled Reactors (SFR), Gas-cooled reactors (GTMHR, PBMR, VHTRR, etc))

Because the purpose of the survey sub-task is to document the participants' viewpoints, the issue of advanced reactor definition is raised as a question in this questionnaire (see Question A.2 below)

2 Questionnaire

The questionnaire consists of four parts: (1) general questions, (2) questions on general technical issues, (3) questions on related research activities, and (4) questions regarding international cooperation. To help respondents answer each question, an appendix is provided. This appendix provides a listing of potentially important topics (tied to the CAPS objectives) and a breakdown of relevant PSA technical topic areas.

IMPORTANT: In answering the following questions, please keep in mind that your answers will provide the basis for a report addressing the CAPS' objectives listed in Section I.B above. Please provide any information you think appropriate and useful for the report authors.

2.1 General questions

(1) What is the current status of the development of advanced reactors in your country?

(2) What types of reactors are considered for this questionnaire?

(3) Preparation of regulator for advanced reactors in your country

- Does your country have ongoing research programs related to the development of regulatory or licensing guidance of advanced reactors? This does not include research activities related to the development of specific methods, tools, and data intended to support advanced reactor PSAs.
- Some countries have developed new, risk-informed regulatory approaches that may be applicable to the licensing and regulation of advanced reactors. (NUREG-1860 prepared by the U.S. NRC, describes a Technology-Neutral Framework for regulation). In your country, what is the current role of PSA in the development, licensing, and regulation of advanced reactors? Are there any plans for near-term or long-term changes in this role? Please discuss.

(4) The current status of advanced reactors' PSA

There are many topics of interest under the broad heading of advanced reactor PSA. In order to help identify topics warranting detailed discussion at the April, 2010 workshop, this questionnaire is limited to two broad issues. The first involves the scope of advanced reactor PSAs, which, in turn, involves hazard types (e.g., internal and external events including fire, flooding, seismic, etc) and operating states (e.g., full power, low power and shutdown). The other involves the PSA framework for advanced reactors. This latter category covers such questions as the applicability of conventional nuclear power plant PSA frameworks (involving the distinction between Level 1, Level 2 and Level 3 PSA), the definition of risk metrics such as plant damage states (including core damage state), and appropriate methods, models, and tools for the analysis (including source term release categorization and consequence analyses).

- Which hazard types are currently being treated in your country's PSAs for advanced reactors?[2] Are there any hazard types that are explicitly excluded? Do these answers depend on whether

[2] In this and following questions, "your country's advanced reactor PSAs" refers both to PSAs performed for advanced reactors (built or planned) in your country, and to PSAs performed in your country for advanced reactors that may be built in other countries.

the site includes currently operating plants or not? For future (near-term and long-term) PSAs for advanced reactors, are there plans to extend the hazard types considered? If so, please describe.

- Which operating modes (full power, low power and shutdown) are currently being treated in your country's PSAs for advanced reactors? For future (near-term and long-term) PSAs for advanced reactors, are there plans to enlarge the implementation scope? If so, please describe.
- One point of controversy for PSAs for advanced reactors (especially non-LWRs) concerns the applicability of the three-stage (Level 1, Level 2, and Level 3) PSA framework currently used for LWRs. This issue is closely related to the definition of the risk metrics or the end states of accident sequences. Does your country have an explicit (e.g., documented standard or guidance) or implicit (revealed through actual PSA models) position on this issue? If so, please describe.
- Are any special (or new) PSA methods and tools (PSA supporting software) being used for your country's advanced reactor PSAs? If so, please describe.

(5) Does your country have any on-going research projects, or plans for research projects, regarding PSA methods, models, tools, or data aimed at or relevant to advanced reactors? If so, please provide a brief, summary-level description. (Note that Section C of this questionnaire requests more detailed information on this subject.)

2.2 General questions on issues

For the following questions, the appendix provides a set of potential topic areas where issues may arise.

(1) What do you think are the essential technical issues that should be addressed to support the effective, confident use of advanced reactor PSAs in risk-informed decision making? Please discuss in sufficient detail to enable the reader to fully understand why the identified issue is an issue, and why the issue needs to be addressed.

(2) What are the essential regulatory issues that need to be addressed to enable the use of advanced reactor PSAs in your country's licensing and regulation processes?

2.3 Research activities on PSA for advanced reactors

The following request is intended to elicit more detail on your answer to Question A.5 above. If your country does not have any research activities aimed at or relevant to advanced reactor PSA, you may skip this section.

(Please, refer to the appendix for specifying your topics). For each research activity aimed at or relevant to advanced reactor PSA, please provide the following:

(1) Technical issue(s) – please describe the issue or issues addressed by the research activity

(2) PSA topic area(s) – please identify the relevant topic area or areas (see appendix). If the activity addresses a topic area not identified in the appendix, please provide a brief (title-level) description of the area.

(3) Technical approach – please provide a summary description of the technical approach planned or being used to address the issue(s).

(4) Lead organizations – please identify the lead technical organization and the sponsoring organization

(5) Milestones and status – please identify key milestones (if established) and the current status of the activity. If milestones have not been established, please provide the expected, general timeframe for the activity.

(6) Regulatory application – please describe the expected regulatory application of the products of the research.

(7) Other – please provide any other information that would be useful to readers of the report, and to the organizers of the April, 2010 workshop.

2.4 International Cooperation

There are a number of international activities regarding PSA for advanced reactors. In the U.S., the American Society of Mechanical Engineers (ASME) has initiated the development of a PSA standard for advanced reactors (Non-LWR) as a part of their efforts to develop PSA standards. The IAEA is developing a proposal for a Coordinated Research Program (CRP) on passive systems reliability. The CNRA has formed a Working Group on the Regulation of New Reactors (WGRNR). The Generation IV International Forum (GIF) conducts an assistance group as the Risk and Safety Working Group (RSWG) for the risk-informed safety evaluation of Gen. IV reactors. It is expected that the understanding of the state of art for the improvement of PSA technology will help to establish and/or refine the research and development roadmaps or programs of each country and will increase the effectiveness of international cooperation.

(1) Are there any PSA-relevant topics that you believe would benefit from increased international cooperation? Please describe.

2.5 Other

(1) If there is any other topics you wish to see discussed in the Task Group report and in the April, 2010 workshop, please identify and briefly discuss these topics.

Annex 1: Supporting Information for Questionnaire

There exist many topics and a broad scope on PSA for advanced reactors to be discussed among the members, but they are roughly divided into as following categories:

1. Definition of advanced reactors
   A. Types of reactors
   B. Essential feature to define advanced reactors
2. Characterization of the ability of current PSA technology
   A. Implementation of PSA for Advanced Reactor
      - Establishment of PSA Procedures or Methodologies
      - Definition of Risk Measures for Advanced Reactor (e.g., VHTR)
      - Identification and Estimation of Initiating Events
      - Definition and Analysis of Accident Sequences
   B. Scope of PSA for Advanced Reactor
      - Severe Accidents Analysis & Consequential Analysis
      - Low Power & Shutdown PSA
      - External PSA & Fire PSA
      - Aging PSA
   C. Use of PSA in Design
      - Implementation of PSA in Conceptual and Design Stage Reactors
      - Feedback to Improvement of a Design
   D. Regulatory Aspect on PSA Framework for Advanced Reactors
   E. Special Topics
      - Need for PSA Standard (Quality assurance program for an implementation of PSA)
      - Usefulness of Technology-Neutral Framework in Regulation & Design
      - Life Cycle Analysis for Advanced Reactors
3. Characterize the potential value of advanced PSA methods and tools
   A. Establishment of PSA Procedures or Methodologies based on advanced PSA methods and tools. There are several available approaches to identify and review the procedures or methodologies, but it is helpful to use a specific frame for these works, which one is an available approach to use ASME classification of PSA scope and areas as the below A.1 summary table (example)

B. Identification and Resolution of New Technical Issues: Some fields in as the above mentioned classification has a difficulty to evaluate the subsidiary-reliability or risk for quantifying an overall risk by using the current available techniques, so they should be discussed among the members for whom are identified and resolved. For examples, following technical issues are already identified as new technical issues on PSA for advanced reactors.

- Reliability of Passive Safety System
- Human Reliability Analysis for Advanced Reactors
- Reliability of Digital I&C System
- Source Term Analysis & Consequence Analysis

C. Development of Reliability Database for Advanced Reactors

4. Develop recommendations to CSNI for any needed developments

A. Consensus of PSA Procedures & Methodologies

B. Roles of International Cooperation

Because divided-topics in these categories are closely related with other topics, they has to be discussed in WGRISK meeting for (1) the current status of each member or member's country and (2) currently attention fields or specific topics around them.

Annex 2: Summary Tables

- **Scope and Area**

Topics
1. Plant Operational States Analysis (POS)
2. Initiating Events Analysis (IE)
3. Event Sequence Analysis (ES)
4. Success Criteria Development (SC)
5. Systems Analysis (SY)
6. Human Reliability Analysis (HR)
7. Data Analysis (DA)
8. Internal Plant Hazards Analysis
A. Internal Flooding Analysis (IFL)
B. Internal Fires Analysis (FI)
C. Internal Hazards Analysis References
9. External Plant Hazards Analysis
A. External Events Screening (EES)
B. Other External Events Analysis (OE)
C. Seismic PRA (SP)
D. High Winds Analysis (HW)
E. External Flooding Analysis (EFL)
F. External Hazards Analysis References
10. Event Sequence Quantification (ESQ)
11. Mechanistic Source Term Analysis (MS)
12. Radiological Consequence Analysis (RC)
13. Risk Integration (RI)
14. Definition of damage states (i.e., core damage)

- **Special topics of technical issues**

Topics
Reliability of Passive Safety System
Human Reliability Analysis for Advanced Reactors
Reliability of Digital I&C System
Source Term Analysis & Consequence Analysis
Reliability Database

APPENDIX 3: ANSWERS TO QUESTIONNAIRE ON PSA FOR ADVANCED REACTORS

Categorization of Respondents (12 countries/16 Organizations)

	Regulatory Body	TSO/Research Institutes Engineering Company	Academia
Organization	BEL V-Belgium USNRC-USA SNSA-Slovenia STUK-Finland SUJB-Czech Republic UJD-Slovakia	TE-Belgium CEA/IRSN-France ENEA-Italy ONR-UK JAEA/JNES-Japan KAERI-Korea VTT-Finland	INET-China

1 General questions

1.1 What is the current status of the development of advanced reactors in your country?

Organization	Responses
BEL V, TE-Belgium	N/A. Because of the phase out law, no new reactors are considered in Belgium. Hence, there are (at least at the regulatory side) no activities related to the development of advanced reactors.
NRC-USA	Currently, several advanced reactor design types are being considered for development in the United States. The design types under consideration in the United States are referred to as small modular reactors and are often grouped into three categories: high-temperature and very-high-temperature gas-cooled reactors, liquid metal reactors (LMRs), and integral pressurized water reactors (iPWRs). The high-temperature gas-cooled reactors (HTGRs) and very-high-temperature reactors (VHTRs) include helium-cooled, graphite-moderated reactor designs. The LMRs include, for example, sodium-cooled fast reactor (SFR) designs that are envisioned to be used for electricity production as well as the management of high-level wastes. The iPWRs include several different designs that are characterized by having reactor modules that produce relatively low power output with the potential to operate multiple modular units together at a site with a centralized control room. Note that these groups are not formal classifications, and some reactor designs may contain attributes of more than one of the groups. At this time, the U.S. Nuclear Regulatory Commission (NRC) has not received any licensing application for an advanced reactor. However, NRC has received information about plans and schedules for possible applications. Advanced reactor designers also have submitted to NRC topical reports and white papers that discuss design features and approaches to performing safety evaluations. NRC expects applications for design certifications or combined licenses (COL) to be submitted as early as 2011. Moreover,

	NRC has had or has scheduled meetings and informal discussions with advanced reactor designers and potential applicants. In addition, NRC has sponsored a workshop on advanced small and medium-sized reactors that was held on October 8, and 9, 2009. During the workshop the NRC met with advanced reactor designers to discuss possible policy and licensing issues for small and medium-sized reactors. Additional information regarding the workshop can be found on the NRC Web site: http://www.nrc.gov/reactors/advanced/public-meetings.html Currently, NRC's work on advanced reactors is largely focused on interactions with the U.S. Department of Energy (DOE) on the development of the needed infrastructure and research to support the Next Generation Nuclear Plant (NGNP) program. Some NRC resources are devoted to preparing to license iPWRs; however, NRC anticipates that much less research and development work will be required for these designs. As a result, this survey response will be focused on preparing to license non-light-water reactors (non-LWRs). The goal of the NGNP program is to develop a prototype VHTR plant that is capable of generating electricity as well as utilizing high-core-outlet temperatures for other process heat applications. NRC has initiated several research tasks to develop the technical knowledge needed to support the NGNP licensing review. NRC's research program to support NGNP spans many technical areas including fuel performance, thermal fluids, high-temperature material performance, and probabilistic safety assessment (PSA). DOE currently plans to submit a licensing application for the NGNP plant in year 2013. The strategy for licensing the NGNP prototype will likely be for the applicant to submit a combined license application under Title 10, Part 52 of the *U. S. Code of Federal Regulations* (10 CFR Part 52), which will require the applicant to submit a description of the plant-specific probabilistic risk assessment (i.e., the PSA) and its results. The licensing review for advanced non-LWR designs will likely provide many regulatory and technical challenges. Several of the applicable regulations are written with requirements that are specific to LWRs. The NRC staffs have proposed to create a risk-informed and performance-based revision to the existing regulations that would be applicable to all reactor technologies. One possible framework for creating this risk-informed and performance-based revision is presented in the NRC document NUREG-1860 "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing." Currently, the NRC staff plans to defer rulemaking for risk-informed and performance-based reactor requirements until the review of a license application for a non-LWR design has progressed far enough to provide useful insights into the application of such an approach to an actual design.
SNSA-Slovenia	N/A. Slovenia does not have any advanced reactors in development.
STUK-Finland	In Finland there are three projects for new nuclear power plants/units waiting for a decision-in-principle by the government and parliament. The type of the reactor has not been decided in any of the projects, but only LWRs are considered. In all three projects there are alternatives which can be considered advanced LWRs. There are no reactor vendors in Finland but some Finnish companies participate in the development of advanced LWRs with vendors. There are also some limited research projects related to Gen IV reactors.
SUJB-Czech Republic	N/A. There is not any development of new type of advanced reactors in the Czech Republic at present. The construction of the new NPP units is being assumed at the moment. This construction will be realized by suppliers. Three suppliers have been

	involved in the tender – AREVA, Westinghouse and consortium Atomstrojexport + Škoda JS + Gidropress. The aim of the tender is to select a suitable supplier of new units.
UJD-Slovakia	N/A. There is no development of advanced reactors in Slovakia.
CEA-France	CEA is currently developing two types of Generation IV reactors: SFR (Sodium cooled Fast Reactor), as a reference technology, and GFR (Gas cooled Fast Reactor) as a long term technology. The current status of development is pre-conceptual design phase. In 2012, a decision is scheduled regarding the building of an industrial prototype to be operational in 2020.
ENEA-Italy	The Italian government has planned the construction of new advanced NPPs (EPR type). In addition, Italy is currently involved in some projects funded by European Union related to Gen. IV (e.g., Lead Fast Reactor, Sodium Fast Reactor, Gas Fast Reactor and High Temperature Gas Reactor).
ONR-UK	Development of advance reactors in the UK is focused on the so-called evolutionary light water reactor designs. Initially the following 4 reactor designs were considered by our generic design assessment (GDA) process: (a) ACR-1000 (AECL), (b) UK EPR (AREVA and EDF), (c) ESBWR (GEH), and (d) AP1000 (WEC). However, by late 2008 the ACR 1000 design was removed from the assessment process by AECL and GEH requested that we temporarily suspend our assessment work on the ESBWR. Therefore, two reactor designs are currently going through our generic design assessment (GDA) process: EPR (AREVA and EDF), and AP1000 (WEC). Further details on GDA can found at: www.hse.gov.uk/newreactors/guidance.htm. Step three of that process has recently been completed (November 2009), with Step 4 now underway. It is expected that Step 4 will be completed by the end of June 2011. In terms of HSE, no other reactor designs are currently being considered.
IRSN-France	In France there are currently two types of new plants: a) The EPR project developed by AREVA/EDF which is an evolutionary design compared to existing plants, although there are some advanced features especially as regards to severe accidents. The building of the first French EPR is presently on going; b) Advanced designs (generation IV) for which studies are in progress.
JAEA-Japan	Japan Atomic Energy Agency (hereafter JAEA) operates two sodium-cooled fast reactors (hereafter SFRs) as part of SFR development in Japan. One is experimental fast reactor Joyo, which has been operating and accumulated operation time longer than 70,000 hours since the first criticality on April 24th, 1977 (JAEA Experimental Fast Reactor Joyo). The other is prototype fast breeder reactor (hereafter FBR) Monju, which attained the first criticality on April 5th 1994 and completed a lot of modification work in May 2007, based on the lessons learned from the sodium leakage accident in 1995. JAEA intends to restart Monju by the end of March 2010 (JAEA Tsuruga Head Office). JAEA is executing "Fast Reactor Cycle Technology Development (hereafter FaCT)" project in cooperation with the Japanese electric utilities. In the FaCT project, both the conceptual design study for JAEA SFR (hereafter JSFR) and the developments of the innovative technologies are implemented with paying attention to the consistency between the design and the innovative technologies. The current target is to start operation of a JSFR demonstration reactor around 2025 (Kotake, Mihara, Kubo, Aoto, & Toda, 2008).
KAERI-	Korea is developing several types of advanced reactors as follows:

<table>
<tr><td>Korea</td><td>
<table>
<tr><th>Reactor</th><th>Status</th></tr>
<tr><td>SFR (Sodium cooled Fast Reactor)</td><td>We are planning to submit the safety analysis report for the Design Certification in 2017, to get the design certification in 2021, and to start the detailed design from 2022 to 2026.</td></tr>
<tr><td>VHTR (Very High Temperature gas cooled Reactor)</td><td>A research for essential features and pre-conceptual design of VHTR are under development. Korea has not yet decided the master plan for a development of VHTR. However, Korean has attended the VHTR's partition in "Generation IV International Forum (GIF)" in cooperation with other relevant countries.</td></tr>
<tr><td>SMART (System-integrated Modular Advanced ReacTor)</td><td>Korea is going to submit the safety analysis report for the Design Certification in 2010, and to get the design certification in 2012.</td></tr>
<tr><td>APR1000 (Korean Advanced Power Reactor, 1000 MWe)</td><td>Korea Electric Power Corporation (KEPCO) has developed the APR1000, which is a two-loop 1000MWe pressurized water reactor (PWR). The APR1000 is an evolutionary reactor based on the proven OPR1000 design constructed and operated at the Shin-Kori nuclear generating station in Korea. APR1000 incorporates a variety of advanced design features designed to provide sufficient reliability and safety margins which can fulfil the need of nuclear customers. Further, design features to address the NRC's Severe Accident and Safety Goal Policy Statements are incorporated into the APR1000 design.</td></tr>
<tr><td>APR1400 (Korean Advanced Power Reactor, 1400 MWe)</td><td>APR1400 was developed in 2002 and received Design Certificate from Korean Government. Currently SKN3&4 has been constructed to start commercial operation in 2013.</td></tr>
</table>
</td></tr>
<tr><td>VTT-Finland</td><td>Finland is involved in several international research activities related to Gen-IV reactors: Gen IV international forum, EURATOM projects, Jules Horowitz Reactor MTR. The Finnish nuclear research organizations have a joint New Type Nuclear Reactors (NETNUC) project focusing on Generation IV reactors. VTT, STUK and the Finnish nuclear industry have a working group for interested parties with Gen IV activities. The Finnish focuses on
a) Most important and feasible reactor concepts with new applications or fuel cycle;
b) Cross-cutting issues e.g. materials, modelling of reactor physics, flow, process simulation;
c) Evolutionary concept, strong knowledge based on LWRs with applicability to boiler technologies.
<u>PSA issues have not been addressed so far.</u></td></tr>
<tr><td>INET-China</td><td>Both AP1000 and EPR, and HTGR type of advanced reactors will be built in the near future.</td></tr>
</table>

1.2 What types of reactors are considered for this questionnaire?

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	The responses to this questionnaire consider advanced reactors to be non-LWR types and iPWR design types. NRC is specifically focused on high-temperature and very-high-temperature gas-cooled reactors, sodium-cooled fast reactors, and iPWRs. The responses to this questionnaire do not consider the so-called "evolutionary" LWR designs. These include designs such as Advanced Boiling-Water Reactor, AP1000, Economic Simplified Boiling-Water Reactor, U.S. Evolutionary Power Reactor, and U.S. Advanced Pressurized-Water Reactor. These reactor designs have advanced through various stages of design certification review at NRC. Several site-specific combined license application reviews have also begun. Since the licensing reviews for these reactor types have matured, NRC has addressed many of the licensing and technical issues. However, some of the evolutionary reactor designs do share common traits with the advanced reactor (non-LWR and iPWR) designs that are considered in this questionnaire (e.g., passive cooling systems). Therefore, some of the responses in this questionnaire could be applicable to the evolutionary reactors, but the responses are written only considering non-LWR and iPWR design types.
SNSA-Slovenia	LWR with advanced safety features
STUK-Finland	LWRs with significant new passive safety systems, e.g., ESBWR, AES-2006 and KERENA (SWR 1000) could be classified as advanced reactors. We do not regard evolutionary LWRs, such as EPR, as advanced reactors from the PSA point of view. Although the evolutionary reactors have advanced features, the same PRA framework can be applied to evolutionary reactors and currently operating reactors.
SUJB-Czech Republic	EPR 1600 (Areva), AP 1000 (Westinghouse), WWER MIR-1200 (Atomstrojexport + Škoda JS + Gidropress).
UJD-Slovakia	N/A
CEA-France	Generation IV reactors (SFR and GFR)
ENEA-Italy	Essentially LWRs. It has to be noted that, despite the fact that there are some ongoing activities on Gen IV reactors (as Lead Fast Reactor, Sodium Fast Reactor, Gas Fast Reactor and High Temperature Gas Reactor), they are mainly devoted to the definition of the concept of these reactors and the PSA aspects are addressed in a very limited way or even only mentioned.
ONR-UK	Evolutionary LWRs, particularly the AP1000 and UK EPR.
IRSN-France	Since the EPR is a new plant but not an advanced design in this questionnaire, it is not included. In particular concerning PSA there are no specific methodology problems in the case of EPR compared to existing plants.
JAEA-Japan	SFRs
KAERI-Korea	SFR, VHTR, APR1400 and APR1000, but not include the SMART because the PSA methodology is almost the same as APR1400 and APR1000.
VTT-Finland	Gen-IV (fission) reactors
INET-China	HTGR only, because the PSA will be provided directly by the American/French designers concerning AP1000 and EPR reactors, the regulatory body and the utilities are more or less enjoying the fruits from the originals.

1.3 Preparation of regulator for advanced reactors in your country

(1) Does your country have ongoing research programs related to the development of regulatory or licensing guidance of advanced reactors? This does not include research activities related to the development of specific methods, tools, and data intended to support advanced reactor PSAs.

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	NRC has research programs in place to address the need for development of regulatory guidance related to the licensing of advanced reactors. A specific research and development plan has been written to support the licensing of the NGNP prototype, and additional research programs are planned for other advanced reactor design types. The NGNP research and development plan includes research tasks to develop the regulatory guidance and also the specific methods, tools, and data to support the licensing review. The NGNP research and development plan includes specific research tasks to support advanced reactor PSA as well as other technical disciplines that may be relevant to PSA such as Human Factors and Digital Instrumentation and Controls. The NGNP research and development plan provides an overview of all of the advanced reactor research that is needed to support the NGNP licensing review. Many of the initial research tasks in the plan are currently underway, and subsequent tasks are planned pending the outcome of the current research. In the area of advanced reactor PSA, NRC is currently pursuing an advanced non-LWR PSA planning study. The goal of this planning study is to identify the research programs that need to be put into place to support the NGNP licensing review. Subsequent follow-on research tasks are likely to include: a) The development of regulatory guidance for PSA technical acceptability; b) The development of tools, methods, and data to support a PSA technical acceptability review; c) The development of a scoping-level PSA model to support the ongoing identification, prioritization, and selection of advanced non-LWR research topics. Although the current research program is focused on supporting the licensing process for the NGNP's VHTR design, many of the research activities could be applicable to other advanced reactor design types. As NRC proceeds with plans to broaden its advanced reactor research program, the results from the NGNP research and development plan will be leveraged as appropriate and necessary.
SNSA-Slovenia	Conditional yes, that is, the regulatory body is actively preparing itself for a possibility that an application for a new build is submitted within the next couple of years. Such a possibility is included in the regulations that are under preparations as well as in the related licensing guidance. The preparations include training of the staff and active participation in related international activities (WENRA, OECD/NEA WGNR ...) but no research program per se.
STUK-Finland	N/A
SUJB-Czech	N/A

Republic	
UJD-Slovakia	N/A
CEA-France	N/A
ENEA-Italy	Actually there are no research plans for regulatory purposes.
ONR-UK	There are ongoing research programs, although these are not necessarily related to the development of regulatory or licensing guidance of advanced reactors. As issues are highlighted that require research, then this may be commissioned as required. We have agreements with other regulators to share results of research, for example a bilateral agreement with NRC on severe accidents – CSARP.
IRSN-France	Reflexions are in progress for defining a regulatory framework for Generation IV reactors. This work could be carried out in cooperation with foreign regulators and TSOs.
JAEA-Japan	Corresponding to the expectation by the Nuclear Safety Commission (hereafter NSC) of Japan, JAEA addresses safety research activities for preparation of regulator with respect to SFRs as follows: a) Pursuing study on sodium-chemical reactions, in particular, burning of sodium leaked and sodium-water reaction, which constitute the basis of the safety design and evaluation of SFRs. b) Developing and improving the evaluation methods of the sodium-chemical reactions with making validation based on insights obtained from operating experiences, experimental studies, etc. c) Applying those insights and methods to the safety design and evaluation and the operation management of sodium-related systems and components. d) Study on the safety evaluation technology of FBR fuel and accumulation of experimental data, which is obtained from experiments in the SFRs (e.g., Joyo) and irradiation test facilities in order to develop and make validation of the safety evaluation method. e) Developing the technologies for prevention of severe accidents and for evaluation of consequences of the severe accidents. Japan Nuclear Energy Safety Organization (JNES) addresses the following safety research activities: a) Analyses to give the reason of AOT (Allowed Outage Time) in the "Safety operation guideline" of Monju: Each AOT of malfunctioned components should have a concrete reason for the value. Some might be simply the identical value that is used in LWR. For the operator's understanding to the situation and even for the transparency of regulation, a rational reason to each AOT should be described as far as possible; b) Preparation of the databases of the Emergency Response Support System (ERSS) for Monju: because in ERSS the on-time analysis on the ongoing progress of accident will be hard to do, it is necessary that the considerable event progresses are all previously prepared.
KAERI-Korea	Yes, the Korean regulatory has performed continuous research program to apply the risk-informed regulatory approach to existing reactors. This R&D program has resulted in practical guidance which includes licensing guidance to review the technical adequacy of Level 1 and 2 PSAs submitted as part of the licensing documents of evolutionary or advanced reactors such as APR-1400 and SMART. In addition, through recent research on technology-neutral framework for non-LWRs such SFR and VHTR, the Level 3 PSA guidance has been proposed for the application.
VTT-Finland	Refer to Answer by STUK

INET-China	YES. A project has been initiated to develop the set of regulatory codes for the HTGR reactors, which is funded within the framework of "Important National Science & Technology Specific Projects" by the National Energy Administration. Within the same framework, the related research projects to the issues induced by the introduction of AP1000 are also be supported, however most of these projects focus on the technical specific issues e.g. material, component design and so on.

(2) Some countries have developed new, risk-informed regulatory approaches that may be applicable to the licensing and regulation of advanced reactors. (See, for example, NUREG-1860, which describes a Technology-Neutral Framework for regulation prepared by the U.S.NRC.). In your country, what is the current role of PSA in the development, licensing, and regulation of advanced reactors? Are there any plans for near-term or long-term changes in this role? Please discuss.

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	The current approach for the licensing of advanced reactors is anticipated to follow the existing regulatory framework that applies to all new reactors. Advanced reactor applications will likely be submitted under 10 CFR 52, which provides a framework for the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities. The existing regulatory requirements of 10 CFR Part 20, Part 50, Part 73, and Part 100 that are applicable and technically relevant will still apply to advanced reactors. Under this existing regulatory framework the applicant is required to submit a description of the plant-specific PSA and its results. Other roles that the PSA may play in the current licensing and regulatory framework include a risk-informed process for categorization of structures, systems, and components (SSCs) and the use of PSA results in monitoring plant performance and maintenance activities. Some of the existing regulatory requirements have been written to specifically address the technology of currently operating LWR designs. An advanced reactor design, such as a VHTR design proposed for the NGNP, may require exceptions to certain requirements or additional review to determine how the requirement can be met. The details of the licensing strategy for the NGNP are described in the report "Next Generation Nuclear Plant Licensing Strategy: A Report to Congress" (available in NRC's Agencywide Document Access and Management System [ADAMS] at ML082290017). The NGNP licensing strategy includes the use of PSA to support risk-informed processes for selection and categorization of licensing basis events (e.g., selection of anticipated operational occurrences, design-basis accidents, and beyond-design-basis accidents) and evaluation of defense-in-depth measures. As subsequent commercial plant applications are pursued for VHTR (or other advanced) designs, revisions to the licensing strategy may be necessary and appropriate. NRC also has given consideration to revising the existing licensing regulations with the development of a risk-informed and performance-based licensing framework, as proposed in NUREG-1860. The framework is also intended to be written in a technology neutral way so as to be applicable to all advanced reactor designs (as well as operating reactors). NRC developed NUREG-1860 to study the feasibility of creating a risk-informed and performance-based licensing framework. However, no near-term

	changes to the existing regulatory framework are planned. At this time, the NRC staffs have deferred rulemaking for risk-informed and performance-based reactor requirements until the review of a license application for a non-LWR design using a risk-informed licensing approach has progressed far enough to provide useful insights into the application of such an approach to an actual design. NRC expects that the regulatory decisions in connection with such a review, permitting greater use of the plant PSA in establishing the licensing basis, will provide practical experience and insights into the development of such rulemaking. In this regard the NGNP prototype reactor license application review will likely provide the first experience. Additionally, the NRC staff plans to test the concepts and methods from a technology-neutral framework during the licensing review of an advanced non-LWR. This test will also likely occur in connection with the licensing review for the NGNP prototype reactor. Any long-term changes to the licensing strategy that may occur in the future will likely depend on the insights gained from the licensing review of an advanced non-LWR.
SNSA-Slovenia	At the moment we are introducing new regulation with much higher PSA content. The old regulation (that will most likely be replaced by the end of 2009) has no PSA requirements. The new regulation has been prepared with necessary consideration for the possibility of a new build, taking into account WENRA and IAEA recommendations and best international regulatory practices. The new regulation has specific quantitative values for CDF and LERF that a new NPP will have to achieve (while operating at power, regarding On-Line-Maintenance (OLM) and modifications etc.). The most limiting factor in the introduction of PSA based requirements is a lack of PSA standards for some uses of PSA.
STUK-Finland	The Finnish regulatory guide YVL 2.8 on PSA requires a preliminary full-scope PSA already in connection with a construction license application. Several risk informed applications, such as RI-ISI, RI-IST, risk informed review of technical specifications, safety classification and emergency operating procedures are also required. The requirements would be applied also to advance LWRs if construction licenses were applied for such types. Special issues related to risk informed regulation of advanced LWRs have not been considered yet.
SUJB-Czech Republic	PSA is not included in the current legislation of the Czech Republic. The amendment of Atomic law and relevant decrees are prepared with respect to future construction of the new units. In this amended legislation will be included PSA and its applications according to WENRA requirements. Use of PSA is also assumed in the licensing process of the new units. PSA will be required from supplier of the new units and will be reviewed by Regulatory Body, as well as deterministic safety analyses.
UJD-Slovakia	N/A
CEA-France	N/A
ENEA-Italy	N/A
ONR-UK	No new regulatory approaches per se have been developed over the last few years since the Safety Assessment Principles (SAPs) (www.hse.gov.uk/nuclear/saps/index.htm) were last updated. The SAPs, which set out relevant good practice for a wide range of nuclear facilities, have been benchmarked against IAEA's nuclear safety standards. The current regulatory standards are being used to license new reactor designs. However, the process for approval has been modified significantly with the introduction of the GDA process prior to licensing; see www.hse.gov.uk/newreactors/newnuclearprogramme.htm for further details.
IRSN-France	The French regulators have required information relating to the PSAs already

	performed for the SFR. Anyway it seems clear that the future regulation will be based on deterministic principles complemented by probabilistic insights.
JAEA-Japan	Current role of PSA in Japan is as follows: a) In the development of SFRs, PSA serves as a useful tool for pursuing rational safety design from the risk point of view (Kurisaka, 15-20 Oct. 2006). b) In the safety review as part of licensing process of LWRs and SFRs, implementation of PSA is not required mandatorily. However, it is strongly recommended for licensees to develop accident management measures for the reactor facilities before fuel loading on a voluntary basis (NSC of Japan, 1992): at this time, PSA would be implemented in order to show quantitative effectiveness of the accident management measures. In addition, when the NSC published the revised version of “Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities” in September 2006 (NSC of Japan, 2006), the NSC issued the NSC decision including the statements that it is important for regulatory agency to ask licensees implementing quantitative evaluation about “residual risk” and to confirm adequacy of its evaluation result not in the safety review. Here, “residual risk" is defined as risk of occurrence of serious damages in the facilities or releases of a large quantity of radionuclide from the facilities caused by earthquake exceeding a design basis earthquake ground motion, or causing hazards by radiological exposures to the surrounding public as their result. Corresponding to this NSC decision, the seismic PSA would be implemented in order to evaluate “residual risk.” As part of promotion of utilization of “risk information” in nuclear safety regulation activities, the Nuclear and Industrial Safety Agency (hereafter NISA) declared that NISA intends to make an examination on utilization of “risk information” widely in nuclear safety regulation and published “Basic concept to apply ‘Risk information’ to nuclear safety regulation”, “Near-Term Implementation Plan for Utilization of ‘Risk Information’ in Nuclear Safety Regulation”, “High-level Guidelines for Utilization of ‘Risk Information’ in Safety Regulation for NPPs -Trial Use-”, and “PSA Quality Guidelines for NPP Applications -Trial Use-”. The Atomic Energy Society of JAPAN (hereafter AESJ) has been developing PSA related standards; AESJ already published the procedure standards for level 1, level 2, and level 3 PSA during power operation, seismic PSA procedure standards, and shutdown PSA procedure standards; the guidelines for the application of “risk information” will be developed soon. Furthermore, a pilot application and/or implementation of risk-informed regulation has just still started primarily in a domain of LWRs having much experience of PSA implementation (Fukuda, Fujimoto, Yamashita, & Sugawara, 23-27 April 2007): e.g., confirmation of adequacy of the allowed outage time from the risk point of view, construction of a graded maintenance plan considering risk importance measures.
KAERI-Korea	PSA report is a supplementary document to be required for licensing new reactors since 1993. For severe accident policy of Korean Government issued in 2000, PSA for all of the operating reactors was required. The basic role of PSA is to verify the safety goals of the NPP and to enhance the safety by identifying and improving the weak points in the views of severe accidents. We are expecting that risk information has a great role in the development and licensing of advanced reactors. So, the regulatory body wants to get practical ways for applying risk-informed regulatory approaches in validating advanced reactors. For example, the PSA documents consisting of Level 1, 2, & 3 scopes were submitted

	for APR-1400 at design certification stage and reviewed by the regulatory. This PSA was voluntarily implemented in accordance with the similar licensing process of 10 CFR 52 in United States. The PSA with similar scope is also expected to do for SMART at DC stage. Research on technology-neutral framework referred to NUREG-1860 and IAEA-TECDOC-1570 has been performed for preparing conceptual design stage of non-LWRs. Furthermore, new R&D has been recently planned to develop regulatory requirements and licensing guidance suitable for specific SFR and VHTR.
VTT-Finland	Refer to Answer by STUK
INET-China	A policy statement about the use of PSA in China will be issued soon. It can be regarded as the clear indication of the regulatory positive attitude to encourage the use of PSA. However due to the lack of detailed technical guidance, the role of PSA in the development, licensing, and regulation of advanced reactors will be discussed case by case in the foreseeable future.

