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Level 1 Probabilistic Safety Assessment Practices for Nuclear Power Plants with CANDU-Type Reactors

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LEVEL 1 PROBABILISTIC SAFETY ASSESSMENT PRACTICES FOR NUCLEAR POWER PLANTS WITH CANDU-TYPE REACTORS

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LEVEL 1 PROBABILISTIC SAFETY ASSESSMENT PRACTICES FOR NUCLEAR POWER PLANTS WITH CANDU-TYPE REACTORS

INTERNATIONAL ATOMIC ENERGY AGENCY
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FOREWORD

Every year the IAEA hosts the meeting of the Canada Deuterium Uranium (CANDU) Senior Regulators Group, which promotes cooperation and information exchange between the regulatory bodies of the Member States with CANDU reactors: Argentina, Canada, China, India, the Republic of Korea, Pakistan and Romania. The CANDU Senior Regulators Group identified probabilistic safety assessments (PSAs) as a topic of interest and as an area for harmonization and information exchange between CANDU regulatory agencies, utilities and designers, with the objective of minimizing the differences among PSAs.

In 2009, the CANDU PSA Working Group was formed to facilitate the harmonization, and its inaugural meeting was held in 2010. The Working Group defined and implemented numerous tasks covering harmonization efforts for different aspects of PSAs, regulatory approaches and their application. The Working Group examined the harmonization of risk metrics for CANDU reactors, compared approaches to the modelling of various initiating events, discussed available data for PSAs, and the modelling challenges of different modes of operation and support from deterministic analysis. The scope of the Working Group's activities covers both Level 1 and Level 2 PSAs. The Working Group has completed its tasks for Level 1 PSA and is currently conducting tasks relating to Level 2 PSA.

This publication summarizes the discussions of Member States with CANDU reactors at a series of technical meetings. The meetings addressed the future harmonization of the CANDU regulatory framework, and the scope, methodologies and tools of Level 1 PSA, and were an opportunity to share actions taken following the accident at the Fukushima Daiichi nuclear power plant.

The IAEA wishes to thank all participants and their Member States for their valuable contributions to this publication. The IAEA officers responsible for this publication were G. Macsuga and S. Poghosyan of the Division of Nuclear Installation Safety.

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1. INTRODUCTION

1.1. BACKGROUND

The status of probabilistic safety assessment varies across Member States operating CANDU-type reactors. In 2009, the CANDU PSA Working Group (CPWG) was formed to provide the mechanism for facilitating the harmonization work in PSA. The CANDU-type reactors are summarized in Table 1 and the status of PSAs of CANDU-type reactors is presented in Appendix I.

TABLE 1. SUMMARY OF CANDU-TYPE REACTORS

MEMBER STATE	OPERATING CANDU/CANDU-TYPE REACTORS
Argentina	• Embalse (PHWR, CANDU-6)
Canada	• Pickering Units 1, 4 and 5–8 (PHWR, CANDU) • Bruce A Units 1–4 (PHWR, CANDU) • Bruce B Units 5–8 (PHWR, CANDU) • Darlington Units 1–4 (PHWR, CANDU) • Point Lepreau (PHWR, CANDU-6)
China	• Qinshan III-1 (PHWR, CANDU-6) • Qinshan III-2 (PHWR, CANDU-6)
India	• Rajasthan 1 and 2 (PHWR, CANDU) • Kaiga 1 and 2 NPP (PHWR, CANDU-type) • Kaiga 3 and 4 NPP (PHWR, CANDU-type) • Kakrapar 1 and 2 (PHWR, CANDU-type) • Madras 1 and 2 (MAPS) (PHWR, CANDU-type) • Narora 1 and 2 (PHWR, CANDU-type) • Rajasthan 3 and 4 (PHWR, CANDU-type) • Rajasthan 5 and 6 (PHWR, CANDU-type) • Tarapur 3 and 4 (PHWR, CANDU-type)
Republic of Korea	• Wolsong 1 (PHWR, CANDU-6) • Wolsong 2 (PHWR, CANDU-6) • Wolsong 3 (PHWR, CANDU-6) • Wolsong 4 (PHWR, CANDU-6)
Pakistan	• Karachi 1 (PHWR, CANDU)
Romania	• Cernavoda 1 (PHWR, CANDU-6) • Cernavoda 2 (PHWR, CANDU-6)

The meetings of the CPWG are organized annually with the purpose of enabling cooperation and information exchange between the PSA specialists of the seven Member States that operate CANDU-type reactors. The objectives are formulated in the Terms of Reference of the CPWG and represent the following items:

- to support regulatory authorities, utilities, and designers in their area of PSA;
- to harmonize regulatory approaches on the use of PSA;
- to exchange information on national practices on PSA (regulatory framework, PSA models and tools, case studies, risk informed decision making application for regulatory purposes, regulatory review on PSA);
- to make recommendations to the CSRG;
- to produce technical reports on selected topics.

Since 2010, the CPWG has met annually to discuss the PSA practices and progress on specific tasks included in the work programme (the list of CPWG tasks is provided in Appendix II).

Thus, the overall scope of the CPWG project includes comparison of CANDU-type PSA practices amongst CPWG Member States to identify the differences and commonalities, understanding and rationalizing differences and harmonization of CANDU-type PSA practices, and providing specific CANDU-type input and clarifications. The participants of the CPWG are the relevant regulatory authorities in countries operating CANDU-type reactors:

- Argentina: Nuclear Regulatory Authority (ARN);
- Canada: Canadian Nuclear Safety Commission (CNSC);
- China: National Nuclear Safety Administration (NNSA);
- India: Atomic Energy Regulatory Board (AERB);
- Republic of Korea: Korea Institute of Nuclear Safety (KINS);
- Pakistan: Pakistan Nuclear Regulatory Authority (PNRA);
- Romania: National Commission for Nuclear Activities Control (CNCAN).

In addition, CPWG activities are also supported by several utilities and design organizations:

- CANDU Owners Group (COG);
- S.N. Nuclearelectrica S.A (SNN) Cernavoda NPP;
- Korea Hydro & Nuclear Power (KHNP);
- Nuclear Power Corporation of India Ltd. (NPCIL);
- Pakistan Atomic Energy Commission (PAEC);
- Third Qinshan Nuclear Power Co., Ltd. (TQNPC);
- Candu Energy Inc.

According to the working method of the CPWG information was exchanged and experiences shared during technical meetings and through emails, as well as through video- and tele-conferences. The main outputs are position papers on relevant tasks which are developed based on the responses to the questionnaires prepared and distributed among CPWG members. The information presented in this publication represents a summary of the position papers and conclusions related to level 1 PSA for CANDU-type reactors.

Furthermore, in the aftermath of the Fukushima Daiichi accident, it was noted that there were some issues in the PSA methodology that needed more emphasis or further development. In this context, the insights gained from the exchange of technical information among various Member States were also considered in CPWG work and taken into account in the development of this publication.

1.2. OBJECTIVE

The objective of this publication is to provide a summary of the comparison analysis performed in the Member States operating CANDU-type reactors with regard to PSA practices, in order to exchange information on national practices and to support harmonization in:

- regulatory review of the PSA to the extent possible taking into account the specific regulatory framework;
- PSA methodologies and tools;
- PSA scope (internal events, external events, and combinations thereof).

In addition, this publication includes information on actions taken in the light of lessons from the Fukushima Daiichi accident, discusses PSA applications, challenges for CANDU-type PSAs and considerations on use of relevant IAEA publications for the level 1 PSA for CANDU-type reactors.

This publication is intended for use by regulatory bodies, operators and designers in Member States operating CANDU-type reactors.

1.3. SCOPE

The publication addresses level 1 PSA for internal initiating events, internal and external hazards, covering various plant operational states (e.g. full power, low power and shutdown). It covers PSA practices, as well as regulatory review of PSA for CANDU-type reactors. Some aspects of full scope PSA are addressed to give a comprehensive overview of the national practices.

1.4. STRUCTURE

Section 2 provides information of regulatory requirements on PSAs and briefly discusses the regulatory framework of PSA review. Section 3 outlines the current practices of PSA development in CPWG Member States, focusing on the CANDU-type PSA issues. Section 4 discusses the national standards on PSA and tools used to develop level 1 PSA in CPWG Member States. Section 5 provides information on the use of IAEA publications in development, application and review of PSAs for CANDU-type reactors. Section 6 discusses in detail the implications of the Fukushima Daiichi accident on CANDU-type PSAs. Information on the current status of CANDU-type PSAs, CPWG tasks, and position papers produced by CPWG is provided in the Appendices and Annexes.

2. REGULATORY CONSIDERATIONS IN THE AREA OF PSA

The regulatory requirements on PSA and PSA review processes of different countries vary based on their legislation process, public involvement, economic conditions and other factors. Overview of regulatory requirements in the area of PSA and brief information on PSA review processes in CPWG Member States are provided below.

2.1. REGULATORY REQUIREMENTS ON PSA

A survey was performed among all the countries operating CANDU-type reactors regarding the regulatory requirements in the area of PSA addressing the PSA documentation, PSA scope, acceptance criteria and new requirements established based on lessons learned from Fukushima Daiichi accident. The summary of the survey results is described in Table 2. More detailed information on survey results is provided in Annex I.

In summary, the countries operating CANDU-type reactors have similarities regarding PSA requirements. However, there are also differences existing in some areas, for instance:

- Safety goals of PSA vary among Member States (more information on risk metrics are presented in Section 3.1);
- Different level of feedback to PSA requirements based on lessons learned from the Fukushima Daiichi accident;
- PSA scope is different among Member States, in some cases level 2 PSA is not required or indicated as desirable, whereas in others level 2 PSA is mandatory. Level 3 PSA is generally not required in CPWG Member States (only for new NPPs in the Republic of Korea);
- Some Member States apply different requirements for existing and new power plants, whereas others use the same requirements for both.

TABLE 2. REGULATORY REQUIREMENTS REGARDING PSA

MEMBER STATE	REQUIRED SCOPE OF THE PSA	SAFETY GOALS	FEEDBACK TO PSA REQUIREMENTS FROM FUKUSHIMA DAIICHI ACCIDENT
Argentina	Level: level 1 and level 2 PSA	There are no formally established criteria for core damage frequency (CDF) and large early release frequency (LERF). However, criteria related to dose limit is defined in the form of a graph	Requirement to implement the stress test. This consists of a reassessment of the NPP safety margins assuming the occurrence of a sequential loss of the lines of defence in depth caused by extreme initiating events
Canada	Level: level 1 and level 2 PSA POSs: full power, low power, shutdown IEs: internal IEs, internal and external hazards	The explicit safety goals for existing NPPs are not given in the regulatory documents. Acceptable safety goals are defined in utility governance: • SCDF < 1E-4/yr; • LRF < 1E-5/yr. For new NPPs: • SCDF < 1E-5/yr; • LRF < 1E-6/yr.	The post-Fukushima requirements include consideration of the following aspects in risk assessment: • multiunit station impacts; • radioactive sources (other than reactor core); • hazard combinations.
China	Level: level 1 and level 2 PSA POSs: full power, low power, shutdown IEs: internal IEs	For existing NPPs: • CDF < 1.0E-4/yr; • LERF < 1.0E-5/yr. For new NPPs: • CDF < 1.0E-5/yr; • LERF < 1.0E-6/yr.	The post-Fukushima requirements include: • level 2 PSA; • spent fuel bay PSA.
India	For existing NPPs: Level: level 1 PSA (mandatory), level 2 PSA (desirable) POSs: full power (mandatory) IEs: internal IEs (mandatory) For new NPPs: Level: level 1 PSA and level 2 PSA (mandatory)	For both existing and new NPPs: • CDF < 1E-5/yr; • LERF < 1E-6/yr.	No new specific requirements related to PSA have been issued. However, deterministic reviews such as ‘stress test’, and required modifications, have been introduced for all NPPs (design stage as well as operating). PSA studies with regard to external event, low power and shut down has taken on priority

TABLE 2. REGULATORY REQUIREMENTS REGARDING PSA (cont.)

