1. EXECUTIVE SUMMARY

1.1 Background

Probabilistic Safety Assessment/Probabilistic Risk Assessment (PSA/PRA) for new and advanced reactors is recognized as an important approach to achieve improved safety and performances of new nuclear power plant (NPP), comparing to the existing plants. However, the application of PSA to these reactors encounters 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 in the last phases of design and commissioning stage, typically within five to ten years of commencing power operations, 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 risk balanced concept etc.) 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. 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 is currently conducting two coordinated tasks: “PSA for Advanced Reactors” and “PSA in the frame of Design and Commissioning of New NPPs". In order to support the objectives of these two tasks, the WGRISK organized a joint workshop entitled “OECD/NEA Workshop on PSA for new and advanced reactors,” which was held at the OCED Conference Center during June 20-24, 2011. The present report summarises the result of this workshop.

1.2 Objective of the workshop

The key objective of the workshop was to support the fulfillment of the two coordinated WGRISK Tasks on “PSA for Advanced Reactors” and on “PSA in the frame of Design and Commissioning of New NPPs." The objectives of the two tasks are:

(1) PSA for Advanced Reactors

- 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 the PSA for advanced reactors.

(2) PSA in the frame of Design and Commissioning of New NPPs

- Identify and characterize current practices regarding the role of PSA in 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.

1.3 Organization of the workshop

The main topics of interest, discussed during the workshop, included the followings: regulatory aspects, risk-informed methods, technical aspects of the PSA for new and advanced reactors, hazards of PSA (internal and external), severe accident/source term/Level 2 PSA, and consequence analysis/Level 3 PSA.

The workshop program is provided in Appendix 1.

The paper presentations and discussions were performed during the first 3 days of the workshop (20-22 June 2011). Two additional days (23-24 June 2011) were dedicated to the preparation of the draft proceeding and conclusions by the Workshop Technical Committee members and Workshop session chairpersons.

Fifty experts from 13 countries and one international organization (IAEA) participated in the present workshop, and 35 technical papers and 2 CAPS task activities were presented as shown in the following:

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

1.4 Results

The joint workshop provided an interesting and useful forum for the participants to share and discuss their respective practices regarding PSA applications for new and advanced reactors.

The workshop discussions indicated that there is currently a wide spectrum of views and practices, key technical and regulatory issues requiring further work, as well as potential areas for future international collaboration. Since the workshop was not designed to achieve consensus, this report does not provide any group recommendation regarding the underlying issues. However, the workshop results are an essential input to these two WGRisk tasks on “PSA for Advanced Reactors” and on “PSA in the frame of Design and Commissioning of New NPPs” and the final report for these tasks will provide recommendations.

Globally, the technical issues discussed during the workshop, like passive systems reliability, reliability data, digital I&C (Instrumentation and Control), HRA (Human Reliability Assessment), and external events (however widely discussed after Fukushima events) etc., are not newly added emerging issues, the workshop was regarded as a good opportunity to discuss the day status and to exchange lessons learned among the participants.

The followings summarize key points of the presentations and discussions made during the workshop.

(1) PSA in the frame of Design and Commissioning of New NPPs:

Current practices:

- For all new reactor projects, the role of PSA is more important and more formalized as compared with the operating plants;

- For new reactors, PSA is used by the industry at all stages of the design for a wide variety of applications, including demonstration of safety level, 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 to identify risk-significant areas for safety reviews and some of them developed requirements for PSAs and applications;
- Regulatory and industry organizations in some countries are supporting the development of standards for PSA development and applications;
- Regulatory PSA models are developed in some countries and are used for a confirmatory check of an applicant's model;
- The participants expressed the need for better advice, based mainly on the lessons learned, on how to use PSA during the different design phases;
- It 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), challenges in ensuring strong interaction between design and PSA teams as the design evolves;
- Despite data, modeling and code limitations, Level 3 PSA was identified as a necessary support for some new reactor applications (e.g. definition of the emergency zones);
- Applicability of PSA for addressing the safety-security interface is an interesting topic.

<u>Key technical issues for further work</u>:

- Better guidance and lessons learned exchange on PSA development and use in the different design phase;
- Data and modeling improvements and clarifications, especially in the modeling of: new components, new severe accident features, passive systems, digital I&C, intersystem CCF (Common Cause Failures), and human actions etc;
- Improve completeness in the modeling of external events, including treating of potential combinations of hazards and their impacts on multi-unit sites.

<u>Potential areas for international collaboration</u>:

- External events modeling;
- Passive systems modeling;
- International reviews of new/advanced reactors PSA and PSA applications.

(2) PSA for advanced reactors:

<u>Current practices</u>:

- PSAs are being used at the conceptual or preliminary design stages, most analyses use currently available PSA methods including the conventional ET (Events Tree) / FT (Fault Tree) and RMPS (Reliability Method for Passive Systems) 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 modeling into the PSA (e.g., RMPS, Dynamic PSA, and DDETs (Discrete Dynamic Event Trees));
- Gen IV RSWG (Risk and Safety Working Group) is developing TNF (Technology Neutral Framework) and ISAM (Integrated Safety Assessment Method) for safety assessment of advanced reactors, which synthesizes the different approaches to safety;
- The USA is developing ASME/ANS standards for advanced non-LWR ((Light Water Reactor) PRA.

Key technical issues for further work:

- Scope of the PSA for advanced reactors, compared to that of the current and new reactors;
- Modeling approaches and tools to assess potential severe accidents at the pre-conceptual design phase, especially non-LWR types of reactor;
- Increasing competence in identifying potential hazards and accident scenarios (what can go wrong), helpful for designers;
- Some guidance to answer the question as to whether the PSA state of the art is adequate to support a specific application;
- Risk metrics and safety goals for advanced reactors;
- Handling the safety-security interface;
- Peer review guidance for the PSA.

Potential areas for international collaboration:

- Guidance to determine technical acceptability of PSA for advanced reactors (including the ASME/ANS non-LWR standard) and provide its implementation process;
- Modeling approaches and tools to support advanced reactor-specific phenomena analysis;
- Assessment of potential severe accidents at the pre-conceptual design phase of advanced reactors;

- Passive safety systems reliability and ISAM under advanced design features;
- Collaboration with ongoing international activities (e.g., GIF-RSWG and IAEA CRP on passive system reliability).

2. SUMMARY OF THE WORKSHOP SESSIONS

The workshop was composed of 9 technical sessions and 3 break-out sessions. The break-out sessions were intended to provide a framework to present summaries of the day presentations and to discuss the day open issues and conclusions. The summaries of the task questionnaire answers were presented during the first two break-up sessions (advance reactors on Monday 20 June and new reactors on Thursday 21 June. The last break-up session on Wednesday 22 June was also dedicated to preparatory discussion for the proceedings development. The following is a summary of these sessions prepared by the corresponding session chairs.

2.1 Day 1 (20 June)

(1) Summary of Technical Session 1 (by Nathan Siu)

This session involved presentations on the performance of PSAs for four advanced reactor designs. (All four designs employ passive decay heat removal systems.) All of the analyses were performed by the authoring organization(s) to support the design process some of the analyses have been discussed informally with regulatory authorities but none have been formally reviewed.

<u>The first presentation</u> ("Level-1 PSA to Support the Design of the KALIMER-600 Sodium-cooled Fast Reactor," S. J. Han, KAERI-Korea) introduced the application of a Level 1, internal events, at-power PSA to support the design of KALIMER-600, a 600 MWe, metallic-fueled, sodium-cooled fast reactor being designed by KAERI. The paper outlined the technical approach used, presented a number of results for CDF (Core Damage Frequency), including the results of sensitivity analyses, and identified important sources of uncertainty.