1.4 The current status of advanced reactors' PSA

(1) Does your country have ongoing research programs related to the development of regulatory or licensing guidance of advanced reactors? This does not include research activities related to the development of specific methods, tools, and data intended to support advanced reactor PSAs.

(2) Which hazard types are currently being treated in your country's PSAs for advanced reactors? Are there any hazard types that are explicitly excluded? Do these answers depend on whether the site includes currently operating plants or not? For future (near-term and long-term) PSAs for advanced reactors, are there plans to extend the hazard types considered? If so, please describe.

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	N/A
SNSA-Slovenia	In principle the regulations require that all hazards are considered. This does not depend on the site of the future NPP. In respect to PSA requirements some exceptions are granted to the existing NPP.
STUK-Finland	PSA shall cover internal events (leaks, transients), area events (internal fires and floods), external hazards (harsh weather, seismic, industrial and transport accidents including oil slick). This applies also to the advanced LWRs if construction license applications are submitted.
SUJB-Czech Republic	As mentioned above, there is not developed any new type of advanced reactors in the Czech Republic at present; from a supplier of the new units PSA level 1 and level 2 will be required; scope of PSA – see next answer.
UJD-Slovakia	N/A
CEA-France	a) For the moment, CEA has only developed a PSA in support to the design of the GFR 2400 MWth (called hereafter GFR PSA). It is likely that a PSA will also be developed in support to the design of the SFR in the near future. b) The GFR PSA includes the external hazard "Loss Of Offsite Power" and the internal

	hazard "turbine deblading" (initiator specific to the GFR 2400 MWth design) for the moment, but no hazard type is explicitly excluded. c) We do not consider the possibility that another operating plant could create an external hazard. d) There is currently no plan to extend the hazard types considered in the GFR PSA.
ENEA-Italy	N/A
ONR-UK	Our expectation for PSA is clearly set out in the SAPs and PSA technical assessment guide - www.hse.gov.uk/foi/internalops/nsd/techasstguides/tast030.pdf. Our expectation for an advanced reactor PSA is that a full-scope PSA is carried out. Therefore we would expect all hazard types to be considered.
IRSN-France	N/A
JAEA-Japan	The risk induced from internal hazard (e.g., random failure of systems, human error) depends on reactor system configuration, in particular, the extent of redundancy and/or diversity in the safety-related systems at a train and/or component level. A hazard primarily to be focused on is an internal hazard in order to confirm adequacy of the safety-related system configuration from the risk point of view for SFRs both already put in operation (Nakai & Kani, 17-21 April 1994) (Ishikawa, et al., January 2009) and under conceptual design work (Kurisaka, 15-20 Oct. 2006). An earthquake is a typical external hazard to nuclear facilities, which are sited in Japan. As part of development of PSA methodology in SFRs, JAEA performed a study on application of a seismic PSA to a model plant of SFRs assuming a specific site (Nakai & Yamaguchi, 1998). A preliminary examination of probabilistic seismic margin analysis has just started for the purpose of understanding safety margin in the earthquake resistance of JSFR early in the seismic design consideration although even site parameters for deterministic seismic design are still under consideration. The risk from other external hazards will be addressed in future. JNES addresses the study on reliability of decay heat removal from the ex-vessel spent fuel storage tank in Monju with focusing on internal hazard, because the fission product inventory would become a considerable amount when the tank is filled with the spent fuels. (Kasagawa, K. Miura, S., Endo, H., December 2009).
KAERI-Korea	In principle, it is necessary to consider all kind of hazards in order to verify the safety of nuclear plants using PSA. Especially, an issue raised for APR1000 on how to treat risk due to the crash of aircraft should be considered. The current operating plants were not required for this issue. During next few years we will develop the methodology to deal with the aircraft crash. SFR and VHTR are being designed under long term project. Currently, we focus on the internal events (excluding fire and flood events) and the research for the passive system reliability for those next generation reactors. The risk of external hazards from the fire, flood and seismic events will be considered in the near future for SFR because we are planning to submit the PSA for the design certificate in 2017. New external events such as aircraft crash are one of issues to be studied in the near future.
VTT-Finland	Refer to Answer by STUK
INET-China	The designer of HTGR and the Safety Authority have come to an agreement that internal flooding, internal fire and external earthquake PSA will be explicitly included in the FSAR. Other hazards such as extreme meteorology, tornado and so on are considered according to the usual deterministic approach. Since HTGR will be the first unit in the site, whether the policy strategy will be updated with respect to the following

	units is not clear yet.

(3) Which operating modes (full power, low power and shutdown) are currently being treated in your country's PSAs for advanced reactors? For future (near-term and long-term) PSAs for advanced reactors, are there plans to enlarge the implementation scope? If so, please describe.

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	Although NRC has not had access to fully review any of the existing advanced reactor PSAs, NRC has received information regarding the current status of these PSAs through pre-application meetings and submittals of topical reports and white papers. Although the information on advanced reactor PSAs has been limited, the communications with advanced reactor designers have given NRC indications for the expected scope of the PSAs. The expectation is that advanced reactor PSAs will cover a full scope of hazard types and operating modes. The term "full scope" as used here suggests a PSA that covers: a) A complete set of initiating events including those caused by internal plant hazards such as fires and floods and external plant hazards such as seismic events and transportation accidents; b) A complete set of plant operating states including all operating and shutdown modes. The designation of a full scope PSA also has implications on the inclusion in the model of radiological releases and offsite consequences. This aspect of PSA scope is discussed further in the following response section. In addition to the limited information that has been shared by advanced non-LWR designers, NRC is also involved in the development of a PSA standard for advanced non-LWRs. Standards addressing different aspects of PSA are developed by the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS), with NRC staff and industry members contributing to the development. The use of PSA consensus standards and independent peer reviews against the standards are important factors in establishing PSA quality. The scope of the relevant PSA standard provides an expectation for what a complete PSA model should include. ASME has developed a draft version of the "Standard for Probabilistic Risk Assessment for Advanced Non-LWR Nuclear Power Plant Applications." Although this standard is not yet finalized, the draft version suggests the standard will establish requirements for a full scope PSA. However, the NRC regulations (specifically 10 CFR 50.71(h)(1)) only require a Level 2 PSA for COL holders.
SNSA-Slovenia	For advanced reactors/new NPPs, all operating modes have to be considered.
STUK-Finland	PSA shall cover all operating modes.
SUJB-Czech Republic	PSA level 1 and level 2 will be required from a supplier of the new units. This PSA should contain full scope of the relevant initiating events (i.e. internal initiating events, loss of off-site power, heavy loads drops, floods and fires and also relevant external hazards) and all operational modes (full power, low power, shutdown).
UJD-	N/A

Slovakia	
CEA-France	The GFR PSA is limited to the full power operating mode. There is no plan to extend the implementation scope.
ENEA-Italy	N/A
ONR-UK	Our expectation for an advanced reactor PSA is that a full-scope PSA is carried out. Therefore we would expect all operating modes to be considered.
IRSN-France	N/A
JAEA-Japan	A full power operation mode has been mainly treated in the SFR PSA study. However, a preliminary study on a shutdown PSA of SFRs was addressed by assuming a scheduled maintenance plan in order to obtain insights into a safety management during shutdown state of a SFR model plant (MIHARA, 13-18 September 1998).
KAERI-Korea	In principle, all operating modes have to be considered for new reactors. Full power and low power/shutdown mode are considered in APR1000 and APR1400 PSAs. Korea is doing PSA for SFR and VHTR under long term project. Currently, we focus on the full power PSA. For SFR PSA, we are planning to do the shutdown PSA by 2017. We have no plan to do shutdown PSA for VHTR in near future.
VTT-Finland	Refer to Answer by STUK
INET-China	PSA for the power operation of HTGR has been finished. The designer of HTGR and the Safety Authority have come to an agreement that PSA for the low power and shutdown mode will be finished one year later after the PSAR review (e.g. 2010).

(4) One point of controversy for PSAs for advanced reactors (especially non-LWRs) concerns the applicability of the three-stage (Level 1, Level 2, and Level 3) PSA framework currently used for LWRs. This issue is closely related to the definition of the risk metrics or the end states of accident sequences. Does your country have an explicit (e.g., documented standard or guidance) or implicit (revealed through actual PSA models) position on this issue? If so, please describe.

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	Based on information from advanced non-LWR designers (especially the NGNP VHTR) and development of PSA standards for such advanced non-LWRs, the expected scope for advanced non-LWR PSAs is to include quantification of radiological source terms and offsite consequences. Using the terminology from the three-stage framework of LWRs, the expectation is to have full Level 3 PSA. However, the LWR framework is not appropriate for many advanced non-LWR designs. Compared to advance LWRs, some advanced non-LWRs can have significantly different phenomena and design aspects with respect to fuel damage states and barriers to prevent radiological releases. Therefore, the surrogate risk metrics that are typically used in Level 1 and Level 2 PSAs for LWRs, such as core damage frequency (CDF), large early-release frequency (LERF), and large release frequency (LRF), cannot be directly applied to all advanced non-LWR designs. NRC relies on the calculation of surrogate risk metrics for a number of regulatory and reactor oversight programs. These include: a) The risk-informed categorization and treatment of plant SSCs;

	b) The calculation of CDF contributions and importance measures to assess increases in risk due to maintenance activities; c) The calculation of risk significance of inspection findings and performance indicators. NRC expects that these risk-informed decision processes will be applied to the licensing and oversight of advanced non-LWRs, and appropriate risk metrics are needed to support these processes. At this time, NRC has not proposed any definitions for new surrogate risk metrics for advanced non-LWRs, although NRC is aware of some definitions proposed for potential non-LWR licensing applications.
SNSA-Slovenia	No. The regulations are as technology-type independent as possible, but they are still to a degree written with LWR-reactors in mind.
STUK-Finland	PSA shall include Level 1 and Level 2. Level 3 is not required. So far we think that the current regulatory approach can be applied also to advanced LWRs.
SUJB-Czech Republic	As mentioned above, PSA level 1 and level 2 will be required from a supplier of the new units.
UJD-Slovakia	N/A
CEA-France	In France, it is likely that a Level 1 and a Level 2 PSA will be recommended.
ENEA-Italy	N/A
ONR-UK	Our expectation is for the standard Level 1, Level 2 and Level 3 PSA approach to be taken. It is our expectation that a full scope Level 1, 2 and 3 PSA is carried out for design acceptance. This is stated in “Guidance to requesting parties” (www.hse.gov.uk/newreactors/ngn03.pdf).
IRSN-France	Refer to Answer by CEA
JAEA-Japan	As for SFRs, Japan has an implicit position that it is applicable but no explicit position on this issue. This is because JAEA has been applying the three-stage PSA framework to a SFRs’ PSA study and implemented a level-1 PSA (Nakai & Kani, 17-21 April 1994) and level-2 PSA (Kondo, Nonaka, Niwa, Sato, Furutani, & Miyake, 12-16 August 1990). As well, there are application instances of the three-stage PSA framework in other countries’ PRAs; e.g., the PRA for the SNR-300 plant (Bayer & Koeberlein, 1984), the Clinch River Breeder Reactor Plant PRA (Gitter & Akhtar, 21-25 April 1985), and the EBR-II PRA (Hill, Chang, Deitrich, Lehto, Ragland, & Schaefer, 12-16 August 1990) (Hill, Ragland, & Roglans, 1998). On the other hand, there are PSA-related procedure standards published by AESJ (AESJ-SC-P008:2008, March 2009) (AESJ-SC-P009:2008, March 2009); those include both basic requirements applicable to any kind of reactor systems and explicit descriptions specific to LWRs such as a judgment criterion about the core damage. Particularly, in the standard for a level-2 PSA, there is a clear description “This standard is for Light Water Reactors (LWRs), however, basic requirements in the standard can be applicable to the other type reactors”.
KAERI-Korea	The regulatory body has not yet explicit position on this issue because the development of non-LWRs is in the state of conceptual design stage. The PSA methodology for non-LWRs such as SFR and VHTR may be a considerable difference from those for existing LWRs. However, the implementation scope and depth of PSA for licensing non-LWRs will be appropriately established through development of new standard or guidance on

	the technical acceptability of the PSA. Accordingly, the regulatory expects to prepare new standard or guidance on the technical acceptability of the PSA for these non-LWRs. In terms of risk metrics, the regulatory has recently suggested quantitative safety goals for existing reactors and new reactors, but, the definition of risk metrics for new-conceptualized reactors, like VHTR, are not considered yet. The PSA method for VHTR is different with the method of PWRs. The inherent safety features of VHTR are classified into the simplified safety functions and the absence of the severe core damage state, i.e., the large release of radioactive materials. This is closely related to the specific fuel performance of VHTR. In consideration of these features, VHTR's PSA has to be implemented in a specific manner. The main reason for this is due to the difficulty of the definition of the robust failure state of a VHTR. This is an essential reason that necessitates a specific PSA procedure for VHTR. Since the ultimate goal of PSA is to estimate a consequential risk due to the behaviour of radioactive materials, the end states of accident sequences have been defined as a few of release categories of radioactive materials, which are called a source term release category in the typical PSA. A tentative PSA procedure for VHTR consists of the two levels of implementation procedure instead of the three levels of the implementation procedure of the typical PSA, which are divided into the sequence level PSA and the consequence level PSA. In the sequence level PSA, the end states of accidents are defined according to release categories of radioactive materials. Thus, new safety target should be developed for VHTR.
VTT-Finland	Refer to Answer by STUK
INET-China	Explicit guidance is available for LWR reactors, e.g. CDF<1E-5 and LRF<1E-6 for new reactors. For non-LWRs, it will be discussed case by case. For example, the recommended objective of HTGR will be likely to use the certain dose limit in the site boundary.

(5) Are any special (or new) PSA methods and tools (PSA supporting software) being used for your country's advanced reactor PSAs? If so, please describe.

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	At this time, NRC has no programs to develop special or new PSA methods and tools that are specifically designed for advanced non-LWRs. However, NRC is pursuing research programs in technical areas that may be applicable to a range of advanced reactor PSAs. These programs include the development of reliability models for digital instrumentation and control (DI&C) systems and advanced modelling techniques for Level 2 and Level 3 PSA. (Additional details on these programs are provided in the responses to Section A, question 5, and Section C.) Although these programs are being pursued with a focus on operating and near-term LWR designs, the results are likely to be applicable to advanced reactor PSAs as well.

SNSA-Slovenia	N/A
STUK-Finland	N/A
SUJB-Czech Republic	N/A
UJD-Slovakia	N/A
CEA-France	No special PSA methods have been used for the GFR PSA except the following: a) CEA has introduced a specific basic event corresponding to the failure of the decay heat removal natural circulation. In order to quantify this probability, it has been necessary to develop a specific method (called RMPS) to assess the performance of passive thermo-hydraulic systems. b) CEA has also developed a method for the modelling of successive time phases with different success criteria for the same safety systems, in Fault Tree Linking PSA tools (as Risk Spectrum for example). The PSA supporting software used by the CEA is Risk Spectrum.
ENEA-Italy	N/A
ONR-UK	N/A
IRSN-France	N/A
JAEA-Japan	There are no special methods and tools developed for a probabilistic analysis (e.g., event tree/fault tree method). However, accident simulation codes for SFRs have been developed and improved, which are used in determining success criteria of safety systems or in developing event trees of level-2 PSA: e.g., Super-COPD (Sagayama, Tanji, & Endo, 21-23 March 1983) (Ohtaki & Ohira, December 1990), SAS4A (Tentner, et al., 21-25 April 1985), SIMMER-III (Kondo, Tobita, Morita, & Shirakata, 1992) (Tobita, et al., March 2006), CONTAIN/LMR (Miyake, Seino, Takai, & Hara, 25-29 October 1992). JNES has developed a new approach called the Phenomenological Relation Diagram (PRD) that is an evaluation technique designed to determine the probability distribution in phenomenological event trees, which are usually constructed in the level-2 PSA (Haga, Endo, Ishizu & Iizuka, December 2009).
KAERI-Korea	No special PSA methods and tools are used for advanced reactors except passive system reliability analysis at this moment. APR1000 and APR1400 follow the same approach used in the operating NPPs. Thus, no special PSA methods and tool are used for APR1400 and APR1000. However, it is necessary to develop the method of HRA during low power and shutdown mode because accident sequences are heavily dependent on the operator actions. Digital I&C reliability analysis becomes a big issue for APR1400 and APR1000. Korea is developing a method for the Digital I&C reliability analysis, but it is not applied to the APR1400 and APR1000 yet. Korea is planning to apply new Digital I&C reliability analysis method to the SMART in few years. We are doing a PSA for SFR based on the typical PSA methodology. We have concluded typical PSA methodology is applicable for the SFR. Thus, the same method and tools can be applied to the SFR PSA.

	For the VHTR PSA, the situation is different. The definition of core damage is hard to be applied for the VHTR. We need a different approach to combine the Level-1, 2 & 3 PSA. The passive system reliability analysis is required for the SFR and VHTR PSAs. We have developed software for the passive system reliability analysis, called MOSAIQUE. We have tested the software and methodology for a passive system of VHTR. It will be described in Section C.
VTT-Finland	Refer to Answer by STUK
INET-China	From the recent practice of HTGR PSA development, it shows that generally the traditional PSA methods (ET/FT) are applicable to HTGR PSA. For some special issues such as passive system reliability and initiating event frequency estimation, new methodologies are developed, e.g. Monte-Carlo Simulation, Dynamic Discrete Event Tree and so on, with the help of commercial mathematical software such as MATLAB.

1.5 Does your country have any on-going research projects, or plans for research projects, regarding PSA methods, models, tools, or data aimed at or relevant to advanced reactors? If so, please provide a brief, summary-level description. (Note that Section C of this questionnaire requests more detailed information on this subject.)

Organization	Responses
BEL V, TE - Belgium	No
NRC-USA	NRC has a number of research projects that are aimed at or may be relevant to advanced reactor PSA. These projects are summarized below. <u>Advanced Reactor Research and Development Plan</u> NRC has developed an advanced reactor research and development (R&D) plan to support the licensing of the NGNP prototype plant. The plan outlines R&D activities in different technical disciplines that are needed to support the staff's review of a VHTR licensing application. The technical disciplines addressed in the R&D plan include: a) Plant safety analysis including thermal-fluids, nuclear analysis, and accident analysis; b) Fuel performance and fission product release and transport; b) High temperature materials performance; d) Graphite performance; e) Safety issues related to process heat applications; f) Structural analysis; g) Digital instrumentation and controls; h) Human factors analysis; i) Probabilistic risk assessment and risk-informed and performance-based licensing infrastructure; The final item in the above list outlines NRC's research tasks that are specifically developed to address advanced reactor PSA. The research tasks in the other technical disciplines, while not specifically directed at PSA, may also influence the development of advanced reactor PSA guidance, methods, tools, and data. <u>Advanced Reactor PSA Planning Study</u> As part of the advanced reactor R&D plan, the NRC staff is currently performing a

	planning study to identify the research needs to determine PSA technical acceptability for a VHTR licensing review. This study involves the following tasks: a) Review literature to understand VHTR safety issues and past operational experience; b) Identify the guidance needed to address the expected expansion in advanced reactor PSA scope with respect to current applications of operating LWR PSA; c) Identify the guidance needed to assess the technical acceptability of a PSA for a reactor in the design phase rather than a PSA for an operating reactor; d) Identify any new and advanced methods, models, tools, and data needed for review of a VHTR PSA; e) Assess the feasibility of creating a scoping-level PSA model to support the ongoing identification, prioritization, and selection of advanced non-LWR research topics. The outcome of this planning study will influence the direction of future research tasks that may include the development of advanced reactor PSA guidance, methods, models, tools, and data. In addition to the research projects being pursued under the advanced reactor R&D plan, NRC has established the following projects that may be relevant to advanced reactor PSA. <u>Digital Instrumentation and Controls Reliability Models</u> NRC is pursuing research with the objective of identifying and developing methods, analytical tools, and regulatory guidance for (a) including DI&C system models into nuclear power plant PSAs and (b) using information on the risks of DI&C systems to support NRC's risk-informed licensing and oversight activities. NRC is now performing a review of quantitative software reliability methods and will attempt to develop one or two technically sound approaches to modelling and quantifying software failures in terms of failure rates and probabilities. Assuming such approaches can be developed, they will then be applied to an example software-based protection system in a proof-of-concept study. <u>Advanced Modelling Techniques for Level 2 and Level 3 PSA</u> NRC has undertaken a long-term research initiative to explore the potential benefits of applying advanced techniques in the areas of Level 2 and Level 3 probabilistic risk assessment. Current efforts are focused on dynamic PSA methods utilizing a phenomenological accident simulator in conjunction with human response modelling. This effort expects to leverage off of similar work being conducted in academia to support operating reactor issue resolutions and high-temperature gas/liquid metal reactor design and licensing.
SNSA-Slovenia	No
STUK-Finland	No
SUJB-Czech Republic	No
UJD-Slovakia	No
CEA-France	CEA has developed a specific approach to: a) Build a reliability database for components of innovative reactors; this approach allows to taking into account the possible similarities between components of other

	reactors, expert judgment or standard industrial values. b) Assess Human Reliability at a pre-conceptual design phase; this approach uses existing HRA methods, but takes into account operational complexity of the considered reactor.
ENEA-Italy	Italy is running research activities on passive system reliability, but not related to a PSA of a specific reactor.
ONR-UK	No ongoing research specifically for this. In terms of plans for research, the GDA process has highlighted the possible need for research in the area of human reliability analysis.
IRSN-France	N/A
JAEA-Japan	a) Reliability data collection for the sodium-fluid components (related to Category 3.C.) b) Use of PSA in design (related to Category 2.C.) c) Implementation of PSA for Advance Reactor (Joyo, Monju) (related to Category 2.A.) d) Development of severe accident evaluation technology (level 2 PSA) for SFR (related to Category 2.B.) e) Development of plant thermal-hydraulic dynamic code f) Development of FP transports code and its validation with experimental data g) Development of core disruptive analysis code h) Development of containment vessel response code i) Preparation of detailed thermal-hydraulic analysis tool for LMFBR j) Research for the level-2 seismic PSA k) Research for statistical safety analysis
KAERI-Korea	Korea has several research activities and plans for advanced reactors' PSA. We can divide these research activities into 2 categories. One is the common technology for evolutionary reactors and GEN-IV reactors. Those topics are: digital I&C reliability and human reliability analysis under the digital I&C environment, improved seismic PSA methodology, the integrated approach of all modes and all scope PSAs (integration of Level-1, 2 & 3 and integration of internal and external PSAs), and the passive system reliability (uncertainty analysis of thermal hydraulic safety analysis). There is one additional research activity for APR1000 and APR1400 PSAs, especially for aircraft crash risk. SFR is being designed under long term project. Currently, we are performing the pilot study for the SFR PSA to verify whether or not the typical PSA method is applicable for the SFR. The PSA is limited to the internal Level-1 PSA. Korea is planning to submit the safety analysis report to the regulatory for DC in 2017. Thus, we have to perform the full scope PSA for the SFR. We expect that more research are required for digital I&C reliability analysis, human reliability analysis under digital I&C environment, seismic risk analysis for the passive decay heat removal system, and passive system reliability. We expect that some topics are hard to be resolved, that are the reliability data for the new component type due to the lack of experience and the level-2 PSA due to the uncertainty in the severe accident analysis Since the Korean VHTR is in an early state of the development, the research plan for PSA focuses on the preparation of PSA methods and procedure. A specific PSA procedure for VHTR has been studied as described in question 4. Additionally, we are

	interested in specific research topics relevant to this reactor. One is a reliability of passive safety system. The other is an evaluation of interfacing events risk between nuclear facility and hydrogen production facility. The regulatory has a plan to develop the PSA review guidance for the licensing purpose of SFR/VHTR.
VTT-Finland	Gen IV aspects have not been addressed in PSA-related research activities.
INET-China	The designer had successfully proposed a research project regarding the PSA methodology for HTGR within the framework of "Important National Science & Technology Specific Projects" in 2008. The original intention of this project was to serve the PSA development for full power operation only, including safety goal of HTGR, technical framework for HTGR PSA, initiating events analysis, reliability data collection and evaluation, and passive system reliability assessment methodology. More difficulties during the LP/SD or external event PSA haven't been covered in this project due to the insufficient understanding at that time.

2 General questions on issues

2.1 What do you think are the essential technical issues that should be addressed to support the effective, confident use of advanced reactor PSAs in risk-informed decision making? Please discuss in sufficient detail to enable the reader to fully understand why the identified issue is an issue, and why the issue needs to be addressed.

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	N/A
SNSA-Slovenia	For the upcoming LWR designs (with the exception of a more distant single phase high pressure LWR design) there seem to be no essential technical issues. For advanced non-LWR reactors those seem to be mostly in the area of implementation of PSA (topic 2A in the questionnaire's appendix) – methodologies, initiating events and definition and analysis of accident (and severe accident) sequences. These are not known in sufficient detail (and could not be known until (most of) designs are not at least a bit more mature than now and corresponding experiments and theoretical (deterministic) analysis completely finished) yet they are prerequisites for having meaningful PSA results.
STUK-Finland	a) Reliability and failure modes of passive safety systems b) Treatment of diverse systems in PRA c) Computer based I & C
SUJB-Czech Republic	a) Treatment of passive features in PSA, e.g. credibility of passive functions, probability of failure of passive components. b) Treatment of risk contributors usually not considered in the current PSAs: It is expected that new plants would have very low CDF/LERF from the "traditional" sources of the risk, such as component failures, CCFs and human failures (almost

	exclusively EOMs). Some potential contributors, such as ECOMs (both as an IE and in the plant response to IE), unintentional violation of Tech Specs (coincidence of repair of redundant equipment), etc., are usually implicitly considered negligible in the current PSAs. But the relative significance of those contributors may increase for NPPs with very low CDF/LERF, unless advanced reactors would have specific built-in features to prevent them (ECOM, etc.) as much as possible.
UJD-Slovakia	N/A
CEA-France	The main technical issues to be addressed are: a) 1.B: Essential feature to define advanced reactors: There must be no "shadow area" regarding the architecture and components of the reactor; in particular, it is very important to have a precise knowledge of components that are difficult to design or are not designed yet (for example: check valves in hot helium environment). b) 2.A: Estimating of Initiating Events: Which approach must be privileged to estimate the initiators frequency? * Experience feedback or development of specific reliability models; * Consistency with deterministic safety approach; c) 2.A: Definition and Analysis of Accident Sequences: As design of the considered reactor progress, it is necessary to get rid of the uncertainties regarding physical consequences of particular sequences. d) 2.B: Severe Accidents Analysis & Consequential Analysis: The Level 2 PSA modelling requires estimating and propagating the uncertainties regarding physical phenomena occurring after core damage. e) 2.C: Use of PSA in design: Developing a PSA in support to the design of a reactor has appeared to be very useful to enhance safety of a reactor; nevertheless, in order to implement this approach as early as possible, the systems have to be sufficiently designed. f) 2.D: Regulatory aspects on PSA Framework for Advanced Reactors: It would be useful to obtain a stronger exchange, especially in tools and data, with the regulatory bodies regarding probabilistic safety approach. Since PSA objectives are required or recommended, the regulators should follow them as close as possible. g) 2.E: Need for PSA Standard: It would be necessary for Level 2 PSA, but it appears to be very difficult since it is specific to the reactor technology. h) 3.B: Reliability of Passive Safety System: Since Passive Safety Systems are considered to be important to enhance the safety level of reactor; a specific assessment of their reliability has to be performed. i) 3.B: Reliability of Digital I&C Systems: It is necessary to model Digital I&C Systems as deep as possible (including software reliability if possible). j) 3.C: Development of reliability Database for Advanced Reactors: It is difficult to choose appropriate reliability data for innovating components, for which there is few or even no operating feedback available.
ENEA-Italy	a) Reliability of Passive Safety System; b) Reliability of Digital I&C System; c) Level 2 PSA. All these issues are still open issues due to the great amount of uncertainties related to the relative assessment (especially as regards the first and the last issue).
ONR-UK	a) Application of human error probability estimation techniques to digital interfaces. b) Failure of passive components and structures now more important in advanced reactor designs. It is essential that passive components are adequately addressed in

	the PSA. c) Modelling of inter-system CCFs. While some types of simplifications in the PSA models may have been acceptable in the past for older reactors with higher core damage frequencies, they may not be justified for newer reactors, in particular to support use of PSA for decision making in design and operational matters.
IRSN-France	N/A
JAEA-Japan	a) We believe that the development of a component reliability estimation method and its relevant data is essential technical issues: in particular, passive system reliability is important such as the occurrence rate of sodium coolant boundary failure in sodium cooling system. The reason is as follows. SFR has passive safety features that it is possible to remove decay heat in a natural circulation of the sodium coolant without depending on active components such as pumps. This can bring about reduction of a sensitivity of failures in pumps and their support systems to the core damage frequency. In turn, it increases relative importance of reliability in the sodium-coolant boundary (e.g., vessels, pipes) constituting cooling path in the decay heat removal system. In general, it is thought that the occurrence rate of static failures is lower than that of dynamic failures; however, the cumulative operating time of SFR components is less than that of LWR ones and is not so much that we cannot say that the occurrence rate of the static failure is sufficient low. Rather, many countries have experienced sodium leakage event due to the coolant boundary failure. It suggests that the reliability in the coolant boundary might not be so high than that expected at the present time in the development phase. In order to demonstrate high reliability of decay heat removal in a natural circulation mode, it is important to accumulate the safe operation record as long as possible in the sodium cooling system. In addition, it might be necessary and effective to study on reliability estimation methods with a different approach: e.g., application of the probabilistic fracture mechanics. b) We believe that basic ideas and requirements in LWR PSA methodology are applicable to SFRs. However, which portion of the models and data in PSA for SFR systems should be treated carefully in detail might differ from LWR systems. It may also depend on the level of maturity in SFR's technology from development to commercial operation. SFR technologies are still under development even though the SFR Joyo has much successful operation history. So, accumulation of PSA implementation/application in SFR systems is also an essential technical issue. c) With regard to the safety analyses against a severe accident of SFRs, computational tools such as SAS4A, SIMMER-III, DEBNET, ARGO and APPLOHS had already been developed. However, these tools were insufficient to systematically assess the whole sequence of core disruptive accidents (CDAs) because the analytical methodologies in the following phase/sequence would still involve large uncertainties: 1) the core-material relocation phase and 2) the ex-vessel accident sequence. In addition, 3) a technical basis for Level 2 PSA should be built up to construct the phenomenological event trees and to determine the branch probabilities in them. Therefore, the three issues are also an essential technical issue: i.e., to establish the analytical methodologies in the core-material relocation phase and the ex-vessel accident sequence to consolidate the Level 2 PSA for SFRs,

	and to build up a technical basis for Level 2 PSA to construct the phenomenological event trees and to determine the branch probabilities in them. (Nakai, et al., 10-14 May 2009) d) One-through analysis by one code from the initiating phase to the transition phase in the core disruptive accident will be favourable. If the initiating phase is solved three-dimensionally and the transition phase is solved two-dimensionally, some geometrical transfer will be needed. In the process, any human judgment will be included resulting in causing an uncertainty band.
KAERI-Korea	Generally, the state-of-the-art in PSA for GEN-IV reactors is less mature than that for LWRs. The GEN-IV reactors have insufficient experience in the development of PSA methodology. It seems that most of PSA models developed for GEN-IV reactors up to now were insufficient for risk-informed decision making. However, we believe that the PSA can be used for risk-informed decision making with some limitation. PSA quality is very important for risk-informed decision making for any reactor type. The following table describes technical issues what we want to address in PSAs for advanced reactors.