MEMBER STATE	REQUIRED SCOPE OF THE PSA	SAFETY GOALS	FEEDBACK TO PSA REQUIREMENTS FROM FUKUSHIMA DAIICHI ACCIDENT
Republic of Korea	For existing NPPs: Level: level 1 PSA and level 2 PSA POSs: full power (for all), low power (only for level 1), shutdown (only for level 1) IEs: internal IEs, internal and external hazards For new NPPs at PSAR stage: Level: level 1 PSA and level 2 PSA (both mandatory) POSs: full power IEs: internal IEs, internal and external hazards For new NPPs at FSAR stage: Level: level 1 PSA, level 2 PSA and level 3 PSA (all mandatory) POSs: full power, low power and shutdown IEs: internal IEs, internal and external hazards	In KINS/RS-16.0 (Rev. 2) • CDF < 1.0E-4/yr; • LERF < 1.0E-5/yr. For new NPPs (after Shinkori Unit 3): • CDF< 1.0E-5/yr; LERF < 1.0E-6/yr.	In June 2015, the Nuclear Safety Act was amended to enhance the regulatory framework on severe accidents. The regulatory body in the Republic of Korea revised the rules, regulatory standards and regulatory guides for severe accidents and PSA. The main changes are as follows: • PSA will be submitted by the law, not administratively; • All existing plants' PSA need to be updated and should be submitted until June 2019; • PSA scope which should be submitted is specified on regulatory standard of KINS; • A quantitative target is included in the Notice of NSSC, i.e. the risk to any individual in the vicinity of NPP should not exceed all the risk from other reason. For considering environment, quantitative target quantitative environmental objective is included in notice of NSSC (i.e. the sum of frequencies of all event sequences that can lead to release to the environment of more than 100TBq of Cs-137 should be less than 1.0E-6/yr)
Pakistan	Level: level 1 PSA (mandatory), level 2 PSA (desirable) POSs: full power, low power, shutdown IEs: internal IEs, internal and external hazards	For both existing and new NPPs: • CDF < 1.0E-5/yr; LERF < 1.0E-6/yr.	PNRA required the following tasks from the licensee: • Re-assessment of natural hazards (i.e. seismic, flood, tsunami, harsh environment, wind, tornado, etc.); • Re-assess and re-analyse the design features for longer station blackout duration; • Re-evaluation of the design features provided at nuclear power plants for controlling and removing hydrogen; • Review of emergency operating procedures and severe accident management guidelines; Re-evaluation of off-site emergency preparedness plan including emergency plan implementing procedures.
Romania	Level: level 1 PSA and level 2 PSA (both mandatory) POSs: full power, low power, shutdown IEs: internal IEs, internal and external hazards	For both existing and new NPPs CDF < 1.0E-4/yr.	No new specific requirements have been issued formally by CNCAN regarding PSA, but after the stress tests, CNCAN developed the national action plan based on conclusions and recommendations of stress tests. This plan contained an activity on updating the PSA in order to include events at the spent fuel bay of Cernavoda NPP.

2.2. REGULATORY REVIEW PROCESS OF PSA

A survey was performed among all the countries operating CANDU-type reactors regarding the review of PSA submissions. The summary of the survey results is described in the following Table 3. More detailed information on survey results is provided in Annex I.

TABLE 3. REGULATORY REVIEW OF PSA

MEMBER STATE	REVIEWING ORGANIZATION	LEVEL OF EFFORTS REQUIRED FOR PSA REVIEW	IAEA PSA REVIEW SERVICE CONDUCTED
Argentina	Review is performed mainly by ARN itself. Sometimes TSO is used for review of specific aspects in PSA	2000–3000 person-hours for internal at power events PSA	PSA study of Atucha (1996)
Canada	Review is performed by RB and supported by TSOs. Review is also supported by other Canadian agencies for external events PSA, such as NRCan for the probabilistic seismic hazard analysis, and Environment Canada for high wind hazard characterization	10 000 to 15 000 person-hours for a full scope PSA submission	Not conducted
China	Review is performed by RB in PSAR and FSAR stages. For other conditions, such as updating the existing PSA model, a TSO and/or consultant review is preferred	Approximately 2500 person-hours (15 person-months)	PSA study of Guandong (1989), PSA study of Day Bay (1998), PSA study of Tianwan (2000, follow-ups in 2000, 2002 and 2004) and PSA study of Qinshan (2003)
India	AERB review is performed in two phases. Phase I is review of initial submission by in-house experts and phase II is a detailed review by the expert committee with members from AERB, BARC (TSO), IGCAR, Utilities and independent PSA experts	Approximately 10 000 to 15 000 person-hours	Not conducted
Republic of Korea	PSA review is performed by regulatory body (NSSC) and Korea Institute of Nuclear Safety (KINS)	The person-hours for each regulatory review of each PSA have not been assessed exactly. But it is estimated that 50–2500 person-hours are required for each review process and 2–5 review processes are going on recently	PSA study of Kori (1991), PSA study of Yonggwang (1994) and PSA study of Ulchin (1997)
Pakistan	Pakistan Nuclear Regulatory Authority (PNRA) has its own TSO named Centre for Nuclear Safety. All regulatory reviews are performed by Centre for Nuclear Safety	5000–6000 person-hours for internal at power events PSA	PSA study of Karachi (1999, 2001), PSA study of Chashma (2007, 2009)
Romania	The review of Cernavoda NPP level 1 PSA has been performed by the CNCAN staff with support of international experts	Approximately 4000 hours (5 persons for 5 months)	PSA study of Cernavoda (1990, 1995, 2001, 2003, 2004, 2005)

Observation has shown that PSA reviews are mainly done by regulatory bodies themselves with support of TSO organizations. The comparison demonstrated that regulatory bodies allocate different level of resources for PSA review. It varies from 500 person-hours to 15 000 person-hours to review full scope PSA.

Similarities have been identified in the regulatory review processes. Thus, PSA regulatory review processes of Canada, India and Pakistan consist of the following two phases:

- Phase-I review: Qualitative review (review of format & contents and methodology);
- Phase-II review: Detailed review of each task.

In Argentina, the review process also includes 2 major steps: the preliminary on-line review of the report during its development process, and the final review is made by the regulatory body after receiving final PSA report.

The survey shows that the regulatory review process for Member States extensively uses IAEA publications such as relevant safety guides, safety reports and TECDOCs (mainly IAEA Safety

Standards Series No. SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [1], IAEA Safety Reports Series No. SRS-25, Review of Probabilistic Safety Assessments by Regulatory Bodies [2], IAEA-TECDOC-1135, Regulatory review of probabilistic safety assessment (PSA) Level 1 [3]). Given the recent developments at IAEA and given the fact that currently the IAEA is working on several publications in the area of PSA, the recently published IAEA-TECDOC-1804 [4] and upcoming publications on multiunit PSA, human reliability analysis, integrated risk informed decision making and seismic PSA are also expected to be used in the regulatory review process.

The majority of CPWG Member States have requested and hosted IAEA PSA review services, i.e. IAEA Technical Safety Review Services in PSA (TSR PSA). Observation shows that the majority of IAEA PSA review services have been conducted in the 1990s and early 2000s, where current IAEA safety standards on PSA were not yet available [1, 5]. Considering the evolvement of PSA methods and approaches in last decades, it is expedient to request and conduct TSR PSA review using current IAEA safety standards.

3. CURRENT PSA PRACTICES OF CPWG MEMBER STATES

This section outlines the current practices of PSA development in CPWG Member States. CPWG discussions have been focused on the specific aspects of PSA found to be emerging topics of current PSA practices in the Member States operating CANDU-type rectors. The following topics have been compared and discussed in detail:

- Risk metrics for level 1 PSA;
- Initiating events (including issues connected with IEs frequencies and treatment of zero occurrence IEs);
- Reliability data (including the data collection, CCFs and issues related to mission times definition);
- Risk insights and use of PSA in CPWG Member States.

The summary of discussions is provided below in subsections 3.1–3.5 and the details for each of above mentioned topics are presented in Annexes II to V.

3.1. RISK METRICS FOR LEVEL 1 PSA

Core damage frequency (CDF) is a representative risk measure in level 1 PSA and has been widely used not only for design and operation of NPPs but also as a key safety measure in almost all countries. According to Ref. [1] criteria have to be specified for the concept of core damage, and these criteria may be different for different reactor designs.

In the case of the level 1 PSA of light water reactors (LWRs), CDF criteria is typically defined in in terms of the fuel parameters (such as the clad temperature) Ref. [1]. This approach is not directly applicable to the CANDU-type PSA because of the difference in reactor design. Subsequently, the CPWG collected definitions of CDF (task 2010-01) from Member States to harmonize the definition of the level 1 PSA risk measure for CANDU-type reactors.

Qualitative definitions of core damage in level 1 PSA of the CPWG Member States:

- Canada: extensive physical damage to the core, fuel bundles and channels would be disassembled;
- Republic of Korea: multiple fuel channel failure;
- Romania: damage to the calandria internal structure and fuel channels;

- India & Argentina: severe overstressing/overheating of reactor core or its components to the point at which loss of core structural integrity and large fraction of fuel melt is expected. Simultaneous structural failure of more than one channel.

Although there are some differences in the expression of 'core damage' in CANDU level 1 PSA, core damage can be interpreted as the condition where extensive physical core damage has occurred with loss of structural integrity and a large fraction of fuel melt.

Some examples of specific system level conditions leading to severe core damage in level 1 PSA of Member States can be found below:

- Canada: no subcooling margin in inlet headers and either moderator level is below highest channels or high plant radiation levels1;
- Republic of Korea: collapsed moderator liquid level is below top of the active fuel (TAF) and no subcooling of primary heat transport system (PHTS) coolant;
- Romania: moderator liquid level is below top of active fuel (TAF) and no subcooling of PHTS coolant.

3.2. INITIATING EVENTS

According to Ref. [1] the definition of initiating event is following:

> "§5.11. … An initiating event is an event that could lead directly to core damage (e.g. reactor vessel rupture) or that challenges normal operation and which requires successful mitigation using safety or non-safety systems to prevent core damage."

Initiating event analysis is a key task in the development of a PSA and the subtasks defined in Ref. [1] include the identification of initiating events, grouping of initiating events and the assessment and determination of initiating event frequencies.

This section describes the initiating event analysis subtasks as identified in Ref. [1], and additional considerations for CANDU-type PSAs. In particular, a comparison of loss of coolant accident (LOCA) frequencies and the assessment of small break LOCAs will be discussed in detail. LOCAs are considered one of the most challenging design basis events for CANDU NPPs because coolant voiding introduces positive reactivity into the reactor core. CPWG task 2010-02, further input on classification and description of LOCA initiating events and LOCA initiating event frequencies, aimed to determine the types of LOCA events considered by Member States in PSA. In addition, CPWG task 2016-06, small break LOCAs (SBLOCAs), aimed to determine how these events due to its possible consequences and due to the specific geometry of the CANDU-type reactors with separate channels and feeders. This section also includes subsection for CPWG tasks 2010-08 and 2016-05, treatment of zero occurrences initiating events. Zero occurrences initiating events are an important consideration for PSA because of the rarity of these events resulting in limited data available for initiating event analysis.