<u>The second presentation</u> ("Study on Preliminary Level-1 PSA for Japan Sodium-cooled Fast Reactor," K. Kurisaka, JAEA-Japan) introduced the application of a Level 1, internal events, at-power PSA to support the conceptual design of the JSFR (Japanese SFR), a 1500 MWe, mixed-oxide fueled, sodium-cooled fast reactor being designed by JAEA. The paper also described a seismic margins analysis performed to show conformance with stricter seismic standards imposed after the 2007 Niigata-ken Chuetsu-oki earthquake. Further, the paper develops reactor-specific CDF and CFF (Containment Failure Frequency) goals based on: a) the point that existing CDF and CFF goals (1E-5/yr and 1E-6/yr, respectively) apply to the site as a whole, and b) a fractionation of these goals to account for the number of reactors on site.

The third presentation ("Level 1 Probabilistic Safety Assessment to Support the Design of the CEA 2400MWth Gas-cooled Fast Reactor," F. Bertrand, CEA-France) introduced the general use of deterministic and probabilistic methods in the design of the GFR 2400, a 1120 MWe ceramic-plate fueled, helium-cooled fast reactor being designed by CEA. Similar to the previous papers, the PSA addresses Level 1 internal events occurring during power operation. The analysis does not include human reliability considerations – these are planned to be addressed later. Some specific technical aspects of the PSA are dealt with in companion papers presented later in the workshop.

The fourth presentation ("Overview of VHTR's PSA Approach in Korea," S. J. Han, KAERI-Korea) introduced the application of an internal events, at-power PSA to support the conceptual design of a KAERI-designed, TRISO-fueled, helium-cooled VHTR (Very High-Temperature Reactor). Due to the non-threshold release characteristics of the reactor's TRISO (TRi-ISOtropic-coated fuel particles) fuel, the paper addressed plant damage states with varying degrees of release as well as a consolidated core heat-up frequency. Due to the stage of the project, the results presented in the paper are tentative.

Discussion: All of the reported analyses employed currently available PSA event tree/fault tree analysis technology. Specific analysis features reported included the use of: Master Logic Diagrams and methods considering heat balance and reactivity balance to support initiating event identification; worldwide experience to estimate the leak frequency-magnitude relationship; documented assumptions regarding PSA model parameters (including initiating event frequencies); explicit models for passive system actuation; multi-state event sequence modeling to address the continuous performance of passive systems; Monte Carlo methods to propagate uncertainties and sensitivity studies to address the potential importance of key uncertainties. The PSA technical issues raised by the papers and subsequent group discussion included: reliability parameter estimation for new SSCs (Structure, Systems and Components) (including CCF) and phenomena (e.g., sodium solidification frequency), passive safety system reliability estimation, digital I&C system reliability, human reliability, and severe accident modeling and source terms.

The PSA uses identified in the papers and subsequent discussion included: the identification of design improvements (e.g., addition of non-safety, electrically-powered blowers; addition of redundant, diverse actuation; rejection of a proposed additional cooling loop to simplify the plant's design); and the demonstration of achievement of design targets (including "balanced design" as well as specific risk-related targets). One of the papers raised the potential need for a common regulatory framework for advanced non-LWRs, and the group also discussed the potential need for different design targets to reflect the specifics of plant damage states.

(2) Summary of Technical Session 2 (by Attila Bareith)

Four presentations were given in Session 2. The focus was on using PSA for advanced reactor designs, although two papers from the four also addressed currently operating as well as new reactors. The first two papers covered safety assessment methodologies, while the last two presentations described examples of specific safety analyses either as full-scope PSAs or as system analyses performed in support of a risk-informed approach to plant design.

The first presentation ("Generation IV Integrated Safety Assessment Methodology," T. Leahy, INL-USA) described the activities of the Gen. IV RSWG that supports the efforts on developing Generation IV systems by promoting a consistent approach to safety, risk and regulatory issues. RSWG develops a technology neutral methodology for safety assessment of advanced reactors. The most important requirements and desired characteristics for the methodology were described. An ISAM is being developed in response to these needs, which synthesizes the different approaches to safety assessment with PSA having a key, central role in the process from pre-conceptual design up to licensing and operation. Some elements of the proposed methodology have been applied to a limited extent to French and Japanese SFR concepts. The methodology document will be finalized in 2011. Its detailed applications are to come afterwards with pilot studies on selected Generation IV systems.

The second presentation ("ASAMPSA2 Project: Appliance of LWR PSA2 Methodology to GEN IV Reactors," H. Bonneville, IRSN-France) included an overview of the European ASAMPSA2 (Advanced Safety Assessment Methods on PSA Level 2) project in the first place. In particular, the harmonized methodology guide on Level 2 PSA was highlighted as the most important result of the project. A draft version is available, improvements are foreseen in 2011. A limited effort was made to examine the relevance and applicability of the guide to actual Generation IV concepts. Four selected concepts were used for the assessment: SFR (Sodium-cooled Fast Reactor), GFR (Gas-cooled Fast Reactor), LFR (Lead-cooled Fast Reactor) and VHTR. The conclusions of the study suggest that Level 2 PSA can be performed at an early stage in the design of advanced reactors to yield risk insights and help prioritize R&D activities. However, compliance/applicability of the guide to Generation IV reactors cannot be assessed yet in detail due to the current status of design, lack of PSA models and other information, like EOP (Emergency Operating Procedure) and SAMG (Severe Accident Management Guideline) necessary for performing Level 2 PSA.

As follow-on to the description of risk-informed support to the design of the CEA's 2400 MWth gas-cooled fast reactor, the third presentation ("Reliability Analysis of 2400 MWth Gas-cooled Fast Reactor Natural Circulation Decay Heat Removal System," M. Marques, CEA-France) discussed the reliability analysis of the decay heat removal system with natural circulation as a specific example of this approach. System reliability was evaluated for two representative scenarios, a LOFA (Loss Of Flow Accident) and a LOCA (Loss Of Coolant Accident) sequence, respectively) based on a complex set of failure criteria following the RMPS methodology. Uncertainties were propagated through thermal-hydraulics model during simulations performed by the CATHARE-2 code. The system failure probability was found very low in the LOFA scenario, while improvements in design parameters and associated uncertainties were seen necessary in case of LOCA. The study yielded insights into making these improvements.

The last presentation ("Level-1 PSA for Internal Events for TAPS3&4 - A Challenge," R. Guptan, NPCIL-India) covered PSA for Indian PHWRs (Pressurized Heavy Water Reactors) including the existing 540MWe reactor, and, to a lesser extent, the new reactor of 700MWe as well as their advanced heavy water reactor. Extended level 1 PSA for internal events was performed to find dominant risk contributors and identify weak links / imbalance using mostly IAEA guidelines and applying a strict quality assurance program. Over and above risk quantification, recommendations were developed to maintain and further improve plant safety by focusing better on important risk contributors (e.g. ensuring risk awareness through training) and also by making modifications (e.g. introduction of staggered testing for ECCS (Emergency Core Cooling System). Risk reduction was also found feasible for the 700 MWe PHWR. Finally, PSA resulted in very low frequency of core damage and core degradation attributable to some salient passive safety design features of the plant. From the point of view of methodology the intention to apply the ATHEANA (A Technique for Human Error Analysis) framework to human reliability analysis was emphasized.

Discussion: The most important aspects from the follow-on discussions that have relevance to the objectives of the WGRISK task for advanced reactors and, to a certain degree, to new reactors can be summarized as follows:

As it is envisaged, ISAM promises to be a methodology that helps ensure the use of appropriate considerations to safety/risk aspects throughout the life cycle of advanced reactors. By putting the use of risk assessment and risk information into the center of the proposed methodology an important aim is to promote risk-informed decisions in the various life cycle phases.

In response to a question it was mentioned that ISAM does not propose particular quantitative safety objectives to be met. It was also discussed that the six System Steering Committees for the Generation IV reactors support ISAM largely. However, there is some concern as far as the level of effort and expertise needed to perform the proposed assessment. These concerns might have important implications for the way forward including the requirements for ensuring PSA quality.