Topic	Problem to be solved
Selection of Initiating Events	We have very limited information what kinds of initiating events may occur for GEN-IV reactors. And we do not have any experience to estimate the frequency for new type of initiating events.
Safety analysis for developing event scenario	Current methods and tools are insufficient to support the safety analysis to develop the event scenario, especially for the GEN-IV reactors. This issue is considered as a part of safety analysis, beyond the PSA. Currently, uncertainty in safety analysis becomes an important role of advanced PSA. We need a method to incorporate the uncertainty from safety analysis into PSA.
Passive system reliability	New reactors extensively adopt the concept of passive systems and inherent features to ensure safety, rather than relying on the active safety systems used by current LWRs. So, the passive system reliability will be an important issue for advanced reactors and especially GEN-IV reactors. The passive system reliability is closely related to the uncertainty of the safety analysis.
Reliability data	We have a little database accumulated for GEN-IV reactors due to the lack of operating experience. Actually we do not have any experience for new designed component. We need to develop a kind of methods to estimate the reliability for those components.
Digital I&C reliability	Digital I&C reliability analysis is very important for the new reactors.
Human reliability analysis	A sound HRA method under digital I&C environment should be developed. We need more research on the HRA under the low power/shutdown mode and extreme case like seismic event.
External event analysis	There is too large uncertainty in seismic risk assessment with the current seismic PSA method. The method and tool

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<tr><td></td><td>are required for new type of external events such as the aircraft crash.</td></tr>
<tr><td>Level 2 & Severe accident analysis</td><td>Current methods and tools for the severe accident analysis are insufficient to support the Level-2 PSA.</td></tr>
<tr><td>Level 3 & Consequence analysis</td><td>There is too large uncertainty in Level-3 PSA.</td></tr>
<tr><td>Consensus of PSA technology</td><td>The consensus of PSA technology in stakeholders is essential technical issues. While traditionally, regulators as well as designers do not fully agree to the PSA results and insights, the adoption of PSA technology to safety goal establishment of nuclear power plants should be consented by them.</td></tr>
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<td>VTT-Finland</td>
<td>Is there need to reconsider risk criteria concepts such as core damage frequency, large (early) release frequency? Is the level 1/level 2 PSA modelling framework some way different for new reactor concepts?</td>
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<td>INET-China</td>
<td>Essential technical issues that should be addressed during the risk-informed decision making may include:

a) Definition of Risk Measures for Advanced Reactor (e.g., VHTR): It is easy to understand why this issue shall be an issue. Without the clear definition of risk measures, PSA cannot be performed.

b) Initiating Events Analysis (IE): The definition level of initiating events is the first major problem in this area. Because initiating event analysis will not serve the PSA only, but also are expected to provide inputs to the deterministic safety analysis, especially for the reactors which don’t have enough operation experience such as HTGR. The definition level of initiating events is not consistent between probabilistic and deterministic safety analysis, even within the existing probabilistic safety analysis. System level and component level failures are often used simultaneously, initial faults and initiating events are more or less mixed and may not be able to be distinguished clearly, and how to deal with the multiple failures are greatly disputed especially when using PSA to support the deterministic safety analysis regardless the low occurrence frequency. It is not always the case that extreme cases are proposed by PSA. How to estimate the initiating event frequency is the second major problem in this area, and the influence from these estimations on the PSA result is much great than that from the hardware reliability data. New identified initiating events will have to estimate its frequency by expert judgment or other subjective quantification methods without any operation data. Development of a systematic and rational approach for initiating event frequency estimation shall be taken for great importance, and is in urgent need.

c) Human Reliability Analysis (HR): HRA is an important and controversial aspect in PSA. Although 1st generation HRA methods such as THERP, HCR are commonly used in LWR PSA, the basis of human behaviour theory determines the limited capability of these methods with respect to the unintended human actions as well as the potential errors in the complex accident scenario. Accident sequences of advanced reactor may have even more features which cannot be well characterized by the current HRA methods, e.g. long time window, digital-based human-machine interface and main control room design.</td>
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	d) External Plant Hazards Analysis: The reason that external hazards analysis is selected as an issue is mainly due to the large portion of the unknown of interest when compared with the internal events analysis. The insufficient knowledge concerning the external hazards themselves such as earthquake, tornado and tsunami will subsequently lead to the large uncertainty of the prospective responses the plant may have. Over conservative approach may not be so welcome because it will greatly conceal the advantages of advanced reactors.

2.2 What are the essential regulatory issues that need to be addressed to enable the use of advanced reactor PSAs in your country's licensing and regulation processes?

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	NRC has participated in a number of activities that have allowed the staff to recognize and discuss the issues related to the development of advanced reactor PSAs. These activities include participation in the development of PSA standards, meetings and information exchanges with potential advanced reactor applicants and industry representatives, and NRC-lead research programs. Through these activities, NRC has identified several issues that should be addressed to support the use of advanced reactor PSA in risk-informed decision-making. Descriptions of the challenges associated with each issue are provided below. Many of the issues present both technical and regulatory challenges. a) The use of PSA consensus standards and independent peer reviews against the standards are important factors in establishing PSA quality. Currently, appropriate PSA standards for U.S. advanced reactors have not been finalized. The absence of appropriate standards presents regulatory challenges to determining the technical acceptability of advanced reactor PSAs. If appropriate standards are not developed prior to submittal of design certification and COL applications, then a suitable method of determining PSA technical acceptability must be established. A method of performing a PSA peer review, in the absence of a PSA standard, also must be established to support the determination of technical acceptability. b) The development of PSA technical elements for advanced reactors must be adequate to justify the use of the PSA to support regulatory programs. To ensure technical acceptability the PSA must use systematic approaches to initiating event selection, event sequence development, end state definitions, and risk metrics. For advanced reactor designs, this may require consideration of non-traditional scenarios. Examples include radiological releases from graphite dust for HTGR/VHTR designs and risk contributions from routine operations (i.e., non-accident conditions). c) The development of appropriate risk metrics for advanced reactor PSA is necessary to support regulatory programs. NRC has several programs in place that rely on PSA results for risk-informed decision-making. The risk metrics CDF and LERF are used as the basis for decisions and are specifically referenced in NRC guidance documents. To support risk-informed decision-making for advanced reactors, appropriate risk metrics must be defined.

	d) The use of a PSA for a plant in the design phase can be quite different than that for an operating plant. NRC is considering how a PSA for a plant in the design phase should be used for regulatory applications such as the selection of licensing basis events (LBEs) or safety classification of SSCs. The appropriate use of the PSA must consider that assumptions are made about plant design and operation during the design phase. Relevant reliability data and operational experience may be lacking, which may necessitate reliance on expert judgment. Proper consideration of these issues and adequate treatment of uncertainties must be made if a PSA for a plant in the design phase is to be used for regulatory applications. e) Many advanced reactor designs have incorporated the use of passive systems, structures, and components to perform safety functions. Yet, insufficient information may exist about the reliability and functional capability characteristics of the passive systems, structures, and components, which may necessitate increased use of expert judgment in the PSA. The reliance on expert judgment may not provide an adequate technical basis to support the use of the PSA for regulatory decision-making. f) For advanced reactors with designs featuring multiple reactor modules, guidance and methods are needed to address how accidents affecting multiple modules should be included in the PSA. Criteria must be developed for determining what accident conditions require considering multiple model effects. Acceptable methods must be developed to perform severe accident analyses, calculate source terms, and determine doses for scenarios involving multiple modules. g) Some features of advanced reactor designs, such as reliance on passive systems, advanced human-machine interfaces, and modularized reactor facilities, will require the development of new concepts of operations and staffing requirements. Considering these features, the development of acceptable methods for assessing human reliability for advanced reactor PSAs may pose technical challenges. h) The use of scenario-specific mechanistic source terms has been proposed for advanced reactor PSA. The development of mechanistic source terms requires that there is sufficient understanding and assurance of plant and fuel performance. Demonstrating that methods to determine the mechanistic source terms are adequate for their purposes could present significant technical challenges. i) The advanced reactor designs have proposed different concepts for containment structures than those of currently licensed U.S. plants. For example, non-pressure-retaining buildings could be used for advanced reactors using advanced fuel types designed to contain fission products. For multiple module facilities, non-traditional small containment structures for each module could be used. The development of acceptable methods to incorporate the containment performance in the PSA is needed for these different containment concepts. j) Considering the intended scope of advanced reactor PSA (e.g., the inclusion of hazards from internal fires and seismic events), concerns have been raised about the appropriate way to aggregate the risk from different contributing hazard groups that may be modelled with different levels of detail. In addition, the development of an acceptable method of assessing the risk due to seismic hazards is necessary for

	advanced reactor PSA. Further development also may be required to establish technically acceptable methods to incorporate fire hazard analysis into PSA models. k) Until such time that significant plant operating experience and PSA experience is established for advanced non-LWR designs, it is expected that conservative deterministic engineering judgment will be needed to compensate for PSA uncertainties and unknowns. The approach to establish advanced non-LWR licensing will use deterministic engineering judgment and analysis, complemented by PSA information and insights. The use of the PSA will be commensurate with the quality and completeness of the PSA presented with the licensing application.
SNSA-Slovenia	The regulation in our country is kept as general as possible. There are none really essential regulatory issues as long as one can have confidence in the PSA and the potential risks of advanced reactors are well understood and quantifiable. Nevertheless some fine tuning of the regulations will be necessary especially for non-LWR designs. It is felt that there is enough time for this as such designs are not yet in a wide commercial use. And as always, a prerequisite for use of PSA in the regulatory domain is existence of corresponding PSA standards and QA programs.
STUK-Finland	The technical issues mentioned in Answer B.1 are also regulatory issues. Specifically regulatory issues have not been identified.
SUJB-Czech Republic	Instead of the fact that PSA studies are elaborated and used for both Czech NPPs, it is necessary to include PSA and its applications into legislation of the Czech Republic. Further it is necessary to elaborate methodologies for the Safety Authority. These methodologies would describe approach of the Regulatory Body to the PSA applications and they would also include relevant acceptance criteria (something like US NRC Regulatory Guides 1.174, 1.175, 1.176, 1.177, 1.178).
UJD-Slovakia	N/A
CEA-France	There is a need for a clear position and a harmonization of the world-wide regulatory requirements upon PSA for advanced reactors. This would facilitate exchanges between regulatory bodies and reactor designers/utilities and would be beneficial to improve nuclear safety. In case of EDF: a) 2.A. There is a great interest but also important difficulties to combine probabilistic sequences and physical uncertainties (see SM2A project). b) 2.E. A Technology-Neutral Framework is certainly very interesting and also very difficult.
ENEA-Italy	N/A, since a PSA framework in a licensing and regulation process is not actually available.
ONR-UK	None specifically identified, although we need state of the art Level 1, 2 and 3 PSAs, which can deal with passive components appropriately.
IRSN-France	N/A
JAEA-Japan	The same grade of safety as LWR should be guaranteed if the nuclear plant generates electricity. Even if we had full understanding the peculiarity of the advanced reactor, no difference exists between advanced reactor and LWR for the local residents.
KAERI-Korea	The Korean regulation does not include the implementation and submission of PSA for the licensing of existing nuclear power plants. Instead of this, Korean Policy Statement on the Severe Accident of Nuclear Power Plants (2001) requires the licensee to perform and submit the Level 1 and Level 2 PSAs for existing nuclear power plants. For advanced reactors, the Level 1 and Level 2 PSAs are going to be submitted by the licensee and reviewed by the regulatory body in accordance with this previous practice.

	However, with respect to this, appropriate performance goals should be improved for the Level 1 and Level 2 PSAs of advanced LWRs. Designers and licensee voluntarily has a tendency to perform the Level 3 PSA for APR 1400 and SMART. So the regulatory needs to develop regulatory guidance on Level 3 PSA to determine whether the technical acceptability of this PSA is adequate. Also it should include appropriate criteria for meeting safety goals for the Level 3 PSA. Regarding the risk informed regulation for light water reactors, we want to discuss two things: 1) currently applied risk-informed regulations are limited such as RI-ISI, and the part of Risk-informed technical specifications. For example, RI-Tech. Spec.4b, 5b are still considered and Option 2 and Option 3 have a long way to go. 2) During the licensing and regulation processes, the essential regulatory issues on PSAs are "implementation of the state of the art technology" and "consistency with previous PSA technology and results." To resolve the issues, PSA standard is considered as a plausible solution. We need to develop more systematic framework and methodology for the various risk-informed regulations, which will be closely related to the quality and uncertainty in PSAs.
VTT-Finland	Refer to Answer by STUK
INET-China	Essential regulatory issues that need be addressed are: a) Regulatory Framework which allows the use of PSA during the licensing and regulation processes: It's better to be stated explicitly, e.g., documented standard or guidance or policy statement. b) Detailed understanding of regulatory staff to the advanced reactor PSA model: Whether the PSA quality is perfect enough to support the role it plays during the licensing and regulation cannot be determined simply by one or two criteria. Only the regulatory staffs know the PSA model quite well e.g. where are the limitations and big uncertainties, can they make the decision to credit the PSA insights or not. The judgment of PSA quality shall be closely bound with the application cases. Online regulatory review may be recommended as a good way to solidify the confidence for both sides from our point of view.

3 Research activities on PSA for advanced reactors

3.1 Technical issue(s) – please describe the issue or issues addressed by the research activity.

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	N/A
SNSA-Slovenia	N/A
STUK-Finland	N/A
SUJB-Czech Republic	N/A

UJD-Slovakia	N/A
CEA-France	There is no ongoing research activity in support to PSA methodology, but only PSA modelling activities in support to the design of advanced reactors.
ENEA-Italy	Passive system reliability; Aging PSA
ONR-UK	Use of THERP data may be questionable. Control room post-fault actions uses digital interfaces. This may be very different to the interfaces assumed in the THERP data-sets.
IRSN-France	N/A
JAEA-Japan	a) Reliability data collection and reliability parameter estimation, in particular, associated with sodium-related components b) Use of PSA in design (in the FaCT project) c) Use of PSA as part of safe operation of the existing SFRs (Joyo, Monju) d) Development of severe accident evaluation technology (level 2 PSA) for SFR e) Development of plant thermal-hydraulic dynamic code f) Development of FP transports code and its validation with experimental data g) Development of core disruptive analysis code h) Development of containment vessel response code i) Preparation of detailed thermal-hydraulic analysis tool for LMFBR j) Research for the level-2 seismic PSA k) Research for statistical safety analysis
KAERI-Korea	N/A
VTT-Finland	N/A
INET-China	Technical issue of the Reliability of Passive Safety System is included in the HTGR research project.

3.2 PSA topic area(s) – please identify the relevant topic area or areas (see appendix). If the activity addresses a topic area not identified in the appendix, please provide a brief (title-level) description of the area.

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	N/A
SNSA-Slovenia	N/A
STUK-Finland	N/A
SUJB-Czech Republic	N/A
UJD-Slovakia	N/A
CEA-France	N/A
ENEA-Italy	Passive system reliability; Aging PSA
ONR-UK	Human Reliability Analysis for Advanced Reactors.
IRSN-France	N/A
JAEA-Japan	a) Development of Reliability Database for Advanced Reactors b) Use of PSA in design

	c) Implementation of PSA for Advanced Reactor d) Scope of PSA for Advanced Reactor (Severe Accidents Analysis & Consequential Analysis) e) Initiating Event Analysis and Event Sequence Analysis f)-j) Event Sequence Analysis k) Seismic PSA
KAERI-Korea	N/A
VTT-Finland	N/A
INET-China	PSA topic areas included in the HTGR research project currently are Initiating Events Analysis (IE), Systems Analysis (SY), Data Analysis (DA), and Definition of damage states (i.e., core damage). We are going to propose a new research project on Seismic PRA (SP) recently.

3.3 Technical approach – please provide a summary description of the technical approach planned or being used to address the issue(s).

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	N/A
SNSA-Slovenia	N/A
STUK-Finland	N/A
SUJB-Czech Republic	N/A
UJD-Slovakia	N/A
CEA-France	N/A
ENEA-Italy	Development of methodologies for evaluating the reliability of thermal-hydraulic passive systems and its inclusion in PSA (e.g. event trees). Inclusion of ageing effects in the reliability models in PSA.
ONR-UK	Issue only recently raised therefore no detailed approaches have been developed. This also applies to the following questions.
IRSN-France	N/A
JAEA-Japan	a) Collection of reliability data required for reliability parameter estimation such as component operating time records and failure instances, in particular, in sodium fluid systems. The data is stored on a computerized relational database management system; the database is named CORDS (Kurisaka, June 1996). Data sources are the Japanese existing SFRs Joyo and Monju, the US SFRs EBR-II and FFTF, and sodium-related test facilities in Japan and US. Reliability parameters are estimated with statistical analysis (e.g. Bayesian method). b) Development of PSA models and parameter for the JSFR system at a level in detail corresponding to progress in the system design work: Utilization of PSA results in confirmation of the level of safety.

	c) Referring to PSA procedures and technical approaches in utilization of "risk information" applied to LWRs, their applicability to SFRs Joyo and/or Monju is examined. d) As for the core-material relocation phase, the MUTRAN code is being developed in order to evaluate the long term behaviour of the materials remaining in the core. The time range to be simulated by MUTRAN is from several hours to several dozen hours. In case of approaching mild re-criticality as the result of the material motion, the material distribution is transferred to the SIMMER-LT code to continue the analysis with calculation of neutronics which is not incorporated into MUTRAN. The development of SIMMER-LT is based on the framework of SIMMER-III. To apply SIMMER-LT to the core-material relocation phase, computational efficiency should be advanced for analyses of rather long-term transients of up to several dozen minutes. Analytical models important in the analysis of the ex-vessel accident sequence are the models CORCON for debris-concrete interaction and VANESA for aerosol and fission products release associated with the debris-concrete interaction, which are installed in the CONTAIN/LMR code. Since these models were originally developed in order to estimate the ex-vessel accident sequence of LWRs, they are not applicable to SFRs. Hence, CORCON and VANESA are being improved taking into account the influence of sodium-pool existence. As for a technical basis for Level 2 PSA, the dominant factors in each phase of SFR's severe accident have been identified through sensitivity analyses. The information obtained from these analyses will be effectively reflected on the technical basis which is indispensable for the construction of phenomenological event trees. (Nakai, et al., 10-14 May 2009) (Sato, Yamano, & Tobita, 10-14 May 2009) (Tobita, Yamano, & Sato, 10-14 May 2009) (Koyama, Yamada, Hayakawa, Watanabe, & Watanabe, 10-14 May 2009) (Ohno, Seino, & Miyahara, 10-14 May 2009). e) Development of plant thermal-hydraulic dynamic codes The ADYTUM code is being developed. The code has so much flexibility as to be able to treat the free surfaces of coolant in vessels. The function enables us to analyze the sodium leak flow-rate in the case of the coolant boundary failure as well as the coolant level displacements in an evaporator and a super-heater in the steam generator system. A series of steam blow-down test is scheduled during the start-up tests of Monju. The preparation for the analysis of blow-down behavior is being performed by using RELAP5 mod3 (Shindo, Y., Endo, H., Inoue, M., 7-11 December 2009). f) Development of FP transport code and its validation test The developed ACTOR code analyzes the behaviour of fission products (FPs) that are released from the fuel plenum or fuel pellets through a fuel cladding breach. The validation tests that have two objectives just started. The first objective is to measure the cesium transferred to the sodium coolant before the cesium appears to the cover gas area. The second objective is to investigate the cover gas heating effect of released gaseous FPs to the cover gas. (Inoue, M., et al., 7-11 December 2009).

	g) Development of core disruptive analysis code An integrated core disruptive accident analysis code, ASTERIA-FBR, is being developed. Now, we are analyzing the core disruptive accident by connecting the pseudo-three dimensional analysis by SAS4A for the initiating phase (IP) with the two-dimensional analysis for the transition phase (TP) by SIMMER-III. Such connection inevitably introduces the following difficulty: - In what timing the results of initiating phase analysis by SAS4A should be transferred to the transition phase analysis by SIMMER-III? - How the fuel-subassemblies should be lumped for the two-dimensional calculation by SIMMER-III; - Continuity between the analytical models; - Continuity in space-time neutronics. To solve these problems JNES started the development of ASTERIA-FBR that analyzes the IP and TP continuously. ASTERIA-FBR consists of a fluid-dynamics calculation module equipped with multi-phase multi-function model, a space-time neutronics module and fuel pin behaviour calculation module. h) Development of containment vessel response code. The containment vessel response code, AZORES, is being modified in the following points; including core melting process model, including a simple primary heat-transfer system model, and including a re-criticality model of molten core (Kawabata, H., Endo, H., Haga, K., 7-11 December 2009). i) Preparation of detailed thermal-hydraulic analysis tool for LMFBR One of the objectives of this project is to fabricate a CFD code with a function of sodium-water reaction analysis. The high-temperature rupture phenomenon of steam generator heat transfer tubes is so complicated due to the combined effects of thermo-hydraulic and chemical events. We considered preparing the analysis tool by adding a function to solve the sodium-water reaction to a CFD (Computational fluid dynamics) code. j) Research for the level-2 seismic PSA It is required in Japan by the regulatory authority, even if the seismic standard was fulfilled in a nuclear plant, to valuate the residual risk. From the point we started the research plan for the level-2 seismic PSA. k) Research for statistical safety analysis Till now, a deterministic analysis approach has been performed on accident progression. However, in many cases the sequence of events in accidents is not unique. Further more, there would be ranges in input parameters. To these situations, a statistical approach would be favourable in the safety analysis. At present, a Continuous Markov Chain Monte Carlo (CMCMC) method has been applied to the plant dynamic analysis. The results gave a rational background for probabilities of blanching points. This CMCMC method is widely applied to PSA.
KAERI-Korea	N/A
VTT-Finland	N/A
INET-China	For the reliability of passive safety system, Monte-Carlo based approach is going to be used with the integrated process of thermal-hydraulic calculation and reliability

	evaluation.

3.4 Lead organizations – please identify the lead technical organization and the sponsoring organization

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	N/A
SNSA-Slovenia	N/A
STUK-Finland	N/A
SUJB-Czech Republic	N/A
UJD-Slovakia	N/A
CEA-France	N/A
ENEA-Italy	European Union and IAEA for the passive system reliability. European Union for the aging PSA
ONR-UK	N/A
IRSN-France	N/A
JAEA-Japan	a) through d): Japan Atomic Energy Agency (JAEA) e) through k): Japan Nuclear Energy Safety Organization (JNES)
KAERI-Korea	KAERI
VTT-Finland	N/A
INET-China	Institute of Nuclear and New Energy Technology in Tsinghua University (INET) is the lead technical organization. The research project is funded by the National Energy Administration.

3.5 Milestones and status – please identify key milestones (if established) and the current status of the activity. If milestones have not been established, please provide the expected, general timeframe for the activity.

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	N/A
SNSA-Slovenia	N/A
STUK-Finland	N/A
SUJB-Czech Republic	N/A

UJD-Slovakia	N/A
CEA-France	N/A
ENEA-Italy	The objective is to harmonize the different approaches that have been developed at a single country level, for passive system reliability, and to work out the still open points related to the issue. The objective is the implementation of the PSA models with the aging effects of systems and components.
ONR-UK	N/A
IRSN-France	N/A
JAEA-Japan	a) Long-term activity: i.e., A current effort of the data collection is focused on Japanese existing SFRs Joyo and Monju, and data will be obtained from Joyo and Monju as long as those reactor systems continue their operation. b) By the end of JFY2010, a judgment whether promising innovative technologies are adopted in the JSFR systems or not is made; e.g., adequacy in adoption of the decay heat removal system concept that is operated in a fully natural circulation mode will be confirmed from various points of view and one of those viewpoints is the occurrence frequency of the core damage sequences via loss of decay heat removal. By the end of JFY2015, PSA models and parameter for JSFR will be developed at a level in detail corresponding to progress in the system design work. In each design phase, it will be confirmed by implementing a PSA whether the JSFR system can meet or not the safety design requirements for SFR in the FaCT project (Kotake, Mihara, Kubo, Aoto, & Toda, 2008). Current status: In the Phase II of the feasibility study until 2005, a level 1 PSA of the JSFR system concepts was implemented and the level of safety was confirmed; it also served to selection of a type of decay heat removal systems (Kurisaka, 15-20 Oct. 2006). c) It is expected to extend gradually an application field of PSA for a safety operation of Joyo and/or Monju, considering a priority according to accumulation of the operating experience (e.g., operating technique, maintenance technology) of the SFR system observing the trend of risk information practical use in domestic LWRs. Current status: In Joyo, JAEA implemented a level-1 PSA associated with internal initiating events in a full power operation to understand the safety characteristics in-depth (Ishikawa, et al., January 2009). In Monju, JAEA implemented a level-1 & -2 PSA associated with internal initiating events in a full power operation to assess the effectiveness of the accident management measures (JAEA Tsuruga Head Office, 2008); JAEA examined adequacy of the allowed outage time in Monju by considering risk indices related to the core damage frequency (Sotsu & Kurisaka, 12-16 July 2009). d) From JFY2006 to now in JFY2009, this research project has been carried out. In this project the analytical methodologies have been developed with the aim of establishing a severe accident evaluation technology for SFRs, and the dominant factors in each phase have been identified through sensitivity analyses. The information obtained from these analyses will be effectively reflected on the technical basis which is indispensable for the construction of phenomenological event trees. After the end of JFY2009, the outcome of this program will be applied

	for the level-2 PSA of JSFR system in research project item 2 that is described in this questionnaire section II.C. e) It will be finished by 2012. During the time, pre- and after-test analyses of steam blow-down operation are included. f), g), and i): It will be finished by 2012. h), j), and k): It will be finished by 2016.
KAERI-Korea	N/A
VTT-Finland	N/A
INET-China	The project will last for two years. In 2010, the phase achievement shall be examined.

3.6 Regulatory application – please describe the expected regulatory application of the products of the research.

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	N/A
SNSA-Slovenia	N/A
STUK-Finland	N/A
SUJB-Czech Republic	N/A
UJD-Slovakia	N/A
CEA-France	N/A
ENEA-Italy	It is advisable that the treatment of these issues is likely to provide insights for regulatory purposes.
ONR-UK	N/A
IRSN-France	N/A
JAEA-Japan	a) Reliability parameters estimated from those data and those data summarized serve to the common technology basis in order to understand PSA results of SFRs. b)-d) Accumulation of SFRs' PSA practices is a direct product of the research, which serves to becoming familiar with SFR's risk in-depth prior to regulatory application. Then it is expected to summarize useful information and to integrate its information into revised PSA-related standards for the purpose of helping both regulators and licensees in understanding how to implement a SFRs' PSA and how to utilize risk information in regulation of SFRs.
KAERI-Korea	N/A
VTT-Finland	N/A
INET-China	The products will be used during the FSAR review of HTGR few years later.

3.7 Other – please provide any other information that would be useful to readers of the report, and to the organizers of the April, 2010 workshop.

Organization	Responses
BEL V, TE - Belgium	N/A
NRC-USA	Advanced Non-LWR PSA Planning Study NRC has developed an advanced non-LWR PSA planning study to identify the research needed to ensure that the staffs have the proper knowledge and tools to support risk-informed licensing activities for advanced non-LWRs. The study addresses many technical issues related to PSA. Item 2 of the appendix of this questionnaire identifies many of the issues that this study is intended to address. The study is to perform a "gap analysis" to identify where guidance, methods, and data are needed to support the technical review of an advanced non-LWR PSA. This study shall identify specific research programs that should be put into place to address the identified areas where guidance, methods, and data are needed. The task shall involve review of the available literature to identify unique design and safety issues associated with advanced non-LWR (specifically VHTR) designs and then to identify if any of the issues could potentially impact the development of a PSA model. The task also involves a review of guidance and PSA standards to identify the need for new regulatory guidance considering the expected expansion in scope of advanced non-LWR PSA. This task will consider the use of PSA during the design, design certification, and licensing phases of the advanced non-LWR life cycle. In addition, because of the unique aspects of an advanced non-LWR design and because there is minimal to no direct operating experience, the methods needed to fully model the plant and the data needed to represent the plant performance may not be available. Consequently, the advanced features of VHTR designs (e.g., pebble bed fuel, fires due to high-temperature gas releases) may require new and advanced PSA methods and data to adequately assess plant risk. The task will identify the methods, models, tools, and data that need to be developed. The study also will perform a feasibility study for developing a scoping-level PSA for an advanced non-LWR. A scoping-level PSA is proposed to be developed to support the identification, prioritization, and selection of R&D topics using appropriate risk metrics that support advanced non-LWR review process. A major aspect of developing this type of PSA will be documenting the assumptions that are made to ensure development of a complete PSA. The study would identify the plant design information needed to develop a scoping-level PSA. In addition, the study would identify what methods, tools, and data are needed to develop and implement a scoping-level PSA. NRC has partnered with experienced PSA technical staff at Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL) to perform the advanced non-LWR PSA planning study. The schedule for completion of the study and transmittal of summary reports is November 2010. This study is not intended to have any direct regulatory application, but it will identify the type of research that should be established to support future regulatory decisions. Advanced Modelling Techniques for Level 2/3 PSA NRC has undertaken a long-term research initiative to explore the potential benefits of applying advanced techniques in the areas of Level 2 and Level 3 probabilistic risk

	assessment. To date, the effort has included an internal scoping study (available in NRC's ADAMS at ML091320447) focused on surveying the spectrum of technologies available and the initiation of a methods development project at Sandia National Laboratories. Current efforts are focused on dynamic PSA methods utilizing a phenomenological accident simulator (MELCOR) in conjunction with human response modelling. The planned demonstration problem for the project may elect to focus on an operating LWR. However, specific aspects of the effort may be of interest to the advanced reactor methods community. In particular, one focus of the effort is to address the consistent treatment of accident progression across the traditional PSA pinch-points (e.g., the Level 1/Level 2 interface). This aspect may be particularly important for advanced reactors where core damage may no longer be the preferred demarcation point. This effort expects to leverage off similar work being conducted in academia to support operating reactor issue resolutions and high-temperature gas/liquid metal reactor design and licensing. Digital Instrumentation Control Reliability Models NRC is pursuing research with the objective of identifying and developing methods, analytical tools, and regulatory guidance for (1) including DI&C system models into nuclear power plant probabilistic risk assessments (PRAs), and (2) using information on the risks of DI&C systems to support NRC's risk-informed licensing and oversight activities. Previous and current NRC research projects have identified a set of desirable characteristics for reliability models of DI&C systems and have applied various probabilistic reliability modelling methods to an example digital system. This work is documented in several NUREG/CR reports that have received extensive internal and external stakeholder review. Recently published reports include NUREG/CR-6962,"Traditional Probabilistic Risk Assessment Methods for Digital Systems"(October 2008); NUREG/CR-6985, "A Benchmark Implementation of Two Dynamic Methodologies for the Reliability Modeling of Digital Instrumentation and Control Systems"(February 2009); and NUREG/CR-6997,"Modeling a Digital Feedwater Control System Using Traditional Probabilistic Risk Assessment Methods" (September 2009). The results of these studies have been compared to the set of desirable characteristics to identify areas where additional research might improve the capabilities of the methods. One specific area currently being pursued is the quantification of software reliability. Given the lack of a consensus on how, or even if, to model software failure in a nuclear power plant PRA, NRC and Brookhaven National Laboratory (BNL) hosted a workshop involving experts with knowledge of software reliability and/or nuclear power plant PRA in May 2009. At the workshop, the experts established a philosophical basis for modelling software failures in a reliability model. The results of the workshop are documented in a BNL technical report (available in NRC's ADAMS at ML092780607). NRC (through a contract with BNL) is now performing a review of quantitative software reliability methods and will attempt to develop one or two technically sound approaches to modelling and quantifying software failures in terms of failure rates and probabilities. Assuming such approaches can be developed, they will then be applied to an example software-based protection system in a proof-of-concept study.
SNSA-Slovenia	N/A
STUK-Finland	N/A

SUJB-Czech Republic	N/A
UJD-Slovakia	N/A
CEA-France	N/A
ENEA-Italy	Passive System reliability has been addressed in Italy since late 90's and the research has continued in the last decade. Main focus has been on the development of methods for the evaluation of the passive systems (that is the ones resting on natural circulation). All these activities have been performed also within international collaborations (EU, IAEA, and OECD). Passive system reliability needs an international effort by research organizations, industries, regulatory bodies in order to reach a common consensus. Many articles are available in open literature about the topic.
ONR-UK	N/A
IRSN-France	N/A
JAEA-Japan	N/A
KAERI-Korea	As described in session A.5, we have several research programs. KAERI is performing a research project "Development of Integrated Assessment Technology of Risk and Performance" to develop a methodologies and tools for PSAs. The project period is from 2007 to 2011, which is funded by MEST (Ministry of Education, Science and Technology). The topics included in the project are: a) All modes all hazards Integrated PSA tool, that consists of integration of event tree and fault tree for Level-1 PSA, integrated approach for external PSA, integrated approach for shutdown PSA, integrated approach for Level-1 & 2 PSA; b) Statistical reliability evaluation software (uncertainty analysis software for thermal hydraulic safety analysis or severe accident analysis); c) Digital I&C reliability and Human reliability analysis under digital environment; d) Improved Seismic PSA methodology. We describe the following research activities in more detail in the supplements: a) Digital I&C reliability and Human reliability analysis under digital environment; b) Statistical reliability evaluation software and reliability analysis of passive safety system. KAERI is performing a research project for the SFR PSA. The first project has been performed from 2006 to 2009. In the project, KAERI has done the pilot PSA for SFR and concluded that the typical PSA method is applicable for the SFR. The PSA is limited to the Level-1 full power internal PSA. From 2010 to 2011, the second project will be initiated. The purpose of the project is to perform the Level-1 and Level-2 PSA for the SFR, and to use the PSA to assess the design alternatives and support the decision making in design process, that is a kind of risk informed design. All projects are funded by MEST. The SFR PSA is also described in more detail in the supplements: Overview of SFR PSA.
VTT-Finland	N/A
INET-China	N/A

4 International Cooperation

4.1 Are there any PSA-relevant topics that you believe would benefit from increased international cooperation? Please describe.