3.2.1. Identification and grouping of initiating events

The value of a PSA is enhanced when the set of initiating events (IEs) considered is comprehensive and complete. There exists a wide spectrum of initiating events considered in

[1] For CANDU-type reactors, the term severe core damage is used also to distinguish from other fuel damage states (e.g. limited, wide-spread). In this case, the risk measure for level 1 PSA is severe core damage frequency.

the various PSAs performed for CANDU-type plant PSAs. Harmonisation of the initiating events would aid in consistency among the diverse developers and users of PSAs, as well as completeness of the PSA scope and quality. This section covers level 1 PSA full power and does not consider initiating events during shutdown operation, internal flood, internal fire and external initiating events.

Information from the following CANDU-type countries has been considered: Pakistan, India, Argentina, Romania, the Republic of Korea, China and Canada.

The systematic process used by the Member States for the identification of the list of initiating events is based on the identification of radioactive sources and the radionuclide displacement mechanisms. Within a CANDU plant, there are several different mechanisms to displace radionuclides from their normal location. The systematic review process starts with the identification of the sources of radioactive material within a CANDU-type plant. For each distinct source of radioactive material, mechanisms that could lead to the displacement of radioactive material from its normal location are identified. The following is a list of radionuclide sources in a CANDU plant:

- Fuel bundles within the core;
- Heavy water in the primary heat transport system and related system;
- Heavy water in the moderator system and related system;
- Structural core components including pressure tubes, calandria tubes, adjuster and shutdown rod;
- Heavy water vapour and hydrogen contained in the moderator cover gas;
- Light water coolant in the end shield and vault cooling system;
- Light water in the liquid zone control system;
- Liquid poison system inventory;
- Carbon dioxide annulus gas;
- Fuel bundles in the fuelling machines, fuel handling and storage systems;
- Heavy water coolant in the fuelling machines;
- Light water in the fuel handling and storage systems;
- Heavy water in the D_2O management systems;
- Solid, liquid and gaseous wastes.

For each radionuclide source, the following displacement mechanisms are considered in order to identify potential initiating events that may result in releases: loss of heat sink from loss of cooling or loss of coolant flow, loss of circulation, loss of inventory or loss of system function. The following is a list of potential sources of radioactive releases that can occur:

- fuel or coolant;
- moderator system and related systems;
- cover gas;
- carbon dioxide annulus gas;
- fuelling machine handling, and storage;
- coolant from fuelling machine;
- fuel handling and spent fuel bay (irradiated fuel bay);
- D_2O Management Systems;
- active wastes.

In addition, support systems failures need to be considered as initiating events. A generic list of initiating events that have been considered in CANDU-type PSAs is provided in Table 4.

TABLE 4. GENERIC INITIATING EVENTS

IE CATEGORY	EVENT GROUP	EVENT DESCRIPTION
Loss of regulation accident	Core power excursion	• Bulk increase in reactivity
	Regional power excursion	• Local increase in reactivity
Loss of coolant accident (LOCA)	Interfacing LOCA containment bypass (V scenario)	• Interface LOCA through ECC system outside containment
	Large LOCA	• LOCA greater than 2.5% RIH, no containment bypass, emergency core cooling (ECC) is automatically initiated, All ECC stages are required
	Small LOCA: multiple steam generator tube rupture (SGTR)	• Multiple steam generator tube ruptures, containment bypass
	Small LOCA: loss of gland seal cooling to all PHTS pumps	• PHTS pump seal failures, ECC automatically initiated
	Small LOCA: pipe break upstream of pressurizer relief/steam bleed valves	• Pressurizer relief line piping upstream of the relief valves or steam bleed valve breaks
	Small LOCA: multiple tube ruptures in any RCW HX	• Tube break in PHTS auxiliary systems with discharge into the recirculated cooling water (RCW) system, containment bypass
	Channel flow blockage	• Blockage of fuel channel flow
	Small LOCA equivalent to 2.5% RIH break	• Any break equivalent of a D_2O feed pump capacity to 2.5% RIH break area, while the isolation of break is not possible, no containment bypass, ECC auto initiated, all ECC stages required to operate, beyond the capacity of PIC pumps
	Pressure tube and calandria tube rupture	• Pressure tube and calandria tube rupture, no containment bypass
	Feeder breaks	• Feeder break without flow stagnation, no containment bypass
	Feeder break with flow stagnation	• An inlet feeder break of certain size leading to flow stagnation in channel and subsequent pressure tube and calandria tube rupture. No containment bypass, similar to small LOCA
	Fuelling machine induced LOCA with no fuel ejection	• FM back off without closure plug, no fuel ejection, FM induced event.
	Fuelling machine induced LOCA with fuel ejection	• FM back off without shield and closure plug, fuel ejection, FM induced end fitting failure, fuel ejection
	Fuelling machine induced end fitting failure	• End fitting leaks outside annulus gas • End fitting leaks into annulus gas • Inadvertent movement of FM ridge (similar to end fitting failure, except that is caused by the FM and has a different frequency)
	HTS leak: within operating D_2O feed pump capacity	• Leaks within two PIC pumps capacity
	HTS leak: heat exchanger single tube rupture into RCW (containment bypass)	• Tube break in PHTS auxiliary systems with discharge into the recirculated cooling water (RCW) system, containment bypass
	HTS leak: SGTR	• Single SGTR, containment bypass
	HTS leak into annulus gas system	• End fitting leaks into annulus gas

TABLE 4. GENERIC INITIATING EVENTS (cont.)

IE CATEGORY	EVENT GROUP	EVENT DESCRIPTION
Loss of flow accident (LOFA)	Total loss of heat transport system pumped flow	• Total loss of flow, thermo-syphoning is established
	Partial loss of heat transport system pumped flow	• Partial loss of HTS
	Single channel flow loss	• Severe flow blockage
Loss of heat sink	Loss of feedwater flow	• All feedwater pumps failed
	Asymmetric feedwater line break inside RB upstream of steam generator check valve	• Break inside reactor building between check valve and containment, the flow to one boiler is lost
	Asymmetric feedwater line break inside RB downstream of steam generator check valve	• Break inside reactor building between check valve and boiler
	Symmetric feedwater line break outside RB	• Main feedwater header breaks, all boilers are affected, all feedwater pumps are lost
	Asymmetric feedwater line break outside RB	• Break outside reactor building after the regulation valve (one boiler is affected)
	Asymmetric steam generator blowdown line break inside RB	• Boiler blowdown line breaks inside reactor building **Note:** blowdown line breaks are divided into symmetric and asymmetric for deterministic analysis.
	Symmetric steam generator blowdown line break inside RB	• Boiler blowdown line breaks inside reactor building **Note:** Blowdown line breaks are divided into symmetric and asymmetric for deterministic analysis.
	Symmetric steam generator blowdown line break outside RB	• Boiler blowdown line breaks outside reactor building, boilers as heat sink affected
	Loss of condensate flow to deaerator	• All condensate pumps failed • Condensate line breaks inside turbine building/powerhouse • Deaerator level control valve failures • Condenser level control valve failures
	Loss of condenser vacuum	• Loss of condenser vacuum, (CSDV and condensate system become unavailable)
	Small condenser cooling water line break	• Small condenser cooling water line breaks, (CSDV and condensate system become unavailable)
	Large condenser cooling water line break	• Large condenser cooling water line breaks, (CSDV and condensate system become unavailable)
	Main steam line leak inside turbine building/ powerhouse	• Large main steam line breaks inside turbine building/powerhouse, (potential to affect all systems inside turbine building/powerhouse)
	Main steam line break inside reactor building	• Large main steam line breaks inside reactor building, (potential to affect all systems inside reactor building)
	Small main steam line failures causing low deaerator level inside reactor building	• Small steam line breaks inside reactor building (affects the deaerator)
	Small main steam line failures causing low deaerator level inside turbine building/powerhouse	• Small steam line breaks inside turbine building/powerhouse, ASDV spurious opened (affects the deaerator

TABLE 4. GENERIC INITIATING EVENTS (cont.)

IE CATEGORY	EVENT GROUP	EVENT DESCRIPTION
Loss of pressure control (LOPC)	HTS pressure control failure low	• Loss of HTS pressure control – low, e.g. spurious pressurizer spray actuation • Pressurizer control valves spurious open • Pressurizer valves open and heaters unavailable • Loss of both feed pumps • Feed valves fail closed and bleed valves fail open • Loop isolation valves fail closed
	HTS pressure control failure high	• HTS pressure control failure – high • E.g. Loss of heat transport control program in both computers • Pressurizer heaters fail on and pressure relief failure • Feed valves fail open and bleed valves fail close • Both feed pumps fail ON
	Pressurizer relief / steam bleed valves fail open	• Pressurizer relief valves spurious open, (loss of PHTS pressure control)
	Heat transport liquid relief valves (instrumentation relief valve) fail open	• Liquid relief valves open spuriously, (loss of PHTS pressure and inventory control)
	Inadvertent closure of three bleed condenser level control valves	• The postulated event is that all PHTS bleed condenser level control valves get closed suddenly
Moderator	Total loss of moderator heat sink	• Total loss of moderator heat sink
	Partial loss of moderator heat sink	• Partial loss of moderator heat sink
	Total loss of moderator flow	• Total loss of moderator flow (e.g. loss of moderator pumps
	Calandria inlet / outlet pipe break outside calandria vault	• Moderator pipe leaks • Moderator pipe breaks outside shield tank • Calandria drain line breaks outside shield tank • Moderator auxiliary system lines break
	Moderator pipe break inside calandria vault	• Calandria vessel rupture • Moderator pipe breaks inside shield tank
	Calandria tube leaks into annulus gas	• Calandria tube leaks into annulus gas, covered by HTS leaks into annulus gas
	Moderator heat exchanger single tube rupture	• Moderator HX tube breaks
	Moderator heat exchanger multiple tube rupture	• Moderator HX multiple tubes break
Moderator cover gas	Loss of moderator cover gas deuterium control	• Loss of cover gas inventory • Cover gas pressure control failed • Hydrogen combustion in cover gas
	Loss of end shield heat sink	• Loss of end shield cooling failure (e.g. heat exchangers)
	Loss of end shield coolant flow	• Loss of end shield cooling flow (e.g. loss of ES pumps)
	End shield cooling system pipe failure	• End shield cooling system pipe failure
Fuelling machine	Fuelling machine D_2O system failures	• Release from cooling machine coolant
	Fuelling machine failures causing mechanical damage to fuel on reactor	• Fuel bundle damaged by fuelling machine • Fuel damaged during fuelling operation
	Loss of cooling to fuel in fuelling machine FM off reactor	• Loss of FM D_2O inventory • Loss of FM D_2O circulation
Fuel handling and storage	Spent fuel transfer system failures	• Spent fuel transfer port failure • Spent fuel transfer cooling system failure • Fuel failure in spent fuel transfer port
	Loss of spent fuel bay heat sink	• Loss of storage bay inventory • Loss of storage bay flow • Loss of storage bay cooling
	Partial loss of storage bay inventory	• Partial of storage bay inventory

TABLE 4. GENERIC INITIATING EVENTS (cont.)