It was clearly spelled out that for Level 2 PSA the core degradation mechanisms specific to the different reactor concepts have to be considered. Although there are noticeable differences in the level of knowledge for the different concepts, in general there is a lack of understanding needed to better model severe accidents due to the lack of experimental data and to the limitations of existing analysis tools. Developments of EOPs and severe accident management guidelines is also seen as an important condition to be able to better assess the capabilities and limitation of current level 2 PSA methodology for advanced reactors.

There were some interfacing discussions with the last workshop session in relation to the modeling of passive systems. In that last session the needs for improving analysis methods, tools and data were discussed at length. On the other hand, the example of the reliability analysis for the decay heat removal system in the 2400MWth GFR reactor witnesses an application and use of currently available techniques and tools (RMPS methodology) up to the level of quantification needed for PSA. Similarly, failure of natural circulation was modeled in the PSA of the advanced Indian PWHR too. Thus some analysts argue that the methodology is mature enough for useful and credible applications. Although a consensus was not aimed at and could not be reached at the workshop, there appears to be an agreement that meaningful analysis of passive systems is possible in the framework of PSA with considerations to the limitations of current methodologies, tools and data. In parallel, the analysis areas in need for further developments have also to be emphasized.

(3) Summary of Technical Session 3 (by J.J. Tong)

In this session, there are four presentations given by different organizations/countries.

The first presentation ("Applying Risk Insights in USNRC Reviews of Integral Pressurized Water Reactor Designs," M. Caruso, NRC-USA) gives the background of the framework which USNRC will use to more fully integrate the use of risk insights for the review of the SMR (Small Modular Reactor), and take the identification of risk-significant SSCs and other aspects of the design that contribute most to safety as the illustration. It concludes that the framework is a graded approach. The four-step process for determining risk significance of SSCs includes: Assembly of Design/Plant-Specific Information; Identification of Plant Systems and system features; Risk Significance Determination for System Functions; and Update Risk Significance Determinations as Necessary. NRC doesn't have any iPWR (integral Pressurized Water Reactor) licensing applications yet. Since there's no pool of experience peer reviewers, self-assessments are expecting. Challenges about the handling of new improvements, e.g., submerged valves, helical SGs (Steam Generators) are foreseen. The objective is to use PSA to help NRC doing its job more efficiently and effectively.

The second presentation ("Development of PSA Audit Guideline and Regulatory Model for SMART," N. Cho, KINS-Korea) summarizes the design features of the SMART (System-Integrated Modular Advanced Reactor, 330MWt) which is under development by KAERI for dual purposes of power generation and seawater desalination, and also the purpose to develop a regulatory PSA model in order to assure the technical adequacy of SMART PSA. Key technical issues identified during the development of SMART PSA include the modeling of PRHRS (Passive Residual Heat Removal System), and the insights from the preliminary quantification results. General standard modeling approach is used for PRHRS, but does have the phenomena related failures in the fault tree.

The third presentation ("Use of PSA in the Development of SMRs," A. Maioli, Westinghouse-USA) shares the Westinghouse experience of using PSA both in the design of AP1000 and IRIS (International Reactor Innovative and Secure, an Integral PWR under development, 200MWe with straight-tube steam generator). It indicates that a risk-informed framework within which PSA and the resulting risk insights can be used in the decision making process at various stages of the design needs to be developed. Challenges of using risk insights in the design include the coordination and interaction among PSA team, design team and the T-H team, epistemic uncertainty treatment, unique modeling problems due to the design (e.g., HRA, code abilities to model other than classic LWR scenarios, external events...), risk metrics and new applications such as risk-informed Emergency Planning Zone. CDF/LRF (Large Release Frequency) may be not of interest in the initial steps of the risk-informed design process other than system reliability.

The last presentation ("Achievement of the Level 1 PSA in Support to the CEA 2400MWth Gas-cooled Fast Reactor," M. Balmain, EDF-France) intended to define what could be the contribution of a level 1 PSA to support the design and safety demonstration by the application for the GFR, and also to exhibit the methodological trends for an efficient PSA model through the project life cycle. It presents with the example of using the relative results and a limited perimeter to support the design, and also another example of using the absolute results and enlarged perimeter. It reports that the design team may put their focuses on the key-points from the feasibility or technical points of view, e.g. the core or some components, so that the preliminary design of GFR presented a heterogeneous level of design for various components and support systems. The nature of the PSA model provides a good frame to link different kinds of knowledge together and balance them. And PSA also led to the consideration of long mission time. It is foreseen that the use of PSA will be continued in the design of a new SFR based on the successful experience.

In general, three of the presentations in this session, including No.1, No.3 and No.4, are focused on the overall roles and applications which PSA played during the design and review of the advanced reactors. General common conclusions which can be drawn from these practices as well as the discussions followed are:

- 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 (T-H, neutronics, and mechanics, etc.) 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;
- 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.

One presentation raised another issue about establishing a regulatory PSA model for confirmatory check of the technical adequacy during the design certification phase. 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.

(4) Summary of Breakout Session 1 (by K.I. Ahn)

This session includes both a summary review of the questionnaire responses and key discussion points of the day’s technical paper sessions.

(4-1) <u>Summary of Questionnaire Answers Survey on PSA for Advanced Reactors by K.I. Ahn</u>

The questionnaire answers on PSA of advanced reactors collected from the twelve countries (16 organizations) spanned wide spectrum of reactor types and associated programs, including new/evolutionary reactors (e.g., EPR, AP1000, ABWR, APWR, ESBWR, APR1400, etc.) as well as advanced reactors in the conceptual or preliminary design stage (e.g., SFR, LFR, GFR, HTGR, iPWR, SMR, SMART, etc.). In order to draw out common-interest PSA issues required specially for advanced reactors, first of all, it seems necessary to more clearly discriminate the underlying difference between new/evolutionary and advanced reactors among member countries. Based on the questionnaire answers, major preliminary observations and findings were presented to the workshop participants and are summarized as follows:

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

- While a few countries are limitedly using for the conceptual or preliminary design stages of advanced reactors, more efforts are currently being focused on indentifying and resolving PSA issues for the 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 that answered 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, and HRA, etc.) is expected to be generally applicable for a wide variety of designs.

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 (a) while some respondents express the applicability of Level 1/2/3 for some reactor types, (b) others have no issue; while some respondents expect treatment of all modes and all hazards, others are more limited; and (c) while some respondents think current ET/FT technology is fine, some are doing research, some are using different methods for certain problems. So far, those items remain possible to discuss (in the conclusions for this workshop) the extent to which workshop discussion helped.

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 (Digital I&C), HRA, and 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 modeling into the PSA (e.g., RMPS, Dynamic PSA, DDETs). While those methods have a potential value for their application to advanced reactors, 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 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 currently exist any consensus on how to take into account those issues for some advanced reactors and on the level of depth to take into account. So far, those items

remain possible to discuss (in the conclusions for this workshop) the extent to which workshop discussion helped.

Based on these results, the potential topics of interest for further discussions in the workshop may include

- Scheme to determine technical acceptability of PSA for advanced reactors (including the ASME/ANS non-LWR standard) and provide its implementation process;
- Establishment of surrogate risk metrics and target for advanced reactors;
- Compliance of the PSA with defense-in-depth principle in the regulation framework;
- Advanced reactors-specific accident phenomena and passive safety system reliability, reliability databases, human and digital system reliabilities, accident sequences and event classifications for PSA modeling and adequacy of current phenomenological models to support the analysis, and aggregation of the risk from different hazard types into the plant risk in an integrated or technology-neutral way;
- Assessment for the possibility of potential severe accidents in the pre-conceptual design phase of advanced reactors;
- More consolidated ways to collaborate with the other international programs (e.g., OECD/CNRA-WGRNR, GIF-RSWG, and IAEA CRP on passive safety systems reliability).