Organization	Responses
BEL V, TE - Belgium	Because we have no activities related to PSA for advanced reactors, we have no opinion on this point
NRC-USA	At this time, NRC has not identified any specific topics that would benefit from increased international cooperation. However, staff from NRC's Office of New Reactors has indicated that participation in the Multinational Design Evaluation Program (MDEP) has been useful in the review of new reactor applications. The MDEP discussions provided opportunities to share experience and understand technical issues encountered in the licensing process of other regulators. Having the same type of interactions focusing on advanced reactor designs may be beneficial.
SNSA-Slovenia	There is no PSA topic that would not benefit from international cooperation. One that could especially benefit from increased international cooperation is regulatory aspect on PSA – regulatory criteria, scope, QA requirements etc.
STUK-Finland	a) Reliability of passive safety systems b) Regarding Level 2 PSA, reliability issues connected with different core catcher designs (also relevant to evolutionary reactors).
SUJB-Czech Republic	Acceptability of probabilistic criteria for RIDM and/or PSA applications: Quantitative probabilistic criteria from US RG 1.174/1.177 may not applicable for new plants with very low CDF/LERF (almost all changes would be allowed). Quantitative probabilistic criteria commonly used in PSA applications, such as risk monitor, RI-ISI, event analysis, etc. should be assessed (generally) to be applicable to very low baseline CDF/LERF.
UJD-Slovakia	N/A
CEA-France	The following PSA topics would benefit from international cooperation: a) Harmonization of the regulatory requirements regarding PSA (safety goals); b) Sharing of reliability databases for components and initiators specific to the different reactor concepts; c) Development of a methodology in order to take into account severe accidents as early as possible during the pre-conceptual design phase. In case of ASN : Common cause failure modelling and quantification
ENEA-Italy	Passive system reliability, although there are some international initiatives about the topic (e.g. IAEA).
ONR-UK	Severe accident analysis is an important area to continue and possibly increase international cooperation.
IRSN-France	N/A
JAEA-Japan	No good idea about topics related to SFR-specific PSA. Although the following is not a proposal for international cooperation, here we add information about current JNES's activity related to international cooperation. In 2008

	JNES reviewed the effectiveness of accident management measures of Monju for the regulatory body, the Nuclear and Industrial Safety Agency (NISA) (Endo, H., et al., 7-11 December 2009), (Nakajima, T., Endo, H., Yokoyama, T., 7-11 December 2009), (Ishizu, T., et al., 7-11 December 2009), (Yamamoto, T., et al., 7-11 December 2009), (JENES, December 2009). JNES asked its peer review to Forschungszentrum Karlsruhe (FZK) in Germany based on the requirement of NISA. In 2009 JNES got the preliminary result from FZK that supported the JNES's work. Following is a part of the report. "The PSA study's for Monju initially performed by JAEA and independently corroborated by JNES are clearly the most detailed PSA study's performed thus far for any sodium-cooled FBR. When made publicly available to a larger scientific community the studies will certainly serve as a basic reference for future PSA studies to be performed for any other FBR reactor." The final peer review report will be issued in mid 2010.
KAERI-Korea	The following PSA topics will benefit from international cooperation: a) Surrogated safety goals (with consensus of countries to have NPPs or to have NPPs); b) Development and enhancement of standard or guidance on technical acceptability for the PSAs of non LWRs; c) Digital I&C PSA and Human reliability analysis under digital environment; d) Seismic PSA (to reduce uncertainty); e) Human reliability analysis for shutdown PSA and external event PSA; f) Collection of possible initiating events for each reactor type or guidance for selecting initiating events; g) Collection of reliability data (that can be used for the PSA of reactor in design stage); h) Aircraft crash risk analysis; i) Passive system reliability; j) Methodology and tool for severe accident analysis (including analysis results); k) Level 3 PSA (to reduce uncertainty).
VTT-Finland	At this development stage of Gen IV reactors, the focus should be on risk and reliability requirements, and on the regulatory licensing framework for new reactors.
INET-China	It is expected that all the international activities will be helpful for the advanced reactor PSA in China, and we do expect the possible and great acceleration of the progresses.

5 Other

5.1 If there is any other topics you wish to see discussed in the Task Group report and in the April, 2010 workshop, please identify and briefly discuss these topics.

Organization	Responses
BEL V, TE -Belgium	N/A
NRC-USA	No other topics have been identified for discussion at this time.
SNSA-Slovenia	N/A
STUK-Finland	N/A
SUJB-Czech Republic	N/A

UJD-Slovakia	N/A
CEA-France	N/A
ENEA-Italy	Aging PSA
ONR-UK	N/A
IRSN-France	N/A
JAEA-Japan	N/A
KAERI-Korea	N/A
VTT-Finland	N/A
INET-China	N/A

Annex 1: References mentioned in Responses by JAEA-Japan

[1] AESJ-SC-P008:2008. (March 2009). A Standard for Procedures of Probabilistic Safety Assessment of Nuclear Power Plants during Power Operation (Level 1 PSA): 2008 [in Japanese]. Atomic Energy Society of Japan.

[2] AESJ-SC-P009:2008. (March 2009). A Standard for Procedures of Probabilistic Safety Assessment of Nuclear Power Plants during Power Operation (Level 2 PSA): 2008 [in Japanese]. Atomic Energy Society of Japan.

[3] Bayer, A., & Koeberlein, K. (1984). Risk-Oriented Analysis on the German Prototype Fast Breeder Reactor SNR-300. Nuclear Safety, Vol. 25, No.1, pp.19-32.

[4] Fukuda, M., Fujimoto, H., Yamashita, M., & Sugawara, M. (23-27 April 2007). Challenges in Introducing Risk Informed Regulation in Japan. International Conference on the Challenges faced by Technical and Scientific Support Organizations in Enhancing Nuclear Safety, IAEA-CN-142. Aix-en Provence, France.

[5] Gitter, J. G., & Akhtar, M. W. (21-25 April 1985). An Assessment of the Clinch River Breeder Reactor Core and Containment Response to Core Disruptive Accidents. International Topical Meeting on Fast Reactor Safety, (pp. 463-470). Knoxville, USA.

[6] Hill, D. J., Chang, Y. W., Deitrich, L. W., Lehto, W. K., Ragland, W. A., & Schaefer, R. W. (12-16 August 1990). An Overview of the EBR-II PRA. 1990 International Fast Reactor Safety Meeting, (pp. 33-42). Snowbird, USA.

[7] Hill, D. J., Ragland, W. A., & Roglans, J. (1998). The EBR-II Probabilistic Risk Assessment: lessons learned regarding passive safety. Reliability Engineering and System Safety, Vol.62, No.1, pp.43-50: Elsevier Science Limited.

[8] Ishikawa, K., Takamatsu, M., Kawahara, H., Mihara, T., Kurisaka, K., Terano, T., et al. (January 2009). Probabilistic Safety Assessment on Experimental Fast Reactor Joyo - Level 1 PSA for Internal Events -. JAEA-Technology 2009-004 [in Japanese].

[9] JAEA Experimental Fast Reactor Joyo. (n.d.). Topics. Retrieved September 9, 2009, from JAEA Experimental Fast Reactor Joyo: http://www.jaea.go.jp/04/o-arai/joyo/english/index.html

[10] JAEA Tsuruga Head Office. (n.d.). Fast Breeder Reactor Research and Development Center. Retrieved September 9, 2009, from JAEA Tsuruga Head Office: http://www.jaea.go.jp/english/04/turuga/info/fbr/fbr.html

[11] JAEA Tsuruga Head Office. (2008, March 17). Press Release [in Japanese]. Retrieved September 9, 2009, from JAEA Tsuruga Head Office:

http://www.jaea.go.jp/04/turuga/jturuga/press/2008/03/p080317.pdf

[12] Kondo, S., Nonaka, N., Niwa, H., Sato, I., Furutani, A., & Miyake, O. (12-16 August 1990). Integrated Analysis of In-Vessel and Ex-Vessel Severe-Accident Sequences. The 1990 International Fast Reactor Safety Meeting, (pp. 1-12). Snowbird, USA.

[13] Kondo, S., Tobita, Y., Morita, K., & Shirakata, N. (25-29 October 1992). SIMMER-III: An Advanced Computer Program for LMFBR Severe Accident Analysis. International Conference on Design and Safety of Advanced Nuclear Power Plant (ANP'92), (paper 40.5). Tokyo, Japan.

[14] Kotake, S., Mihara, T., Kubo, S., Aoto, K., & Toda, M. (2008). Development of Advanced Loop-Type Fast Reactor in Japan (1): Current Status of JSFR Development. ICAPP'08, June 8-12, 2008, Paper 8226. Anaheim, CA USA.

[15] Koyama, K., Yamada, Y., Hayakawa, S., Watanabe, M., & Watanabe, O. (10-14 May 2009). Development of Severe Accident Evaluation Technology (Level 2 PSA) for Sodium-cooed Fast Reactors (4) Identification of Dominant Factors in Core Material Relocation and Heat Removal Phases. ICAPP'09, (p. Paper 9126). Tokyo, Japan.

[16] Kurisaka, K. (June 1996). Development of Component Reliability Database for an LMFBR. PNC Technical Review No.98, pp.18-31, PNC TN1340 96-002[in Japanese].

[17] Kurisaka, K. (15-20 Oct. 2006). Probabilistic Safety Assessment of Japanese Sodium-cooled Fast Reactor in Conceptual Design Stage. 15th Pacific Basin Nuclear Conference. Sydney, Australia.

[18] Mihara, T. (13-18 September 1998). Level-1 PSA during shutdown for an LMFBR plant. 4th International Conference on Probabilistic Safety Assessment and Management, (pp. 2577-2582). New York City, USA.

[19] Miyake, O., Seino, H., Takai, T., & Hara, H. (25-29 October 1992). Development of CONTAIN Code for FBR Severe Accident Analysis. International Conference on Design and Safety of Advanced Nuclear Power Plants (ANP'92), (paper 41.4). Tokyo, Japan.

[20] Nakai, R., & Kani, Y. (17-21 April 1994). Utilization of Insights gained from the level-1 PSA for an LMFBR plant. ARS '94 International Topical Meeting on Advanced Reactors Safety, (pp. 407-415). Pittsburgh, USA.

[21] Nakai, R., & Yamaguchi, A. (1998). Study of rationalized safety design based on the seismic PSA for an LMFBR. Reliability Engineering and System Safety 62, pp.221-234: Elsevier Science Ltd.

[22] Nakai, R., Suzuki, T., Yamano, H., Seino, H., Ishikawa, H., Kamiyama, K., et al. (10-14 May 2009). Development of Severe Accident Evaluation Technology (Level 2 PSA) for Sodium-cooled Fast Reactors (1) Overview of Evaluation Methodology Development. ICAPP'09, (p. Paper 9117). Tokyo, Japan.

[23] NSC of Japan. (1992, May 28). NSCRG: L-AM-Ⅱ.01 Accident Management for Severe Accidents at Light Water Power Reactor Installations. Retrieved September 9, 2009, from Nuclear Safety Commission of Japan: http://www.nsc.go.jp/NSCenglish/guides/lwr/L-AM-II.01.pdf

[24] NSC of Japan. (2006, September 19). NSCRG: L-DS-I.02 Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities. Retrieved September 9, 2009, from Nuclear Safety Commission of Japan: http://www.nsc.go.jp/NSCenglish/guides/lwr/L-DS-I_02.pdf

[25] Ohno, S., Seino, H., & Miyahara, S. (10-14 May 2009). Development of Severe Accident Evaluation Technology (Level 2 PSA) for Sodium-cooled Fast Reactors (5) Identification of Dominant Factors in Ex-vessel Accident Sequences. ICAPP'09, (p. Paper 9110). Tokyo, Japan.

[26] Ohtaki, A., & Ohira, H. (December 1990). Development of Super-COPD Code for Plant Dynamics. PNC Technical Review No.76, pp.27-36, PNC TN1340 90-004[in Japanese].

[27] Sagayama, Y., Tanji, M., & Endo, H. (21-23 March 1983). Whole-Plant Simulation for Loop-Type LMFBR Dynamics. Power Plant Dynamics, Control and Testing Symposium, (pp. 16.01-16.18). Knoxville, USA.

[28] Sato, I., Yamano, H., & Tobita, Y. (10-14 May 2009). Development of Severe Accident Evaluation Technology (Level 2 PSA) for Sodium-cooled Fast Reactors (2) Identification of Dominant Factors in Initiating Phase of Unprotected Events. ICAPP'09, (p. Paper 9132). Tokyo, Japan.

[29] Sotsu, M., & Kurisaka, K. (12-16 July 2009). Evaluation of Monju Core Damage Risk with Change of AOT Using Probabilistic Method. 17th International Conference on Nuclear Engineering, (pp. ICONE17-75263). Brussels, Belgium.

[30] Tentner, A. M., Weber, D. P., Birgersson, G., Bordner, G. L., Briggs, L. L., Cahalan, J. E., et al. (21-25 April 1985). The SAS4A LMFBR Whole Core Accident Analysis Code. International Topical Meeting on Fast Reactor Safety, (pp. 989-998). Knoxville, USA.

[31] Tobita, Y., Kondo, S., Yamano, H., Morita, K., Maschek, W., Coste, P., et al. (March 2006). The Development of SIMMER-III, An Advanced Computer Program for LMFR Safety Analysis, and Its Application to Sodium Experiments. Nuclear Technology, Vol. 153, No.3, pp.245-255: American Nuclear Society.

[32] Tobita, Y., Yamano, H., & Sato, I. (10-14 May 2009). Development of Severe Accident Evaluation Technology (Level 2 PSA) for Sodium-cooled Fast Reactors (3) Identification of Dominant Factors in Transition Phase of Unprotected Events. ICAPP'09, (p. Paper 9127). Tokyo, Japan.

[33] Kasagawa, K. Miura, S., Endo, H., (7-11 December 2009), Reliability Analysis for a Cooling System of a typical FBR Plant's Ex-vessel Fuel Storage Tank (EVST) Using the PSA Method, (Paper IAEA-CN-176-03-20P), Kyoto, Japan.

[34] Haga,K., Endo, H., Ishizu, T., Iizuka, H. & Iizuka H. (7-11 December 2009). Development of Integrated Analytical Tools for Level-2 PSA of LMFBR, (Paper IAEA-CN-176-03-07), Kyoto, Japan.

[35] Shindo, Y., Endo, H., Inoue, M., (7-11 December 2009). Validation of Two-Phase Flow Model in the RELAP5/3D Code for Steam-Generator Blow Down Analysis Using the Test Data of Monju, (Paper IAEA-CN-176-03-25P), Kyoto, Japan.

[36] Inoue, M., Endo, H., Haga, K., (7-11 December 2009). Plans of verification Tests for the ACTOR Code Analyzing Fission Products Behavior in Primary Heat Transportation System of LMFBR, (Paper IAEA-CN-176-03-18P), Kyoto, Japan.

[37] Kawabata, H., Endo, H., Haga, K., (7-11 December 2009). Sever Accident Containment-response and Source Term Analyses by AZORES Code for a typical FBR Plant, (Paper IAEA-CN-176-03-16 P), Kyoto, Japan.

[38] Endo, H., Kawabata, T., Ito, T., Ishizu, T., Haga, K. Inoue, M, Nagase, H., (7-11 December 2009). Level-2 PSA for the Prototype Fast Breeder Reactor MONJU Applied to Accident Management Review, (Paper IAEA-CN-176-03-25P), Kyoto, Japan.

[39] Nakajima, T., Endo, H., Yokoyama, T., (7-11 December 2009). Analysis of Core Physics Test Data and Sodium Void Reactivity Worth Calculation for MONJU Core with ARCADIAN-FBR Computer Code System, (Paper IAEA-CN-176-06-14), Kyoto, Japan.

[40] Ishizu, T., Endo, H., Haga, K., Inoue, M., Shirakawa, N., (7-11 December 2009). Study of Energy Release Mechanism during ULOF Initiating Phase of LMFBR, (Paper IAEA-CN-176-03-17P),

Kyoto, Japan.

[41] Yamamoto, T., Nakajima, T., Endo, H., Yokoyama, T., (7-11 December 2009). A Fundamental Study on Criticality Evaluation of Damaged Core under Disruptive Accidents of LMFBRs, (Paper IAEA-CN-176-03-19P), Kyoto, Japan.

[42] JNES/SAE09-0011, Evaluation of the Effectiveness of Accident Management Measures to the Prototype FBR Monju Plant," December 2009.

APPENDIX 4: QUESTIONNAIRE ON PSA FOR NEW REACTORS

OECD/WGRisk

Survey on

PSA in the frame of Design and Commissioning of New NPPs

The task is coordinated by IRSN (France) and the project core group includes: IRSN (France), NUBIKI (Hungary), STUK (Finland) and NRC (USA).

New power plant projects are ongoing or expected in many countries. In all these countries, the new power plant project actors (safety authorities, technical support organisations, designers, constructors, operators) generally recognize that using probabilistic methods is necessary in order to achieve improved nuclear power plant safety and performances compared to the existing plants.

The objectives of the WGRisk task are:

- identify and characterize current practices regarding the role of probabilistic safety assessment (PSA) in the frame of design, construction and commissioning of new nuclear power plants,
- identify key technical issues, current approaches for dealing with these issues and associated lessons learned, as well as issues requiring further work,
- develop recommendations regarding the use of PSA by different actors in the frame of new nuclear power plant projects, i.e. appropriate PSA scope and level of details, pertinent PSA applications and decision-making process,
- identify future international cooperative work on the identified issues.

This activity focuses on technical issues related to PSAs for nuclear plants in the final design, construction, testing or commissioning phases. Plants in these phases have a stable general design configuration and are typically within one to ten years of commencing power operations. More advanced reactor designs (e.g., Generation IV), are generally in the earlier conceptual or preliminary design stage and are outside the scope of this effort.

The task covers PSA development and uses during the final design, construction, testing and commissioning pre-operational phases of a new power plant project, by all actors involved in new NPP design and commissioning. It refers to all PSA levels (1, 2, 3) and to all PSA scopes (reactor, fuel pool, internal events, hazards, etc.).

The main steps of the task are:

- March 2011: survey of participants' practices and needs, as well as past PSA development and using experience. This step includes the answering of the present questionnaire.
- June 2011: workshop on lessons and best practices. The questionnaire answer summaries will be also presented and discussed during the meeting. The workshop will be organized in common with CSNI/WGRISK task on PSA for Advanced Reactors,
- June 2012: CSNI task report.

The task is complimentary to the ongoing WGRISK task on advanced reactor. Since many of the survey questions are related, it has to be noted that the survey is focused on the new/evolutionary designs. The advanced designs are being addressed by the specific advanced reactor PSA task survey.

The following questionnaire addresses to individual organization rather than countries.

Instructions:

- the questions refer to the NPP project(s) in which you are involved or future projects,
    - *if the respondent is not yet involved in a project to design/build/acquire/license/etc. a new NPP, she/he can refer to the requirements regarding the future project.*
- if you are involved in more than one NPP project (reactor type) please duplicate the C, D, E, F, and G sections,
- if a specific question has not yet been addressed in the PSA you deal with, then please respond to the question by indicating how you feel the technical aspect should be addressed,
- if a specific and important question is not included in the questionnaire, please add a comment,
- please add a comment if you think that there are specific aspects that should be emphased by the WGRisk Task or/and it should be specifically included in future WGRISK activities

A. Identification

Please identify your organisation:

A 1. name: ………………………………………………………………………

A 2. address:….…………………………………………………………………

A 3. country: .………………………………………………………………………

A 4. contact person:………………………………………………………………

A 5. e-mail:……………………………………………………………………..

Are you?

A 6. international organization YES/NO

A 7. regulatory body YES/NO

A 8. supporting organisation to a regulatory body YES/NO

A 9. utility YES/NO

A 10. vendor YES/NO

A 11. manufacturer YES/NO

What is your involvement in the development and using of the PSA?

A 12. please specify (do you develop PSA yourself, do you review PSA, do you develop PSA procedures / guides / standards, etc.), (Level 1, Level 2, Level 3)

B. Workshop organisational aspects

A common workshop with CSNI/WGRISK task on PSA for Advanced Reactors, on PSA lessons and best practices, will be organized in June 20 - 24, 2011 at OECD Conference Centre in Paris, France.

B 1. do you intend to participate to the workshop? YES/NO

B 2. do you intend to present a paper to the workshop? YES/NO

B 3. intended title of the paper:………………………………………..

(for abstract submission, please follow the instruction in the attached "ANNOUNCEMENT and CALL for ABSTRACTS")

B 4. Topics of most interest to you?......................

C. New NPP PSA and Risk informed applications

C 1. Specify the NPP project you refer to?

- name, reactor type, NPP project status, etc.
- not yet decided

C 2. What is the regulatory role of PSA in your country?

C 3. Was the use of a PSA and PSA results one of the criteria used for selecting the specific NPP design for this project?

C 4. Please provide a general description of the design stage PSA:

- generic (reactor type) or site specific

- who is the PSA developer? (if is not you, do you have access to the computerized PSA model?)
- PSA scope (internal events, internal hazards, external hazards, power states, shutdown states, etc.)
- releases from other than the reactor core were considered? (ex: fuel pool)
- PSA Levels (Level 1, Level 2, Level 3, or extended Level 1)

C 5. What is/was the role of the PSA during the development of the plant design?

- safety demonstration, supporting the choice of design options, well-balanced safety concept, defence in depth assessment (multiple failures initiating events identification…), appreciation of the improved safety level compared to existing plants, etc.
- please explain PSA roles during conceptual design, detailed design, construction, commissioning and initial operation stages

C 6. Which risk-informed applications are used in the project? Are the applications based on regulatory requirements?

- safety classification of SSCs
- program for technical specifications
- program for inservice testing
- program for inservice inspection
- program for on-line preventive maintenance
- cost-benefit
- development of emergency operation procedures

C 7. How are uncertainties and limitations addressed?
please discuss the difficulties in using the PSA for a NPP in design stage, uncertainties, etc.

C 8. PSA guidelines used? Are the guidelines specific for new reactors?

- reference to the guidelines used
- the scope of the guidelines (Level 1, 2, 3)

C 9. Are you involved in an international group dealing with that specific PSA / type of reactor?
ex: MDEP

D. Internal Initiators PSA Level 1 technical aspects

D 1. How was the list of initiating events developed?

- based on existing reactor initiating events list, specific systematic method, etc.

- were new initiating events considered that were not considered in existing PSAs?

D 2. What types of supporting studies are used?

- Safety Report studies, specific studies for PSA, etc.

D 3. What kind of reliability data and CCF data are used?

- sources of data for the components of the same type as in the existing NPP PSAs?
- sources of data for evolutionary components / components with limited operating experience?
- if data were not available for certain components, then what methods were used to address component reliability?

D 4. How are the new / evolutionary design features treated?

- what is the contribution of the new / evolutionary design features to PSA results
- potential new initiating events
- inadvertent actuation of the new automatic actions

D 5. Which is the level of detail of the considered TechSpecs and preventive maintenance procedures?

D 6. How is the HRA performed?

- method used to model and quantify the post-accidental human errors? (screening, detailed, generation 1/generation 2, etc.)
- the detailed accident procedures should be available?
- is (should be) a simulator available? (is it used somehow in the frame of HRA?)
- method used to model and quantify the pre-accidental human errors?

E. PSA Level 1 hazards technical aspects

E 1. How the external hazards are treated?

- what method is used for seismic events? (seismic PSA, seismic margins, PSA based seismic margins, etc.)
- which are the extreme weather hazards considered? (extreme cold, heat wave, wind, storms, snow storms, lightning, sand and dust storms etc.)? How the modeling is done?
- which are the other external hazards considered (loss of ultimate heat sink, external flooding, tsunami, industrial, transportation accidents etc.)? How the modeling is done?
- are the possible future evolutions of some hazards (frequency, intensity) taken into account in the PSA?

E 2. How site specific information are treated in the design stage (e.g., bounding assumptions, representative site selected, site was known, etc.)?

E 3. How the internal hazards are treated?

- what method is used for fire PSA? (fire frequencies data source, which fire propagation code was used, considering of the internal explosions)
- what method is used for internal flooding PSA?
- which are the other internal hazards considered ? (heavy loads drops, etc)

F. Severe Accident/Source Term/Level 2 PSA

F 1.What kind of study is available for the Severe Accidents / Source term assessment?

- please describe the interface between Level 1 PSA and Level 2 PSA and precise how the dependencies are considered (human errors, shared components, I&C, treatment of containment bypasses, etc.)
- releases from other than the reactor core were considered? (ex: fuel pool)
- are the internal and external hazards initiating events considered?

F 2.What severe accident progression support studies are used?

- what computer codes are used? How detailed are the studies? Have experiments been performed? Thermal hydraulics and fission product releases?

F 3.What new severe accident reactor features are modelled in detail in the PSA?

- phenomena covered, reliability

G. Consequences analysis / PSA Level 3 technical aspects

G 1. Do you agree that consequence analysis and Level 3 PRA can be treated in a design independent fashion? If not, how design may impact the consequence analysis?

G 2. What kind of study is available for the offsite consequences analysis? (Level 3 PSA, Consequences analysis, etc.)

- please describe the interface with Level 2, methods, computer code, etc.
- releases from other than the reactor core were considered? (ex: fuel pool)
- are the internal and external hazards initiating events considered?

G 3. How the sites specific aspects are considered?

- meteorological data
- targets data (population, agriculture, land, economic)

G 4. How are the emergency actions considered?

- sheltering, evacuation, decontamination, etc.

- is the off-site emergency response plan available?

APPENDIX 5: ANSWERS TO QUESTIONNAIRE ON PSA FOR NEW REACTORS

	A. Please identify your organisation
AREVA	A 1. name: AREVA, A 2. address: Tour AREVA, 1 place Jean Milier, 92084 Paris La Défense Cedex A 3. country: FRANCE A 4. contact person: Mr. François BOUTEILLE A 5. e-mail: francois.bouteille@areva.com A 10. vendor A 11. manufacturer
BARC	A 1. Name: Bhabha Atomic Research Centre (BARC) A 2. Address: Room No. 104, DHRUVA Complex, BARC, Mumbai – 400085 A 3. Country: INDIA A 4. Contact person: Dr. P. V. Varde A 5. e-mail: varde@barc.gov.in A 8. supporting organization to a regulatory body Note: BARC is a technical support organization to the Atomic Energy Regulatory Board (AERB). The response to this questionnaire has been prepared in consultation with AERB and the utility, Nuclear Power Corporation of India Limited (NPCIL).
Bel V	A 1. name: Bel V A 2. address: Rue Walcourt 148, B-1070 Brussels A 3. country: Belgium A 4. contact person: Pieter De Gelder A 5. e-mail: pieter.degelder@Bel V.be A 7. regulatory body A 8. supporting organisation to a regulatory body (Bel V is the TSO of the FANC (Federal Agency for Nuclear Control; FANC and Bel V are considered to constitute together the Belgian regulatory body)
EDF	A 1. name: EDF SEPTEN A 2. address: 12-14 avenue Dutriévoz 69628 Villeurbanne Cedex A 3 country: FRANCE EPR FA3 A 4. contact person: RAVEL Cédric A 5. e-mail: cedric.ravel@edf.fr EPR UK A 4. contact person: BODY Guillaume A 5. e-mail: guillaume.body@edf.fr A 9. utility
ENEL	A 1. name: Enel Ingegneria e Innovazione SpA A 2. address: Viale Regina Margherita 125, 00198 Roma A 3. country: Italy A 4. contact person: Federica C.V. Mancini A 5. e-mail: federicaclaudiavaleria.mancini@enel.com A 9. utility
ENSI	A 1. name: Swiss Federal Nuclear Safety Inspectorate (ENSI) A 2. address: Industriestrasse 19, 5200 Brugg A 3. country: Switzerland

	A 4. contact person: Rainer Hausherr A 5. e-mail: rainer.hausherr@ensi.ch A 7. regulatory body
IRSN	A 1.name: …IRSN A 2. address:…BP17 92262 Fontenay aux Roses A 3. country: France A 4. contact person: Gabriel Georgescu A 5. e-mail: gabriel.georgescu@irsn.fr A 8. supporting organisation to a regulatory body
JNES	A 1. name: Japan Nuclear Energy Safety Organization (JNES) A 2.address: 3-17-1 Toranomon Minato-ku Tokyo A 3.country: Japan A 4. contact person: Hitoshi MUTA A 5. e-mail: muta-hitoshi@jnes.go.jp A 8. supporting organization to a regulatory body
MHI	A 1. name: Mitsubishi Heavy Industries , Ltd. (MHI) A 2. address 16-5, Konan 2-Chome, Minato-ku, Tokyo, 108-8215 A 3.country: Japan A 4. contact person: Takayuki Nirasawa A 5. e-mail: takayuki_nirasawa@mhi.co.jp A 10. vendor A 11. manufacturer
NRC	A 1.name: United States Nuclear Regulatory Commission (NRC) A 2. country: United States of America A 3. contact person: Jeffery Wood A 4. e-mail: Jeffery.Wood@nrc.gov A 6. regulatory body
NRI	A 1. name: Nuclear Research Institute Řež plc A 2. address: Husinec-Rez 130: Czech Republic A 4. contact person: Stanislav Hustak A 5. e-mail: hus@ujv.cz
NUBIKI	A 1. name: NUBIKI Nuclear Safety Research Institute A 2. address: Konkoly-Thege M. út 29-33., Budapest, H-1121 A 3. country: Hungary A 4. contact person: Attila Bareith A 5. e-mail: bareith@nubiki.hu A 8. supporting organisation to a regulatory body. Note: NUBIKI provides technical support to the nuclear safety authority (Hungarian Atomic Energy Authority) but it is independent from both the regulatory body and the utilities in Hungary. Due to this position NUBIKI is also involved in technical support activities to the utility (Paks NPP) on a contractual basis.
ONR	A 1. name: Office for Nuclear Regulation (ONR) A 2. address: Redgrave Court, Merton Road, Bootle, Merseyside, L20 7HS A 3. country: UK A 4. contact person: Shane Turner A 5. e-mail: shane.turner@hse.gsi.gov.uk A 7. regulatory body
SNSA	A 1. name: Slovenian Nuclear Safety Administration A 2 address: Železna cesta 16, P.O Box 5759 SI-1001 Ljubljana A 3.country: Slovenia

	A 4.contact person: Djordje Vojnovič A 5. e-mail: Djordje.Vojnovic@gov.si A 7. regulatory body
STUK	A 1. name: Radiation and Nuclear Safety Authority (STUK) A 2. address: Laippatie 4, FIN- 00881 Helsinki A 3. country: Finland A 4. contact person: Reino Virolainen A 5. e-mail: reino.virolainen@stuk.fi A 7. reulatory body
UNISTAR	A 1. name:UNISTAR NUCLEAR ENERGY A 2. address:750 E. Pratt Street BALTIMORE MD, 21202 A 3. country: USA A 4. contact person: Vincent Sorel A 5. e-mail: vincent.sorel@unistarnuclear.com A1. utility
	What is your involvement in the development and using of the PSA? A 12. please specify (do you develop PSA yourself, do you review PSA, do you develop PSA procedures / guides / standards, etc.), (Level 1, Level 2, Level 3)
AREVA	AREVA develops reviews and maintains PSA level 1, 2 and 3 for the utilities and its own internal needs.
BARC	We review Level-1 and Level-2 PSA developed by the utilities. We also develop PSA procedures / guides / standards, etc.
Bel V	Being part of the regulatory body, we review the PSAs of the Belgian NPPs, that are performed by Tractebel Engineering on behalf of the utility Electrabel (GDF-Suez).
EDF FA3	The FA3 PSA is developed by EDF (Levels 1 and 2).
EDF EPR UK	Co-applicant: the UK EPR PSA is developed jointly by EDF and AREVA (Level 1, Level 2 and Level 3).
ENEL	We are developing PSA Level 1 and 2.
ENSI	ENSI developed two PSA guidelines, one concerning the quality and scope, the other concerning applications of PSA Level 1 and Level 2. ENSI reviews the plant specific PSA Level 1 and 2 from the licensees and develops PSAs for Level 1.
IRSN	IRSN develops PSA for French power reactors (existing and new): • Reactors: 900 MWe, 1300 MWe, N4, EPR (including the spent fuel pool for EPR reactor) • Scope: mainly internal events and internal fire (900 MWe and 1300 Mwe) • Projects under development for seismic PSA and internal flooding PSA • Level: 1 and 2 • Level2 for 900 MWe, 1300 MWe and EPR.

	IRSN performs the licensee's PSA review, as the French Safety Authority (ASN) support organization. The PSAs considered as "Reference PSA" are the PSAs developed by the licensees. In addition IRSN develops independent PSAs in order to perform the review of the reference PSA and independent studies. IRSN was and is involved in the development of all French guides regarding the PSA.
JNES	Developing PSA by ourselves, reviewing PSA developed by utilities, committing to develop PSA guides in accordance with regulatory body. Scoping Level 1-3 PSA.
MHI	MHI developed Level 1, Level 2 and Level 3 PSA ourselves.
NRC	NRC participates in several aspects of PSA development. For new reactors the primary role of the NRC is to review a description of the PSA and its results included with an application for design certification (DC) or combined license (COL). This review may include and audit of the PSA in the applicant's offices. NRC also develops its own PSA models; however, these PSA models do not have a formal function in the licensing reviews of new reactors. The development of the NRC's PSA model for a new reactor generally occurs after the final design has been completed and the licensing review process has commenced. In addition, the NRC contributes to the development of PSA standards. PSA standards are developed by the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS). NRC staff members, along with industry representatives, contribute to PSA standards by participating in writing groups and serving on development committees. The NRC staff uses information from a new reactor PSA to develop risk insights for that new reactor design to help focus that staff's overall review of the proposed design, construction and operation of the facility in a risk informed manner. As mentioned above, NRC also develops independent PSA models. These are known as SPAR (Standardized Plant Analysis Risk) models. The SPAR models are primarily Level 1 PSA models, although the NRC has developed a limited number of feasibility models for operating plants that extend beyond Level 1 to include aspects of Level 2 PSA modelling. For new reactor designs that are currently under review or have recently completed design certification, the NRC's SPAR models are limited to only Level 1 modelling. In the past, NRC has performed Level 3 PSA studies (e.g., NUREG-1150). NRC continues to foster knowledge and expertise pertaining to Level 3 PSA modelling methods, but NRC does not regularly develop or maintain Level 3 PSA models.
NRI	NRI Rez developed and maintains Living PSA for NPP Dukovany in the Czech republic. It participated in development of PSA for NPP Temelin (Czech Republic) and NPP Bohunice (Slovakia). NRI Rez is also participating in development and review of PSA Level 1 and level 2 guidelines (national, IAEA, EUR).
NUBIKI	We develop PSA (Level 1 and Level 2) and we are occasionally engaged in the development of PSA guides too.
ONR	Regulatory body – reviews licensee PSAs.
SNSA	At the Slovenian Nuclear Safety Administration (SNSA) we review the PSA model developed by the utility (the Krško NPP), we use it to make our own calculations (e.g. for event analysis, inspection findings,…) and to eventually check the NPP's, to prepare inspection plans, etc.
STUK	PRA level 1 and 2 model, performed by the licensee and reviewed by STUK, is used for resolution of safety issues by both parties. For this purpose the licensees provide STUK with the PRA model in electronic form and regularly maintain and update it. STUK accepts the models after a thorough regulatory review process. STUK develop the PRA guides which sets forth the guidelines on how to perform PRA and its application for risk informed decision making. Examples of safety issues for which the PRA insights give an improved basis for decisions, are approvals of plant modifications and resolution of testing, inservice inspection and maintenance programs. Examples of such requirements are details of safety classification and many Technical Specification requirements.