IE CATEGORY	EVENT GROUP	EVENT DESCRIPTION
Support/service	Total loss of instrument air	• Total loss of instrument air reactor operating
	Total loss of service water	• Total loss of service water, some stations high pressure and low pressure service water
	Partial loss of service water	• Partial losses in service water system
	Total loss of class IV	• Total loss of class IV power
	Partial loss of class IV	• Partial loss of class IV power
	Total loss of class III power	• Total loss of class III power
	Partial loss of class III power	• Partial loss of class III
	Total loss of class II power	• Total loss of class II power
	Partial loss of class II power	• Partial loss of class II power
	Total loss of class I power	• Total loss of class I power
	Partial loss of class I power	• Partial loss of class I power
	Dual computer control failure	• Dual computer control failure
	General transient	• General transient: reactor trips from unknown causes not included in other initiating events (also known as forced shutdown)
	Loss of bulk electrical supply	• Loss of bulk electrical supply
Multiunit	Loss of switchyard (off-site power)	• Loss of power from switchyard
	Loss of forebay	• Loss of forebay resulting in loss of circulating and service water
	Loss of common instrument air	• Loss of common instrument air

Comparing the list of initiating events considered in the various CANDU-type reactors, there is general consensus on releases from fuel or coolant due to a power or cooling mismatch. This may include loss of regulation, loss of heat sink or loss of coolant, release from the moderator and related systems. For example, this could be due to direct or consequential breach of the system boundaries, and releases resulting from the consequences of support system failures such as service water, electrical power, and instrument air. Some initiating events reflect design features that are provided in select CANDU-type reactors. One example is the spurious closure of main steam isolation valves initiating event, which are included in Wolsong and Qinshan PSAs, but not in the Canadian PSAs because main steam isolation valves are not installed in Canadian plants. In addition, some plants have considered additional support systems failures such as loss of active and/or non-active process water systems which would lead to loss of production. There is broad agreement among CPWG Member States to not analyse radionuclide releases from the annulus gas system, D_2O management systems, and active wastes because the consequences are negligible.

3.2.1.1. Release from moderator cover gas

A release from the cover gas system may arise as a result of a pressure control failure or a direct breach of the system boundary. Hydrogen/deuterium deflagration or detonation in the cover gas, following a failure in the deuterium control, is an event affecting the core as a whole. An initiating event related to the deuterium control has been considered in most of the CANDU-type PSAs.

3.2.1.2. Release from fuelling machine, handling, and storage

A release may arise from: failures in the fuelling machine, failures in the fuel transfer system, or failures during storage. Fuelling machine cooling failures may result from a loss of inventory, or a loss of coolant circulation. Direct mechanical damage to fuel bundles may also occur during fuelling machine operation. Spent fuel transfer system failures are categorized as: transfer port failure or a cooling failure during transfer. Fuel failures during spent fuel storage are categorized as: a loss of storage bay heat sink, or direct failure of the fuel bundle in storage,

either from chemical or mechanical action, or a defective bundle. Loss of heat sink events are further categorized as loss of storage bay inventory, loss of storage bay flow, or loss of storage bay heat removal. As per Refs [5] and [6], the PSA needs to take account the potential for releases from other radioactive sources from outside the core, such as irradiated fuel, stored radioactive waste, fuelling machine. Some PSAs have considered these events.

3.2.1.3. Release of coolant

Losses of fuel coolant are categorized as loss of coolant accidents (LOCA) affecting the core. A generic list of LOCA initiating events that have been considered in CANDU-type PSAs is provided in Table 4[2]. These LOCA initiating events are categorized according to location and size; in-core, out-of-core, interface LOCA or LOCAs via the fuelling machine.

In-core LOCAs are further categorized as pressure tube leaks, pressure tube ruptures or feeder stagnation break. Pressure tube ruptures may lead to subsequent calandria tube ruptures. Out of core LOCAs may occur from failures of the primary heat transport (PHT) pipes, steam generator tubes, or failures elsewhere in the pressure boundary.

Smaller LOCA events may be equivalent of a D_2O feed pump capacity to 2.5% reactor inlet header (RIH) break, or leaks. An example of a small LOCA is an end-fitting break.

Leaks may be outside or inside of the reactor building. Examples of leaks outside the reactor building include: heat exchanger tube ruptures into the recirculated cooling water (RCW) system via the PHTS purification circuit, the PHTS gland seal circuit, and the PHTS pressure and inventory control system degasser condenser cooler, the PHTS D_2O sampling system, or the fuelling machine D_2O heat exchangers. Leaks inside the reactor building include any break with an initial discharge less than D_2O feed pump capacity.

3.2.1.4. Release of coolant from fuelling machine

Fuelling machine LOCAs are categorized as shield plug failures, closure plug failures, end fitting leaks, or an inadvertent movement of the fuelling machine bridge. A total loss of inventory from the fuelling machine is categorized with events affecting the fuel in the fuelling machine. Initiating events solely related to release of coolant from the fuelling machine off reactor are of no consequences.

3.2.1.5. Release from fuel handling and spent fuel bay

A radionuclide release may arise from a direct breach of the storage bay with consequent partial loss of inventory or a failure of the active ventilation system. A loss of water inventory in the storage bay or a failure of fuel discharge mechanism could represent a loss of shielding for the spent fuel. The initiating events related to radionuclide releases from fuel handling and spent fuel bay have been considered in some Canadian PSAs. Furthermore, in light of recent events at Fukushima Daiichi Unit 4, initiating events related to spent fuel bays need to be considered in future PSAs.

[2] The IE frequencies used in this publication for different NPPs are based on IE report updates at different years as a result the number of reactor-years' experience credited in calculations are not exactly the same.

3.2.1.6. Multiunit initiating events

Multiunit initiating events result from an event that simultaneously creates a disturbance in more than one unit and has the potential to lead to core damage. A multiunit initiating event may be the result of an external initiator such as a seismic event, internal initiator such as internal flood, or may also occur as a result of a significant amount of shared systems between multiple units on a site. Multiunit initiating events have been considered by Canadian plants. The design features of the multiunit stations also have an impact on the number of initiating events leading to site specific events.

3.2.2. Initiating event frequencies

For initiating events frequency assessment, Ref. [1] promotes use of plant specific data (whenever it is possible). The plant specific data are to be supplemented by data from similar plants, if it can be shown that this is relevant. In case of lack of plant specific data (e.g. for new NPPs) Ref. [1] envisages use of data from similar plants or generic data. In addition to above mentioned techniques Ref. [1] also envisages use of fault trees that can provide a logic model of all the component failures and human errors that can potentially lead to the initiating event. However, use of fault tree technique for initiating event frequency assessment needs to be supplemented by analysis of consistency with operating experience.

Another important source for initiating event frequency evaluation could be IAEA initiative related to introduction of Initiating event module in PRIS (Power Reactor Information System) which will allow analysts to gather data related to the occurrence of particular initiating events for different type of reactors (e.g. CANDU) worldwide. Currently initiating event module for PRIS is developed and is in data input process.

Survey shows that initiating event frequency assessments for CANDU-type reactors are currently based both on fault tree analysis and operating experience evaluation. In some cases, depending on the initiating event type, generic pipe failure rates with the site specific pipe length have been used. Typically, Bayesian methods have been used to combine generic and site specific data. It was concluded that the fault tree technique used for CANDU-type reactors provides conservative results.

In order to standardize initiating event frequency assessment for CANDU-type reactors, the following considerations are prudent:

1. Use of fault tree analysis technique for initiating event is related to a process or support system.
2. Application of pipe failure rate data and site specific pipe length when analysing break-type initiating event.
3. When using operating experience, the site specific data is combined with generic data using the Bayesian method.

According to the CPWG observations, LOCAs represent one of the most challenging design basis events from the point of view of consequences, in terms of radionuclide releases for CANDU-type reactors, due to the pressure tube design. The special safety systems design requirements have been established for the reactor power control, PHTS heat sinks, and the ability to contain radioactivity following accidents.

3.2.3. Comparison of LOCA initiating events

The major LOCA-type initiators that have been considered in PSAs of different CANDU-type reactors are:

- Small LOCAs;
- Large LOCAs;
- Feeder breaks;
- In-core LOCAs;
- HTS leaks;
- Fuelling machine failures leading to LOCAs;
- Out-of-core LOCAs (in different parts of HTS piping);
- Interfacing LOCA.

Comparing the list of initiators considered in the various CANDU-type countries, there is general consensus on some LOCA initiating events, such as:

- Large LOCA with no containment bypass;
- Small LOCA equivalent to less than 2.5% RIH break (beyond D_2O feed capacity);
- Pressure tube and calandria tube rupture and: in-core LOCA;
- Feeder size breaks;
- HTS leak: within operating D_2O feed pump capacity and no containment bypass;
- Small LOCA: multiple/single steam generator tube rupture and containment bypass;
- HTS leak: heat exchanger single tube rupture into RCW and containment bypass.

Some differences between various CANDU-type countries exist, such as:

- Small LOCA: pipe break upstream (at the top) of pressurizer relief/steam bleed valves;
- Small LOCA: multiple tube ruptures in any RCW HX containment bypass;
- Feeder break with flow stagnation (FBS);
- Fuelling machine induced LOCA with or without fuel ejection;
- HTS leak: leak into annulus gas system.

3.2.4. Comparison of LOCA initiating event frequencies

The frequency of different LOCA categories has been obtained from the CPWG participants and a summary of these results is presented in Table 5. It can be observed that the majority of the initiating events have similar frequencies (same order or magnitude) while for some others have significant differences (more than one order of magnitude). A wide difference in initiating event frequencies was observed for the large LOCA, loss of gland seal cooling to all PHTS pumps and fuelling machine induced LOCAs.

Overall, the differences of initiating event frequencies can be attributed to the use of different methods of calculations (i.e. fault tree analysis; operating experience (most common); pipe failure rate data). As well, the difference of initiating event frequencies resulted from the use of different methods of calculating/considering the operating experience (e.g. zero occurrences, chi-squared; Bayesian; use of site specific station data or other methods).

TABLE 5. RANGE OF LOCA INITIATING EVENTS FREQUENCIES

IE GROUP	FREQUENCY RANGE, Y^{-1}	IE DERIVATION METHOD
Large LOCA: containment bypass into MPECC (V scenario)	6.1E-9 to 7E-12	fault tree analysis
Large LOCA: no containment bypass	2E-4 to 1.2E-6	operating experience fault tree analysis
Small LOCA: multiple SG tube rupture	1E-5 to 3.5E-5	operating experience fault tree analysis
Small LOCA: loss of gland seal cooling to all PHT pumps	6.6E-2 to 4E-4	operating experience fault tree analysis
Small LOCA: pipe break upstream of pressurizer relief/steam bleed valves	2.3E-4 to 1E-4	fault tree analysis pipe data calculation
Small LOCA: multiple tube ruptures in any RCW HX (containment bypass)	2.3 E-5 to 1.0E-5	operating experience fault tree analysis
Small LOCA equivalent to 2.5% RIH break	5.7E-3 to 6.5E-4	operating experience fault tree analysis
Pressure tube and calandria tube rupture	2.9E-3 to 5E-4	operating experience fault tree analysis
Feeder breaks (no stagnation)	2E-3 to 1.14E-3	operating experience
Feeder break with flow stagnation (FBS)	5.0E-4 to 1.20E-4	operating experience fault tree analysis
Fuelling machine induced LOCA with no fuel ejection	6.9E-2 to 5.3E-5	operating experience fault tree analysis
Fuelling machine induced LOCA with fuel ejection	7.3E-4 to 4.2E-9	operating experience fault tree analysis
Fuelling machine induced end fitting failure	6.2E-3 to 3.3E-6	operating experience fault tree analysis
HTS leak: within operating D_2O feed pump capacity (no containment bypass)	2.4E-1 to 3.71E-2	operating experience fault tree analysis
HTS leak: heat exchanger single tube rupture into RCW (containment bypass)	3.2E-2 to 4.4E-4	operating experience fault tree analysis
HTS leak: steam generator tube rupture	3.5E-3 to 9.38E-4	operating experience fault tree analysis

3.2.5. Modelling of small break LOCA

3.2.5.1. Introduction and definition

Small break LOCA (SBLOCA) is one of the initiating events heavily analysed in PSA studies due to its possible consequences and due to the specific geometry of the CANDU-type reactors with separate channels and feeders. In order to evaluate the differences observed in the results obtained by different countries, a specific questionnaire was prepared, and the answers were presented in detail in Annex II.