(4-2) <u>Summary of the Day's Presentations and Key Discussion Points</u>

Summaries and highlights of the Day 1 Sessions (especially dedicated to PSA for advanced reactors) were introduced by each session chair (Session 1 by N. Siu, Session2 by A. Bareith, and Session 3 by J.J. Tong), and key points made in the discussion of those sessions are as follow:

- PSA community needs to challenge itself in identifying scenarios on what can go wrong for new designers;
- The targets of risk for advanced reactors should be determined as soon as possible;
- Quantitative PSA results may not be very useful at the conceptual design stage of advanced reactors but PSA is still reasonable in providing a systematic approach;
- Security issues should be considered for advanced reactors;
- Some groups are wrestling with the question as to whether the PSA state of the art is adequate to support a specific application. Guidance would be helpful;
- Some surprise that there was not much discussion on issues and phenomena where tools used did not do very well. This would indicate where more work would be helpful.

2.2 Day 2 (21 June)

(1) Summary of Technical Session 4 (by J.M. Lanore)

In this session three papers were presented: two papers relating to I&C modeling in PSA, and a more general paper relating to the application of Design-Reliability Assurance Program.

The first presentation ("Application of Fault Tree Methodology to Modeling of the AP1000 Plant Digital Reactor Protection System," David Teolis, Westinghouse-USA) introduced a real example, the characteristics for modeling of I&C by fault tree methodology, taking into account hardware and software failures, and CCF corresponding to these different failure modes. The paper mentions the difficulties for finding data. However the model is being developed and is intended to be integrated in the PRA model.

The second presentation ("I&C Modeling in the IRSN EPR Level 1 PSA" J. Delache and G. Georgescu, IRSN-France) introduced a real example of I&C modeling by fault tree methodology, taking into account hardware, software, CCF, support systems (electrical and ventilation). The model was reviewed by I&C specialists; this model is rather simple but still has hundreds of fault trees. CCF contribution is dominant. The objective of this work is to analyze the PSA proposed by EDF for Flamanville3 EPR design. A particular objective is to assess the importance of a backup system.

The discussions following these two papers were very similar: 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 modeling 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 CCF failures.

A general comment is that although there is no real methodology consensus for I&C modeling and quantification, some tentative approaches are developed and integrated in PSAs. Another general comment is that digital I&C is not really specific to new plants, but due to the improved general safety level the role of I&C is increasing and becomes a potentially dominant issue.

The third presentation ("Analysis of Design-Reliability Assurance Program in ACP600 Application," Huang Zhichao, CNPE-China) introduced D-RAP (Design-Reliability Assurance Program), which is a formal management system of plant performance and safety information, applied to ACP600 (a new GEN III advanced reactor). The conclusion is that D-RAP is a useful tool for new reactor development, combining deterministic and probabilistic analyses advantages.

Moreover an answer to a question indicates that this tool contributed to the definition of several safety improvements. The discussion was also related to the treatment of passive systems, and this point was discussed more widely during other sessions.

(2) Summary of Technical Session 5 (by K. Ahn and G. Georgescu)

Four papers were presented in this session: two papers addressing new and improved modeling employed in a severe accident analysis code, MAAP5, and two papers related to the Level 2 PSA (one for Japanese SFR and the other for PHWR employing some partly uncommon features).

The first presentation ("In-Vessel Retention Modeling Capabilities – IVR - in MAAP5," Quan Zhou, FAI-USA) introduced new models and improvements made in MAAP5.01 specifically to address complex phenomena important for IVR (In-Vessel corium Retention). 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. These models important to the IVR evaluation were implemented in MAAP5.01 and were successfully tested for the AP1000® passive plant. He said that the AP1000® plant results demonstrate how MAAP5.01 can be used to evaluate IVR and to gain insight into responses of the lower head during a severe accident. Finally, he emphasized that this was one of the first integrated, transient calculations for evaluation of IVR and the MAAP5.01 will be used as a platform of severe accident simulator for the AP1000®.

The second presentation by Quan Zhou ("MAAP5 Modeling Capabilities for Initial Plant Transients and Shutdown States, and Application to Shutdown PSA and Full Scope SAMG Covering All Plant States for Operating and New Plants") introduced major improvements made in the latest MAAP5.0.1 code to simulate initial transients and shutdown conditions in nuclear power plants, status of development of the full scope SAMGs covering all plant operating states including Shutdown SAMG (SSAMG) integrated into at-power WOG (Westinghous Owners Group) SAMG to form a complete symptom-based SAMG package applicable to all POSs (Plant Operational States), and capabilities of the MAAP code simulating non-severe accident transients demonstrated with selected AP1000 transients. He also emphasized capabilities of MAAP5 as an appropriate tool for severe accidents and key steps of the Shutdown SAMG development, and to execute long term non-severe accident transients and PRA Level 1 success criteria calculations. Nevertheless, a few challenges and uncertainties still remain for the followings:

- There are specific challenges to thermal-hydraulic codes for low power and Shutdown plant states; verification of codes, model modifications and improvements;
- Regarding the severe accident phenomenology, the remaining uncertainties, and also the diversity of accident scenarios considered, the development of Shutdown SAMG is still a very complex activity;
- For SSAMG validation, operator and TSC (Technical Support Center) training exercises and upgrades to Full Scope Simulators are required to support high fidelity simulation of shutdown states, including low reactor inventory states, open reactor and open containment states, refuelling operations, and spent fuel pool accidents.

The third presentation ("Development of level 2 PSA Methodology for Sodium-Cooled Fast Reactors," T. Suzuki, JAEA-Japan) introduced the current status of evaluation-technology development of Level 2 PSA for SFRs made by JAEA to systematically assess the phases/sequences to be evaluated in Level 2 PSA: development and verification of computational tools for the material-relocation phase and the ex-vessel accident sequence, and consolidation of the technical basis for constructing phenomenological event trees, in which the information on the related analyses/experiments were compiled so as to determine the event progression and branch probabilities within the event trees. The conclusions of the paper are summarized as follows:

- JAEA newly developed MUTRAN and SIMMER-LT codes in order to evaluate the long term behaviors of the material-relocation in the degraded core. These tools enabled systematic simulations of the material-relocation phase;
- JAEA also improved CONTAIN/LMR code taking into account the feature of SFRs and verified the analytical models in CONTAIN/LMR by utilizing the new experiments such as sodium-concrete reaction test. As a result, the CONTAIN/LMR code with the improved models enabled appropriate simulations of the ex-vessel accident sequence taking into account the feature of SFRs;
- The general information needed for the Level 2 PSA of SFRs (including the construction of event trees) was compiled as a technical data basis, in which the dominant factors having significant effects on the event progression were corresponded to the related analytical/experimental results.

The last presentation ("PSA Level 2 as Element of an Integral Safety Assessment before Plant Commissioning," H. Löffler, GRS-Germany) introduced the experience with CNA II (PHWR reactor under near completion in Argentina) with respect to PSA Level 2 for new and advanced reactors. He emphasized that although CAN II is not a recent design, it has some partly uncommon features, so that there is a certain resemblance to performing a PSA for a new design. Additional focus was on the definition of the two interfaces to PSA Level 1 and Level 3, and key issues taken into account in defining release categories. For the Level 2 PSA, both MELCOR code and EVNTRE methodology were used for deterministic accident simulation and Probabilistic accident progression analysis, respectively. Experience gained from the PSA Level 2 for CNA II was highlighted as follows,

- Existing deterministic (MELCOR) and probabilistic (EVNTRE) methods and PSA guidelines in general are flexible enough to analyze new or especially uncommon reactor designs;
- Plant specific design details may require specific analyses or estimates beyond present code capabilities, and they can largely determine the PSA results;
- If PSA level 3 is required, significant uncertainty exists regarding the definition of source terms and the selection of representative or most "challenging" cases;
- In particular the behaviour of Iodine is still not covered satisfactorily by state-of-the-art models in MELCOR. Additional effort was needed in the event tree to represent gaseous iodine;
- A precise definition of interfaces between the PSA levels supports understanding among different PSA teams and enables parallel work on the different levels. A direct transfer of MELCOR results to the PSA Level 3 team is a simple and useful approach.