UNISTAR	UNISTAR was responsible to submit a site-specific PRA for the EPR on the Calvert Cliff Unit3 (CC3) site.
	B. Workshop organizational aspects
AREVA	B 1. Do you intend to participate to the workshop? YES B 4. Topics of most interest to you? New plants (GEN3): General topics: PSA level 1 and level 2 during design phases (basic design, detailed design, commissioning) / use of PSA in decision making / Assessment of internal and external hazards / Risk-Informed application for new plants / Probabilistic safety criterion Specific topics: new HRA approach for NPP with advanced computerized HMI / realistic modeling of software failure in the PSA / Long term analyses / assessment of PSA limitations
BARC	B 1. do you intend to participate to the workshop? YES B 2. do you intend to present a paper to the workshop? YES B 3. intended title of the paper “Overview of the Probabilistic Safety Assessment Activities in India and Future R&D Programmes “ B 4. Topics of most interest to you? Reliability analysis of passive systems and digital C&I and its integration into PSAs for advanced reactors
Bel V	B 1. do you intend to participate to the workshop? Will depend on the program of the workshop B 4. Topics of most interest to you? no specific suggestions
EDF	B 1. do you intend to participate to the workshop? YES B 4. Topics of most interest to you? hazards PSA
ENEL	B 4. Topics of most interest to you? Seismic PSA, probabilistic analysis for the I&C and external hazards modelization
ENSI	B 1. do you intend to participate to the workshop? YES B 4. Topics of most interest to you? Modeling of new design features in particular reliability and thermohydraulic behaviour of passive systems, severe accidents, reliability of passive components, digital I&C, software reliability, and dynamic PSA
IRSN	B 1. do you intend to participate to the workshop? YES B 2. do you intend to present a paper to the workshop? YES B 3.intended title of the paper: • Paper 1: Lessons learned form IRSN review of Flamanville 3 Level 1 PSA

	• Paper 2: I&C modeling in the IRSN EPR Level 1 PSA • (Paper 3: Level 2 PSA (outcome of the ASAMPSA2 project for Gen IV reactors)) B 4. Topics of most interest to you? External Events PSA, Digital I&C modeling, Common Cause Failures.
JNES	-
MHI	B 1. do you intend to participate to the workshop? YES
NRC	B 1. do you intend to participate to the workshop? YES B 2. do you intend to present a paper to the workshop? YES B 3.intended title of the paper: • Paper 1. NRC Activities Concerning PSA for New and Advanced Reactors • Paper 2. Risk-Informing the NRC Standard Review Plan for Small Modular Reactor Designs B 4. Topics of most interest to you? • Topics of interest to NRC include: • Use of PSA and risk results in regulatory review and licensing processes. • Use of PSA and risk results in other regulatory programs (e.g., risk-informed in-service inspection, risk managed technical specifications.) • Assessment of external hazards. • Use of PSA throughout the reactor design cycle (i.e. pre-conceptual, conceptual, preliminary, final design, construction and operation.) • Assessment of site risk for multi-reactor sites. • Uncertainty quantification methods and communication of uncertainty results. • Quantification of digital instrumentation and control system risk. • Design features credited in the PSA for the prevention and mitigation of severe accidents.
NRI	B 1. do you intend to participate to the workshop? YES B 4. Topics of most interest to you? SW reliability, new types of IE in PSA for new designs, human-induced IEs for new designs, credibility of passive design.
NUBIKI	B 1. do you intend to participate to the workshop? YES B 4. Topics of most interest to you?

	Role of PSA in licensing new nuclear power plants.
ONR	ONR intends to attend the workshop and will present at the workshop.
SNSA	-
STUK	B 1. do you intend to participate to the workshop? YES B 2. do you intend to present a paper to the workshop? YES B 3.intended title of the paper: - B 4. Topics of most interest to you? Role of PRA in licensing new nuclear power plants.
UNISTAR	B 1. do you intend to participate to the workshop? Not yet decided B 2. do you intend to present a paper to the workshop? Not yet decided
	C. New NPP PSA and Risk informed applications
	C 1. Specify the NPP project you refer to?
AREVA OL3	EPR™ in Olkiluoto/Finland 4-LOOP PWR Plant under construction.
AREVA TSN	EPR™ in Taishan, 2 units/Public Republic of China 4-LOOP PWR Plant under construction.
AREVA US	EPR™ Design Certification in the USA 4-LOOP PWR Under the regulatory review.
BARC	TAPS-3&4 (540 Mwe twin units of Pressurized Heavy Water Reactors (PHWRs) Tarapur Atomic Power Station) under operation.
Bel V	At present, there are no plans in Belgium to construct new power reactors (because of the present phase-out law).
EDF FA3	EPR FA3, 4-LOOP PWR, plant under construction.
EDF EPR UK	UK EPR, Generic Design Assessment in the UK, 4-LOOP PWR. Under the regulatory review by ONR
ENEL	The NPP project is the Italian Nuclear Project for the construction of EPR reactor.
ENSI	There are three General Licence Applications (dealing mainly with site characteristics and external hazards) for new NPPs in Switzerland. The public vote on the projects is expected in 2013. The reactor design/type is not yet decided.

<table>
<tr><td>IRSN</td><td>FA3 – Flamanville 3, EPR, under construction

PY3 – Penly 3, EPR, construction license under review (PY3 project is not enough advanced to be included in the answers).

IRSN is involved in the review of the EDF PSA.</td></tr>
<tr><td>JNES</td><td>Review of Accident Management (AM) strategies for Shimane Nuclear Power Station Unit 3 developed by utility before commissioning

Reactor type: ABWR

NPP project status: Review of AM review is almost finished.</td></tr>
<tr><td>MHI</td><td>MHI refer to the US-APWR. Reactor type is a Pressurized Water Reactor (PWR). The U.S. NRC had accepted the design certification application, and the design of the US-APWR is under review of the design certification in the U.S.</td></tr>
<tr><td>NRC</td><td>NRC is currently participating in new NPP projects related to the review of applications for design certifications (DCs) and combined licenses (COLs). NRC is also developing SPAR models for new NPPs. The specific NPP designs that are used in these projects are discussed below.

In addition to the new NPP designs discussed below, NRC is anticipating DCs and COLs for advanced small modular designs, including integral pressurized light water reactors (also referred to as iPWRs) and High Temperature Gas Reactors (HTGRs). At this time, applications for these designs have not yet been submitted to NRC.

<u>Design Certification Application Reviews</u>

NRC has issued design certifications for the following new designs and applicants:
<table>
<tr><td>Design</td><td>Applicant</td></tr>
<tr><td>Advanced Boiling Water Reactor (ABWR)</td><td>General Electric (GE) Nuclear Energy</td></tr>
<tr><td>Advanced Passive 1000 (AP1000)</td><td>Westinghouse Electric Company</td></tr>
</table>
NRC is currently reviewing the following design certification applications:
<table>
<tr><td>Design</td><td>Applicant</td></tr>
<tr><td>ABWR – amendment to the design certification rule</td><td>South Texas Project Nuclear Operating Company</td></tr>
<tr><td>ABWR – renewal and update to the design certification rule</td><td>GE-Hitachi Nuclear Energy</td></tr>
<tr><td>ABWR – renewal and amendment to the design certification rule</td><td>Toshiba Corporation Power Systems Company</td></tr>
</table>
</td></tr>
</table>

	<table><tr><td>AP1000 – amendment to revise the design control document</td><td>Westinghouse Electric Company</td></tr><tr><td>Economic Simplified Boiling-Water Reactor (ESBWR)</td><td>GE-Hitachi Nuclear Energy</td></tr><tr><td>U.S. EPR</td><td>AREVA Nuclear Power</td></tr><tr><td>U.S. Advanced Pressurized-Water Reactor (US-APWR)</td><td>Mitsubishi Heavy Industries, Ltd.</td></tr></table> <u>Combined License Applications</u> At the time of preparing this response, NRC has received 18 COL applications to construct and operate a nuclear power plant (NPP) at a specific site. COL applications have been received for the following reactor designs. The table below also includes the number of NPP sites and number of reactor units associated with each reactor design. <table><tr><td>Design</td><td>NPP Sites</td><td>Reactor Units</td></tr><tr><td>ABWR</td><td>1</td><td>2</td></tr><tr><td>AP1000</td><td>7</td><td>14</td></tr><tr><td>ESBWR</td><td>4</td><td>5</td></tr><tr><td>U.S. EPR</td><td>4</td><td>4</td></tr><tr><td>US-APWR</td><td>2</td><td>3</td></tr><tr><td>Sum:</td><td>18</td><td>28</td></tr></table> <u>Development of SPAR Models</u> NRC has commenced work on Level 1 SPAR models for new NPP designs. Initial versions of models for the AP1000 and ABWR designs have been completed. Development of a SPAR model for the US-APWR design is currently underway. NRC also plans to develop SPAR models for U.S. EPR and ESBWR. The new NPP SPAR models are based on information submitted by the applicants for design certification. Generic reliability data for the models are taken from NRC's operating experience and data analysis program.
NRI	not yet decided

NUBIKI	A project for new nuclear builds in Hungary is under preparation. It aims at extending nuclear power production capacity at the site of the Paks NPP with one or two PWR units. The specific PWR reactor type is not yet decided. NUBIKI is involved in preparatory work for site permit, environmental permit and preparation.
ONR	Office for Nuclear Regulation (ONR – an Agency of the UK's Health and Safety Executive) uses a pre-licensing process known as the Generic Design Assessment (GDA) to evaluate a new reactor design in advance of a nuclear site license being made. If the design is judged to be satisfactory, a Design Acceptance Confirmation will be issued. The following designs are currently under evaluation: • UK AP1000, PWR with passive systems designed by Westinghouse (WEC) • UK EPR, evolutionary PWR designed by AREVA
SNSA	• the possible project of a new unit of the Krško NPP • general legislation requirements
STUK	A project for a new build in Finland is in progress (OL3-EPR) and two further new builds have received a positive decision in principle in the Parliament. The specific reactor types for the new builds are not yet decided.
UNISTAR	EPRTM, 4 LOOP PWR, Generic Design Certification and Combined License Application (COLA) for CC3 in progress in the USA.
	C2. What is the regulatory role of PSA in your country?
AREVA OL3	PSA Level 1 and Level 2 is licensing requirement in Finland, both for construction and operation.
AREVA TSN	PSA Level 1 and Level 2 is licensing requirement in People Republic of China, both for construction and operation.
AREVA US	PSA Level 1 and Level 2 is licensing requirement in the USA, both for design certification and operating license.
BARC	The submission of Level-1 PSA for internal events at full power before the first criticality is a mandatory requirement for new NPPs.
Bel V	For the existing NPP, PSA is used in the framework of the Periodic Safety Review (PSR) to evaluate the safety of the NPP, by using PSA as a complementary tool to the deterministic safety analysis. In view of our reply to question C1 above, and since all further questions are related to PSA for new NPP, we consider all following questions as "not applicable" for our PSA-review work at Bel V. Hence, no further answers are given.
EDF FA3	In France, a specific chapter of Safety Report is dedicated to PSA, with quantified objectives in term of core melt-down and releases. These objectives are not safety limits but indicative values to check the design.

EDF EPR UK	In the UK, it is a regulatory requirement for nuclear facilities to respect quantified objectives in term of releases (numerical targets of the SAPs3). PSA is used to demonstrate that the objectives are met. Additionally, the SAPs FA.10 to FA.13 gives the general regulatory requirements of the PSA. PSA is used to inform the design process and help ensure the safe operation of the site and its facilities. According to SAPs FA14, appropriate use of PSA is made in activities such as: • designing the facility; • supporting modifications to the design and operation during the design phase and life of the site and its facilities; • testing, inspection and maintenance planning, and management of plant configuration; • investigating significant abnormal occurrences; and • developing and changing operating procedures and associated training programmes for managing incidents and accidents (including severe accidents).
ENEL	The regulatory role of the PSA is not yet established.
ENSI	Based on the Nuclear Energy Act and its accompanying ordinance, full-scope, plant-specific Level 1 and Level 2 PSA for all relevant operational modes are required. A Level 3 PSA is not required in Switzerland. The Nuclear Energy Ordinance also enacts a number of PSA applications. Furthermore, the ordinance authorizes the Inspectorate to issue two PSA guidelines: • Guideline ENSI-A05: PSA quality and scope • Guideline ENSI-A06: PSA applications Although both guidelines are in general written for existing and new reactors, they are currently focussing on existing reactors. In particular ENSI-A06 may be changed in the future in order to reflect the more stringent requirements for new reactors. PSA applications: Evaluation of the safety level and the identification of potential plant-specific vulnerabilities. Corresponding evaluation criteria are given in the regulatory guideline ENSI-A06. This evaluation is performed within the framework of plant-specific licensing actions and/or the periodic safety review. The balance among the risk contributions from initiating event categories, accident sequences, components and human actions shall be evaluated. If any of the initiating event category accident sequences, components or human actions are found by PSA to have a remarkably high contribution, measures to reduce the risk shall be identified and – to the extent appropriate – implemented. The regulatory guideline ENSI-A06 provides criteria for the evaluation of the balance of the risk contribution of the various initiating event categories. According to the ordinance on “Hazard Assumptions and Evaluation of Protection Measures against Accidents in Nuclear Installations” the plant shall be designed against natural hazards such as earthquakes, flooding and extreme weather conditions. In particular, sufficient protection against natural hazards with a frequency greater than or equal to 1E-4 per year shall be demonstrated. The corresponding hazard curves are taken from the PSA. The impact of a plant modification on the risk shall be assessed. This applies to all PSA-relevant structural or system-related plant modifications as well as to changes of the technical specification involving PSA-relevant components. Criteria are given in the regulatory guideline ENSI-A06. In defining the allowed outage times, it shall be ensured that components shown to be significant to safety from the PSA point of view (ENSI-A06) are considered in the technical specifications (completeness), and assigned to correspondingly short allowed outage time categories (balance). Based on the risk measures CDF and LERF, a review

[3] HSE - Safety Assessment Principles for Nuclear Facilities - 2006 Edition , Version 1

	of the completeness and the balance of the allowed outage times shall be carried out in the course of the periodic safety review. At the beginning of every year, the licensees submit ENSI a probabilistic evaluation of the operational experience of the previous year. In this study initiating events as well as component unavailabilities due to planned or unplanned maintenance or tests are considered. The study involves among other things the determination of the probabilistic safety indicators (maximum annual risk peak and incremental cumulative core damage probability) and the risk contribution of the online maintenance. PSA is one element in the integrated decision-making. Therefore, PSA is also used to classify reportable events (provided the event affects a PSA-relevant structure, system, component or operator action). Since ENSI classifies all reportable events by the INES-Scale, regulatory guideline ENSI-A06 provides a relationship between the cumulative conditional risk of an event and the INES-Scale. Insights from the plant-specific Level 2 studies are used as part for the technical basis of the Severe Accident Management Guidance (SAMG) in order to provide information on the possible accident progressions and plant states. Furthermore, Level 2 PSAs are also used for the preparation of emergency exercises dealing with severe accidents. The Nuclear Energy Ordinance also enacts the requirement of a PSA Level 1 and Level 2 for the construction permit and the operation permit of a new plant to show the safety level, the balance of risk contributors, and the balance and completeness of the technical specifications (only for operation permit).
IRSN	As indicated in the French PSA Fundamental Safety Rule (2002-01), the safety of French nuclear reactors is based essentially on a deterministic approach. PSA supplements the conventional deterministic analyses. For the new generations of reactors, PSA is used as a supplemental tool in safety assessment during the design phase. The contributions of these assessments include the following: • help for the design of safety systems, particularly in terms of redundancy and diversification, • verification of a balanced conception of reactor safety related to the absence of scenarios having a predominant contribution to the frequency of core damage, • estimation of the deviations with respect to the safety requirements applied to operating reactors, • comparison of the level of safety of the future reactor with that of operating reactors or of other reactors under development, • help with the definition of operating conditions related to multiple failures, • preliminary assessment of the safety improvement resulting from the planned measures in the case of a severe accident, • help in the demonstration that the sequences leading to large (and/or early) releases are practically eliminated.
JNES	Not a mandatory regulatory requirement, but considered to be important information to regulate NPP safety. In some area, especially in inspection of NPPs, risk-informed regulation is put in place.
MHI	In the U.S., PSA is required as one of the regulatory requirements in the design certification application.
NRC	For DC and COL applications submitted according to Title 10, Part 52 of the U.S. Code of Federal Regulations (10 CFR Part 52), the applicant is required to submit a description of the design-specific (for DC) or plant-specific (for COL) probabilistic risk assessment (i.e., the PSA) and its results. The applicant is required to develop a Level 1 and a Level 2 PSA. The PSA must cover those initiating events and modes for which NRC-endorsed consensus standards on PSA exist one year prior to the scheduled date for initial loading of fuel. The uses of the PSA during reactor licensing are discussed in NRC's Regulatory Guide 1.206. These uses include: • Identify risk-informed safety insights based on systematic evaluations of the risk associated with the design, construction, and operation of the plant.

	• Demonstrate how the risk associated with the design compares against the Commission's goals of less than 1x10-4/year for core damage frequency, less than 1x10-6/year for large release frequency, and that the conditional containment failure probability be less than approximately 0.1 for the composite of all core damage sequences assessed in the PRA. • Demonstrate whether the plant design, including the impact of site-specific characteristics, represents a reduction in risk compared to existing operating plants. • The PSA results and insights are used to support other regulatory programs as follows: o Support the process used to demonstrate whether the Regulatory Treatment of Non-Safety Systems (RTNSS) is sufficient and, if appropriate, identify the SSCs included in RTNSS. o Support the regulatory oversight processes for operating reactors, e.g., the Mitigating Systems Performance Index (MSPI) and the significance determination process (SDP), and programs that are associated with plant operations, e.g., technical specifications (TS), reliability assurance, human factors, and Maintenance Rule (10 CFR 50.65) implementation. o Identify and support the development of specifications and performance objectives for the plant design, construction, inspection, and operation, such as Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC); the Reliability Assurance Program (RAP); TS; and COL action items and interface requirements.
NRI	The regulatory legislation does not currently contain explicit requirements on the use of PSA for the safety evaluation of the NPP in the Czech Republic.
NUBIKI	PSA is mandatory according to Hungarian regulatory requirements laid down in the Nuclear Safety Codes. Level 1 and 2 PSAs are required for a nuclear power plant covering all plant operational states, modes and initiating events (internal and external). PSA quality is addressed in a regulatory guide on PSA. No explicit requirements exist on PSA applications. More details on the role of PSA in regulation are given in CSNI WGRISK report NEA/CSNI/R(2007)12.
ONR	As part of the site licensing process ONR requires that a Site Specific Pre-Construction Safety Report (PCSR) submitted prior to legal permission being given for the start of major safety related construction activities. This PCSR should include a full scope Level 1, Level 2 and Level 3 PSA. The PSA should be used to help show that the design satisfies As Low As Reasonable Practicable (ALARP) requirement. For the pre-licensing GDA process the Generic PCSR should also include a full scope Level 1 and level 2 PSA. A Level 3 PSA is also required but as details are site specific a high level outline analysis is acceptable. Where numerical targets are given in the SAPs, ONR will seek sufficient information for it to judge that the target is likely to be achieved and the overall risk is ALARP. Further guidance on ONR's (HSE) expectations relevant to PSA can be found in the SAPs (Ref. 1) and in the PSA TAG (Ref. 2). GDA is following a "Claims – Arguments – Evidence" structure: • Step 2 was "claims" and for PSA these were interpreted as approach, outline scope, criteria and output of the PSA. • Step 3 was "arguments" which were broadly interpreted as being the methods, techniques and detailed scope (Ref. 3 and 4). • Step 4 has concentrated on the "evidence" and for PSA this is the detailed implementation of the methods and techniques, and the data and parameters used to quantify the PSA. Step 4 GDA reports were published in ONR website in December 2011..As well as the detailed review of all the technical areas of the PSA, during GDA Step 4 a Risk Gap Analysis (RGA) has been undertaken. The RGA was designed to meet the following objectives: • Support to GDA conclusion whether the EPR or AP1000 are reactors that can be built and operated safely in the UK • Evaluation of the importance of the findings of the GDA review in the various PSA technical areas.

	• Evaluation of the overall gap between the plant design risk claimed by RP and risk contributors that may have been underestimated or omitted
SNSA	In our regulation we have some general requirements for the use of PSA in design and operation of NPPs, like: • the NPP must be designed so as to assure that the total CDF is less than 1E-5 /yr and the LERF is less than 1E-6 /yr; If the CDF is less than 1E-5 /yr, but greater than 1E-6 /yr or if the LERF is less than 1E-6 /yr, but greater than 1E-7 /yr per year, investor or operator shall substantiate that further risk reduction is not possible or reasonable, • the licensee must establish the PSA analysis as a part of the Final Safety Analysis Report, • the PSA analysis must be used in decision making process, for plant modifications, periodic safety review, online maintenance, development and verification of programs, event analysis,…
STUK	In the Finnish regulatory guides the Living PRA is formally integrated in the regulatory process of NPPs already in the early design phase as a part of the licensing documentation. PRA and its applications are to run throughout the construction and operation phases of the plant service time. A plant specific, design phase level 1 and 2 PRA is required as a prerequisite for issuing a positive statement for an application of the construction license for a new NPP design and a complete level 1 and 2 PRA for issuing a positive statement for an application of a operating license. The plant specific level 1 and 2 PRA includes internal initiators, fires, flooding, harsh weather conditions and seismic events for full power operation mode and for low power and shutdown model. STUK will review the PRA and makes an assessment of the acceptability of the design phase and construction phase PRAs prior to giving a statement about the construction license and operating license applications, respectively.
UNISTAR	The Regulator (the USNRC) defined the role of PRA in nuclear activities:in USNRC Policy Statement 60FR42622 ("…the use of PRA technology should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach." The Regulator provides Risk-Informed approach requirements and supporting guidance to achieve PRA Quality. The review of PRA quality and technical adequacy and their use in support of risk-informed decision making are carried out against those regulatory requirements.
	C 3. Was the use of a PSA and PSA results one of the criteria used for selecting the specific NPP design for this project?
AREVA OL3	Yes
AREVA TSN	No
AREVA US	No
BARC	No
Bel V	-
EDF FA3	No
EDF EPR UK	Yes.

ENEL	Yes, the EPR is a Gen III reactor and we would improve the design with a probabilistic assessment.
ENSI	Specific NPP design is not yet selected.
IRSN	As mentioned before, the use of PSA during the design stage and the results of PSA is one of the important criteria for the licensing of new reactors.
JNES	Utility decides accident management countermeasures and facilities to be installed based on the results and insights obtained from the plant specific PSA. JNES reviews effectiveness of these countermeasures using PSA.
MHI	The PSA results were extensively utilised in the US-APWR design in order to improve the design for better safety.
NRC	NRC is not involved in selecting the specific NPP designs. NRC reviews those NPP designs that are selected and submitted by the applicants. The PSA is required for all applicants that submit a DC or COL application in accordance with 10 CFR Part 52.
NRI	We have no information about that.
NUBIKI	As given in the answer to question C1, the specific NPP design is not selected yet in the project. It is only decided that the new plants will be PWRs. This decision was not driven by PSA considerations but other technical aspects and country specific features (lack of experience in operating reactor types other than PWRs). The specific design will be selected during the evaluation of bids and biding requirements include PSA related requirements too.
ONR	PSA results are one element of “Claims” that were examined in Step 2 of GDA, though “selection” was not undertaken on any ranking, only that targets were claimed to be met.
SNSA	The project of a new NPP in Slovenia has not yet started. The design has not been selected.
STUK	No, not at all!
UNISTAR	No
	C 4. Please provide a general description of the design stage PSA • **generic (reactor type) or site specific** • **who is the PSA developer? (if is not you, do you have access to the computerized PSA model?)** • **PSA scope (internal events, internal hazards, external hazards, power states, shutdown states, etc.)** • **releases from other than the reactor core were considered? (ex: fuel pool)** • **PSA Levels (Level 1, Level 2, Level 3, or extended Level 1)**
AREVA OL3	a) PSA for construction license: Full scope level 1 and level 2

	• Internal events • External events • Hazards for at-power and shutdown states / including spent fuel pool cooling b) PSA for operating license : Detailed full scope level 1 and level 2 • Internal events • External events • Hazards • seismic for power and shutdown states including spent fuel pool cooling
AREVA TSN	a) PSA for construction license: Full scope level 1 and level 2 • Internal events • External events • Hazards for at-power and shutdown states / including spent fuel pool cooling b) PSA for operating license: Detailed full scope level 1 and level 2 • Internal events • External events • Hazards • seismic for power and shutdown states including spent fuel pool cooling Under progress (to be issued in 2012)
AREVA US	a) PRA for design certification includes: Full scope Level 1 and Level 2 for power and shutdown states for: • Internal events • Internal hazards • External hazards (screening analysis)

	b) PRA based seismic margin assessment (PRA "before fuel load" should include a full scope seismic PRA)
BARC	Site-specific The utility is the PSA developer. We do not have access to the computerized PSA model. PSA scope: Internal events (excluding internal hazards), full power state Releases from other than the reactor core (ex: fuel pool) were not considered PSA Levels: Level 1
Bel V	-
EDF FA3	EDF developed a full scope level 1 and level 2 PSA for a specific FA3 EPR. All reactor states are addressed in the PSA (from full power operation to shutdown states including accidents in the spent fuel pool) Internal events as well as Internal and External hazards are studied. Releases from the Spent Fuel Pool were considered.
EDF EPR UK	EDF and AREVA developed a full scope level 1 and level 2 PSA for a generic UK EPR (a simplified level 3 was performed): • All reactor states are addressed in the PSA (from full power operation to shutdown states including accidents in the spent fuel pool) • Internal events as well as Internal and External hazards are studied. • Releases from other than the reactor core were considered: accidents in the Spent Fuel Pool, fuel handling accidents, accidental aircraft crash on non fully protected buildings where radioactive containing systems are located.
ENEL	The PSA will be site specific for the EPR. The development will be based on collaboration between ENEL and AREVA. PSA scope: internal events, internal hazards, external hazards, power states, shutdown and low power states. We will consider releases from fuel pool, reactor building and the bypass without core fusion. PSA Levels: Level 1 and Level 2.
ENSI	Specific NPP design is not yet selected. According to the relevant guideline (ENSI-A05), the PSA has to be site specific to the extent possible. The applicant has to develop the PSA. If necessary ENSI can require the computerized PSA model (viewer or even the model). The Nuclear Energy Act and its accompanying ordinance require a plant-specific, full-scope (internal events, internal hazards, external hazards, power states, shutdown states, etc.) PSA. The risk of radioactive release involving the spent fuel pool for the NPP at full-power operation shall be evaluated. If it can be shown based on conservative assumptions that the risk of radioactive release involving the spent fool pool is negligible (contribution to the Total Risk of Activity Release, TRAR less than 1%), no further analysis is

	necessary. Otherwise, a PSA shall be performed for the spent fuel pool, which follows the same requirements as set forth for NPPs. PSA Level 1 for full-power and shutdown, PSA Level 2 for full-power. PSA Level 2 for shutdown will be additionally required for operating license.
IRSN	IRSN EPR PSA is a generic EPR PSA. IRSN develops its own EPR PSA (the FA3 PSA developer is EDF, based on the previous PSA developed by AREVA). As EDF does not provide the PSA computerized model and to perform the better review possible, IRSN uses its own PSA for the FA3 PSA review The IRSN PSA scope is internal events (FA3 EDF PSA scope: internal events, internal fire and internal flooding, limited assessment of common failure of external grid and ultimate heat sink) Releases from other than the reactor core were considered? (ex: fuel pool): Reactor and fuel pool (as EDF) PSA Levels (Level 1, Level 2, Level 3, or extended Level 1): Level 1 and Level 2 (as EDF) (Note the IRSN development of EPR L1 and L2 PSA has started while EPR Fla3 was already under construction. This is more a “verification of design study” than a “design” PSA.
JNES	(JNES comment: It is inappropriate to limit “design stage PSA.” We consider this question is applicable to all of the scopes that this questionnaire covers.) Plant specific PSA is developed. (JNES comment: This classification should be one of options.) JNES develops PSA to be used for the evaluation of effectiveness of AM measures. This PSA includes internal event during power operation. Releases from other than the reactor core were considered? (ex: fuel pool): NO. PSA Levels (Level 1, Level 2, Level 3, or extended Level 1) : Level 1 and 1.5 PSA. (JNES comment: The meaning of extended Level 1 PSA is not clear. Level 1.5 PSA means that the scope of PSA does include evaluation of frequency of containment vessel failure but does not include source term evaluation. If the term of “extended Level 1 PSA” is used for this meaning, you may substitute by “extended Level 1 PSA instead of “Level 1.5 PSA.”
MHI	PSA for the design certification application review is for a design-specific. MHI has developed the PSA model for the design certification application review. Basically, full scope is considered for the design stage PSA except for the site-specific external hazards. The release from a spent fuel pool was qualitatively considered as a part of shutdown PSA. PSA Levels (Level 1, Level 2, Level 3, or extended Level 1): Level 1, Level 2, and Level 3 PSA were considered.
NRC	The PSA is developed by the applicant for a design certification (DC) or combined license (COL) in various stages. An applicant for a DC develops a design-specific PSA. Representative site characteristics may be assessed to support this PSA. An applicant for a COL develops a plant-specific PSA including assessment of specific site characteristics. At the final stage before commercial plant operation, the COL holder must develop a PSA that covers those initiating events and modes for which NRC-endorsed consensus standards on PSA exist one year prior to the scheduled date for initial loading of fuel. The use of consensus PSA standards is discussed in further detail in NRC’s Regulatory Guide 1.200, which refers to the PSA standard ASME/ANS RA-Sa-2009. The standard provides both process and technical requirements for an at-power Level 1 and limited Level 2 PSA for internal events, internal flood, internal fire, seismic, wind, external flood and other external events. The scope of the PSA typically does

	not consider sources other than the reactor core. Throughout the design process the applicant may choose to use the PSA to assist in developing insights and considering design options. The NRC only reviews a description of the PSA and its results during the review of a licensing application, which occurs when the final design phase has been reached.
NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	Since the project is only in a preparatory phase, not details are available in the design stage PSA yet.
ONR	UK AP1000 PSA: • The UK AP1000 PSA submitted for GDA is based on the standard small event tree / large fault tree approach, and is a Level 1 and Level 2 PSA. A simplified and generic Level 3 PSA has been performed (not site specific). Westinghouse submitted a separate PSA for the Spent Fuel Pool • The scope of the PSA includes consideration of internal initiated events and internal hazards and includes low power and shutdown operating states. Releases associated with non-core damage sequences and fuel pool has not been evaluated and integrated with the overall PSA results. • The PSA quantification for both Level 1 and Level 2 is carried out using the CAFTA software. The computerized PSA model in CAFTA was provided as part of the safety submission for GDA. UK EPR PSA: • The UK EPR PSA submitted for GDA has been carried out at Level 1, 2 and 3. In the PSA, fault trees are used to estimate the failure probability of the system missions. Event trees are used for estimating the Core Damage Frequency (CDF) due to each initiating event in the Level 1 PSA and further event trees are used to analyse potential failure sequences that could give rise to releases in the Level 2 PSA. • The scope of the PSA includes consideration of internally initiated faults, internal and external hazards and includes non-power operating states. The sources of radioactive releases considered are the reactor core, the spent fuel storage pool, the spent fuel handling facilities and the radioactive waste storage tanks. • The RiskSpectrum model, provided as part of the safety submission, has been specifically developed to deal with both level 1 and level 2 PSA that are fully integrated.
SNSA	The PSA model shall be site and plant specific. The development of PSA is the responsibility of the utility (it would probably be developed by vendor), but the regulator shall have access to the computerized PSA model and all of its regular updates. The regulation requires that the PSA includes all relevant modes of operation (from power to cold shutdown and refueling), all relevant initial events and hazards (internal and external), human reliability analysis, relevant dependencies, sensitivity and uncertainty analysis. The regulation regarding releases is not explicitly limited to those from the reactor core damage, thus any relevant releases must be assessed. For the NPP at least Level 1 and 2 are required.
STUK	Reactor type specific including site characteristics. The vendor in cooperation with the applicant of the license. The plant specific level 1 and 2 PRA includes internal initiators, fires, flooding, harsh weather conditions and seismic events for full power operation mode and for low power

	and shutdown mode. Fuel pools are included. A plant specific, design phase level 1 and 2 PRA.
UNISTAR	The site-specific CC3 PRA incorporates by reference the U.S. EPRTM generic PRA developed by AREVA-NP. WARNING: AS UNISTAR'S CC3 PRA WAS DEVELOPED BY AREVA-NP, UNISTAR'S ANSWERS TO THIS QUESTIONNAIRE SHOULD BE ANALYZED WITH THE AREVA'S ANSWERS FOR THE U.S. EPRTM The PRA is developed by the vendor (AREVA-NP). UNISTAR has not access to the computerized model. Internal initiating event, fire and flood for all states. Screening analysis for external hazards (Airplane crash, high wind, industrial activities). PRA-based SMA Releases from other than the reactor core were considered? (ex: fuel pool) No Level 1 and 2 in the SAR, Probabilistic analysis of severe accident consequences (Level 3-) in the Environmental Report
	C 5. What is/was the role of the PSA during the development of the plant design? • **safety demonstration, supporting the choice of design options, well-balanced safety concept, defence in depth assessment (multiple failures initiating events identification…), appreciation of the improved safety level compared to existing plants, etc.** • **please explain PSA roles during conceptual design, detailed design, construction, commissioning and initial operation stages**
AREVA OL3	Demonstration of meeting probabilistic safety goals Provide probabilistic insights for the system design, especially for support systems Compare probabilistically design alternatives
AREVA TSN	Demonstration of meeting probabilistic safety goals
AREVA US	Demonstration of meeting probabilistic safety goals Providing probabilistic insights for the system design
BARC	Safety demonstration, supporting the choice of design options, well-balanced safety concept, Defence in depth assessment (multiple failures initiating events identification…), appreciation of the improved safety level compared to existing plants, etc.
Bel V	-
EDF FA3	designing and optimizing the facility during the design phase and life of the site, confirm the well-balanced risk profile of the design, broaden the deterministic design scope of systems, with specific beyond-design analysis (multiple failures conditions),

	justify the maintenance planning, support the severe accidents analysis, confirm the protections from external and internal hazards, assess the safety level increase compared to existing plants.
EDF EPR UK	In addition to its use in assessing the current design, the PSA has provided insights throughout the design process, leading to a number of design improvements, and has provided input to other aspects of the Pre-Construction Safety Report (PCSR). More specifically, it has been used: To extend the deterministic design basis (Beyond Design Basis Analysis) in order to achieve a balanced design, ensuring that there are no ‘cliff edge’ effects. To confirm the appropriateness of protection of the plant against certain internal and external hazards. To verify that the design of the severe accident mitigation features allows the risk of radioactive product release to be reduced to an acceptably low level. To calculate the plant seismic capacity in order to demonstrate that the plant has sufficient margin beyond the safe shutdown earthquake. To assess the improvement in the safety level in comparison with existing reactors. To assess the impact of preventative maintenance. To design a non-computerized safety system To support some specific cost-benefit analyses, i.e. assessments stating that the residual risk is as low as reasonably practicable (ALARP demonstration)
ENEL	During the plant design the PSA helps the identification of the design fragilities and it supports the choice of design options.
ENSI	Based on the ordinance on “Hazard Assumptions and Evaluation of Protection Measures against Accidents in Nuclear Installations” it is required to design NPPs against earthquakes and external floods having a frequency of exceedance equal or higher than 1E-04/year. The specific acceleration/flood level is derived by the use of probabilistic methods. Additionally, the PSA will be used to demonstrate a sufficient level of safety (CDF lower than 1E-5/yr as required by the Nuclear Energy Ordinance) and a well-balanced safety concept. PSA roles during conceptual design, detailed design, construction, commissioning and initial operation stages: • General license: The same probabilistic methods and concepts as in the PSA shall be used for the evaluation of the external hazards. • Construction license: A Level 1 PSA for all operational states (full power, low power, and shutdown) and a Level 2 PSA for full power is required. Acceptance criteria are based on CDF, FDF, and LERF. • Operational license: The PSA for the construction license has to be updated. A Level 2 PSA for low power and shutdown is required. Additionally, the balance among the contributors to risk shall be investigated.
IRSN	PSA was used by EDF for early design verification of EPR Reactor, several design improvement being defined based on these PSA insights and following the discussions with the French and German safety authorities. At IRSN, in the FA3 Project today context (anticipated instruction of the application for commissioning), the PSA plays an important role for the:

	• verification of the NPP safety level, • design verification, • demonstration of “practical elimination” of the Large Early Releases and that other accident sequences would conduct to limited impact (in time and space) • Risk Reduction Categories (RRC-A) definition.
JNES	In ABWR concept design stage, PSA insights were used to decide the configuration of engineering safety features, especially ECCS design. In specific ABWR plant design stage, detail design of accident management countermeasures and facilities was developed based on PSA results, and effectiveness of the accident management features to reduce risk of the plant were evaluated using level 1 and 1.5 PSA after detail design finished.
MHI	The PSA result is utilised for selecting the advanced design options and safety demonstration. The PSA result is utilised for selecting the advanced design options. Also planning the use of PRA for detail design and operations.
NRC	The PSA summary and results are submitted to NRC after the final design is completed. Applicants may report on the role of the PSA during earlier design stages and how the PSA results and insights influenced the design development. The role of the PSA during the licensing process is described in the response to C2. These regulatory uses of PSA can assist NRC reviewers and inspectors in assuring the safety of the design during the licensing, construction, and commissioning phases of the plant. During the operational phase of the plant, the licensee must conform to regular PSA maintenance and upgrades as specified in the endorsed PSA standard. Also during the operational phase, NRC uses its internal PSA models, the SPAR models, to support the reactor oversight process, assessment of generic issues, assessment of precursors to core damage, and scoping studies. NRC is currently developing SPAR models for new reactor designs, which may be used to support licensing reviews and inspection activities during pre-operational phases.
NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	All these aspects will be included as bidding requirements. The actual role of PSA in plant design can only be described in detail when the specific design have been selected based on the evaluation of bids. please explain PSA roles during conceptual design, detailed design, construction, commissioning and initial operation stages: See the answer to the previous question.
ONR	UK AP1000 PSA: • The Probabilistic Safety Analysis was performed by Westinghouse to support the design of the AP600 in the 1990s. This practice was carried over to the design of the AP1000. • Westinghouse has developped a SAMDA (Severe Accident Management Design Alternatives) analysis which is intended to support the ALARP demonstration. Potential modifications (design alternatives) have been assessed in this analysis. UK EPR PSA: • EDF and AREVA claim that the developed PSA has allowed a well-balanced system and process design to be achieved. They also consider that it has also provided a reasonable assurance that the plant complies with the stated safety objectives.