A SBLOCA can be initiated by a break in the PHTS headers, feeders, end fitting or by rupture of a pressure tube discharging directly to containment. Breaks in the line connecting the top of the pressurizer to degasser-condenser or in the gland seals of PHTS pumps are also included in this category.

Some degree of variability has been noted in the definition of the SBLOCA initiating event between the CPWG Member States regarding the break location and break dimension. The specific case represented by a break in the pressure tube break followed by a consequential calandria tube break was usually observed as a separate initiating event, called in-core LOCA. Other initiating events affecting a single channel (i.e. channel flow blockage or feeder stagnation breaks) are also usually covered by the in-core LOCA type of initiating event by most of the CANDU-type owner countries except India.

The maximum or minimum break dimension of the SBLOCA type of events are defined as a percent of the RIH area, continuing, at the lower side, the dimensions of the large LOCA category.

The largest dimension of a SBLOCA initiating event domain is defined as 5% of the RIH area. The break dimension is determined by the capacity of the reactor regulating system to maintain constant the reactor power during the event.

The smallest break dimension defining a SBLOCA was found to vary among the different PSA studies, from 0.16% RIH to 0.8% of the RIH area. The differences are mainly determined by the credited capacity of the primary pressurizing pumps (D_2O feed pumps) or by the domain covered by the automatic initiation of the emergency core cooling injection system. In the case of Romania, a supplementary initiating event (called very small LOCA) is covering the remaining break area domain (down to 0.16% of the RIH area).

3.2.5.2. Comparison of plant behaviour for a SBLOCA event

Based on the responses provided, the event progression for a SBLOCA event was noted to be similar between the participant countries. A general description of the event is presented below.

Following a postulated SBLOCA event, the gradual loss of the PHTS inventory is expected at a rate depending on the effective break dimension. Part of the inventory lost through the break is expected to be compensated by the inventory available in the pressurizer and by the D_2O pressurizing feed pumps. However, PHTS voiding will induce a reactor power increase that may be compensated by the reactor regulating system. In the long term, due to inventory depletion from the pressurizer or due to PHTS pressure decrease, the reactor will trip.

Loop isolation and steam safety valves opening are expected to take place when the PHTS pressure reaches the associated design set points. Emergency core cooling (ECC) injection may also be automatically initiated due to low PHTS pressure, if the conditioning signal is present. For some designs and for a certain domain of the break dimensions, the ECC injection is initiated manually.

Depending on the initial discharge, the containment pressure may increase to a level that containment is isolated and dousing initiated. Dousing spray may cycle few times until dousing inventory is depleted, or the energy discharged through the break is not enough to increase the containment pressure.

Following a successful high pressure/medium pressure ECC injection, the PHTS loops will be quickly refilled, decay heat being removed through the steam generators to the atmosphere. Adequate fuel cooling and no fuel failures are expected for the respective sequences. Long term fuel cooling is ensured by ECCS low pressure stage, using the water recovered from the sump and cooled through the ECC heat exchanger, as well as through the steam generators to the atmosphere. No fuel failures for SBLOCA with failure of loop isolation are expected since both loops are refilled by ECC injection in a similar way as for the case with loop isolation available.

In case of ECCS impairment (other than loop isolation) a limited number of fuel failures are expected as the PHTS inventory in the loop is depleted. The pressure tubes may heat up and contact the associated calandria tubes by ballooning or sagging. Heat transfer to moderator and adequate heat removal from this system will ensure integrity of these channels.

3.2.5.3. Comparison of SBLOCA initiating event frequencies

Comparison of the SBLOCA initiating event frequency was part of the questionnaire. The responses received identified that the methodology adopted for frequency estimation among CPWG Member States is different.

In the case of Canada and the Republic of Korea, generic values with the use of chi-square method and Bayesian update respectively have been used as these CPWG Member States have observed zero failure occurrences. Lognormal distribution is used for uncertainty analyses by the majority of the CANDU-type owner countries. Even the methodology adopted for the frequency estimation of SBLOCA in CPWG Member States is different, these methods are statistically equivalent.

The SBLOCA initiating event frequency varies between 1.3E-2/yr to 6.5E-4/yr. The differences in the mean value of the SBLOCA frequency are associated to the different break size definition and different method used to calculate it, as presented in the sections above.

3.2.5.4. Comparison of the SBLOCA quantification results

The comparison was initiated to understand the possible attributes of the variations observed in the earlier evaluation of PSA results regarding SBLOCA events. Towards this initiative, a detailed analysis and comparison of the response from the Member States was conducted as a specific task. The results obtained indicated that even though the nomenclature used is different, there are similarities observed in the function events and the subsequent accident progression.

In the SBLOCA contributions to CDF, large variations were observed among CPWG Member States. Such variations are conditioned by the differences in the initiating event frequencies and frontline line systems unavailability among the CPWG Member States.

3.2.6. Treatment of zero occurrence initiating events

3.2.6.1. Introduction and definition

The initiating events with occurrences expected to occur only a few times throughout the world nuclear industry over many years (i.e. less than 1E-4/yr) (Ref. [7]). A survey of Member States with CANDU-type NPPs was conducted to determine the methods used. Various methods are used by different level 1 PSA studies for the calculation of rare initiating event frequencies.

3.2.6.2. Methodology for derivation of zero occurrence initiating events

Annex III has the information collected from CPWG Member States on treatment of zero occurrences for calculation of initiating event frequencies. The methods used include Bayesian methods, Jeffery's non-informative prior, chi-squared approximation, the use of piping data where applicable and, fault tree analysis. In some cases, initiating event frequencies from past PSA studies or international publications were used.

The selection of data sources is critical. Surveyed Member States used a variety of data sources such as old safety design matrices to recent US NRC NUREGs. The difference in initiating event frequencies may be attributed to the use of different data sources. All Member States use Bayesian techniques and fault tree analysis depending on the initiating event. Respondents to the survey in Annex III identified several data sources for use.

There are several methods for calculation of zero occurrence initiating event frequencies depending on the type of events. There are two main calculation methods: (i) based on operating experience, and (ii) based on fault tree analysis.

A Bayesian approach is normally used when basing calculations on operating experience. In some cases, chi-square approximation can also be used. A non-informative Jeffrey's prior distribution is used when no instances are observed in the generic data. Non-informative prior distributions are a class of prior distributions that minimize the relative importance of the prior distribution in generating a posterior estimate. Another method of calculation involves chi-squared approximation. This is a standard technique for evaluation of rare events.

For initiating events that solely reflect piping failure modes, piping failure rate data may be used, for example, main steam line breaks and feedwater line breaks. Selection of an appropriate pipe data source is dependent on the temperature and pressure of the piping as well as the size of piping (diameter).

For application in CANDU-type PSAs, it is preferred to apply operating experience specific to CANDU-type reactors. In some cases, if the CANDU-type operating experience is not sufficiently large, LWR time periods can be used as long as the operating experience is applicable. For example, LWR time periods can be used for determining a large LOCA initiating event frequency.

The CANDU Owners Group operating experience database is a good source for identification and review of recent operating experiences when conducting updates of PSA initiating events. The latest publications of other data sources may also be used. For some initiating events with no events, fault tree analysis is used. This is for initiating events that are based on systems or subsystems of the plant. For example, the frequency of interfacing LOCA or loss of class IV power can be calculated by fault tree analysis.

3.2.7. Observations

The comparative analysis and discussions held in CPWG regarding initiating events allowed to come up with number of observations, which are summarized below.

3.2.7.1. Identification and grouping of initiating events

The list of initiating events has been compiled and has been found to be consistent among the Member States operating CANDU-type reactors. There are some examples in the list of specific initiating events depending on the differences in design. It is prudent that each Member State ensure that a systematic review process is in place for the identification of initiating events and verifies the completeness of its own list of initiating events against the list of initiating events presented in Table 4.

3.2.7.2. Initiating event frequencies

There exists a wide spectrum of LOCA initiating events considered in the various PSAs performed for CANDU-type reactors, such as: small LOCA, in-core LOCA, PHTS leakages in the capacity of the P&IC system, interface systems LOCAs, FM induced LOCA. There is relatively good agreement in the most LOCA initiating event frequencies, but there are still differences in some initiating event frequencies, given the different methods of calculations. It is prudent that CPWG Member States consider standardizing the calculation method with regards to initiating event frequencies in general and LOCA frequencies in particular. Once

finalized, the IAEA PRIS database could be used as an effective source for initiating event frequency evaluation for CANDU-type reactors.

3.2.7.3. Small break LOCA

The CWPG noted that there is a consensus on the definition of SBLOCA among the Member States, with a slight variation. A good harmonization in understanding of the event sequences was observed between different PSA studies. Some variations were noted in the event tree headers due to differences between CANDU-6 design and Indian PWHRs. Since the PSA study reflects the respective design, harmonization is expected to be limited. Also, variations observed in the minimal cut sets list and SBLOCA contributions to overall CDF is expected, considering the differences noted in the initiating event frequency, the event tree and fault tree models, component failure data, treatment of CCF, and human reliability analysis used by different studies.

3.2.7.4. Treatment of zero occurrence initiating events

It was observed that the methods for the treatment of the zero occurrence events for calculating initiating event frequencies were consistent among the Member States operating CANDU-type reactors. However, due the use of different data sources, the calculations of specific initiating event frequencies varied among the participants. It is prudent that Member States of CANDU-type reactors share initiating event frequency data and their basis such that significant variations between similar events across all Member States can be rationalized and lessons learned where appropriate.

3.3. SUCCESS CRITERIA

3.3.1. Introduction and definition

According to IAEA-TECDOC-1804 the success criteria are:

> "Criteria for establishing the minimum number or combinations of systems or components required to operate, or minimum levels of performance per component during a specific period of time, to ensure that the safety functions are satisfied" [8].

The success criteria of a system may be different depending on the function that the system needs to perform in response to a certain transient or accident, and the effects produced by the transient or accident itself. In general, the success criteria of a safety system have to be independent from the success or failure of any other individual plant system.

The success criteria are presented as the minimum level of performance for the support systems that are meant to perform a safety function during an initiating event and are taking into account the specific of the respective initiating event sequence.

3.3.2. Success criteria analysis

The scope of task 2010-07 of the CPWG working plan was to evaluate and understand the differences in the success criteria used for the systems in CANDU 6 type reactors, which are credited to perform a function in the mitigation of initiating events in level 1 PSA. For this purpose, a table for the collection of data related to systems success criteria from the Member States was prepared. The information requested and received as part of the first questionnaire is presented in Annex IV.

The information was received from Argentina, Canada, China, India, the Republic of Korea, and Romania. The information received from Pakistan was not considered due to the design differences in comparison with CANDU 6 reactors.

The evaluation revealed that almost all of the respondent countries reported the success criteria for the same frontline and support systems credited in the PSA. However, differences related to the crediting of digital control computers (DCC) and the reactor control functions (setback and step back) were identified. Argentina, Canada and the Republic of Korea credit the DCC computers in the PSA while the other participant countries do not. Romania did not give credit in the PSA to the reactor control functions despite of the fact that the control programs' function was credited.