(3) Summary of Technical Session 6 (by Reino Virolainen)

The first presentation ("Regulatory Assessment of the PSAs for the UK-EPR and AP1000 Reactors in the UK," A. Gomez Cobo, NII-UK) gave a detailed procedure to conduct a reactor type specific regulatory assessment, "Generic Design Assessment (GDA)" which includes also a PRA review. The GDA is running through four steps, from general to detail.

PRA is mainly reviewed during GDA step 3 focused on methods, techniques and scope. This corresponds to a generic review step in a typical PRA review procedure. The main review is performed during GDA step 4 but the review is not focused on each event tree, fault tree and data in detail but a representative sample is selected. This corresponds partly to a detailed review step in a typical PRA review procedure. However the assessment does not pursue an in- depth review level but the Risk Gap Analysis plays an important role.

Because of the Fukushima aftermath the GDA is exposed to changes. The presentation posed also interim results and conclusions of AP 1000 and UK EPR assessments. Because of the Fukushima accident the results of the GDAs are exposed to changes. Potential needs for changes in the GDA are foreseen.

The GDA resembles a kind of reactor type specific pre-licensing. However before start of nuclear safety related construction, the licensee shall ensure that the existing PRA is representative and sufficient insights into the vulnerabilities and strengths of the unit are presented.

Before delivery of the mechanical, electrical and I&C systems, structures and components to the site, the licensee shall provide updated PRA including all technical assessment findings.

- Before fuel into the site: Updated PRA including as-built and as operated information;
- Before power rise: Full scope risk monitor. The paper also noted that UK-ONR (Office for Nuclear Regulation) has developed a technical Assessment Guide on PRA that has proven valuable on their reviews. This Guide was developed using available ASME standards and IAEA TECDOC 1511.

<u>The second presentation</u> (“Lessons Learned from IRSN Review of Flamanville 3 Level 1 PSA,” G. Georgescu, IRSN-France) raised several important findings:

- The design phase PRA was based on partial design information;
- Updated design information documentation is necessary for better traceability;
- CCF: the assumptions of full diversification of redundant components should be better justified;
- Loss of ultimate heat sink: independence between main heat sink and secondary heat sink should be better justified;
- I&C modeling: Better justification of FMEA (Failure Mode and Effect Analysis) for different sub-systems and components;
- HRA: Correct identification and justification of dependencies;
- Sump clogging: Should be modeled in PRA;
- Spent fuel pool: the assumptions used in quantification of recovery actions should be traceable and better documented;

- For the decision making: Impact of incomplete information should be assessed and iterative approach should be applied.

STUK had similar conclusions in its regulatory review of OL3 PRA and this was noted in the discussion.

The third presentation ("Role of PRA in New NPP Projects in Finland," Ari Julin, STUK-Finland) gave a generic survey of the risk informed licensing process and insights into the plant changes performed in the OL3 EPR design based on risk information. As set forth in the legislation, a mandatory pre-requisite for a Construction license is acceptable with full-scope Design Phase PRA and few risk-informed applications and for an Operating Licensee Application an acceptable full-scope PRA and several risk-informed applications. The presentation highlighted some key point in the PRA reviews:

- Staff instead of contractors perform the PRA reviews;
- Importance of checking the PRA results as design evolves;
- Aim to have balanced design;
- The quantitative design objective includes also spent fuel pool.

The last presentation ("Introduction of PSA Team Works in CNPE," Zhao Bo, CNPE-China) gave a general survey of NPP projects, documents and standards used and PRA techniques. PRA team was introduced too.

- In China, there are 28 new build projects in progress;
- IAEA , ASME, and NUREG documents, reports and standards are used to support PRA projects;
- In modeling small event trees and large fault tree are applied;
- PRA data is generic;
- Post-Fukushima conclusion is that more attention has to be paid to external events. So far PRAs include only internal initiators.

CDFs of analyzed NPPs are in range of $5.0x10^{-6}$/y ~ $2.0x10^{-5}$/y including power and shut down operations.

(4) Summary of Breakout Session 2 (by G. Georgescu)

This session involved a summary presentation of the answers on the survey on "PSA in the frame of Design and Commissioning of New NPPs, the presentation of the summary of the technical presentations of the day and general discussions.

The questionnaire was answered by 16 organizations, referring to the EPR, AP1000, ABWR, APWR, ESBWR new designs as well as to advanced small modular reactors (iPWR, HTGRs).

The topics of most interest for the workshop mentioned by the respondents were:

- 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 NPPs;
- Reliability analysis of passive systems.

Regarding the regulatory role of PSA the answers show that in general, the Level 1 and Level 2 PSA are considered essential or mandatory for construction and operation of new reactors.

In general, the design stage PSA is a full scope level 1 and level 2 (internal events, internal and external events hazards) for power and shutdown states. The spent fuel pool is not always included. The design PSA is not always site specific. The computerized PSA model is not always available to the regulatory or TSO (Technical Safety Organizations). Some regulatory organizations develop own PSA models.

The main roles of the PSA during the development of the plant design identified by the respondents are:

- Safety demonstration;
- Supporting the choice of design options;
- Well-balanced safety concept;
- Defense in depth assessment /multiple failures conditions;
- Appreciation of the improved safety level compared to existing plants.

The main fields were risk-informed applications are performed are:

- Technical specifications;
- Safety classification of SSCs;
- In-service testing;
- Online preventive maintenance;
- Emergency operation procedures;
- In-service inspection;
- Operator training program / simulator training.

In general, the risk-informed applications are not required by safety authority. Only 4 countries have legislation in this respect.

The respondents recognized that in general the design PSA has important uncertainties. The data uncertainties are analyzed quantitatively using uncertainties propagation methods. For the others non-quantifiable uncertainties, sensitivity analyses are generally performed. Always the limitations of the PSA and main assumption are indicated and discussed. Conservative safety margins are in general considered in the design PSA. The design PSA is developed in general by using international guidelines are used

IAEA, NUREG, ASME, etc. In some cases, country specific guidelines are also used. No specific guidelines for new reactors were mentioned in the answers.

The international groups dealing with that PSA for new reactors mentioned by the respondents are: MDEP (Multinational Design Evaluation Program) (EPR, AP1000), EPR Family Group, WGRISK, WANO, and CANDU Senior Regulator Groups.

The summary of the answers to the questions on PSA Level 1 technical aspect are the followings:

- Initiating events list. The initiating events list is mostly developed based on similar reactors PSA and on generic lists. Some design specific analyses are also performed (system analyses/FMEAs and Master Logic Diagram). In general, few new initiating events specific for the new design were identified;
- PSA supporting studies. In general, specific support studies are performed. Also the Safety Report analysis / design basis reports are used as well as studies of similar reactor PSAs;
- Reliability data. Various sources of data (NUREG/CR-6928, T-Book, EIReDA, NUREG/CR-5497, IAEA-TECDOC-478, IEEE-500, ZEDB, etc.) were mentioned. For evolutionary / limited experience components, ancient components data, supplier's information, supporting reliability evaluations or expert judgment are used;
- New / evolutionary design features modeling. In general these features are reflected by safety improvement. Some new initiating events (like spuriously I&C) were also identified;
- Availability of Tech-Specs and preventive maintenance procedures. In general Tech. Spec. and preventative maintenance procedures are not available, the PSA being based on simplified information;
- HRA. In general, generation 1 methods are used for HRA (THERP (Technique for Human Error Prediction), ASEP (Accident Sequence Evaluation Program)). In some cases, the using of generation 2 methods is foreseen in the future. The accident procedures are not available for the actual design PSA development the respondents identified the need of detailed accident procedures for the next stage PSA. Simulators are not available; in the future the using of simulators is in general expected;
- External hazards. In general, screening analyses were performed. The answers show that the lists of hazards to analyze are different for different projects. The future possible hazards evolution, generally, is not taken into account. Only few external hazards PSA are available, the hazards being generally treated with other methods (bounding analyses, simple quantification). In general, seismic margins assessments are available at the design stage. Full seismic PSA is expected later (requested by some countries);
- Internal hazards. Internal fire and internal flooding are generally modeled in the design PSA. NUREG/CR-6850 is used generally for fire PSA. Heavy load drops were assessed by one project.