	• ALARP (As Low As Reasonably Practicable) arguments are also presented as part of the GDA submission.
SNSA	Legislation requirements are: • the design must meet quantitative limits, • PSA must be used to determine the postulating initiating events (and design basis), • PSA shall assure well-balanced safety concept. please explain PSA roles during conceptual design, detailed design, construction, commissioning and initial operation stages: The roles of PSA use are not specified in such detail. For use of PSA in the design phase, the answer to the former question is applicable.
STUK	Safety demonstration, well-balanced safety concept, defense-in-depth assessment, appreciation of the improved safety level compared to existing plants, A plant specific, design phase level 1 and 2 PRA is required as a prerequisite for issuing a positive statement for an application of the construction license for a new NPP design and a complete level 1 and 2 PRA for issuing a positive statement for an application of a operating license. In addition the design phase PRA is important to ensure that the new build can meet the numerical safety goals.
UNISTAR	Respect of regulatory safety objectives and appreciation of the improved safety level compared to existing plants. The site-specific PRA confirmed the well-balanced risk profile of the design. Conceptual design: input for the Reliability Assurance Program Detailed design/ construction/ commissioning: Support License Amendment including design and Tech spec. changes. Support the Risk-informed decision making process
	C6. Which risk-informed applications are used in the project? Are the applications based on regulatory requirements? • **safety classification of SSCs** • **program for technical specifications** • **program for inservice testing** • **program for inservice inspection** • **program for on-line preventive maintenance** • **cost-benefit** • **development of emergency operation procedures**
AREVA OL3	PSA is used • In the review of the technical specifications • In the review of safety classification • in the working up of programs for testing and preventive maintenance

	• in the working up of disturbance and emergency operating procedures. • as an input for planning of simulator training for selection of accident and incident sequences. • As support for riskinformed in-service inspection (RI-ISI) These applications are required by the safety authority in their regulation (YVL guides)
AREVA TSN	No Risk-Informed application during the construction phase
AREVA US	No risk-Informed application were proposed during the design certification phase
BARC	PSA insights were used for development of Technical Specifications for safety systems.
Bel V	-
EDF FA3	Program for technical specifications.
EDF EPR UK	No Risk Informed application at this stage. Note that some PSA-based ALARP demonstration were developed.
ENEL	Our preliminary general plan would consider the risk-informed approach for the program for inservice testing and for inservice inspection. Perhaps the development of the emergency operation procedures will be on risk informed basis. This preliminary program is not based on regulatory requirements because the Nuclear Energy Agency has not established any regulation yet.
ENSI	• safety classification of SSCs: According to the guideline ENSI-A06, components with a Fussel-Vesely ≥ 0.001 or a Risk Achievement Worth ≥ 2 are regarded as significant to safety from the PSA point of view. According to the guideline ENSI-G01 these components shall be classified. • program for technical specifications According to ENSI-A06, in defining the allowed outage times, it shall be ensured that components shown to be significant to safety from the PSA point of view (see above) are considered in the technical specifications (completeness), and assigned to correspondingly short allowed outage time categories (balance). • program for inservice testing It should be ensured that components significant to safety are assigned to a correspondingly short test interval. • program for on-line preventive maintenance According to ENSI-A06, maintenance work in an existing NPP shall be planned in such a way that: o no component unavailability configuration resulting from maintenance will result in a Conditional Core Damage Frequency (CCDFi) greater than 1 · 10-4 per year, and o the total cumulative maintenance time for components shall be limited such that the portion of the Incremental Cumulative Core Damage Probability (ICumCDP) resulting from maintenance is less than 5 · 10-7.

	Compliance with the above mentioned requirements shall be demonstrated either by a previous enveloping analysis along with an additional probabilistic evaluation of operational experience or assessed with the help of a risk monitor. Any deviations from the requirements on maintenance planning mentioned shall be justified. For new NPPs other criteria may be defined in the future. • cost-benefit There exists no specific cost-benefit criterion in Switzerland. However, there are cost-benefit considerations when regulatory decisions (e.g. concerning backfits) are made. • development of emergency operation procedures There are no specific requirements.
IRSN	• safety classification of SSCs; not yet, but is foreseen • program for technical specifications; not yet, but is foreseen (methodology under assessment) • program for inservice testing; not yet but is foreseen (methodology under assessment) • program for inservice inspection; no • program for on-line preventive maintenance; some PSA role is expected • cost-benefit; not discussed for EPR at the moment (concern only Gen II reactors during periodic safety review), • development of emergency operation procedures; some PSA role There are no regulatory requirements regarding the above PSA applications
JNES	Development of accident management countermeasures and emergency procedures. This was done based on recommendation by the regulatory body.
MHI	Regarding the PSA for the design certification application review, these risk-informed applications are not used in the design since they are not required by the regulation.
NRC	As discussed in the response to C2, the PSA results and insights are used to support several regulatory programs, including • Risk-informed categorization and treatment of SSCs (as defined in 10 CFR 50.69) • Development of the Reliability Assurance Program (RAP) • The Regulatory Treatment of Non-Safety Systems (RTNSS) • Development of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) • Development of Technical Specifications (TS) • Maintenance Rule (10 CFR 50.65) implementation. NRC is currently engaging with external stakeholders to review risk-informed regulatory programs and their application to new reactors. Additional information on this effort can be found in the paper SECY-10-0121 Modifying the Risk-Informed Regulatory Guidance for New Reactors and the staff requirements memorandum response to SECY-10-0121 (available in NRC's Agencywide Document Access and Management System [ADAMS] at ML102230076 and ML110610166 respectively).

NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	No risk informed applications are used in the current, preparatory phase of the project. Expectations are that applications will be as follows in future developments of the project: • safety classification of SSCs: It is expected that safety classification of SSCs will be risk-informed, although currently no specific regulatory requirements exist. • program for technical specifications: It is expected that technical specifications will be risk-informed, although currently no specific regulatory requirements exist. • program for inservice testing: This is under discussion and no decision has been made yet as far as the role of PSA in in-service testing is concerned. • program for inservice inspection: This is under discussion and no decision has been made yet as far as the role of PSA in in-service testing is concerned. • program for on-line preventive maintenance: The whole area of allowing on-line preventive maintenance is under discussion including the role of PSA in an on-line preventive maintenance program. The current safety regulation generally does not allow on-line preventive maintenance. • cost-benefit: No consideration has been given to this aspect yet. • development of emergency operation procedures: It is expected that emergency operating procedures will be risk-informed, although currently no specific regulatory requirements exist.
ONR	PSA has provided or will provide input to the development of the Safety classification of SSCs for both reactors. Cost-benefit methods have been used for both reactors to identify design improvements. In addition, ONR expects that the PSA would be used to provide input to the development of technical specifications, in-service testing, in-service inspection, preventive maintenance procedures, emergency operation procedures etc. in due course.
SNSA	The legislation requires that PSA must be used for: • event analysis, • on-line maintenance (if licensee performs it), • identification of needed modifications to the plant and plant's operating procedures as well as identifying the needs and developing severe accident management procedures, • modification implementation (if applicable); also to track cumulative risk due to changes, • periodic safety review, • verification of maintenance, inspection and testing programs, • development and verification of operator training program. Also PSA can be used for other applications not required by the legislation.
STUK	• safety classification of SSCs: Risk informed safety classification of SSCs used for OL3. A specific regulatory requirement exists. • program for technical specifications: Risk informed application for OL3. A specific regulatory requirement exists.

	• program for inservice testing: Risk informed application for OL3. A specific regulatory requirement exists. • program for inservice inspection: Risk informed application for OL3. A specific regulatory requirement exists. • program for on-line preventive maintenance: Risk informed application for OL3. A specific regulatory requirement exists. • cost-benefit: No requirement on this. • development of emergency operation procedures: Risk informed application for OL3. A specific regulatory requirement exists.
UNISTAR	• safety classification of SSCs: not yet used but foreseen • program for technical specifications: not yet used but foreseen • program for inservice testing: not yet used • program for inservice inspection: not yet used but foreseen • program for on-line preventive maintenance: not yet used but foreseen • cost-benefit: kind of (SAMDA) • development of emergency operation procedures: not yet used but foreseen
	C 7. How are uncertainties and limitations addressed? **please discuss the difficulties in using the PSA for a NPP in design stage, uncertainties, etc.**
AREVA OL3	Data uncertainties are analyzed quantitatively using log-normal distributions For some uncertainties sensitivity analyses are performed For important assumptions and limitations a qualitative uncertainty analysis is performed
AREVA TSN	Data uncertainties are analyzed quantitatively using log-normal distributions For some uncertainties sensitivity analyses are performed Limitations and assumption are indicated.
AREVA US	All inputs uncertainties are analyzed quantitatively using: • Lognormal • Constrained Non-Informative • Beta • Gamma

	• Uniform • Double-Delta State-of –Knowledge Uncertainty is considered. Limited modeling uncertainties are analyzed and multiple sensitivity analyses are performed
BARC	At the design stage, plant-specific data are not available. However, the operating experience of other Indian NPPs of the similar design was used for estimating some of the initiating event frequencies. The software contribution in reliability analyses for digital systems was not addressed in PSA, however, extensive Verification and Validation of the software was carried out to ensure its reliability in qualitative manner.
Bel V	-
EDF FA3	Propagation of parameter uncertainties in the final core melt-down frequency.
EDF EPR UK	Parameter uncertainties are propagated to obtain the uncertainties for the intermediate and final results of the analysis, these results, along with their uncertainties, are compared to the acceptance criteria. Model uncertainties are dealt with sensitivity analysis. Limitations and assumptions are listed and discussed.
ENEL	There are many uncertainties in using the PSA for NPP in design stage, the first problem is related to the site dependent phenomena, it is also necessary to evaluate different models with uncertainties and sensitivity study.
ENSI	The PSA for the new plants are not yet available. The question can not be answered in detail now. However, since mean values are used in the PSA, uncertainties in the reliability of components, in estimating hazard frequencies, human error probabilities etc. are considered. To accommodate the modelling uncertainty, Switzerland has a robust reactor oversight process – the integrated decision making process: If deterministic safety analysis, probabilistic safety analysis or operational experience lead to different requirements, the most restrictive requirement applies.
IRSN	Some uncertainties about assumptions or modeling related to new design are mainly addressed by the comparison between the independently PSAs developed by IRSN and EDF. IRSN performs and also requests to EDF to perform sensitivity studies when certain contributions are shown to be important and uncertain. Always the deterministic aspects prevail and design margins are preferred when high uncertainties are highlighted. For L2 PSA, IRSN has developed a specific methodology for uncertainties assessment applied on Gen II reactor. It will be used for Gen III (and IV …) reactors but experience is missing in the context of EPR (a safety margin approach for severe accident may although be preferred for Gen III reactors)
JNES	Conservative assumptions were used for emergency procedures, e.g. emergency power supply from the adjacent unit, which were not issued at the time of evaluation. Uncertainties of latest technologies, e.g. digital I & C or touch panel display for operations, were evaluated in comparison with conventional equipments.
MHI	In order to estimate the impact of uncertainties on PRA insights and to identify important insights and assumptions, various sensitivity analyses and importance analyses were performed.
NRC	NRC has developed guidance on the treatment of uncertainties associated with PSA in NUREG-1855 (available in NRC's ADAMS at ML090970525.) In addition, as part of the licensing reviews for new designs, NRC may request additional calculations and studies be performed in order to assure that the uncertainties associated with novel aspects

	of the designs are adequately addressed. These may include, for example, studies demonstrating the appropriate use of conservatism in developing PSA success criteria, the use of bounding parameters for PSA supporting calculations and sensitivity studies, and testing programs to validate calculations.
NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	As given in the answers to questions C3 to C5, no information is available on the role of PSA in the design stage, because no specific plant design has been selected yet. Bidding requirements will include the requirement for addressing and evaluating uncertainties and limitations in PSA.
ONR	Limitations due to the unfinished design stage have been addressed in the PSAs using generic conservative assumptions. Reconsideration of these assumptions will be needed in due course. Uncertainty, importance and sensitivity analyses have been performed for the Level 1 and Level 2 PSA.
SNSA	The legislation requires that sensitivity and uncertainty analysis must be provided together with PSA itself. Like in general PSA use, uncertainties in the design stage shall be determined and assessed and PSA results shall be used in accordance with that.
STUK	Because the safety systems are not yet designed in all details, the design phase PRA includes uncertainties related to the systems configuration. There may be also uncertainties involved in the data used if the plant unit in question is a prototype. In the design phase PRA, operating experiences collected from similar plants or corresponding applications shall be used. As to the PRA of an operating plant, the plant specific data and if necessary, combined with data received from other similar plants or corresponding applications shall be used, and in the absence of such a data, general data shall be used. The feasibility and uncertainty of the data shall be justified. Provided that no adequate design, site and reliability data are available for the design phase PRA or if some safety related systems are constructed using a technology such that there are no well established methods available for computing the system reliability estimate, expert judgment, experiences and information from corresponding applications and corresponding sites can be used. In that case the estimation procedure must be justified.
UNISTAR	The reactor basic design is mostly based and safety-justified by deterministic studies. The PSA can be used complementarily for basic design optimizations or specific safety analyses (e.g. well-balanced design or beyond design safety analysis). For new reactors, risk metrics are generally very small with high uncertainties and should be manipulated carefully in the decision-making process.
	C 8. PSA guidelines used? Are the guidelines specific for new reactors? **• reference to the guidelines used** **• the scope of the guidelines (Level 1, 2, 3)**
AREVA OL3	Based on international PSA guides and own experience with many PSA methodology report for the various tasks of this PSA were written.
AREVA TSN	Based on international PSA guides such as Safety Series No 50-P-4 and IAEA-TECDOC-1144. Guidance from ASME RA-S 2008 PRA Standard And local regulation HAF102 and a specific PSA guide HAF.J008, (January 2008), Technical Document for Nuclear Safety Regulations, Standard formats and Contents of

	Nuclear Power Plant Probabilistic Safety Assessment Report. These guides are not specific for new reactors.
AREVA US	Guidance from ASME RA-S 2008 PRA Standard and the US NRC Regulatory Guide 1.200 – that are non-specific for new reactors.
BARC	AERB/NPP&RR/SM/O-1 and IAEA-SS-50-P4 were used.
Bel V	-
EDF FA3	The PSA are included in the design process according to French RFS 2002-01 and EPR Technical Guidelines.
EDF EPR UK	The UK EPR PSA follows the UK and international practices as principally described in the HSE Safety Assessment Principles, the relevant Technical Assessment Guide from HSE, the IAEA Safety Series and EUR generic requirements: • UK Health and Safety Executive (HSE). Safety Assessment Principles for Nuclear Facilities. 2006 Edition Revision 1. January 2008. • UK Health and Safety Executive (HSE). Technical Assessment Guide, Probabilistic Safety Anaysis, T/AST/030 issue 03, February 2009 • Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (level 1). Safety Series No 50-P-4. IAEA. August 1992. • European Utility Requirements for LWR Nuclear Power Plants, Volume 2: Generic Nuclear Island Requirements, Chapter 17: PSA Methodology, Revision B. EUR Document. November 1995.
ENEL	IAEA, NRC, guidelines from French and English Regulatory bodies.
ENSI	ENSI-A05 (quality and scope of the PSA) and ENSI-A06 (PSA-Applications), are not specific for new reactors. Although both guidelines are valid for new plants from a formal point of view, they focus on the existing plants and they may be subject to changes in the future. the scope of the guidelines (Level 1, 2, 3) Level 1 and level 2 (full-power and non-full-power).
IRSN	French PSA fundamental safety rule (2002-01), ASAMPSA2 guideline will be used in the future for L2 PSA. the scope of the guidelines (Level 1, 2, 3): Level 1, internal events for the French PSA fundamental safety rule (2002-01) although some rules are applicable for all PSAs. ASAMPSA2 guideline is not really specific for new reactors but provides some elements.
JNES	Guidelines for PSA quality for NPP prepared by NISA, Japanese regulatory body, and JNES, the technical supporting organization for NISA, are used. The level 1 and 2 PSA standards prepared by Atomic Energy Society of Japan are also referred. The scope of PSA is level 1 and 1.5 for the internal events during power operation.
MHI	The latest revision of U.S. NRC Regulatory Guide 1.200 and ASM/ANS PRA Standard (ASME/ANS RA-S-2008, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications” and associated addendum) are used.

	Bellows are also used. • PRA Procedures Guide, NUREG/CR-2300 • Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150 • EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, NUREG/CR-6850 • American National Standard External-Events PRA Methodology, ANSI/ANS-58.21-2007 the scope of the guidelines (Level 1, 2, 3): Full scope is covered by these guidelines.
NRC	The current guidelines are outlined in the NRC's Regulatory Guide 1.200 and the PSA standard, ASME/ANS RA-Sa-2009. The standard was not developed specific for new reactors. An additional PSA standard document is currently under development to address PSA for pre-operational reactors. The standard provides both process and technical requirements for an at-power Level 1 and limited Level 2 PSA for internal events, internal flood, internal fire, seismic, wind, external flood and other external events. PSA standard documents are also under development in the areas of Level 2, Level 3, and low power/shutdown.
NRI	The Czech Regulatory has prepared at this time guideline for Level 1 PSA BN-JB-1.6. It is now released for trial use and comments. PSA for new reactors is not distinguished in this guideline.
NUBIKI	It is expected that the PSA for the new reactors in Hungary will be in agreement with Regulatory Guide 3.11. on PSA of the Hungarian Atomic Energy Authority. This guide does not differentiate between existing/operating and new reactors. There are initiatives in place to improve the guide so that it can better represent the state of-the art and can be applied to existing and new reactors too. the scope of the guidelines (Level 1, 2, 3): The guide addresses Level 1 and Level 2 PSA in accordance with the regulatory requirements on PSA in Hungary.
ONR	The main standards and criteria used are ONR's Safety Assessment Principles (SAPs) (Ref. 1). The PSA Step 3 assessment strategy (Ref. 5) identified SAPs FA.10 to FA.14 and Numerical Targets 7 to 9 as the relevant parts of that document. Attention has also been paid to relevant parts of the International Atomic Energy Agency (IAEA) standards (Ref. 6) and the Western European Nuclear Regulators' Association (WENRA) reference levels (Ref. 7). The above PSA related SAPs, IAEA standards and WENRA reference levels are embodied and enlarged in ND's Technical Assessment Guide (TAG) on PSA (Ref. 2) and it is this guide that provides the principal means for assessing the PSA in practice (Level 1, 2 and 3 and use of PSA to support decision-making. All for conceptual design, detailed design, construction, commissioning and operation stages).
SNSA	At the SNSA we don't have our own guidelines. Instead we use the IAEA as well as the U.S. NRC's guidelines.
STUK	International PSA Guides. See AREVA C8. General requirements are given in the Regulatory Guide YVL 2.8 "Probabilistic safety analysis in safety management of nuclear power plants " used for OL3. the scope of the guidelines (Level 1, 2, 3): The guide YVL 2.8 addresses Level 1 and Level 2 PSA in accordance with the regulatory requirements on PRA.
UNISTAR	No specific guideline for PRA for new reactors in the U.S. but the PRA has been reviewed against the requirements of the ASME/ANS Ra-S ANSI/ANS-58 21-2007 standards on PRA the scope of the guidelines (Level 1, 2, 3): Level 1 (IIE, fire, flood) and LERF in power states for ASME/ANS RA-S 2008 standard and external hazards for the ANSI/ANS-58

	21-2007
	C9. Are you involved in an international group dealing with that specific PSA / type of reactor?
AREVA	Not formally –but the work inside the AREVA presented an opportunity to compare different international PRA experiences and approaches as used in the EPR™ PSA model developments (as presented in this table: German, French and US)
BARC	Yes, CANDU Senior Regulator Groups' PSA activities.
Bel V	-
EDF	No
ENEL	I'm involved in the WGRISK group and I participate to the WANO workshops.
ENSI	No
IRSN	IRSN supports the ASN participation to EPR group of MDEP.
JNES	NO
MHI	No
NRC	NRC participates in MDEP to gain a better understanding of new reactor design regulatory issues.
NRI	No
NUBIKI	No.
ONR	UK AP1000 PSA: Participation in Multinational Design Evaluation Programme (MDEP) Advanced Passive 1000 Working Group which will be held on 12-13 May 2011 is expected although there is not a specific PSA topic group. UK EPR PSA: ONR represents the UK in the Multinational Design Evaluation Programme (MDEP) - In the PSA assessment, insights from other regulators looking at EPR variants have been gained through the MDEP. ONR have shared assessment views and findings with the MDEP partners (USA, France, Finland, Canada and China) and contributed to joint working. GDA Step 3 review main inputs from MDEP were from the Finnish nuclear safety authority, STUK and US NRC: ONR has commissioned work to examine the questions raise by STUK on L2 PSA modelling suitability and possible quantification errors.

	US NRC has provided a list of their questions and the responses they received related to the US EPR and in an effort to make use of this information ONR has looked at the US EPR PSA to gauge the extent to which the NRC views are directly relevant to the UK EPR.
SNSA	At the moment the SNSA is not involved in such groups.
STUK	YES (MDEP)
UNISTAR	No, but participation to internal EDF "EPR Family Group"
	D. Internal Initiators PSA Level 1 technical aspects
	D 1. How was the list of initiating events developed? **• based on existing reactor initiating events list, specific systematic method, etc.** **• were new initiating events considered that were not considered in existing PSAs?**
AREVA OL3	The list of initiating events was developed plant and site (external events) specific. Bases were list of initiating events for similar PWRs (reference plants), Master Logic Diagram, system analyses/FMEAs
AREVA TSN	Process performed according to IAEA-TECDOC-719 • Analysis of generic lists of IEs (IAEA-TECDOC-719, NUREG/CR 3862, NUREG/CR 6928) • Analysis of previous PSAs: PSAs of German and French operating NPP and Previous EPR™ PSAs including specific PSA studies. • FMEA • Analysis of Common Cause Initiators
AREVA US	• Analysis of generic lists of IEs (NUREG/CR 3862, NUREG/CR 6928) • FMEA • Analysis of the plantspecific/ support system related initiators • Guidance from ASME RA-S 2008 PRA Standard and the US NRC Regulatory Guide 1.200 were used.
BARC	Based on master logic diagram, the list of initiating events from earlier PHWR PSAs and engineering evaluation, relevant international literature. No new initiating event was considered in PSA.
Bel V	-
EDF FA3	The list of initiating events was developed based on French PSA list, taking into account French and international feed-back, completed by specific analysis of spurious CC.

EDF EPR UK	The following methods and sources have been used to ensure a systematic and exhaustive search for potential initiating events. This is consistent with the approach discussed in IAEA-TECDOC-719 "Defining Initiating Events for Purposes of Probabilistic Safety Assessment": • Engineering evaluation or technical study of the plant, • Previous PSAs such as the supporting document for FA3 PSAR Chapter 18.1 [Ref], the Basic Design Report or French 1300 MWe and 900MWe PSA, • Lists of initiating events such as NUREG/CR 3862, NUREG/CR-6928 and NUREG-1829, • Analysis of operating experience for actual plant discussed in FA3 PSAR Chapter 18.1, • FMEA of EPR systems.
ENEL	We suppose to develop the initiating event list on the basis of the past experience with the necessary modifications related to different design of the plant.
ENSI	Guideline ENSI-A05 describes the methods (e.g. system analysis, FMEA …) how a comprehensive list of potential initiating events shall be identified. Additionally, the guideline lists specific initiating events which have to be considered. No PSA for a new plant available.
IRSN	The IRSN PSA initiating events list is adapted from EDF EPR PSA and based on French and German PWR experience. Some new initiating events are considered for EPR PSA, mainly related to spurious I&C signals. Loss of digital I&C systems will be also addressed in the IRSN PSA.
JNES	based on existing reactor initiating events list were new initiating events considered that were not considered in existing PSAs? NO.
MHI	The first step is to comprehensively identify the potential initiating events that result in a reactor trip. And then screening evaluation is performed by gathering information in other PSA studies. Failure mode and effects analyses (FMEA) are performed to identify the potential initiating events as a result of the structure, system, and components (SSCs) failure. The second step is to categorise the initiating events into several groups to define the minimum set of initiating events identified in the first step. were new initiating events considered that were not considered in existing PSAs? Yes. By performing the FMEA and confirming that the initiators are properly mapped to the initiating event categories, a complete set of initiating events has been developed taking into account the plant design features. As the result, newly considered initiating events are all screened out.
NRC	Applicants determine which specific systematic method is used for developing the initiating events list in accordance with existing PSA standards and guidance. This typically involves referencing a generic list, e.g., NUREG/CR-6928. New initiating events are considered based on any unique or novel plant design features.
NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	It is noted that no details are available yet because the PSA in question is not in place yet. However, some of the issues listed in the questions below can be addressed at the

	level of expectations/requirements based best practices as follows. Existing initiating event lists as well as design specific systematic analyses are seen necessary to develop the list of initiating events. were new initiating events considered that were not considered in existing PSAs? New initiating events need to be addresses if required by the specifics of the plant design.
ONR	UK AP1000 PSA: The identification process for "generic" initiating events relies almost completely on NUREG/CR-3862. In addition, a number of plant specific initiating events (specific to AP1000 systems) were identified in the AP1000 PSA documentation. UK EPR PSA: EDF/Areva approach based on IAEA Tecdoc 719 and includes: • Previous PWR PSAs • lists of IEs of PWR plants such as NUREG/CR 3862 • Probabilistic Analysis of French and German operating experience for actual PWR plant • FMEA of EPR systems
SNSA	There are no detail requirements regarding how to develop initiating events. We would use the IAEA and US NRC criteria and guides.
STUK	Existing initiating event lists from reference plants. Site specific characteristics included. See AREVA D1. were new initiating events considered that were not considered in existing PSAs? No.
UNISTAR	based on existing reactor initiating events list, specific systematic method, etc.: YES: US NRC NUREG reports were new initiating events considered that were not considered in existing PSAs? No
	D 2. What types of supporting studies are used? **• Safety Report studies, specific studies for PSA, etc.**
AREVA OL3	Specific support studies performed for the use of OL3 EPR™ PSA
AREVA TSN	Specific support studies performed for the use of TSN EPR™ PSA
AREVA US	Specific support studies performed for the use of US EPR™ PSA
BARC	Safety analysis reports, design basis reports and PSA reports of other PHWRs.

Bel V	-
EDF FA3	Specific studies for PSA.
EDF EPR UK	Specific studies for PSA (best estimate)
ENEL	We think that we will use the past studies developed for the PSA and we will develop the necessary new models related to the new design.
ENSI	PSA is not yet available. Safety Report among other reports must be available. The scope of the reports will be defined in a new guideline.
IRSN	IRSN Level 1 PSA uses the results of EDF studies when applicable: • Safety Report studies, • Specific PSA support studies.
JNES	PSA procedure manual and PSA report of Japanese representative ABWR are used. Both were done by JNES.
MHI	ASME/ANS PRA Standard (ASME/ANS RA-S-2008) and its associated addenda (ASME/ANS RA-Sa-2009) are used.
NRC	Applicants determine which specific studies are used to support PSA development in accordance with existing PSA standards and guidance. These may include, for example, system engineering analyses and thermal hydraulic analyses for determining success criteria for mitigating systems.
NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	No information is available yet.
ONR	Specific studies for PSA have been developed in both cases.
SNSA	There are no detail requirements regarding the supporting studies. Basically the safety report studies would be used complemented by the special hazard assessment studies.
STUK	Safety reports specific to OL3 design. Success criteria and accident progression analyses.
UNISTAR	specific studies for PSA
	D 3. What kind of reliability data and CCF data are used? **• sources of data for the components of the same type as in the existing NPP PSAs?** **• sources of data for evolutionary components / components with limited operating experience?** **• if data were not available for certain components, then what methods were used to address component reliability?**

AREVA OL3	Reliability data: • T-Book, Reliability Data of Components in Nordic Nuclear Power Plants, 6th edition, 2005 • VGB PowerTech e.V. Zentrale Zuverlässigkeitsund Ereignisdatenbank (ZEDB), Zuverlässigkeitskenngröß en für Kernkraftwerkskomponent en, Dezember 2007, ISSN 1439-7498 • EIReDA 1998, European Industry Reliability Data Bank, Joint Research Centre of the European Commission, 1998 • EGG-SSRE-8875, Generic component failure data base for light water and liquid sodium reactor PRAs, February 1990 • NUREG/CR-6928, Industry-Average Performance for Components and Initiating Events at U.S., Commercial Nuclear Power Plants, January 2007 CCF: • European Utility Requirements, Volume 2: Generic Requirements, Chapter 17 PSA Methodology, November 1995 CCF: • European Utility • Requirements, Volume 2: • Generic Requirements,
AREVA TSN	-
AREVA US	Reliability Data: • ZEBD data EG&G data (EGG-SSRE- 8875) • EIReDA (very little) • NUREG/CR-6928 and NUREG/CR-5750 (for initiating events) CCF Data: • NUREG/CR-5497
BARC	The generic data from IAEA-TECDOC-478 and IEEE-500 std. were used for the components reliability data in PSAs. The operating experience of other Indian NPPs of the similar design was used for estimating some of the initiating event frequencies. The generic CCF parameters were used from NUREG/CR-5801. The limited operating experience of equivalent components was used with expert judgment for the components where data were not available to address component reliability.
Bel V	-
EDF FA3	EPR component are compared to existing components in French EDF or German ZEDB2000 and 2004 databases. In case of lack of data, the information comes from American EG&G or NUREG database. CCF parameters are taken from EDF or NRC database.

EDF EPR UK	Reliability data are derived mainly from operational experience feedback from France (EDF internal reports), and Germany, supplemented by data from the EG&G generic reliability database. • VGB PowerTech e.V., Zentrale Zuverlässigkeits- und Ereignisdatenbank (ZEDB), Zuverlässigkeitskenngrößen für Kernkraftwerkskomponenten, Dezember 2007, ISSN 1439-7498 • EGG-SSRE-8875, Generic component failure data base for light water and liquid sodium reactor PRAs, February 1990 CCF parameters are taken from the European Utility Requirements. • European Utility Requirements, Volume 2: Generic Requirements, Chapter 17 PSA Methodology, November 1995
ENEL	The sources of data for the components are the same as in the existing PSA, the idea is to use the international industrial and nuclear databases. The experience obtained during the other plant life is a fundamental database and it will be supported by the supplier's data. For the evolutionary components probably we will use the results of the ancient components with the supplier's information.
ENSI	Not yet decided. Probably data from existing NPP PSAs (more or less generic data) will be used. sources of data for evolutionary components / components with limited operating experience? Not yet decided. if data were not available for certain components, then what methods were used to address component reliability? Not yet decided. The current approach is to calculate reliability data for such components based on experiments (and/or expert judgment).
IRSN	IRSN PSA uses data from EDF or German operating experience data, as applicable (similar with EDF EPR PSA) In the IRSN PSA the data for new/evolutionary experience are generally assigned by expert judgment, based on similarities with components of existing NPPs. For components for which data are not available in the EDF operating experience (or available/applicable German data) IRSN PSA uses data from international available sources (NUREG, IAEA, etc.)
JNES	Japanese component failure data based on the domestic operating experiences. sources of data for the components of the same type as in the existing NPP PSAs? YES. sources of data for evolutionary components / components with limited operating experience? Failure data of digital I & C components derived from IAEA documents are used. There is no component failure data with limited operating experienced. if data were not available for certain components, then what methods were used to address component reliability? Data of similar component in the viewpoint of mechanism would be used.
MHI	Following sources are utilised for the reliability data: • NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," Idaho National Laboratory, February 2007. • Institute of electrical and electronic engineers (IEEE) Std. 500 "Guide to the Collection And Presentation of Electrical, Electronic, Sensing Component, And Mechanical Equipment Reliability Data For Nuclear power Generating Stations," Appendix D, 1984.