Although the level of detail in the information provided influenced the depth of the evaluation, the general conclusion related to the success criteria themselves was that there are differences between the participating countries for most of the systems considered, including the special safety systems. Due to the limited information available, it was difficult to determine the reasons behind the differences in the success criteria of systems credited in the PSA.

A supplementary list of questions was prepared in order to justify and provide the rationale for the differences in the success criteria. The questions were provided in a table format in order to harmonize the answers from the participants. Both the break size and break discharge were included in the table in order to corroborate different approaches/definitions of the LOCA type initiating events.

The list of questions and summary of the answers are presented in Annex IV. The list of questions prepared in order to justify and provide the rationale for differences in the success criteria is presented below. For each question the function was also included. One key success criterion is the number and type of valves required to open for the boiler crash cooldown function in order to ensure an effective PHTS heat sink. The steam is released to atmosphere and the increase in the heat transfer removes the energy stored in PHTS. The differences observed came from the design (valves capacity itself) or from refined safety analyses.

Another item was the number of headers required to be supplied with water by ECC injection, in order to ensure coolant inventory control in case of a LOCA type event. Differences in responses came from the CANDU 6 (C6) design CANDU versus different design Enhanced CANDU 6 and/or interpretation for India. The results are based on different, and in some cases, refined safety analyses between C6 in support of the less restrictive success criteria (see in-core LOCA event).

The number of ECC heat exchangers required in order to ensure long term decay power removal after LOCAs was identified. The results observed indicate that the success criteria are similar for CANDU-6 plants. Supplementary deterministic safety analyses demonstrate that the Emergency Water Supply can be made available and can provide an effective heat sink for ECC heat exchanger cooling in a special case (in-core LOCA). The time available allows for the required operator action.

The number of steam generators required per loop required to ensure decay power removal is another item. Differences are determined by the conditions required for thermo-syphoning process. The number is determined based on specific safety analyses assumptions and results. More refined deterministic analyses have identified that not all steam generators may be needed for decay heat removal for some specific initiating events.

The number of moderator heat exchangers and pumps required to ensure the functionality of the moderator as an ultimate heat sink, moderator temperature control, cover gas functionality is a further item. Differences are determined by the differences in design and by specific deterministic safety analyses.

With regard to DCC operation, control programs have been considered in the PSA models to represent the plant operation. Program failures due to hardware or software failures were included. In addition, DCC failure is considered as a separate initiating event (Argentina, Canada, China, the Republic of Korea and Romania). Differences related to the crediting of the reactor control functions (setback and step back) that have been identified are determined by the specific deterministic analyses.

3.3.3. Observations

Several potential reasons were identified for differences in the success criteria, such as:

- Differences in the systems design (especially for India);
- Differences in initiating events grouping and definitions;
- Differences in the safety margins resulting from potential differences in the deterministic safety analyses;
- Existence of supplementary PSA specific deterministic analysis for evaluation of the accident progression;
- Differences in the abnormal operating procedures, or other accident mitigation procedures used by the plant;
- Differences in the general and specific assumptions used in the development of the PSA models.

Except for the differences in the plant design, a different set of deterministic safety analyses (supplementary analysis in support of PSA) is considered the main basis for the differences observed in the success criteria for similar plants.

In order to harmonize the success criteria and to reduce the conservatism, supplementary sensitivity analyses cases or refined analyses may be conducted, as needed, to reconcile the success criteria between similar plants.

3.4. RELIABILITY DATA

Reliability data analysis is an important PSA task with various subtasks including calculating frequencies of initiating events and component failure rate probabilities Ref. [1]. This section provides a brief summary of the results from relevant CPWG task on raw data collection and sharing among CPWG Member States. In addition, this section includes information about CPWG task 2010-04, Common Cause Failure Data Collection, and task 2010-06, Definition of Mission Time.

3.4.1. Raw data collection

Component failure data forms the essential input for PSA study. The CANDU Senior Regulators Group (CSRG) suggested the possibility of sharing the component data among Member States to serve as the priori information and can be updated with plant specific data (i.e. likelihood function) using Bayesian or equivalent statistical techniques to generate a posterior probability of component failure. The task has been investigated, however presently sharing of data is not considered feasible.

3.4.2. Common cause failure

The Member States operating CANDU-type reactors generally have consensus on the procedural guidance for common cause failure (CCF) analysis in PSA for NPPs. However, an estimate of CCF parameters on the basis of CANDU-type specific component database would be beneficial. It was concluded that Member States may use the International Common Cause Failure Data Exchange database.

3.4.3. Mission time

According to IAEA-TECDOC-1804 the mission time is:

> "The time period that a system or component is required to operate in order to successfully perform its function" [8].

Current practice in CPWG Member states implies application of 24 hours or 48 hours. The use of a standard mission time[3] (for full power, internal events PSA) is generally based on judgment, balancing the potential optimisms and pessimisms in the approach. Although decay heat removal is required after this time, the following factors are considered to support the use of a standard mission time:

- No claim is made for repair actions within the standard mission time, although it would be reasonable to expect that such actions would be effective in many cases;
- In the event of failures after the standard mission time, more time is available for corrective actions, including repair, as a consequence of the reduced decay heat level;
- The calculation of the probability of failure of a redundant operating component and its 'backup' over the standard mission time is typically pessimistic, since only one of the components is required to operate at any point in time.

For CANDU-type multiunit PSAs with shared containment structures a duration of 72 hours is considered to be appropriate as the standard mission time since CANDU-type reactors take longer for many accidents to evolve to severe core damage or other end states. The CANDU-type design allows for many ways or combinations with which to remove decay heat (including the use of the moderator system or end-shield cooling system as heat sinks).

Based on response from Member States that have developed PSA for CANDU-type NPPs with individual containment structures for each reactor, the standard mission time is typically defined as 24 hours to evaluate accident evolution to severe core damage and other end states, which is often also applied to also evaluate containment performance via level 2 PSA. In some Member States, a mission time of 72 hours is used for level 2 PSA. In summary, for CANDU PSA purposes there are three types of missions that can be considered as follows:

- Short term mission could range from zero to several hours. Systems with short term mission for example are (i) systems required to trip the reactor or reduce power, (ii) systems with multiple phases of operation (e.g. emergency coolant injection phase), systems with multiple redundancy (standby generators, diesel generators) where excess capability would be shut down, and (iii) generators that supply power only to put a safety/mitigating system in a required configuration;
- Standard (default) mission is used for most of the PSA systems by default;

[3] In this context standard mission time is the mission time assumed for most PSA systems by default.

- Long term mission is normally for systems in level 2 PSA to support the containment function.

A survey of the Member States was carried out to study how mission time is defined. The details of the survey can be found in Annex V.

3.4.4. Observations

The comparative analysis and discussions regarding the reliability data, CCF and mission times allowed to come up with a number of observations, which are summarized below.

3.4.4.1. Reliability data

In terms of reliability data collection, it was observed that Member States use a generic database in conjunction with Bayesian update procedure and are also collecting plant specific data. Since the collection of plant specific data is in various stages, presently sharing of data is not feasible.

3.4.4.2. Common cause failures

It would be beneficial to estimate CCF parameters on the basis of a CANDU-type specific component database. The International Common Cause Failure Data Exchange database or any standard database may be used.

3.4.4.3. Mission time

In the context of mission time definition for systems modelled in PSA, the following observations were made:

- The multiunit CANDU-type PSAs with shared containment structures have a longer standard mission time (72 hours) as opposed to most CANDU-type PSAs (24 hours). As the standard mission time value could have a significant impact on quantified core damage frequencies, this needs to be considered when interpreting and comparing core damage frequencies between Member States;
- Short term or long term mission times for specific systems were applied to some special cases in the PSA modelling (e.g. in CANDU PSAs ECCS or moderator as heat sink usually apply long term mission times, shutdown systems or boiler emergency cooling system typically apply short term mission time);
- Long term mission time were defined for level 2 PSAs as a different value than the standard mission time defined for level 1 PSA.

In general, mission times need to be supported by a specific justification and accepted by the national regulator in a particular Member State.

3.5. RISK INSIGHTS AND USE OF PSA

3.5.1. Introduction

The purpose of task 2014-04 was to identify the similarities between Member States in their use of PSA and to identify areas of interest for other Member States. The method to identify similarities included the use of a questionnaire of survey submitted to each Member State with CANDU-type reactors.

3.5.2. Summary of task

The list of questions was prepared and separated in two categories: the first set of questions was directed to NPP operators and included questions related to design and plant configuration; operation; and, accident management. The second set of questions was submitted to national regulators of the Member States and was focused on the use of PSA within regulatory processes. All Member States in the CPWG (Argentina, Canada, China, the Republic of Korea, India, Pakistan and Romania) provided answers in response to the questionnaire on risk insights from PSA and use of PSA studies.

While the specific objectives of PSA can be diverse and varied, the main objectives of the PSA are to confirm the robustness of the plant design and to identify vulnerabilities, which can then be addressed, if necessary.

3.5.2.1. NPP operators

Information received from participants confirmed that the PSA results have been used in accordance with the general objective presented above. The ranking of different proposed modifications has been evaluated based on PSA results. The proper design changes have been introduced for implementation and new PSA models reflecting updated safety analysis and success criteria demonstrated CDF reduction. Examples of PSA study results used to regard NPP improvement include:

- Improvement of ECC initiation by including a supplementary logic (Argentina, the Republic of Korea, Romania);
- The PSA support safety analyses have been used as a basis for CDF reduction through success criteria and requirements (all);
- PSA results and associated effects were used for plant modifications ranking (all);
- PSA results have been used for technical operability evaluations (Argentina, Pakistan, India, Romania);
- Optimization of redundancy level and surveillance testing internal of critical components (India).

PSA results have been used in the evaluation of safety issues and for technical operability evaluations to confirm continuous plant operation within risk targets. Critical components identification based on PSA results has been confirmed as a current practice for all Member States. Examples of PSA study results used regarding evaluation of safety issues:

- Possibility to check certain ECC manual valves status after ECC initiation (Argentina);
- Identification of specific critical components (all);
- Insights on critical containment components that may benefit from restoration of power in a severe accident condition (Canada);
- High risk related to use of specific system during plant outages (China);
- Identification in PSA of operator actions leading to possible initiating events (India).

The day-to-day use of PSA in maintenance activities is a limited practice based on the information received. However, PSA studies identified the need for supplementary verification of the critical components after maintenance/outages. PSA can be used to establish the basis for frequency of maintenance activities that are also credited for surveillance of component failure modes. PSA can also be used to examine maintenance scheduling to ensure that overall quantified plant risk is maintained within risk targets.

PSA results have been used for identification of new operating procedures, setting of testing and surveillance frequencies, verification of emergency procedures, and for implementation of measures to limit accident consequences. Utilizing PSA, procedures have been revised and updated to improve likelihood of human action success to mitigate accidents, and to improve overall training activities. Examples of PSA study results used in the area of accident mitigation and management include:

- Abnormal plant operating procedures related to human actions have been improved based on PSA results (Argentina, Canada, Pakistan, India, Romania);
- PSA has identified hook up arrangements in support of severe accident management guidelines (SAMG) actions (India);
- Used for operators training activities (Argentina, Canada, the Republic of Korea, Pakistan, Romania);
- Basis for specific supplementary plant procedures (Romania);
- Improvements in SAMG for defining scenarios for emergency drills (Canada, India, Romania);
- PSA results have been used and contributed to the optimization of surveillance tests requirements in technical specification and outage cycle optimization (China).