The summary of the answers to the questions on PSA Level 2 technical aspect are the followings:

- Generally a Level 2 PSA is available. The Level 2 is integrated with Level 1 PSA, but generally does not include the spent fuel pool. Some internal hazards are included in the Level 2 PSA;
- Severe accident progression support studies. In general, specific support severe accident progression studies are performed (MAAP, Crystal Ball, MELCOR, ASTEC, GOTHIC, TEXAS-V, LS-DYNA, and FLOW-3D). Very limited experiments were performed;
- New severe accident reactor features modeling. In general, the new severe accident reactor features are considered in the Level 2 PSA.

The summary of the answers to the questions on Consequences analysis /PSA Level 3 technical aspect are the followings:

- Several Level 3 PSA were performed for reactor at the design stage;
- Sites specific aspects: In general bounding assumption are used;
- Emergency actions. In general bounding assumption is used, i.e. the emergency actions are not typically considered in the design stage PSA.

The discussions pointed-out the following important aspects:

- The need for phenomenological code developments to address design specific issues (e.g., potential thermal radiation to the containment);
- Challenges of using long running code models in the PSA;
- GDA represents practices used in UK and USA but not necessarily in other countries;
- Some regulators and TSO may not have a direct access to the PSA models; this is a challenge;
- UK ONR highlighted the value of the using of a generic Level 3 PSA in early stages of the design cycle;
- IAEA sees Finland's use of PSA in construction process is very efficient and effective;
- One participant asked if the Level 3 goals should be used to demonstrate the safety of new reactors. The group has widely varying viewpoints.

2.3 Day 3 (22 June)

(1) Summary of Technical Session 7 (by G. Georgescu)

The first presentation ("Probabilistic Modeling of Passive Design Features," F. Sassen, Westinghouse-Germany): The current PSA-guidelines in Germany can in some aspects only be applied to LWRs with active safety systems. The application of the relevant PSA-rules on modern future reactor types (e.g. high temperature reactors) demands an interpretation of these rules. The PSA for advanced light water reactors with passive safety features has increased demands on data for the event tree and fault tree modeling. The PSA-model for a "passive" safety system not only needs to model the few active powered components of the system, but all passive components must be consider in fault tree, if they contribute to the same extent to unavailability of the system. The new challenge for the modeling of passive systems thus is that the failure rates for both the active components and passive components must be defined. For the passive systems for example the failure of pipes and tanks at the start of the system function should be considered as well as the wrong standby system alignment and the damage mechanisms that cause moving parts, which must keep their original position, change. As an example, the model of the AP 1000 Core Makeup Tank system was presented. The paper addresses potential failure of hardware components but does not addresses potential contributions due to uncertainty in phenomena arguing that the design has large safety margins.

The second part of the presentation referred to the requirements for PSA modeling of other reactor types than light water reactors. Valid PSA guidelines apply to a large extent to plants which are analyzed "as-built-as-operated". Since for non-LWR plants usually even during the conception phase a PSA is required, a corresponded data base is missing. This leads to demanded scopes for uncertainty and sensitivity analysis which may not be fulfilled with a reasonable effort. Because of the utilization of new materials and due to incomplete knowledge on phenomena partially success criteria can only be established with some uncertainty, as is known from Level 2 PSAs. Besides the question whether such systems are fit for licensing from the deterministic point of view, this causes increased demands for the PSA.

<u>The second presentation</u> (“Uncertainty Analysis Methods for Estimation of Reliability of Passive System of VHTR,” S. J. HAN, KAERI-Korea): An estimation of reliability of passive system for the VHTR PSA is under development in Korea. The essential approach of this estimation is to measure the uncertainty of the system performance under a specific accident condition. The uncertainty propagation approach according to the simulation of phenomenological models (computer codes) is adopted as a typical method to estimate the uncertainty for this purpose. The paper introduced the uncertainty propagation and discussed the related issues focusing on the propagation object and its surrogates. To achieve a sufficient level of depth of uncertainty analyses results, the applicability of the propagation should be carefully reviewed. For an example study, Latin-hypercube sampling (LHS) method as a direct propagation was tested for a specific accident sequence of VHTR. The presentation discussed the obtained insights (benefit and weakness) to apply an estimation of reliability of passive system.

<u>The third presentation</u> (“Problems Facing the Use of Passive Safety Systems,” L. Burgazzi, ENEA-Italy) referred to the current state of the art in the reliability of passive systems and identify the critical issues which need further consideration.

The paper stress that, 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 strong need for the development and demonstration of consistent methodologies and approaches for evaluating their reliability. Recently, the development of procedures suitable for establishing the performance of a passive system has been proposed. In order to get confidence in the achieved results, it is necessary to reduce the level of uncertainty pertaining to the passive system behavior, and in particular the phenomenological uncertainty. The determination of the dependencies among the relevant parameters adopted to analyze the system reliability is also essential. The study of the dynamical aspects of the system performance, because the inherent dynamic behavior of the system should to be characterized, is another important aspect. It is also necessary to compare the passive systems against the active systems, in order to evaluate the economical competitiveness, while assuring the same level of safety.

Summary of discussions: Many new reactor designs use passive safety systems. On the other hand, in the available design PSA 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. Some group members expressed the opinion that the scenario dependent situations which can lead to a combination of conditions for which the passive system function cannot be performed should be identified and modeled explicitly in the PSA. Some other group members expressed the opinion that parametric models can be used. The modeling of passive systems in the PSA raised also the question of the impact on other PSA aspects. For example, the functioning of the passive systems for long term accident scenarios should carefully be 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 group highlighted also the fact that the existing thermal hydraulic codes may not be completely applicable for the analysis of the passive systems behaviour in the context of developing design PSA support studies. The international activities performed up to now on the passive systems reliability didn't treat explicitly the modeling of the passive systems in the PSA. For the group, recognizing that have been no 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 modeling in the PSA is an open issue which needs more efforts.

(2) Summary of Technical Session 8 (by T. Leahy)

This session consisted of four presentations focused on relevant activities by national and international organizations, and on the topic of PSA technical adequacy for advanced light water reactors. Because the presentations were quite diverse in their scope and focus, broad, cross-cutting conclusions were not apparent. Nonetheless, presenters made a number of important points, and several notable results and conclusions arose from the presentations and subsequent discussion during the session.

The first presentation ("NRC Activities Concerning PSA for New and Advanced Reactors," N. Siu, NRC-USA) outlined the status of new and advanced reactor design reviews, relevant guidance regarding the use of PSA in these reviews, and certain related NRC research activities that are in progress. It was noted that the licensing process explicitly calls for the use of risk insights, and for comparison of risk with established safety goals and with existing plants. In this regard, it was specifically noted that the NRC has determined that existing safety goals and risk guidance are sufficient for new plants. In the longer term, NRC will be developing a new "risk informed framework" for advanced reactors. Significant NRC research programs are being conducted in the areas of High Temperature Gas Reactor issues, Digital Systems, and Human Factors Analysis for new and advanced Reactors.