	Following sources are utilised for the CCF data: • NUREG/CR-4780, “Procedures for Treating Common Cause Failures in Safety and Reliability Studies,” U.S. Nuclear Regulatory Commission, Washington, DC, January 1988. • NUREG/CR-5485, “Guidelines on Modelling Common Cause Failures in Probabilistic Risk Assessment,” U.S. Nuclear Regulatory Commission, Washington, DC, November 1998. • NUREG/CR-5497, “Common Cause Failure Parameter Estimations,” U.S. Nuclear Regulatory Commission, Washington, DC, November 1998. It is assumed that the reliability data of similar components can be applicable. For example, gas turbine generators (GTGs) are adopted for the emergency power source. The reliability data of emergency diesel generators reported in NUREG/CR-6928 are applied to that of GTGs since no reliability data are available for GTGs employed for nuclear power plants. In addition, sensitivity analyses for the reliability data were performed to clarify the impact on risk because the data involves several uncertainties. if data were not available for certain components, then what methods were used to address component reliability? The answer is same as above question.
NRC	Reliability data are based on United States operating experience of similar reactor designs and supplemented with appropriate generic data often taken from the Advanced Light Water Reactor Utility Requirements Document developed by the Electric Power Research Institute (EPRI). For evolutionary components and components without available data, generic data for similar components are considered with supporting reliability evaluations and expert judgment.
NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	Relevant and up-to-date data from existing NPPs with same or similar components is indispensable. sources of data for evolutionary components / components with limited operating experience? No information is available yet. Applicable data sources are an open issue. if data were not available for certain components, then what methods were used to address component reliability? No information is available yet.
ONR	UK AP1000 PSA: The sources of data used for components and CCF are generic and in most of the cases are dated. For GDA stage, generic data have been also used in general for components for which there is a limited operating experience (e.g large SQUIB valves). UK EPR PSA: Component data is based mainly on EDF operating experience supported by generic data bases. CCF is based on the Beta factor approach in EUR, but the Betas translated into Multiple Greek Letter parameters for use in Risk spectrum
SNSA	There are no detail requirements regarding reliability and CCF data. For the components without data, generic numbers would be used accompanied by sensitivity analysis to verify that there are no cliff-edge effects.
STUK	sources of data for the components of the same type as in the existing NPP PSAs? See AREVA D4-OL3 sources of data for evolutionary components / components with limited operating experience? See C7.

	if data were not available for certain components, then what methods were used to address component reliability? See C7.
UNISTAR	sources of data for the components of the same type as in the existing NPP PSAs? Yes sources of data for evolutionary components / components with limited operating experience? No if data were not available for certain components, then what methods were used to address component reliability? Generic data: EG&G, ZEDB and EPRI LWR Utility Requirement Document)
	D 4. How are the new / evolutionary design features treated? **• what is the contribution of the new / evolutionary design features to PSA results** **• potential new initiating events** **• inadvertent actuation of the new automatic actions**
AREVA OL3	The most important evolutionary design features are the dedicated primary depressurization (bleed & feed) and the core melt mitigation system. Its risk reduction capability is analyzed in the Level 2 PSA.
AREVA TSN	Specific functional (FMEA) and reliability analyses (FTs) are developped for new systems, however as EPR™ is an evolutionary PWR, set of IEs identified from the generic lists and the French and German PSAs covers the most frequent initiating events.
AREVA US	New redundancies, systems and functions are analyzed using the existing PRA methods
BARC	• The contribution of new evolutionary design i.e. Pressuriser features to PSA results is insignificant because of the low frequency of the related initiating event (IE frequency is less than 1E-4) • The new initiating event is Pressurizer heaters get fully on along with pressuriser bleed valves get stuck closed • Not applicable
Bel V	-
EDF FA3	Specific functional (FMEA) and reliability analyses (FTs) are developed for news systems. Spurious automatic actions are analyzed.
EDF EPR UK	Specific functional (FMEA) and reliability analyses (FTs) are developed for news systems.
ENEL	In this preliminary phase the idea is to include new initiating events and we would like to extend the study covering more possible situations.
ENSI	PSA is not yet available. Question can not be answered now.
IRSN	In general the new /evolutionary design features contribute to the decrease of the core damage frequency. Some new initiating events were identified, mainly related to inadvertent actuation of the new automatic actions.

JNES	Among new design features of ABWR, digital I & C is one of the most difficult features to evaluate reliability in system analysis. Common cause failures among instrumental or actuate logic channels are considered as well as software CCF.
MHI	The improved design features are thoroughly evaluated in the PSA. For example, initiating event frequency of total loss of component cooling water is quantified using the fault tree analysis reflecting the plant design feature (e.g., two independent subsystems which contains two trains). In-containment refuelling water storage pit (RWSP) contributes to eliminate the interfacing system LOCA (ISLOCA) from the initiating event. potential new initiating events: Failure mode and effects analyses (FMEA) are performed to identify the potential initiating events. inadvertent actuation of the new automatic actions: Spurious actuation of automatic systems is appropriately addressed in the model.
NRC	NRC may request additional calculations and studies be performed in order to assure that new and evolutionary design features are adequately addressed. These may include, for example, studies demonstrating the appropriate use of conservatism in developing PSA success criteria, the use of bounding parameters for PSA supporting calculations and sensitivity studies, and testing programs to validate calculations.
NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	No information is available yet for any of the above features.
ONR	UK AP1000 PSA: In-Vessel Retention (IVR) mitigates the consequences of some of the low pressure sequences and therefore decrease the large release frequency. Reactor Pressure Vessel (RPV) depressurization and new passive features as Passive Core Cooling System or Passive Containment Cooling System have a major role to reduce the risk associated with Internal Events and Hazards. There are potential new initiating events associated with Reactor Pressure Vessel (RPV) depressurization or some features of the Passive Core Cooling System due to spurious actuation of some automatic actions or other failure modes. UK EPR PSA: The Corium stabilisation system ("core catcher") mitigates the consequences of some of the low pressure sequences. Also the EPR includes an extension of the primary system depressurization to cover severe accident depressurisation. Improvements on the redundancy and (or) diversity of systems as feedwater systems, cooling systems, blackout diesel generators, etc. would contribute to reduce the risks. No potential new initiating events have been identified at this stage. However, the FMEAs supporting IE derivation may need to be completed when further design detail will be available and potential initiating events due to actuation of the new automatic actions and others should then be investigated.
SNSA	There are no detail requirements regarding new/evolutionary design features. Surely the consideration of new design features would be looked at in detail. Support analysis would be needed to assess the uncertainties and to verify no-cliff-edge effects.
STUK	See the Answer from AREVA-D2-OL3.
UNISTAR	what is the contribution of the new / evolutionary design features to PSA results: CCF on redundant trains (electrical, ECCS, Aux. Feed Water, cooling chain) and unavailability due to preventive maintenance

	potential new initiating events No inadvertent actuation of the new automatic actions No
	D 5. Which is the level of detail of the considered TechSpecs and preventive maintenance procedures?
AREVA OL3	TechSpecs are train-specific for safety systems. Preventive maintenance procedures follow the recommendations of the manufacturer of the concerned component.
AREVA TSN	TechSpecs and Preventive maintenance procedures are not available for the PSA for construction and operating licenses. Thus assumptions regarding unavailability and maintenance errors are taken into account in the PSA. Assumptions are based on the reference plant TechSpecs, if available.
AREVA US	Because Tech Specs and preventive maintenance procedures are not available in the design certification phase, assumptions were made on preventive maintenance and on corrective maintenance durations. These assumptions were based on the anticipated Tech Specs. and the industry experience.
BARC	The system failure criteria, test and maintenance schedules and practices are considered in development of PSA.
Bel V	-
EDF FA3	Preliminary preventative maintenance procedures have been modeled with point values in Fault Trees, for system unavailability.
EDF EPR UK	TechSpecs and preventative maintenance procedures are not available at this stage. Thus assumptions regarding unavailability and maintenance errors are considered in the PSA.
ENEL	In this phase we are using the international directives on preventive maintenance, in the future we will develop something more detailed and plant specific following vendor’s technical specification.
ENSI	TechSpecs and preventive maintenance procedures are not yet available; question can not be answered now.
IRSN	The TechSpecs and preventive maintenance information provided by EDF on EPR is preliminary. For TechSpecs information, IRSN considers that it is acceptable to use preliminary information. Nevertheless, as the preventive maintenance is foreseen during power operation, IRSN considers that detailed maintenance information (mainly related to the configuration management) should be available.
JNES	Tech. Specs. and preventive maintenance procedures of preceding ABWR plants are referred.
MHI	The US-APWR specific technical specification is established based on the standard technical specification, and the PSA model is developed in accordance with the US-APWR Tech Spec.
NRC	NRC staff evaluates the design stage Tech Specs to confirm that they will preserve the validity of the plant design by ensuring the plant will be operated with the required design conditions, and with operable equipment that is essential to prevent accidents and to mitigate consequences of accidents. In some instances, detailed design information, equipment selection, allowable values, or other information are needed to establish the information to be included in the Tech Specs. These plant-specific values must be

	provided when a combined license application is submitted for a specific plant.
NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	No information is available yet. If technical specifications are risk-informed, then they should be considered in an appropriate level of detail.
ONR	TechSpecs and preventive maintenance procedures are not yet available; assumptions regarding unavailability and maintenance errors are considered in the PSA. ONR's (HSE) expects that the PSA would be used to provide input to the development of TechSpecs, preventive maintenance procedures, etc. in due course.
SNSA	N/A
STUK	See the Answer from AREVA-D5-OL3.
UNISTAR	System unavailability for maintenance introduced as point values in Fault Trees
	D6. How is the HRA performed? **• method used to model and quantify the post-accidental human errors? (screening, detailed, generation 1/generation 2, etc.)** **• the detailed accident procedures should be available?** **• is (should be) a simulator available? (is it used somehow in the frame of HRA?)** **• method used to model and quantify the pre-accidental human errors?**
AREVA OL3	A HRA methodology report was written for the analysis and quantification of human preaccident and post-accident. The methodology is mainly based on THERP.
AREVA TSN	The methodology is mainly based on ASEP for pre-accident and post-accident HRA. Actions are evaluated based on the relevant EOP, when available. A HRA chapter is included in the PSA report for quantification of human pre-accident and postaccident.
AREVA US	The HRA was performed using EPRI HRA Calculator®: ASEP method for pre-accident human errors and SPAR-H method for post-accident operator errors.
BARC	Generation 1 method was used to model and quantify the pre-accidental and post-accidental human errors using emergency operating procedures. Simulator is available, however the same is not used for HRA.
Bel V	-
EDF FA3	The methodology is based on ASEP for pre and post-accidental HRA. Some global accident procedures are available. Human actions assessment will be updated in the future when lessons from simulator observations are available. The use of more advanced method such as MERMOS is planned.
EDF EPR UK	The methodology is mainly based on ASEP for pre-accident and post-accident HRA.

ENEL	In this preliminary phase we think that we will use generation 2 methods in collaboration with our partner in the PSA development. We would like to prepare detailed accident procedures. We expect that a simulator will be available and requested by the regulatory body. The method to model and quantify the pre-accidental human errors in this phase is the traditional based on the international guidelines and on the first generation methods.
ENSI	method used to model and quantify the post-accidental human errors? (screening, detailed, generation 1/generation 2, etc.) : Not yet decided. Guideline ENSI-A05 describes the HRA methods accepted by ENSI. the detailed accident procedures should be available? Not yet decided. is (should be) a simulator available? (is it used somehow in the frame of HRA?) No, not necessary for the construction license. method used to model and quantify the pre-accidental human errors? Guideline ENSI-A05 describes the HRA methods accepted by ENSI.
IRSN	The IRSN Level 1 PSA considers generally the same HRA as EDF EPR design PSA (adapted Swain screening model) The detailed accident procedures were not available for the design PSA (IRSN considered that this was acceptable for a design Level 1 PSA) The EPR simulator was not yet available during the design phase. For IRSN, such a simulator should be used in the future to support PSA assumptions. The Preaccidental human errors are quantified by IRSN by using the same method as for the operating reactors (as in the EDF PSA): 3 10-2 x non-recovery factor, function of individual component specific requirements: periodical test, monitoring, administrative lookout).
JNES	THERP method is used to evaluate pre- and post- accidental human error probabilities. The detail accident procedures were not available at the time of evaluating PSA, but the ones of preceding ABWR plant were referred. Plant simulator is not available.
MHI	THERP (Technique for Human Error Rate Prediction) and ASEP (Accident Sequence Evaluation Program) described in NUREG/CR-1278 and NUREG/CR-4772, respectively, are used to quantify the post accidental human error. In addition, the dependency between human actions is considered for the consecutive actions, using the methodology described in NUREG/CR-6883. Emergency operating procedures and severe accident management guidelines are currently under development, in which the operator actions modelled in the PSA are fully addressed. The simulator for the design basis accidents is currently under development by human factor engineering section. HFE features are extensively addressed in the simulator development and the feedback from the HFE factor is considered in the frame of HRA. ASEP based on NUREG/CR-4778 is used to quantify the pre accidental human error.
NRC	The methods used to assess post-accident human errors may vary for each applicant. The methods used are consistent with existing PSA standards and guidance. During the design phase detailed accident procedures and the use of a simulator to support HRA quantification typically are not available. Pre-accident human errors are considered in the PSA using established methods, such as NUREG/CR-4772, ASEP.
NRI	PSA for new units in the Czech Republic has not been developed.

NUBIKI	Generation 2 methods are preferred. Mismatch between context and cognition is to be addressed. Rough analysis based on poorly developed can be misleading. Thus availability of detailed procedures is preferred. Simulator is a very useful aid to HRA but is not necessary initially. method used to model and quantify the pre-accidental human errors? No information is available yet. The general tendency is still the use of THERP in most PSAs. It is envisaged that reference to operating experience from applicable existing reactors will be made in addition to the use of a sepcific HRA method for identifying and quantifying pre-initiator human errors.
ONR	UK AP1000 PSA: • The Human Error Probabilities (HEP) used in the AP1000 PSA HRA have been extracted from the Technique for Human Error Rate Predication (THERP) method in NUREG/CR-1278. • Although pre-accident HFEs have not been included in the AP1000 PSA, an identification of this type of human error and a human error probability analysis is presented in the Human Factors submission to GDA. WEC has selected HEART to support pre-accident human errors probability evaluation for the Human Factors submission. • HRA is assumption based since task analysis is not fully complete, written procedures and detailed information on operating practices were still in development during GDA. UK EPR PSA: • The Human Reliability Analysis has been carried using the Accident Sequence Evaluation Programme (ASEP) method. The method includes pre-accident tasks and post-accident tasks. • Pre-initiating fault HFEs include errors in positioning actuators (valves, circuit breaker racked-out). Failure to perform a critical step in a calibration procedure (calibration of C&I or of an actuator, pressure setting of a relief valve) is not assessed as pre-accident human error in the PSA. The calibration errors are assessed in the failure rate of the instrumentation part. • HRA is largely assumption based since there is a lack of task analysis, written procedures and detailed information on operating practices available for GDA. At GDA Step4/PCSR ONR’s (HSE) expectations are: • extensive identification, quantification & inclusion of potentially significant pre-fault errors (Type A) in the PSA i.e. errors impacting on significant safety systems/functions • identification & quantification of human errors leading to initiating events (Type B) • identification & quantification of post-fault claimed safety actions On HRA methods for post and pre-fault errors/actions, ONR is not prescriptive but expects the HRA quantification to be based on a sufficiently detailed qualitative task analysis of the task & design. TAG on HRA (Ref.8) gives detailed expectations. On detailed accident procedures, ONR does not expect these to be necessarily available but the 'best' procedure basis possible should be used as part of the qualitative task analysis. ONR expects that the HRA qualitative and quantitative analyses are used to help develop the detailed accident procedures.

	Where an appropriate simulator is available ONR would expect it to be used to help support the qualitative analysis. Additionally, ONR expects that qualitative arguments are made about post-fault mis-diagnosis i.e. that it has been reduced to ALARP. Where appropriate, ONR would expect to see explicit assessment of mis-diagnosis for key post-fault decisions via suitable methods (e.g. confusion matrices).
SNSA	There are no detail requirements regarding development of HRA. For our existing plant THERP method is used to assess the HEPs which is also supported by simulator gained data. Detailed accidents procedures as well as a simulator shall be available.
STUK	See the Answer from AREVA-D6-OL3.
UNISTAR	method used to model and quantify the post-accidental human errors? (screening, detailed, generation 1/generation 2, etc.) screening: SPAR-H (NUREG/CR-6883) + EPRI HRA calculator. the detailed accident procedures should be available? NO is (should be) a simulator available? (is it used somehow in the frame of HRA?) NO method used to model and quantify the pre-accidental human errors? ASEP- Swain
	E. PSA Level 1 hazards technical aspects
	E 1. How the external hazards are treated? • **what method is used for seismic events? (seismic PSA, seismic margins, PSA based seismic margins, etc.)** • **which are the extreme weather hazards considered? (extreme cold, heat wave, wind, storms, snow storms, lightning, sand and dust storms etc.)? How the modeling is done?** • **which are the other external hazards considered (loss of ultimate heat sink, external flooding, tsunami, industrial, transportation accidents etc.)? How the modeling is done?** • **are the possible future evolutions of some hazards (frequency, intensity) taken into account in the PSA?**
AREVA OL3	A detailed external events screening analysis for all kind of potential external events was performed. The screening analysis is based on the SKI report “Guidance for External Events Analysis”. Single and Multiple External Events are considered. External events which were not screened out were analyzed as initiating events in the PSA model.
AREVA TSN	A detailed external events screening analysis for all kind of potential external events was performed. The screening analysis is based on the SKI report “Guidance for External Events Analysis”. Single and Multiple External Events are considered. External events which were not screened out were analyzed as initiating events in the PSA model. Seismic risk will be analyzed with a simplified approach.
AREVA US	A detailed external events screening analysis for all potential external events were performed, based on NUREG 0800.

BARC	External events are not in the scope of this PSA.
Bel V	-
EDF FA3	A simplified PSA based seismic margins assessment has been used. It will be updated in the future with a more detailed seismic PSA. The extreme weather hazards have been screened out by simplified assessment or justification. Loss of ultimate heat sink is combined with loss of external power, for a specific long term PSA analysis.
EDF EPR UK	To be included in the PSA scope, an external hazard must be able to impact on plant structures, systems or components and degrade one or more plant safety functions, challenging plant safety systems that act to maintain or bring the plant to a safe state. The process adopted for the probabilistic analysis of external hazards involves the following steps: • a screening analysis of the initial external hazard list (which is as exhaustive as possible). • a probabilistic analysis of the 'screened in' external hazards. The seismic margin of the UK EPR is assessed by a PSA-based SMA, following a methodology developed by the US NRC
ENEL	We would like to use the seismic PSA to evaluate the seismic events. Seismic hazard is the most important event that could affect the plant then other hazards are not considered in this preliminary phase. In the definitive PSA perhaps we will include more external hazards.
ENSI	• what method is used for seismic events? (seismic PSA, seismic margins, PSA based seismic margins, etc.) The Swiss licensees carried out a large-scale project "PEGASOS" – a German acronym for "Probabilistic Assessment of Seismic Hazards for Swiss Nuclear Power Plant Sites" – in response to a requirement that came out of the Inspectorate's PSA review process. In order to achieve a thorough quantification of the uncertainty of seismic-hazard estimates, the licensees conducted an extensive elicitation process involving technical experts, scientific institutions and engineering organisations from Europe and the USA. The project was conducted in full compliance with the Senior Seismic Hazard Analysis Committee (SSHAC) Level 4 methodology. In 2008, Swiss licensees initiated a follow-up project, the "PEGASOS Refinement Project" (PRP). The project takes advantage of the most recent findings in earth sciences and new geological and geophysical investigations at Swiss NPP sites. A particular objective is to reduce the uncertainty range of the PEGASOS results. In 2009 the scope of the PRP was extended to include the sites for the proposed new Swiss NPPs. The Inspectorate is following the study closely through a system of continuous peer reviews similar to that for PEGASOS. In the PRP full compliance with the SSHAC Level 4 methodology is maintained. The design of the new NPPs and their seismic PSA must be based on the results of this project. • which are the extreme weather hazards considered? (extreme cold, heat wave, wind, storms, snow storms, lightning, sand and dust storms etc.)? How the modeling is done? Guideline ENSI-A05 defines which external events have to be modelled in the PSA. This includes high winds and tornadoes. In addition, ENSI-A05 includes a list of hazards, which shall be considered (drought, high summer temperatures, ice cover lightning, low winter temperatures and snow (drift) …) • which are the other external hazards considered (loss of ultimate heat sink, external flooding, tsunami, industrial, transportation accidents etc.)? How the modeling is done? Guideline ENSI-A05 defines which external events have to be modelled in the PSA. This includes aircraft crash and external flooding. Additionally, ENSI-A05 includes a list

	of hazards, which have to be analysed. Methods and criteria for screening are also defined. For external flooding, in the applications for the general license (early site permissions), a two dimensional model was used to quantify extreme river flow rates and the corresponding water levels. • are the possible future evolutions of some hazards (frequency, intensity) taken into account in the PSA? Not yet decided. In the general license applications, effects of climate change were considered in the assessment of external hazards.
IRSN	IRSN do not develop yet PSA for internal/external hazards for EPR reactor. • what method is used for seismic events? (seismic PSA, seismic margins, PSA based seismic margins, etc.) For IRSN any method can be used, if the application is capable to indicate, with a high degree of confidence, that the seismic contribution to core damage frequency is acceptable (low). • which are the extreme weather hazards considered? (extreme cold, heat wave, wind, storms, snow storms, lightning, sand and dust storms etc.)? How the modeling is done? IRSN considers that the WENRA statements are fully applicable for new reactors ("Additionally, external hazards such as severe weather conditions and seismic events shall be addressed in the PSA so that the overall risk of a plant is assessed realistically - This means that these two hazards shall be included in the PSA, except if a justification is provided for not including them, based on site-specific arguments on these hazards or on sufficient conservative coverage through deterministic analyses in the design, so that their omission from the PSA does not weaken the overall risk assessment of the plant.") • which are the other external hazards considered (loss of ultimate heat sink, external flooding, tsunami, industrial, transportation accidents etc.)? How the modeling is done? A limited study for a long term common mode failure of the external grid and ultimate heat sink was done by EDF. The study is under review by IRSN. • are the possible future evolutions of some hazards (frequency, intensity) taken into account in the PSA? Not in the actual studies. However, IRSN thinks that the possible future evolutions of some hazards should be addressed in the context of EE PSA
JNES	No external events are considered in the introducing the accident management countermeasures.
MHI	• what method is used for seismic events? (seismic PSA, seismic margins, PSA based seismic margins, etc.) PRA based seismic margin method is used. • which are the extreme weather hazards considered? (extreme cold, heat wave, wind, storms, snow storms, lightning, sand and dust storms etc.)? How the modeling is done? Extreme weather hazards are site specific. The bounding design parameter requirements for meteorology are considered in the design certification application. The site specific meteorology hazards will be assessed in accordance with the requirements for screening and conservative analysis of other external hazards of ASME/ANS RA-Sa-2009. • which are the other external hazards considered (loss of ultimate heat sink, external flooding, tsunami, industrial, transportation accidents etc.)? How the modeling is done? The other external hazards are site specific. The bounding design parameter requirements for other external hazards are considered in the design certification application. The

	nearby industrial, transportation and military facilities, hydrologic engineering, and geology, seismology and geotechnical engineering will be assessed in accordance with the requirements for screening and conservative analysis of other external hazards of ASME/ANS RA-Sa-2009. • are the possible future evolutions of some hazards (frequency, intensity) taken into account in the PSA? No further activities have been intended so far.
NRC	• what method is used for seismic events? (seismic PSA, seismic margins, PSA based seismic margins, etc.) Seismic hazards are typically addressed by a seismic margins assessment during the design phase. A full seismic PSA is required prior to initial fuel load if a standard for seismic PRA has been endorsed by the NRC one year prior to the date of initial fuel load. • which are the extreme weather hazards considered? (extreme cold, heat wave, wind, storms, snow storms, lightning, sand and dust storms etc.)? How the modeling is done? High wind hazards are considered. Hazard frequencies are typically based on historical data for a representative site region. Frequently, PSAs that support design certification will treat the hazard in a bounding fashion so that it may apply to all applicants for a COL referencing that design certification. • which are the other external hazards considered (loss of ultimate heat sink, external flooding, tsunami, industrial, transportation accidents etc.)? How the modeling is done? Other hazards that may be considered include external flooding, transportation accidents, and nearby facility accident. Additional hazards may be considered if warranted by site specific considerations. • are the possible future evolutions of some hazards (frequency, intensity) taken into account in the PSA? Future evolutions are not typically modeled directly. External hazard analyses typically involve bounding assessments that are meant to demonstrate the design margin for these hazards.
NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	Similarly to question group D, no details are available yet but some expectations can be addressed based on existing PSA related requirements in Hungary. • what method is used for seismic events? (seismic PSA, seismic margins, PSA based seismic margins, etc.) Existing regulatory requirements call for a seismic PSA. Indications are that short-term future modifications of the Nuclear Safety Codes will allow seismic margin assessment instead of a detailed seismic PSA. • which are the extreme weather hazards considered? (extreme cold, heat wave, wind, storms, snow storms, lightning, sand and dust storms etc.)? How the modeling is done? Safety regulation requires a PSA for all natural and man-made hazards that are relevant for the site. So the actual hazards to be considered must be selected on a plant specific basis. Bidding requirements will include a listing of hazards to be considered in PSA Modeling details are not available yet.. • which are the other external hazards considered (loss of ultimate heat sink, external flooding, tsunami, industrial, transportation accidents etc.)? How the modeling is done?

	See the answer to the previous question. • are the possible future evolutions of some hazards (frequency, intensity) taken into account in the PSA? This is a regulatory requirement. So all plant PSAs must meet this.
ONR	UK AP1000 PSA: • A Seismic Margins Analysis (SMA) has been submitted in GDA to address seismic risk. • A set of hazards listed up front was looked at, including, high winds, tornadoes, external floods and transportation and nearby facility accidents and have been screened out from the PSA submitted for GDA. • Site specific work on this area will be needed. UK EPR PSA: • A Seismic Margins Analysis (SMA) has been submitted in GDA to address seismic risk. • For the GDA PSA, the majority of external events are screened out on deterministic or on probabilistic grounds. Frazil ice, solid or fluid impurities released into the water from a ship (e.g., oil spill) and the effect of organic material on the water intake water are included in the loss of ultimate heat sink (LUHS) initiating event and included in the Riskspectrum PSA model. • Site specific work on this area will be needed.
SNSA	Again, there are no detail requirements regarding external hazards. For existing plant: For seismic events a detailed PSHA is prepared. The results are incorporated in the internal events PSA model. Extreme weather hazards are included in the other external events (together with external flooding, external fires, aircraft crash hazard, etc). For the extreme weather most of the hazards are included in other initiating event frequencies (e.g. lightning is included in loss of off-site power and external fire, intense precipitation included in the external flooding, high temperatures, low river water included into drought, etc). For extreme winds (which also include hurricanes and tornadoes) a bounding study was performed. External events considered are: aircraft impact, avalanche, biological events, coastal erosion, drought, external flooding, extreme winds, tornadoes and hurricanes, external fires. Fog, frost, hail, hazardous materials, high tide, high river stage, high summer temperatures, ice cover, industrial and military facility accidents, internal fires, landslide, lightning, low river water level, low winter temperatures, meteorite, pipeline accident, intense precipitation, release of chemical in on-site storage, river diversion, sandstorm, seiche, snow, seismic activity, storm surge, transportation accidents, tsunami, toxic gas, turbine-generated missile, volcanic activity, waves. None of the external events, except seismic activity, internal fires and floods, are modeled in the PSA model, but are added to the final PSA results based on performed PSA analysis.
STUK	See the Answer from AREVA-E-OL3. The Finnish Regulatory Guide YVL 2.8 requires Seismic PRA to be performed.
UNISTAR	The hazards considered were extensive and came from ANSI/ANS-58.21-2007, “External-Events PRA Methodology.”

	• what method is used for seismic events? (seismic PSA, seismic margins, PSA based seismic margins, etc.) PRA-based seismic margins assessment • which are the extreme weather hazards considered? (extreme cold, heat wave, wind, storms, snow storms, lightning, sand and dust storms etc.)? How the modeling is done? Analysis of High winds and tornadoes for screening them out • which are the other external hazards considered (loss of ultimate heat sink, external flooding, tsunami, industrial, transportation accidents etc.)? How the modeling is done? Analysis of Airplane crash and industrial transportation for screening it out • are the possible future evolutions of some hazards (frequency, intensity) taken into account in the PSA? No
	E 2. How site specific information are treated in the design stage (e.g., bounding assumptions, representative site selected, site was known, etc.)?
AREVA OL3	Site conditions, e.g. air temperatures, water temperatures, sea levels, were defined by the owner and respected as design values for buildings and systems.
AREVA TSN	Site conditions, e.g. air temperatures, water temperatures, sea levels, were defined by the owner and respected as design values for buildings and systems.
AREVA US	In the design certification phase site specific information was treated through the selected bounding assumptions
BARC	Not applicable
Bel V	-
EDF FA3	As the facility is under construction, site specific information was available.
EDF EPR UK	As the PSA is performed for a generic design (no site available), the site data is treated mainly through bounding assumptions
ENEL	The site will influence the design of the plant but it is not detailed yet.
ENSI	Site specific information is the imperative background for the general license. The general license is valid for a specific site.
IRSN	The specific site information is treated by EDF mainly by bounding assumptions. IRSN considers that the site specific aspects should be further investigated.
JNES	N/A
MHI	See responses to Question E1.
NRC	In the design stage and prior to site selection, representative site parameters are selected. These typically include bounding assumptions.
NRI	PSA for new units in the Czech Republic has not been developed.

NUBIKI	No information is available yet. PSA in the current project will have to be specific to the Paks site.
ONR	Generic assumptions expected to be bounding. Reconsideration of these assumptions will be needed when site specific work will be carried out in due course.
SNSA	The site for the new unit will be known as well as all relevant site information that are needed for the design.
STUK	See the Answer from AREVA-E2-OL3. The experiences from operating plants were used.
UNISTAR	When feasible, which was often, bounding assumptions were used. For example, this is case for seismic spectra, min/max ambient temperature, max speed for high winds, etc. When not feasible, site-specific analyses were done for the COLA, for example, seismic soil effects, airplane crash, tornados, nearby facilities, etc.
	E 3. How the internal hazards are treated? **• what method is used for fire PSA? (fire frequencies data source, which fire propagation code was used, considering of the internal explosions)** **• what method is used for internal flooding PSA?** **• which are the other internal hazards considered ? (heavy loads drops, etc)**
AREVA OL3	Internal hazards like fire, internal flooding were analyzed as initiating events.
AREVA TSN	Internal hazards like fire, internal flooding were analyzed as initiating events. Analysis is based on NUREG/CR-6850 and ANSI/ANS-58.23–2007.
AREVA US	Internal hazards, internal flooding and fires, were explicitly analyzed as initiating events
BARC	Internal hazards are not in the scope of this PSA
Bel V	-
EDF FA3	For the fire PSA, based on analysis of French fire feedback and of EPR components installation, the initiating frequencies are assessed. After functional analysis of consequences in a fire volume, the core damage frequencies are assessed by fire volume (methodology from EPRI 1011989 NUREG/CR-6850). The internal flooding PSA is quite similar to the fire PSA with floor propagation of flooding instead of total loss by fire volume. For the internal explosion PSA, specific rooms are selected based on explosion risk criteria. With mitigating functions failures, the explosive atmosphere frequencies are assessed. Heavy load drops are also considered.
EDF EPR UK	The main components of the fire PSA are: • identification of fire events relevant to safety

	• estimation of the initiation frequencies for these fire events and • calculation of the resulting core damage frequencies. Guidance is provided for this analysis by NUREG/CR-6850 The main steps in the simplified flooding PSA are: • the evaluation of the scope of flooding events relevant to safety, • the estimation of the initiation frequencies for these flooding events and • the calculation of the resulting core damage frequencies.
ENEL	Fire and internal flooding will be probably studied with specific PSA but it is not decided the method yet.
ENSI	• what method is used for fire PSA? (fire frequencies data source, which fire propagation code was used, considering of the internal explosions) PSA is not yet available. Question can not be answered now. Accepted methodologies for the fire PSA are described in guideline ENSI-A05. • what method is used for internal flooding PSA? PSA is not yet available. Accepted methodologies for the flooding PSA are described in guideline ENSI-A05. • which are the other internal hazards considered ? (heavy loads drops, etc) Guideline ENSI-A05 defines the initiators which have to be considered in the plant specific PSA (turbine missiles, explosions, toxic gases). The decision-making process to what extent these requirements shall already be met for the PSA for the construction license is still ongoing.
IRSN	For FA3, EDF developed a Fire PSA. The study is not yet available at IRSN. For FA3, EDF developed an internal flooding PSA. The study is not yet available at IRSN. which are the other internal hazards considered ? (heavy loads drops, etc)? For FA3, EDF developed a reliability study for reactor building polar crane. The study is under review by IRSN.
JNES	No internal hazards are considered in the introducing the accident management countermeasures.
MHI	• what method is used for fire PSA? (fire frequencies data source, which fire propagation code was used, considering of the internal explosions) The methodology of NUREG/CR-6850, "EPRI/NRC RES Fire PRA Methodology for Nuclear Power Facilities" for fire PRA was applied. CFAST code is used for fire propagation analysis. • what method is used for internal flooding PSA? The internal flooding PRA was performed in accordance with the requirements for the internal flood PRA of ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications." The used data source is EPRI 1013141, Rev.1 "Pipe Rupture Frequencies for Internal Flooding PRAs."