3.5.2.2. Nuclear regulators

From a nuclear regulatory perspective, the PSA has been commonly used in NPP licensing activities and in support of regulatory decision making processes. Operating experience involving plant transients was used in enhancing knowledge of potential initiating events included in PSA. The PSA was used as a basis for regulatory staff training, both from on-site inspections and regulatory review perspective. Examples of PSA use by nuclear regulatory organizations include:

- Licensing and decision making (Argentina, Canada, Pakistan, India, Romania);
- Evaluation of the impact of the experienced transients and for identification of possible precursors;
- On-site inspectors training on special systems, components, human actions (Romania).

3.5.3. Observations

The evaluation emphasized that PSA is used both in support of plant operations and national regulators within their regulatory processes. The responses to the questionnaire submitted by Member States showed similarities in the use of PSA results. To provide a greater harmonization in the use of PSA results across Member States, specific applications for risk monitoring may be benchmarked, developed, supporting plant maintenance activities. From regulatory perspective, the PSA have been used for NPP licensing activities and decision making processes, as well as for supplementary staff training on different accident sequences, inspections and review of operational documents.

4. RISK METRICS AND SOFTWARE TOOLS

PSAs are being used in various regulatory applications in all Member States with CANDU-type reactors. Sharing information on PSA applications to risk-inform various aspects of NPP operation is viewed as beneficial towards harmonizing approaches across Member States. In this regard, there are two major elements: (i) the development of probabilistic goals; and, (ii) development of the guidelines for use in risk informed decision making. Hence, it was recognized that apart from other harmonization issues, the CPWG needs to collect and compare

the information on different probabilistic measures, which are used in the CPWG Member States for different PSA applications.

This Section presents the results of the CPWG tasks 2010-09/2010-05, the objective of which was to collect information on the procedure and methods for PSA applications as well as numerical acceptance guidelines for PSA results. A survey was circulated across Member States with CANDU-type reactors to consider the PSA applications mentioned in IAEA-TECDOC-1200, Applications of PSA for NPPs [9] with only responses provided by Argentina, Pakistan and India.

In addition, this section provides information on the software tools used for the development of level 1 PSA models among the CPWG Member States.

4.1. RISK METRICS

There is variability between Member States in the regulatory framework for specifying time-averaged safety goals for probabilistic risk metrics and on the scope of PSA (level 1 versus level 2 versus level 3). In all cases, the time-averaged PSA risk metrics reflect a best estimate approach, which is interpreted that risk metrics are generally expressed as an arithmetic mean value. In terms of applying time-averaged probabilistic risk metrics to the definition of individual system unavailability targets, predefined target values for special safety systems (shutdown systems, containment, and emergency core cooling) have generally been defined at 1E-03 years/year. There appears to be little consensus on application of PSA for defining the target of other modelled mitigating systems[4].

The scope and quality of a PSA determines the applications to which it can be applied (Ref. [4]). If the quality of the PSA is limited, how and to what applications the probabilistic risk metrics may be applied undergoes scrutiny to ensure insights from the PSA are appropriately weighed and that uncertainties in the PSA are not driving decision making.

Provided that sufficient quality and scope of a PSA is assured, the probabilistic measures of primary importance include severe core damage frequency (SCDF), large (early) release frequency (LRF or LERF), and individual mitigating system unavailability targets that may be established. Often, these measures are applied to, but are not limited to:

- defining or optimizing test and surveillance intervals included in technical specifications or plant programs;
- defining allowable outage times through risk monitors or assessments by applying instantaneous risk thresholds or criteria when equipment is removed from service;
- supporting graded approaches to define the scope of deterministic safety analysis;
- identifying and categorizing initiating events;
- defining operator action times and end states in case of unplanned equipment failure;
- evaluating risk impact as part of operational events precursor analysis;
- supporting reliability-centred maintenance processes.

Where Member State regulatory processes also require compliance with quantitative health objectives via level 3 PSA (limited or full scope), applicable probabilistic measures may also include individual early or late fatality risk, assurance that predicted cancer rates fall within

[4] Note that although not included in the survey responses, one NPP in Canada has followed a methodology for defining individual system unavailability targets directly from the PSA, which has received regulatory acceptance.

some percentage of background rates, or some other variation of this requirement, depending on specific regulatory objectives for a Member State. Such probabilistic measures can be utilized to assist in emergency planning and in support of environmental risk assessment.

4.2. SOFTWARE TOOLS FOR LEVEL 1 PSA

A number of verified and validated computer codes and software packages are currently used for performing PSA in CPWG Member States. Typically, an integrated software package is used in the level 1 PSA analyses for the development and storage of system models, accident sequence models, failure data, and accident sequence quantification. Other computer codes are used to ensure deterministic support for the PSA models (e.g. TH calculations to support success criteria analysis). The summary of the computer codes used for the development of level 1 PSA models among the CANDU-type operating countries are presented in Table 6.

TABLE 6. COMPARISON OF LEVEL 1 PSA TOOLS

MEMBER STATE	COMPUTER CODE USED FOR LEVEL 1 PSA	VERIFICATION/VALIDATION STATUS
Argentina	RISK SPECTRUM	Verified and validated by Lloyd's Register Consulting
Canada	CAFTA, FTREX, ACUBE, FRANX	Verification and validation of the codes is done by EPRI and vendors prior to use of a new code version
China	CAFTA	Same as Canada above
India	RISK SPECTRUM	The regulatory requirement is brought in AERB safety guides
Republic of Korea	AIMS PSA (quantification engine: FTREX) by regulatory body and the utility uses SAREX (quantification engine: FORTE) for level 1 PSA	Verified and validated by vendor KAERI and KEPCO E&C
Pakistan	RISK SPECTRUM	Same as Argentina above
Romania	CAFTA	Same as Canada above

4.3. OBSERVATIONS

In general, all Member States with CANDU-type reactors that responded to the survey apply similar probabilistic measures to support PSA applications such as SCDF, LRF, LERF or system unavailability targets. Some Member States also require probabilistic risk metrics applicable to level 3 PSA.

The definition of system unavailability targets for specific safety systems is consistent; however, the Canadian methodology of defining unavailability targets for other mitigating systems directly from PSA using time-based probabilistic measures can be shared.

While instantaneous probabilistic measures can be applied to determine AOT, there is an opportunity to provide a consistent definition of instantaneous risk threshold across all Member States operating CANDU-type reactors such that all CANDU-type reactors are controlling acceptable operational risks in a similar manner.

In terms of software tools, it could be concluded that CPWG Member States mostly use Risk Spectrum and CAFTA. The Republic of Korea uses different computer software called AIMS PSA and SAREX. The Member States indicated that the computer codes used for PSA are validated by the vendors and users.

5. IAEA PUBLICATIONS FOR PSA DEVELOPMENT, APPLICATION AND REVIEW

This section describes the results of the CPWG tasks aimed to review the IAEA publications from PSA perspective to facilitate their use for CANDU type reactors. The focus of this study was made on the PSA development and application process, as well as to the regulatory review aspects. Therefore, the following IAEA publications have been considered:

- IAEA Safety Standards Series No. SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants (Ref. [1]).
- IAEA Safety Reports Series No. SRS-25, Review of Probabilistic Safety Assessments by Regulatory Bodies (Ref. [2]).

The results of the study are presented in subsections below.

5.1. SPECIFIC CONSIDERATIONS IN USING THE SSG-3

This section describes the results of the CPWG task 2011-03/2013-01, Develop CANDU-Specific PSA Guide. The objective of this activity was to look through Ref. [1] to facilitate its use for development and application for PSA of CANDU type reactors.

As per the defined scope of IAEA safety guides, those are intended to be technology neutral to the extent possible and therefore Ref. [1] is applicable to the CANDU-type reactors. During the CPWG discussions it was concluded that when using Ref. [1], the following aspects require special considerations from the point of view of CANDU PSA development and application.

Definition of the core damage frequency. Currently, Ref. [1] provides clarification on core damage definition in paragraphs 5.42 and 5.43, which is supplemented by an example for pressurized water reactor (PWR). In CANDU PSA it is necessary to keep in mind the concept of severe core damage which, for instance, could be expressed as a condition with an extensive physical damage of the multiple fuel channels due to overheating leading to loss of core structural integrity.

End states of the event tree. For CANDU-type reactors, the different accident sequences representing the end states of the event tree could be clearly defined as fuel damage categories, e.g. rapid loss of core structural integrity (FDC1) and slow loss of core structural integrity (FDC2) or plant damage states (PDS0, PDS1, etc.). The examples of fuel damage categories for CANDU-type reactors are presented in Table 7.

TABLE 7. EXAMPLES OF END STATES OR FUEL DAMAGE CATEGORIES

DEFINITION	ACCIDENT CATEGORY	EXAMPLE
Early (rapid) loss of core structural integrity	Severe core damage	Fast reactivity increase (severe power excursion) + failure to shutdown reactor
Late loss of core structural integrity with high heat transport system (HTS) pressure	Severe core damage	Loss of feedwater + loss of shutdown cooling (SDC) + loss of emergency core cooling (ECC) + consequential loss of moderator with high HTS pressure
Late loss of core structural integrity with low HTS pressure	Severe core damage	Loss of feedwater + loss of shutdown cooling (SDC) + loss of emergency water system + loss of moderator due to loss of service water with low HTS pressure (ECC provides makeup, but is not available as heat sink service water is lost)

CANDU-type plant operational states (POS). The examples of typical POSs for CANDU-type reactors are presented in Table 8 on the basis of Ref. [6]. For a full scope PSA study, it is necessary to clearly outline the specifics of the outage types available at CANDU-type reactors and specification of the plant operational states.

TABLE 8. TYPICAL CANDU-TYPE PLANT OPERATIONAL STATES

PARAMETER	POS A	POS B	POS C	POS D	POS E
Guaranteed shutdown state	Over poisoned guaranteed safe shutdown	Drained guaranteed shutdown state	Over poisoned guaranteed safe shutdown	Over poisoned guaranteed safe shutdown	Over poisoned guaranteed safe shutdown
HTS inventory	FULL	FULL	Drained*	FULL	FULL
HTS boundary	Closed	Closed	Open	Closed	Closed
Typical HTS pressure	≤200 kPa(g)	≤200 kPa(g)	0 kPa(g)	≤200 kPa(g)	≥2.7 MPa(g)
Primary heat sink (circulation)	SDC pumps (even/odd)	SDC pumps (even/odd)	Convection	SDC pumps (even/odd)	HTS pumps
Primary heat sink (heat removal)	SDC heat exchangers (even/odd)	SDC heat exchangers (even/odd)	ACU + ESC + moderator	Feedwater + boiler blowdown	SDC heat exchangers (even/odd)
Backup heat sink (circulation)	SDC pumps (odd/even)	SDC pumps (odd/even), convection	SDC pumps	SDC pumps (odd/ even)	SDC pumps
Backup heat sink (heat removal)	SDC heat exchangers (odd/even)	SDC heat exchangers (odd/even), boiler blowdown ACU + ESC	SDC heat exchangers (odd/even)	Boiler blowdown (re-heater drains pump)	SDC heat exchangers (odd/even)
Emergency heat sink	Emergency water supply	Emergency water supply	Emergency water supply	Emergency water supply	Emergency water supply

* Drained is a state with coolant partially drained to certain level to allow the inspection of steam generators and/or the maintenance of some components (e.g. main pumps). It cannot be completely drained, because fuel is still located in the reactor core and the coolant in HTS still needs to provide the cooling function.

Initiating events in shutdown PSA for CANDU-type reactors. Annex III of Ref. [1] presents the examples of initiating events in shutdown PSA for PWR type reactors, which does not cover initiating events specific for CANDU-type reactors in shutdown state, such as Loss of Moderator Inventory and failure of operating D_2O pump. The example list of typical initiating events for shutdown PSA for CANDU-type reactors is presented in Table 9 on the basis of Ref. [6]. Plant operating states presented in the Table 9 correspond to the ones presented in Table 8.