The second presentation ("ASME/ANS Standards for ALWR and Advanced Non-LWR PRA: Status and Challenges," N. Siu, NRC-USA) provided an overview of the status of developing guidance and standards for advanced reactor PSAs. It was noted that key aspects of useful standards cover the risk assessment application process, PSA technical requirements, configuration control, and peer review to assure technical quality. Echoing a major theme of the entire workshop, early use of risk insights in design, construction, and "pre-operation" is desirable. Several approaches to develop the ALWR standard are under consideration, but no consensus has yet emerged. The non-LWR standard is further developed, and a draft standard has been developed. Review and comment is in progress. Both technical and non-technical challenges were discussed.

IAEA activities were discussed in the third presentation ("Insights from Recent Activities on PSA Being Pursued by the IAEA," I. Kuzmina, IAEA). This presentation provided updates on several recent or ongoing IAEA activities including further work in the definition and application of the principle of "defense in depth." The use of PSA in assessing defense in depth was briefly described, and the importance of PSA quality and independent peer review was emphasized. The availability of IAEA peer review services was also noted. This presentation also reported on the IAEA's April 2011 Technical Meeting on Safety Goals in Application to Nuclear Installations. Results, conclusions, and agreements established during that meeting will be outlined in a meeting report that is currently under preparation. Finally, it was noted that the IAEA is developing guidance on the use of Integrated Risk Informed Decision Making (IRIDM).

The final presentation of the session ("Assuring PSA Technical Adequacy for New Advanced Light Water Reactor Designs," A. Maioli, Westinghouse-USA) described some of the challenges of using PSA to design and license new technologies, and emphasized the importance of PSA technical adequacy in light of those challenges. It was recommended that in the design stage, PSA is best used to identify vulnerabilities and to help identify opportunities for possible design improvements. In the construction stage, PSA is used for equipment procurement, development of technical specifications, and training of plant personnel. In pre-operational and early operational stages, it was recommended that the PSA be used to develop regulatory oversight programs and to validate design assumptions. When used for these important purposes, PSA quality and technical adequacy are essential. Practical difficulties of finding knowledgeable experts to conduct peer review for novel technologies were noted.

Perhaps the main unifying thread of Session 6 was the idea that PSA quality and technical adequacy are critically important when the PSA is to be used in reactor design and licensing. Because new and advanced 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. Collectively, presenters in this Session made the point that appropriately focused research in selected areas, careful thinking about both qualitative and quantitative safety goals, well conceived PSA standards, and independent peer are important allow PSA to fulfill its potential in supporting design and licensing.

(3) Summary of Technical Session 9 (by L. Burgazzi)

This session included 4 papers which addressed different areas:

- Lessons learnt from New and Advanced Reactors in Russia, presented by V. Morosov (Atomenergoproekt, Russia),
- Risk-informed, Performance based Safety-Security Interface, presented Farouk El-Tawila (FANR, UAE),

- Automatic fault tree generation in the EPR PSA project, presented by Natalie Villatte (EDF, France),
- Investigations of inter-system common cause failures, presented by Philippe Nonclercq (EDF, France).

The first paper focused on new evolutionary VVERs which feature new systems, basically passive systems such as hydro-accumulators, passive residual heat removal system and fast injection boron system to achieve redundancy and diversity of main active systems. Preliminary analysis revealed a decrease in both CDF and LERF (Large Early Release Frequency) values as compared with current operating VVERs.

Some of the open issues highlighted in the presentation concern (i) adequacy of regulatory approach, (ii) probabilistic safety targets definition, (iii) PSA scope, (vi) uncertainties induced by new designs, (v) consistency and need of interaction between design and PSA teams leading to an iterative PSA process, (vi) longer mission time, (vii) human reliability during longer time, (viii) safe end states, (ix) homogeneity and consistency of data coming from different sources as there are no specific data for new designs. The paper mentioned also some considerations following the Fukushima accident to be taken into account in PSA such as combined external events, multi-units scenarios and extension of SAMG.

During the discussion, contribution of passive systems to the risk reduction was confirmed. Also the importance of communication between PSA and design teams was noted.

The second paper addressed the safety-security interface which refers to the actual or potential interactions that may adversely affect security due to design or operation activities (e.g., maintenance) or vice versa. This interaction is recommended to be included in a PSA framework at the early stage during site selection and design in order to optimize it in compliance with safety and security requirements. Means to enhance synergy of safety, security and emergency preparedness are recommended in order to assure protection of the workers, public and environment. A risk-informed security analysis framework was proposed aiming at broadening security-related event scenarios, increasing the number of mitigation alternatives, adding greater realism to the construction of the sequences of events and considering similar elements as those addressed in the safety assessment.

During the discussion of this paper concern was raised on whether PSA should be a confidential security tool not open to every analyst and hence limiting the safety culture. One proposal to overcome this difficulty is to foresee two versions, a restricted one including the safety-security interface and the other one limited to safety.

The third paper presented a tool (KB3) developed by EDF for automatic fault tree generation to be incorporated within the main tool (RiskSpectrum). This tool which is used in conjunction with a knowledge basis (electrical, thermal-hydraulics and I&C) is particularly suitable to overcome some difficulties such as complexity of the systems, different level of experience of PSA practitioners and dispersion of PSA teams in different places. The tool was stated to be flexible, effective and beneficial to the safety analysts as it provides diagrams, homogeneity between the fault trees, and allows better traceability and control of modeling. However, some improvements are needed in terms of fault tree readability, export to RiskSpectrum and handling of the tools.

During the discussion, concern was raised regarding the loss of learning and knowledge which might be induced by the use of automatic tools as the act at model construction is important in helping analysts understand how a system works and how it can fail.

The fourth paper presented investigations of inter-system CCF through three studies on EDF 900 MWe pumps and motor operated valves, and EPR 10 kV breakers. These investigations used a methodology based on NUREG/CR-5485 and the related French PSA fundamental safety rule. The methodology includes 4 steps: (i) analysis of similarities in design, operation, environment and cause of failure, (ii) search for CCF in the operating experience feedback, (iii) qualitative and quantitative analysis of the possible impact of multiple failures with Risk Achievement Worth Cumulative Factor (RAWC), (iv) modeling of new CCF groups in the model. The results for EDF 900 MWe pumps and motor operated valves, and EPR 10 kV breakers showed a small risk impact of inter-system CCF.

(4) Summary of Breakout Session 3 (by A. Amri)

During this session, the participants were invited to ask questions or to provide any comment related to the papers presented during the day. The main questions, comments and insights are highlighted below:

- The reliability of passive systems may be impaired by external hazards (e.g., external high temperature); hence, one participant recommended to include the effect of external hazards on passive systems;
- As for the failure of passive systems by physical phenomena change or degradation, one participant expressed whether we may be able to include such a failure in a fault tree. Some of these failures may be addressed by design; for others, there is a need for Thermal-hydraulic community and PSA community to work together in order to get better understanding and to develop an appropriate method to address them;
- Another participant wondered whether we need a specific PSA and specific tools to address long time scenarios including later recovery actions;
- One participant underscored that there is no consensus to address the reliability of passive systems which may fail through physical phenomena degradation (category B); he expressed his personal opinion on how to address and for which important uncertainties remain by considering a fault tree and by assuming the failure of the system/ sub-system impacted by this degradation (e.g., effect of non-condensable gases = assume failure of venting, degradation of heat transfer coefficient = assume heat exchanger plugging). Others did not agree to say in the report that the issue is difficult and there is no consensus, so we cannot do anything as we can at least try to have some order of magnitude. They see in the method of uncertainty propagation method a promising one which should be further explored, e.g., through an international effort to address major issues in this method (robustness of system codes, uncertainty methods, etc.);

- A participant raised the concern whether too much standardization, harmonization, consensus model might prevent us to learn more as being outside the standard mode pushes us to investigate more and to learn more;
- Several comments were provided regarding data, e.g., use generic vs. specific data and influence on CDF; use of data from different sources and the induced uncertainties;
- In general, the participants found the workshop very interesting and useful as it had shown that studies and data exist, and that they may use its outcomes in their guidelines.