	• which are the other internal hazards considered ? (heavy loads drops, etc) The internal hazards beyond design basis were screened out.
NRC	• what method is used for fire PSA? (fire frequencies data source, which fire propagation code was used, considering of the internal explosions) Internal fire hazards for new designs are typically assessed using the Fire Induced Vulnerability Evaluation (FIVE) methodology developed by EPRI. The NRC report NUREG/CR-6850 describes the fire PSA methodology developed jointly by EPRI and NRC, which includes improvements over previous methods such as FIVE. During the design stage, the NRC staff may find that the FIVE methodology is acceptable to satisfy the applicable regulatory requirements. Because the as-built configuration cannot be assessed until construction is complete, the staff may find it acceptable to update the internal fire analysis if the need to do so is identified when walkdowns are performed after the plant is built. • what method is used for internal flooding PSA? The internal flooding analysis typically includes a systematic approach to identify potential internal flooding sources and their impacts on the plant. Because the as-built configuration cannot be assessed until construction is complete, NRC staff may find it necessary to update the internal flood analysis if the need to do so is identified when walkdowns are performed after the plant is built. • which are the other internal hazards considered ? (heavy loads drops, etc) Typically, other internal hazards are not modeled, but they may be considered if warranted by unique design considerations.
NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	No details on modeling are available yet. As to the scope of internal hazards, all hazards must be subject to PSA that cannot be screened out on the basis of low occurence frequency and/or consequences on plant operation.
ONR	UK AP1000 PSA: • The methodology used for the AP1000 Fire PSA is EPRI's FIVE method and data with some enhancements. Generic US data provided in the FIVE methodology have been used. No detailed fire modeling has been performed (apart from the analysis of the Main Control Room). • The approach to Flooding Analysis in the AP1000 PSA followed a systematic process. • Apart from internal fire and internal flood no analysis or screening of additional potential internal hazards (e.g. turbine missile, dropped loads) had been presented in the GDA PSA documentation. UK EPR PSA: • Currently a coarse fire and flood analyses based on major buildings. • Evaluations are based on the assumption that flood or fire makes all of the equipment in a building unavailable and demands safety functions. NUREG-CR 6850 is used for the evaluation of fire frequencies. • Other internal hazards considered but screened out or excluded from GDA are high energy components breaks, internal missiles, internal explosions and dropped loads, though these will be revisited.
SNSA	There are no detailed requirements regarding internal hazards. For the new NPP the state of the art method shall be used (the newest PSA standards) with site specific data. All

	relevant hazards shall be taken into account, including heavy load drops if applicable. For existing plant the methods used are in compliance with US NRC Generic Letter 88-20, Supplement 4.
STUK	See the Answer from AREVA-E3-OL3. A lot of emphasis on FRNC cables based on the VTT research.
UNISTAR	• what method is used for fire PSA? (fire frequencies data source, which fire propagation code was used, considering of the internal explosions) USNRC NUREG data, no fire propagation code, bounding assumptions, no internal explosion • what method is used for internal flooding PSA? USNRC NUREG data, bounding assumptions • which are the other internal hazards considered ? (heavy loads drops, etc) No
	F. Severe Accident/Source Term/Level 2 PSA
	F1. What kind of study is available for the Severe Accidents / Source term assessment? **• please describe the interface between Level 1 PSA and Level 2 PSA and precise how the dependencies are considered (human errors, shared components, I&C, treatment of containment bypasses, etc.)** **• releases from other than the reactor core were considered? (ex: fuel pool)** **• are the internal and external hazards initiating events considered?**
AREVA OL3	A level 2 PSA was performed based on the PSA level 1 results. Source terms and their frequencies were determined.
AREVA TSN	Integrated PSA level 1 and Level 2 is developed for TSN EPR™. In the frame of the PSA Level 2 a complete review of the plantspecific phenomenon applicable to the TSN EPR™ was performed. Each phenomenon is supported by a dedicated engineering document and corresponding MAAP runs and Crystal Ball excel files. The source term calculation is based on dedicated MAAP runs.
AREVA US	A full scope Level 2 analysis is developed for the U.S.EPR™ with a linked Level 1/ Level 2 PRA model. Use is made of Core Damage End States to propagate boundary conditions from the Level 1 to the Level 2 to capture systems and plant state dependencies. A Level 2 HRA is developed based on SAPR-H and dependencies between the Level 1 and the Level 2 operator actions are treated explicitly. The probabilities derived from the phenomenological analyses are used as split fraction inputs to the containment event trees. The accident sequences are grouped into release categories that represent; intact containment, large and small source term releases. Releases from the spend fuel pool were not analyzed.

BARC	Severe accident/source term/Level-2 PSA has not been performed.
Bel V	-
EDF FA3	The L1/L2 PSA is integrated in the internal events model, with containment bypasses included. The accidents in the spent fuel pool are included.
EDF EPR UK	Integrated L1/L2 PSA is developed for the UK EPR (internal and external hazards modelled as initiating events in the level 1 PSA are thus treated in the level 2 PSA) In the frame of the L2 PSA a complete review of the plant-specific phenomenon applicable to the UK EPR was performed. Each phenomenon is supported by a dedicated engineering document and corresponding MAAP runs and Crystal Ball excel files. The source term calculation is based on dedicated MAAP runs.
ENEL	There are no studies available yet, the interface between level 1 and level 2 PSA probably will be done with the plant damage condition. The releases can be from fuel pool, reactor building or due to a bypass without core fusion. We think that internal and external hazards initiating events will be considered
ENSI	PSA is not yet available. Question can not be answered now. Good practice is to use an integrated project for Level 1 and Level 2 PSA. The risk of radioactive release involving the spent fuel pool for the NPP at full-power operation shall be evaluated. If it can be shown based on conservative assumptions that the risk of radioactive release involving the spent fool pool is negligible (contribution to the Total Risk of Activity Release, TRAR less than 1%), no further analysis is necessary. Otherwise, a PSA shall be performed for the spent fuel pool, which follows the same requirements as set forth for NPPs. Internal and external initiating events have to be considered.
IRSN	EDF has initially developed a Level 1+ PSA and then a Level 2 PSA, which will be reviewed by IRSN in 2011. Both are integrated L1-L2 models. IRSN develops an integrated L1/L2 PSA model for EPR with the objective to have a good description of systems and human dependencies (before and after start of core degradation). This is seen as a crucial point. Nevertheless it supposes that IRSN L2 PSA team develops specific methodology. In existing L2 PSA modeling at IRSN (900 and 1300 MWe PWR – Gen II), the L1 and L2 PSA are not integrated and the dependencies were modeled through the interface variables used to define the PDS. The high number of dependencies taken into account induces a very high number of PDS (150 to 200). Advantage is the readability of the modeling for L2 PSA, inconvenient is the difficulty to build the list of PDS (and minimal cut-sets) with the L1 PSA. The EDF EPR L2 PSA only consider reactor core. Fuel pool accidents are only addressed in L1 PSA (consequences are “unacceptable” from a L2 PSA point of view). The IRSN Level 2 PSA is under development only for power state (internal events, core).
JNES	For systems and components, dependencies between level 1 and 2 such as recovery of failed ECCS pumps, EDGs, power supply and so on, are considered by means of conditional failure probabilities. For containment bypass events, containment vessel failure frequency (CFF) is just same as core damage frequency (CDF) because of no mitigation of this event.
MHI	The failure end states of the Level 1 PRA event trees result in accident classes (ACLs) that are initial conditions of the containment event tree (CET). ACLs are classified as a combination of (1) initiating event and primary system pressure, (2) containment intact or failed at core damage, (3) accident progression in containment, and (4) loss of support system initiating events. In total 28 ACLs are defined for the US-APWR PRA.

	The CET is developed to model each ACL and track the potential influence of accident progression in the containment. The CET consists of two portions, the containment system event tree (CSET) and the containment phenomenological event tree (CPET). The interface between CSET and CPET is defined as the plant damage state (PDS), which forms the end states of the CSET and the initial conditions of the CPET. The CSET models the containment systems and functions that are provided to mitigate the consequences of an accident and to prevent containment failure. The CPET models the physical phenomena in containment that influence to containment failure and fission product release to the environment. The CSET quantification is performed by the computational code, RiskSpectrum®, employed for the Level 1 event tree quantification by the linking of the CSET with the Level 1 PRA event tree model. This is done because fault trees used in the CSET are the same trees already modelled in the Level 1 PRA. Additionally, the CSET has the same support systems and HRA dependencies considered in the Level 1 PRA fault trees. These dependencies between Level 1 PRA and CSET are simultaneously modelled and quantified by employing Riskspectrum® code. releases from other than the reactor core were considered? (ex: fuel pool)? Not in the Level 2 PSA. Release from the SFP is screened out for the Level 1 PSA. Internal fire and internal flood are considered.
NRC	Applicants develop the interface between Level 1 PSA and Level 2 PSA in accordance with existing PSA standards and guidance. The approach must account for the Level 1 model characteristics leading to core damage and ensure that dependencies between the Level 1 and Level 2 models are properly treated. The PSA is typically limited to releases from the reactor core. Releases from other than the reactor core, such as spent fuel pools, are addressed in other studies. The Level 2 PSA addresses internal and external events.
NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	No information is available yet. The requirement is to cover all potential sources of large releases including the fuel pool and any other sources that apply. The requirement is to include in the Level 2 PSA aIl the initiating events that are considered in the Level 1 analysis, i.e. the scope is the same for the two levels.
ONR	UK AP1000 PSA: • Each Level 1 sequence is mapped to the relevant PDS by using a set of attributes. The frequency of each PDS is calculated by adding the frequencies of all the Level 1 sequences. Each PDS is connected to a Containment Event Tree (CET). Westinghouse's methodology is implemented to transfer the information from the Level 1 sequences to the Level 2 CETs by explicitly linking the Level 1 and Level 2 sequence fault trees in Cafta. • Radiological releases and consequences from AP1000 accident sequences without core damage are not considered (fuel ponds, fuel handling facilities, waste storage tanks, etc). • Sequences from hazards are not linked to the Level 2 PSA model and neither are they added to overall results. • Sequences from the shutdown models are not linked to the Level 2 PSA. The shutdown assessment for Level 2 PSA is simplified being a scaling of the AP600 analysis to the AP1000 CDF. UK EPR PSA: • UK EPR PSA is an integrated level 1 and level 2 PSA analyses which allows for accurate transfer of dependency information. L2 PSA is a large model with about 70

	Containment event trees. L1 results are grouped into Core Damage End States (more discrimination than standard Plant damage states). • In addition to at-power conditions, the Level 2 study considers shutdown plant states, internal fire and flood, external hazards as treated in Level 1 and accidents occurring in the Spent Fuel Pool.
SNSA	There are no detailed requirements regarding Level 2 PSA, so it is up to licensee (and vendor) to develop the Level 2 PSA which will be in compliance with IAEA and US NRC guides and standards. Releases from other probable sources shall be considered as well as internal and external hazards of which contribution is mot negligible.
STUK	See the Answer from AREVA-F1-OL3
UNISTAR	Interconnected Level 1/ Level 2 models: CDESs to bin the core damage accident sequences and linked system fault trees; Limited analysis for shutdown states and for internal hazards. HRA: SPAR-H (NUREG/CR-6883) adapted for SAMG. Containment bypasses: IS LOCA, non-isolated SGTR, induced SGTR and containment hatch open in most POS with an operator action to close it. releases from other than the reactor core were considered? (ex: fuel pool) no are the internal and external hazards initiating events considered? Internal hazards
	F 2. What severe accident progression support studies are used? • **what computer codes are used? How detailed are the studies? Have experiments been performed? Thermal hydraulics and fission product releases?**
AREVA OL3	MAAP analyses were performed as support for the assessment of containment event tree branch probabilities and for the determination of source terms.
AREVA TSN	MAAP 4.0.7 was used to determine branch probabilities for the containment event trees and for the determination of source terms.
AREVA US	The Level 2 success criteria and phenomenology runs are performed with MAAP 4.0.7. For uncertainty analyses the results of published studies are considered (NRC, EPRI, IAEA, OECD)
BARC	Not applicable
Bel V	-
EDF FA3	Specific support studies are developed for L2 phenomena, with MAAP and Crystal Ball analysis for the event trees and the source term calculations.
EDF EPR UK	MAAP 4.0.7 and Crystal Ball excel files were used to determine branch probabilities for the containment event trees. The source term calculation is based on dedicated MAAP runs.
ENEL	It is not defined yet
ENSI	PSA not yet available, question can not be answered now. The current practice is to use at least one of the integral codes (MELCOR, MAAP,or ASTEC). The structural failure

	probability of the containment and penetrations is usually assessed by using finite element codes. In addition, insights from research are used for the quantification of the containment event tree.
IRSN	For IRSN EPR L2 PSA, ASTEC is used as severe accident integral code. Some other tools are used in addition (MC3D for DCH, TONUS for hydrogen distribution …). A very fast running code (MER) will be developed for the source term calculations. For Level 2 PSA, IRSN tries to develop all studies independently of the utility and uses effort on severe accident (research program, development of tools like ASTEC …).
JNES	MELCOR is used for level 1.5 PSA. The studies for a validation of analyzing code, MELCOR and CFD code, are being conducted based on the insights from international severe accident programs.
MHI	General accident progression analysis including source term evaluation: MAAP 4.0.6 Hydrogen burn: GOTHIC 7.2a-p5(QA) Steam explosion: TEXAS-V Containment structure: LS-DYNA ver. 970 Molten core spreading behaviour: FLOW-3D ver. 9.0 Several sensitivity studies are performed to find the bounding case for each physical phenomenon. Sufficient safety margin is therefore considered for all studies. No specific experiments have been performed.
NRC	Applicants typically use computer codes, such as MAAP, to support severe accident progression studies. NRC develops evaluation models for new plant designs using MELCOR and other computer codes. Both the applicant and NRC may perform experiments to support severe accident studies. An example of experiments sponsored by NRC for the AP1000 reactor design is described in the report NUREG-1826 (available in NRC's ADAMS at ML052980061.)
NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	No information is available yet. Thermal hydraulics and fission product releases should be calculated by an integral severe accident code.
ONR	UK AP1000 PSA: • For the Level 2 PSA WEC has used MAAP4 to support the severe accident progression analyses and a stand-alone model to support the analysis of In Vessel Retention (IVR). • The accident scenarios examined with MAAP4 calculations were based on the PDS analysis from the Level 1 PSA. • The PSA presents a set of Release Categories associated with containment failure at different nodes in the CET. A common Source Term for each Release Categories is calculated with MAAP4 based solely on the mode (time) of containment failure without any discrimination among accident sequence characteristics or severe accident phenomena. UK EPR PSA:

	• MAAP 4.0.7, with an EPR-specific parameter file and EPR models (e.g. core catcher), was used to model accident progression. In support of the Level 2 PSA, MAAP calculations were performed to represent the CDES sequence, sensitivity to the calculation, evaluate specific phenomena, and to confirm if a particular scenario results in core damage. • There are 29 unique release categories established with attributes defined based on the status of the following: o containment bypass; o timing of containment failure; o containment failure mode; o core melt arrested in-vessel; o core concrete attack; o core debris flooding ex-vessel; and o mitigation by containment sprays. The radionuclide releases are based on the standard set of 12 fission product groups. MAAP analysis is then described to calculate the source terms for the release categories with clearly stated post-processing rules to adjust for; 1) Iodine chemistry, 2) scrubbing in ventilation systems, 3) submergence of ISLOCA piping, and 4) SGTR scrubbing.
SNSA	Again since there are no legal requirements, the international standards shall be taken into account. All assumptions shall be supported by appropriate studies.
STUK	See the Answer from AREVA-F2_OL3
UNISTAR	MAAP4.07 for SA calculations and source term analysis
	F 3. What new severe accident reactor features are modelled in detail in the PSA? **• phenomena covered, reliability**
AREVA OL3	In the level 2 model the core melt stabilization system, the hydrogen recombiners, the severe accident containment heat removal system and the containment filtered venting system are modeled.
AREVA TSN	In the MAAP and RiskSpectrum® model all of the EPR™ systems which impact safety were modeled. This includes the core melt stabilization system, the hydrogen recombiners and the severe accident containment heat removal system. The annulus ventilation is partially represented in the probabilistic assessment and integrated into the source term calculation. Additionally as the level of risk is assessed with an integrated model, the safety system used in the level 1 are included in the RiskSpectrum® model which enable to carry out all dependency between level 1 and level 2.
AREVA US	The new evolutionary features of the U.S.EPR™ are modeled in the PSA model and MAAP and are credited for long term pressure control and source term mitigation. These features are: • Passive Autocatalytic Recombiners,

	• Core Melt Stabilization System • Severe Accident depressurization and heat removal
BARC	Not applicable
Bel V	-
EDF FA3	The RCS depressurization valves, the catalytic recombiners, the core catcher and the containment heat removal system are modeled.
EDF EPR UK	The most important evolutionary design features are the dedicated primary depressurization and the core melt stabilization system (including the Severe Accident spray system including all functions: i.e., passive debris flooding, containment spray and active recirculation cooling).
ENEL	It is not defined yet
ENSI	PSA is not yet available. Core catcher (or similar safety features) should be modelled.
IRSN	EPR severe accident features (primary heat transport depressurization, hydrogen recombination, core catcher) will be modeled in the L2 PSA. It will have some impact on ASTEC functionalities (especially the possibility to model the whole reactor with ASTEC (primary, secondary circuits plus safety system and core-catcher).
JNES	NO.
MHI	The rocket mode reactor vessel failure is newly added as part of the high pressure melt ejection in the model. The US-APWR eliminates the ICIS penetrations at lower plenum so that this new energetic phenomenon is additionally considered.
NRC	The new phenomena that are modeled depend on the unique features of the reactor design being assessed.
NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	No modeling details are available yet. It is expected that all reactor features that are important to severe accident behaviour for the given design be addressed and evaluated in the Level 2 PSA.
ONR	UK AP1000 PSA: • Severe accident reactor features such as In Vessel Retention (IVR), igniters, Passive Containment Cooling (PCS), deflagration to detonation transition DDT (Hydrogen) are represented in the containment event tree. In some cases conservative assumptions made to simplify the CET. • Parametric sensitivity to selected assumptions is examined in some areas, for example: RPV depressurization, IVR and hydrogen combustion. The analyses are usually specific to AP-1000. However, some information is drawn from earlier analysis of AP-600. Scaling arguments or adjustments to calculations have been made, when necessary. The original ULPU experiments for IVR, for example, were extended to capture the power and geometric differences between the two designs. UK EPR PSA:

	• The severe accident phenomena and challenges considered are the following: o Induced RCP [RCS] ruptures including hot leg and SG tube ruptures, o Fuel-Coolant Interactions (FCI), in-vessel and ex-vessel, o Hydrogen generation, distribution and combustion, including flame acceleration and Deflagration to Detonation Transition (DDT), o In-vessel recovery (i.e., evaluation of potential in-vessel corium quench before vessel failure) o Direct Containment Heating (DCH), and vessel rocketing, o Long-term containment challenges, including Molten Core-Concrete Interaction (MCCI) and long-term overpressurisation of the containment. • The quantification of individual split fractions is based on a Monte Carlo evaluation.
SNSA	N/A
STUK	See the Answer from AREVA-F3_OL3
UNISTAR	SA RCS depressurization valves (High pressure core melt, DCH), Core catcher for core melt stabilization (MCCI), SA Containment Heat Removal System (long term containment failure), Passive autocatalytic recombiners (H2 combustion or burning)
	G. Consequences analysis / PSA Level 3 technical aspects
	G 1. Do you agree that consequence analysis and Level 3 PRA can be treated in a design independent fashion? If not, how design may impact the consequence analysis?
AREVA OL3	No. Level 3 PRA needs source terms and their frequencies as input. Both are design dependent.
AREVA TSN	No level 3 PRA was performed.
AREVA US	The Level 3 needs the following inputs which are design dependent: • Source term and • Accident scenarios frequencies The design is reflected in the PRA model that produces the accident scenario frequencies and in the MAAP model that produces the source term.
BARC	The opinion on this issue is subjective and needs discussion.
Bel V	-
EDF FA3	Yes but consequence analysis needs source terms as input, design dependant.

EDF EPR UK	No. Level 3 PRA needs source terms and their frequencies as input. Both are design dependant.
ENEL	If a level 3 PSA will be developed it will be design dependent because in the design phase an objective would be the limitation of the releases, this aspect is not analyzed yet.
ENSI	No. Core inventory, containment failure probability and modes as well as height and energy of release may affect consequences and Level-3 risk measures.
IRSN	As for Gen II reactors, IRSN will not develop for EPR a L3 PSA but only L2 +PSA. L2+ PSA include a simplified assessment of radiological impact of the release for a standard meteorology. A specific paper is presented at the PSA2011 conference. For the EPR reactor, it may be particularly relevant for the verification of the design objectives regarding accident consequences. IRSN has no experience of consequence analysis achieved independently of the design. A comparison of L2 PSA for 900 MWe (single containment with steel liner) and 1300 MWe PWR (double containment without liner) consequences calculations show different level of consequences due to the containment design difference (e.g failure of the reactor building conduct to different impact for the aerosols). In general, we have the impression that the accident consequences for 2 different plants for one similar release path are equivalent only if the design is very similar (same section, same retention …). For that reason, we would not endorse that application of generic data for consequence analysis is a relevant approach in general. Nevertheless, it can be acceptable in a very preliminary approach.
JNES	PSA study of the plant doesn't include level 3 PSA or consequences analyses. So there is no answer for following questionnaires.
MHI	Although that consequence analysis and Level 3 PRA should be treated in a design independent fashion, it is difficult to ignore them. Because extreme risk would not be accepted, design would be changed in such case.
NRC	In general, the Level 3 analysis is treated independent of the reactor design. However, it is important to note that the regulations for nuclear power plants licensed under 10 CFR 52 do not require a Level 3 PSA.
NRI	It depends on the level of details required for the purpose of consequence analysis/Level 3 PSA. The different leakage paths and the layout of the buildings may impact the consequence analysis.
NUBIKI	In general, Level 3 PSA is not required by nuclear safety regulation in Hungary, so this aspect is not planned to be addressed in detail he project. On the other hand off-site effects (consequences) will have to be considered in relation to the environmental permit, although not in the for of a Level 3 PSA. Based on these facts, the answers below represent NUBIKI's position rather than the actual status or plans in the newbuils project in Hungary. Yes, only the source term (size, timing and position) depends on the design.
ONR	Source terms characteristics are design specific. However, it is agreed that within any Level 3 PSA, there is a large volume of input data that is not directly related to the power plant design and is either specific to the site or to the critical group habits. For GDA, the studies have been done specifically for each design. However, as many parameters needed in Level 3 PSA are site specific (for example weather and population distribution), the risks derived for GDA have been recognized to be indicative.

	A site and design specific L3 PSA is expected in due course.
SNSA	The Level 3 PSA analysis is not required by our legislation, still some consequence analysis will have to be done even for a sitting phase. We believe that this kind of analysis can be done in some part on the generic design, if possible releases (and releases frequencies) are known, i.e. assessed.
STUK	No. Source term analysis from level 1 and 2 PRA is necessary for performing level 3 PRA. Hower the level 3 PRA can be performed independently based on postulated source terms.
UNISTAR	No answer
	G2. What kind of study is available for the offsite consequences analysis? (Level 3 PSA, Consequences analysis, etc.) **• please describe the interface with Level 2, methods, computer code, etc.** **• releases from other than the reactor core were considered? (ex: fuel pool)** **• are the internal and external hazards initiating events considered?**
AREVA OL3	No level 3 PRA was performed.
AREVA TSN	No level 3 PRA was performed.
AREVA US	A Level 3 PRA was not a part of the Design Certification application. However, a Level 3 PRA, using the MACCS2 code, has been performed for each site where a COLA application for the US EPR™ is proposed. This level 3 PRA uses the results directly from the Level 2 PRA, combined with plant specific information to provide estimates of early fatalities, early and latent cancers, population doses, and economic impact.
BARC	The radiological impact assessment is addressed in the existing safety analysis reports. However, the level-3 PSA has not been performed.
Bel V	-
EDF FA3	Source term results were used to calculate the effective dose to a hypothetical individual that remains at a fixed location 500 m downwind of the reactor for a period of seven days after an accident. This was performed with the computer codes COSAQUE, CORRA and ASTRAL.
EDF EPR UK	At this stage, only a simplified Level 3 PSA is performed. The radiological consequence which is considered is the unmitigated effective dose to a child at 500 m downwind from the point of release during the first 7 days following the release, with standard weather condition “DF2”
ENEL	There are no study available yet.
ENSI	No level-3-PSA is required.
IRSN	As explained above, this is not achieved for IRSN EPR L2 PSA. A L2+ PSA will be developed.

	For FA3 Project deterministic consequence analyses provided by EDF are available (covering the PCC and RRC accident categories). IRSN is developing an integrated L1-L2 PSA with Risk-Spectrum (the progress is too limited to have any positive or negative opinion on the tool, the practical methodology is being established). Previous experiences with risk-Spectrum for L1 PSA and KANT for L2 PSA are very positive (900 and 1300 MWe PWRs separated L1-L2 PSA). The fuel handling accident is considered in the EDF deterministic studies. Fuel pool accident is not modeled in detail in L2 PSA. are the internal and external hazards initiating events considered? Not yet for IRSN L2 PSA and for EDF L2 PSA.
JNES	N/A
MHI	• please describe the interface with Level 2, methods, computer code, etc. The fission product source terms for the release categories (six release categories were defined for the US-APWR Level 2 PSA) were evaluated by using the MAAP 4.0.6 code. These release category conditions include the fission product release fractions, release height, release energy, and release duration. MACCS2 (Ver. 1.13.1) code is used for Level 3 PSA. • releases from other than the reactor core were considered? (ex: fuel pool) The release only from the reactor core was considered because releases from other sources are negligibly small relative to the release from the reactor core. • are the internal and external hazards initiating events considered? Internal fire and flood are considered.
NRC	• please describe the interface with Level 2, methods, computer code, etc. NRC uses the MACCS2 computer code to assess offsite consequence analyses. Applicant may use other computer codes, such as CRAC. • releases from other than the reactor core were considered? (ex: fuel pool) The PSA is typically limited to releases from the reactor core. Releases from other than the reactor core, such as spent fuel pools, are addressed in other studies. • are the internal and external hazards initiating events considered? The Level 3 PSA typically includes consequence analyses from internal event accidents.
NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	• please describe the interface with Level 2, methods, computer code, etc. No informatiom is available. • releases from other than the reactor core were considered? (ex: fuel pool)

	We would expect that all potential sources of large releases and severe consequences be considered. • are the internal and external hazards initiating events considered? Both kinds of hazards should be taken into account.
ONR	UK AP1000 PSA: • Limited scope Level 3 PSA analysis which uses the Level 2 PSA results, frequency and Source Term for each RC each with source terms characterized by environmental release rates and timing, together with isotopic content, as input. The Level 3 PRA code used is MACCS2 (MELCOR Accident Consequence Code System). • The Level 3 as the Level 2 PSA does not cover all sources of radioactivity (only the reactor core is included; fuel ponds, fuel handling facilities, waste storage tanks, etc, are not included). Low consequence sequences (Level 1 non-core damage sequences) are not included in the scope meaning that their radiological risk contribution is not taken into account. • As in the Level 2 PSA, fires, floods and external hazards are not included. UK EPR PSA: • See Answer from EDF EPR UK • The consequences analysis considers at-power conditions, shutdown plant states, internal fire and flood, external hazards as treated in Level 1 and accidents occurring in the Spent Fuel Pool.
SNSA	Our legislation does not require Level 3 analysis. Also for the existing plant this kind of analysis was not prepared.
STUK	Level 3 PRA not required
UNISTAR	Consequences analysis • please describe the interface with Level 2, methods, computer code, etc. Consequence analysis for the Release Categories (23) of the Level 2 PRA. Dose evaluations with MACCS2 • releases from other than the reactor core were considered? (ex: fuel pool) No • are the internal and external hazards initiating events considered? Yes: internal fire and flood
	G 3. How the sites specific aspects are considered? **• meteorological data** **• targets data (population, agriculture, land, economic)**

AREVA OL3	No level 3 PRA was performed.
AREVA TSN	No level 3 PRA was performed.
AREVA US	Plant specific information for the COLA sites included: • Population numbers and distribution • Year round meteorological data • Agricultural information – crops, growing season, location of cultivation • Economic data – farmland and non-farmland property value
BARC	Not applicable
Bel V	-
EDF FA3	Bounding assumption (e.g. standard weather condition "DF2").
EDF EPR UK	Bounding assumption (e.g. standard weather condition "DF2")
ENEL	The site specific aspects evaluated in the preliminary phase are meteorological data, population and land use.
ENSI	No level-3-PSA is required.
IRSN	This issue is not addressed in France for severe accident analysis. Some studies exists but are generic (some analysis have been performed by IRSN in the context of review of cost-benefits method proposed by the utility).
JNES	N/A
MHI	• meteorological data In the analysis for the standard design, the meteorological data of the Surry site has been used as the "typical", which is accessible in the MACCS2 code sample input file attached to the MACCS2 code. • targets data (population, agriculture, land, economic) In the analysis for the standard design, the population data of the Surry site in the MACCS2 code sample input file has been adjusted to be representative of about 80% of the U.S. nuclear plant sites in NUREG/CR-2239, "Technical Guidance for Siting Criteria Development".
NRC	Representative site parameters are selected to perform the consequence analysis. The representative site parameters may be based on existing site data for an operating reactor. The representative site parameters are typically selected to bound many possible reactor sites in the United States.

NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	No information is available. As a requirement, we consider that all relevant site specific historical and data (including that of the above domain) must be taken into account with state-of-the-art consequence assessment codes.
ONR	UK AP1000 PSA: • These analyses consider 'generic' assumptions about weather and are conducted to estimate the whole-body dose and acute red bone marrow dose, both at the site boundary (0.5 miles). The population whole-body dose out to 80.5 kilometres and the downwind, centerline, ground-level thyroid dose at the site boundary (0.5 miles) are also calculated for information. • The analysis carried out is biased towards US methodology and regulatory requirements. Consequently, inferences have to be made when comparing the assessment results with SAP targets as direct comparisons are not possible. Compliance is claimed with Numerical Targets 5 to 9 of the SAPs (corresponding to risk from accidents) on this basis without additional analysis. UK EPR PSA: • See Answer from EDF EPR UK • Consequences are calculated in terms of long term doses for comparison with Target 8. Results do not include the specific calculation of early or late health effects based on organ doses. Based only on these doses an estimate of risk of death is made for comparison with Target 7 and a screening method is adopted to identify releases that are likely to result in significant off-site consequences for comparison with Target 9.
SNSA	N/A
STUK	Level 3 PRA not required
UNISTAR	• meteorological data The meteorology data used in the analysis was hourly data for one year that includes wind velocity (speed and direction), stability class, and rainfall. • targets data (population, agriculture, land, economic) To assess human health impacts, the analysis determined the expected number of early fatalities, expected number of latent cancer fatalities, and collective whole body dose from a severe accident to the year 2050 population within a 50-mile radius of the plant. Economic costs were also determined, including the costs associated with short-term relocation of people, decontamination of property and equipment, and interdiction of food supplies.
	G 4. How are the emergency actions considered? **• sheltering, evacuation, decontamination, etc.** **• is the off-site emergency response plan available?**
AREVA OL3	No level 3 PRA was performed.

AREVA TSN	No level 3 PRA was performed.
AREVA US	Actions dictated by the Emergency Operating Procedures and Severe Accident Management Guidelines are included in the Level 1 and Level 2 PRA. The: Level 3 PRA includes the sheltering and/or evacuation of populations affected by the plume(s), depending on the severity of the release.
BARC	Not applicable
Bel V	-
EDF FA3	Consequence analysis doesn't credit any emergency actions.
EDF EPR UK	Bounding assumption (no evacuation)
ENEL	In this preliminary phase the emergency action like sheltering, evacuation, decontamination, stable iodine intake are considered An off-site emergency plan will probably be developed with site specification.
ENSI	No level-3-PSA is required.
IRSN	This issue is not formally addressed in the L2 PSA in France (no requirement by the Safety Authority). IRSN uses L2+ PSA approach for Gen II to have some indication on the compatibility of emergency plan with the consequences of potential accident, but conclusions are used in a context of continuous plant improvement. EDF calculated the total frequency of accident that may not be manage by existing emergency plans. For Gen II reactors, a target of 1.10-6 /y.r was proposed by EDF during the 3rd PSR of 900 MWe PWRs (value not endorsed by the Safety Authority). For EPR, L2 PSA will be used to check that accidents with low RCS pressure would have a limited impact in the environment. This point will be discussed in 2011 during the EDF EPR (Fla3) L2 PSA review by IRSN.
JNES	N/A
MHI	• sheltering, evacuation, decontamination, etc. No emergency actions are considered in Level 3 PRA for the design certification application. In the analysis for the standard design, Level 3 PRA is calculated for the severe accident mitigation design alternative (SAMDA) calculation. The evacuation is not assumed, and therefore the population dose risk is conservatively evaluated. • is the off-site emergency response plan available? In the analysis for the design certification application, Level 3 PSA is calculated for the SAMDA calculation. The off-site emergency response plan is not assumed, and therefore the population dose risk is conservatively evaluated.

NRC	Emergency actions are not typically considered in the design stage PSA. For applicants that have selected a specific site and are applying for a combined license, a discussion of the emergency plans is required.
NRI	PSA for new units in the Czech Republic has not been developed.
NUBIKI	No information is available but emergency actions need to be considered for a credible consequence analysis and sensitivity studies. This assumes the availability of developed emergency plans.
ONR	UK AP1000 PSA: • The site boundary study doesn't take into account emergency actions. • A site specific L3 PSA is expected in due course. UK EPR PSA: • See Answer from EDF EPR UK • A site specific L3 PSA is expected in due course.
SNSA	For the existing plant the off-site emergency response plan is available. It also uses meteorological data as well as inputs regarding releases to support decision making (sheltering, evacuation, use of potassium iodine).
STUK	Level 2(3?) PRA not required but emergency operating procedures and emergency plans, and severe accident management guidelines are followed
UNISTAR	sheltering, evacuation, decontamination, etc. yes for sheltering and evacuation, depending of the severity of the release. is the off-site emergency response plan available? no