3. CONCLUSIONS

A wide spectrum of presentations and discussions on the PSA for new and advanced reactors were made during the three days workshop of June 20-22 2011, including current practices among member countries (e.g., analysis methods, tools and relevant data), efforts to improve technical problems, and potential challenges to the use of the risk-informed decision makings in design improvements of new and advanced reactors and related potential regulation processes, and technical practices according to the basis of these challenges.

The workshop participants also paid particular attentions to the necessity of more guidance to ensure the quality of the PSA for new and advanced reactors (like ASME/ANS LWRs and non-LWR standard) and peer-review process. The capability of the current PSA methods and applications in the frame of new and advanced reactors PSAs, as well as the potential areas for future improvements were also addressed by the participants, whose major concerns included passive safety system reliability, human reliability, and digital system reliability, new and advanced reactors-specific risk assessment methodology and relevant computational tool, and safety-security interface, etc).

The workshop played a great role in sharing the current state-of-the art on the PSA of new and advanced reactors, and points of interest among member countries. Main findings obtained from the workshop are as follows:

- For new reactors, the role of PSA is more important and more formalized comparing with exiting reactors. The PSAs are being used in the design of new reactors for purposes such as balance between accident prevention and mitigation features of the design, demonstration of safety, identification of design vulnerabilities and improvements, and comparison with the risk of existing plants, etc.
- For advanced reactors, the role of PSA is also being regarded as an essential tool for safety improvement and comparison of reactor designs at the conceptual or preliminary stages, but their practical use is at the early stage to gain the insights of risk into these reactors. This is due that the use of PSA for advanced reactors is mainly focused on indentifying the PSA issues for designer and regulatory bodies, and developing the relevant methodologies and tools. Differently from the PSA for the existing and new reactors, moreover, there seems to not currently reach any general consensus on how to take into account several technical issues discussed in the workshop and on what scope and level of depth to take into account them in implementing the PSA for advanced reactors.

General conclusions of this workshop are as follows:

- 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;
- Better guidance and peer review process on the PSA of new and advanced reactors was identified by both industry and regulatory as an important aspect;
- 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 workshop.

All the findings and insights obtained in the workshop will be fed back into the respective task report on "PSA for new reactors and PSA for advanced reactors."

Appendix 1: Daily Program of the Workshop

20 June	21 June	22 June
9:00 – 10:40 (N.SIU)	**9:00 – 10:40 (JM. LANORE)**	**9:00 – 10:40 (G. GEORGESCU)**
Introductory remarks	Modeling of the AP1000® Digital Instrumentation and Control Systems, David S. Teolis, Westinghouse, USA	Probabilistic modeling of passive features, F. Sassen, Westinghouse, Germany
Level-1 PSA to support the design of the KALIMER-600 Sodium Cooled Fast Reactor, Sang Hoon HAN, KAERI, Korea		
Study on preliminary level-1 PSA for Japan sodium-cooled fast reactor, K. Kurisaka, JAEA, Japan	I&C modelling in the IRSN EPR level 1 PSA, J. Delache, IRSN, France	Comparison of two uncertainty analysis methods for the estimation of reliability of passive system of VHTR, Seok Jung HAN, KAERI, Korea
Level 1 probabilistic safety assessment to support the design of the CEA 2400MWth gas-cooled fast reactor, F. Bertrand, CEA, France	Fire PSA Development in CGNPC, X. Jun, CGNPC, China	Problems Facing the Use of Passive Safety Systems, L. Burgazzi, ENEA, Italy
Overview of VHTR's PSA approach in Korea, Seok Jung HAN, KAERI, Korea	Analysis of Design-Reliability Assurance Program in ACP600 Application, Huang Zhichao, CNPE, China	
11:00 – 12:40 (A. BAREITH)	**11:00 – 12:40 (R.J. LUTZ)**	**11:00 – 12:40 (T. LEAHY)**
Generation IV Integrated Safety Assessment Methodology, Timothy Leahy, INL, USA	In-Vessel Retention Modeling Capabilities in MAAP5, C.Y. Paik, Westinghouse, USA	NRC Activities Concerning PSA for New and Advanced Reactors, N. Siu, NRC, USA
ASAMPSA2 project : appliance of LWR PSA2 methodology to GEN IV reactors, H. Bonneville, IRSN, France	Development of level 2 PSA Methodology for Sodium-Cooled Fast Reactors, T. Suzuki, JAEA, Japan	ASME/ANS Standards for ALWR and Advanced Non-LWR PRA: Status and Some Challenges, N. Siu, NRC, USA
Reliability analysis of 2400 MWth gas-cooled fast reactor natural circulation decay heat removal system, M. Marques, CEA, France	PSA Level 2 as Element of an Integral Safety Assessment before Plant Commissioning, H. Löffler, GRS, Germany	Insights from Recent Activities on PSA Being Pursued by the IAEA, Irina Kuzmina, IAEA
Level-1 PSA for internal events for TAPS3&4 - A Challenge, Rajee Guptan, Nuclear Power Corporation of India Ltd, India	MAAP5 Modeling Capabilities for Initial Plant Transients and Shutdown States, and Application to Shutdown PSA and Full Scope SAMG Covering All Plant States for Operating and New Plants, Chan Y. Paik, Fauske and Associates, Westinghouse, Belgium	Assuring PSA Technical Adequacy for New Advanced Light Water Reactor Designs, R.J. Lutz, Westinghouse, USA
14:00 – 15:40 (J.J. TONG)	**11:00 – 12:40 (R. VIROLAINEN)**	**11:00 – 12:40 (L. BURGAZZI)**
Applying Risk Insights in USNRC Reviews of Integral Pressurized Water Reactor Designs, M. Caruso, NRC, USA	Regulatory Assessment of the PSAs for the UK EPR and AP1000 Reactors in the UK, A.G. Cobo, NII, UK	Lessons Learnt from PSAs for New and Advanced Reactors in Russia, V. Morozov, Atomenergoproekt, Russia
Development of PSA Audit Guideline and Regulatory Model for SMART, Namchul Cho, KINS, Korea	Lessons learned form IRSN review of Flamanville 3 Level 1 PSA, G. Georgescu, France	Safety-Security Interface, Bruce Mrowca, Information System Laboratories, USA
Use of PSA in the Development of SMRs, Andrea Maioli, Westinghouse, USA	The Role of PRA in New NPP Projects in Finland, Ari Julin, STUK, Finland	Automatic fault tree generation in the EPR PSA project, P. Nonclercq, EDF, France
Achievement of the level 1 PSA in support to the CEA 2400 MWth Gas-cooled Fast Reactor, M. Balmain, EDF, France	Introduction of PSA team works in CNPE, Zhao Bo, CNPE, China	Existence and impact on safety of inter-system common cause failures : a method, P. Nonclercq, EDF, France

16:00 – 17:40 (K. AHN)	16:00 – 17:40 (G.GEORGESCU)	16:00 – 17:40 (A. AMRI)
Breakout session (*) PSA for advanced reactors - Session summary by each session chair - CAPS on PSA for advanced reactors: summary of questionnaires and answers, K. Ahn, KAERI, Korea	Breakout session (*) PSA for new designs - Session summary by each session chair - CAPS on PSA for new reactors: summary of questionnaires and answers, G. Georgescu, IRSN, France	Breakout session (*) Common discussion - Session summary by each session chair - **Closing remarks**.

(*) The breakout sessions are intended to provide a framework to present summaries of the day presentations and to discuss the day open issues and conclusions. A presentation of the summaries of the task questionnaire answer is foreseen during the first two breakout sessions (advance reactors on Monday and new reactors on Thursday. The last breakout session is also dedicated to preparatory discussion for the proceedings development

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