TABLE 9. TYPICAL INITIATING EVENTS OF A SHUTDOWN PSA

IE DESCRIPTION	POS A	POS B	POS C	POS D	POS E
INTRINSIC SYSTEM FAILURES FOR PRIMARY HEAT SINK					
Failure of primary heat sink (main HT pumps and boiler blowdown)					X
Failure of primary heat sink #4 (main HT pumps + SDC HXs)					X
Failure of primary heat sink (SDC pumps and SDC HXs)	X	X			X
Failure of primary heat sink (SDC pumps and SDC HXs)					
Failure of primary heat sink (SDC pumps, MBFP, condensate pumps and boiler blowdown)					
Failure of primary heat sink (SDC pumps and boiler blowdown using re-heater drains pump)					X
Failure of primary heat sink (convection ACUs, moderator and ESC)	X	X		X	X
HT SYSTEM BOUNDARIES					
Non-isolatable HTS leak due to maintenance induced causes or single ice plug failure (within the capacity of two D_2O feed pumps)	X	X	X	X	

TABLE 9. TYPICAL INITIATING EVENTS OF A SHUTDOWN PSA (cont.)

IE DESCRIPTION	POS A	POS B	POS C	POS D	POS E
Non-isolatable HTS large leak due to load drop or feeder damage from inadvertent fuelling machine movement (beyond the capacity of D_2O recovery)	X	X	X	X	
Non-isolatable rupture within the capacity of two D_2O feed pumps (initial discharge rate 1–40kg/s)					X
Non-isolatable breaks inside containment from a pressurized HTS, beyond the capacity of two D_2O feed pumps (initial discharge rate >40 kg/s)					X
Failure of liquid nitrogen supply to all ice plugs	X	X		X	
Pressure tube failures					
Pressure tube failure resulting in an initial discharge rate in excess of 1kg/s					X
Pressure tube failure resulting in an initial discharge rate of less than 1kg/s	X	X	X	X	X
STEAM GENERATOR TUBE RUPTURE					
Steam generator tube break					X
MODERATOR LOSS OF INVENTORY					
SDC HX tube break within the capacity of two D_2O feed pumps	X	X	X	X	X
SDC SYSTEM BOUNDARY					
Loss of moderator inventory	X		X	X	X
Isolatable leak in piping within the SDC system	X	X	X	X	
Isolatable break in piping within the SDC system within the capacity of D_2O feed pumps					X
Isolatable leak in piping within the SDC system					X
Adjacent unit secondary side line breaks (only for multi units)					
Adjacent unit large secondary side line break outside containment (initial discharge rate >1000 kg/s)	X	X	X	X	X
Adjacent unit intermediate steam line break outside containment (initial discharge rate 100–1000 kg/s)	X	X	X	X	X
Adjacent unit small secondary side line break outside containment (initial discharge rate 10–100 kg/s)	X	X	X	X	X
LOSS OF HEAT TRANSPORT PRESSURE AND INVENTORY CONTROL SYSTEM (LEADING TO HTS HIGH PRESSURE)					
Any HTS bleed valve fails closed					X
Any D_2O feed valve fails open					X
Bleed condenser level control valves fail closed					X
LOSS OF HEAT TRANSPORT PRESSURE AND INVENTORY CONTROL SYSTEM (LEADING TO HTS LOW PRESSURE)					
Spurious opening of both HTS liquid bleed valves					X
Spurious opening of one HTS liquid bleed valve					X
Operating D_2O feed pump fails					X
Any D_2O feed valve fails closed					X
Bleed condenser spray valve fails open					X
PIPE BREAKS IN THE PRESSURE AND INVENTORY CONTROL SYSTEM					
Pipe break in D_2O feed system upstream of check valve					X
FAILURE OF SUPPORT SYSTEMS					
Loss of bulk electricity supply	X	X	X	X	X
Loss of switchyard	X	X	X	X	X
Total loss of unit Class IV power	X	X	X	X	X
Loss of unit Class I 250 V dc buses (odd and even)	X	X	X	X	X
Adverse fore bay conditions	X	X	X	X	X
Total loss of low pressure service water	X	X	X		X
Total loss of high pressure service water	X	X	X		X
Total loss of recirculated cooling water system flow	X	X	X		X
Total loss of instrument air	X	X	X	X	X

5.2. REGULATORY REVIEW OF PSA

This section describes the results of tasks 2011-04 and 2013-02, which had the objective to look through Ref. [2] from CANDU PSA perspective to facilitate the use of Ref. [2] for regulatory review of PSA for CANDU-type reactors. Reference [2] was analysed in order to provide supplemental information for regulators in Member States operating CANDU-type reactors. Regulatory review of level 1 PSA for full power operation is presented in section 3 of Ref. [2]. CPWG analysis shows that Ref. [2] is technology neutral, provides general information on regulatory PSA review for different types of reactors and it is well applicable to the CANDU-type reactors. However, in order to enhance the use of Ref. [2] for CANDU-type reactors CPWG provided some supplemental information to assist CANDU PSA users. The details are provided in Table 10.

TABLE 10. SUPPLEMENTAL INFORMATION FOR CANDU-TYPE REACTORS

SECTION, TITLE	COMMENTS (APPLICABILITY TO CANDU-TYPE REACTORS)
3.1.1. Identification of initiating events	Reviewers need to check that a systematic procedure has been used to identify the set of initiating events for PSA. The CANDU type reactor generic initiating events have been discussed in detail in Section 3.2.1 of this TECDOC and a list of example generic initiating events for CANDU type reactors can be found in Table 4. The reviewer may include the comparison of the initiating events with the generic CANDU-type reactor initiating events for completeness.
3.1.2 Grouping of initiating events	Initiating events which cause a containment bypass are not grouped with other LOCAs where the containment would be effective. For a CANDU PSA, in addition to bypass events, it is also important that the in-core LOCAs should not be grouped with other out-core LOCAs.
3.1.3 Further guidance on initiating events	Reference [2] presents typical categories of initiating events based on PWR experience. For a CANDU-type reactor, due to different design features, the categories of initiating events are different and could be found in Table 4 of this publication (see details in Section 3.2.1).
3.1.3.1 LOCAs	For CANDU-type reactors, in addition to the success criteria of the safety systems, the location is also an important factor to be considered in the LOCA identification and grouping. The reviewer needs to ensure that in-core LOCA (failure of pressure tube and calandria tube), end-fitting failure, stagnation LOCA, fuelling machine induced LOCA, etc., are identified and grouped properly. The list of generic LOCAs for CANDU-type reactor can be found in Table 4 of this publication.
3.1.3.2. Transients	Typical examples of transients for a PWR are given in Ref. [2]. For CANDU-type reactors, due to the different design features, the typical transients are quite different. The reviewer should refer to the list of generic initiating events (Table 4 of this publication) for the categories of transients.
3.1.3.3 Loss of grid power/station blackout	CANDU-type reactors typically have two sets of onsite AC power systems, Class III Power System and Emergency Power System (EPS). The station blackout is defined as the loss of all external internal Class IV power supply, the standby Class III power supply and the EPS and considered as a consequential scenario of Loss of Offsite Power initiating event.
3.2.1. Success criteria	Reviewers should check that criteria have been developed for what constitutes core damage. The typical definitions of core damages for CANDU-type reactors are described in Section 3.1 of this publication and could be used by reviewer.
	The reviewer needs to verify carefully the justifications provided for mission times in accordance with Section 3.4.3 of this publication.
	As for the other type reactors the success criteria for a CANDU PSA typically expressed as the number of major components that need to be operated for a system or for a safety function. The example success criteria for CANDU-6 type reactors are provided in Section 3.3 and Annex IV.
3.2.2. Event sequence analysis	It is mentioned that the reviewer should check the important operator actions are adequately modelled in the event trees. Some examples of typical post-accident human interactions for CANDU-type reactors are: (i) initiation of emergency power system and emergency water for the steam generators and for the calandria; (ii) latching open steam relief valves prior to reservoir air depletion.
3.12. Results of PSAs	In addition to the general issues to be considered in the review of the level 1 PSA results, the reviewer also needs to consider CANDU specific PSA results and insights. For a CANDU PSA, the CDF (or SCDF) is usually presented as two categories FDC1 and FDC2 (see section 5.1). Due to CANDU specific characteristics and design features, the reviewer needs to check if FDC1 value is low and it makes a very small contribution to CDF (FCD1 + FCD2).

6. IMPLICATIONS OF FUKUSHIMA DAIICHI ACCIDENT TO EXTERNAL EVENT SCREENING

The accident at Fukushima Daiichi in March 2011 had broad worldwide impact in terms of prompting all Member States of the IAEA to review the safety of their NPPs to withstand beyond design basis external hazards; to re-examine the efficacy of regulatory systems to enhance emergency preparedness and, to examine if improvements to safety assessment were necessary, including scope of PSA. The Fukushima Daiichi accident showed that external natural hazards can have an impact on the safety of nuclear installations, specifically the potential combinations of these hazards and the potential multiunit impacts. The international nuclear community identified combinations of external hazards and multiunit impacts as potential areas where the PSA scope could be expanded.

The objective of task 2011-05 *External Event Screening* was to learn how the Member States conduct external hazards screening, and to determine the extent of how combinations of external hazards are being assessed in CANDU-type PSAs. The purpose of the task was to compare and learn about the different approaches adopted in each Member State for external event assessment. A questionnaire was sent to Member States of the CPWG to poll insights on their approach to external event assessment and identify common aspects. Responses to the survey were assessed, and the findings were documented in the following sections.

This section provides a summary of the External Event Screening questionnaire that was sent to all CPWG Member States. Annex VII provides detailed responses to the external event screening questionnaire that is summarized in section 6.2 and provides Canadian regulatory and industry perspectives regarding the implications of the Fukushima Daiichi accident.

6.1. EXTERNAL EVENT SCREENING

6.1.1. Questionnaire sent to participating countries

The questionnaire sent to the participating countries consisted of questions as well as supporting definitions to clarify what information was requested. The basis for most of the questions were taken from items raised in the text below, extracted from Ref. [1].

> "The bounding analysis is performed with the aim of reducing the list of external hazards subject to detailed analysis, allowing focus on the most significant accident scenarios. With this objective bounding analysis should be performed in such a way that it provides assurance that the core damage associated with the specific external hazard is insignificant compared with other hazard sources.
>
> In the bounding analysis, all potential impacts of each non-screened external hazard on the nuclear power plant should be considered.
>
> The cumulative contribution of the external hazards subject to the bounding analysis should be calculated and retained in the final results of the level 1 PSA.

Screening of hazards

At the starting point of level 1 PSA for external hazards, all available information specifically related to the plant in question should be collected. This information should include, as a minimum:

- Design information relating to external hazards as considered in the safety analysis report;
- List and layout of plant buildings, structures, systems and components;
- Plant layout and topography of the site and surroundings;
- Information on the location of pipelines, transportation routes and on-site and off-site storage facilities for hazardous materials;
- Location of industrial facilities in the vicinity of the site;
- Historical information on the occurrence of any internal and external hazards at the site, in the region, etc.

The initial information should be updated and expanded in the course of the internal and external hazards level 1 PSA preparation, depending on the necessary level of detail for the screening analysis, bounding assessment or detailed analysis for each hazard.

The task of hazard identification should aim to generate a comprehensive list of potential internal and external hazards.

Bounding analysis

The bounding estimations should be based on models and data that are either realistic or demonstratively conservative. Such models and data include:

- Assessment of the frequency of hazards (i.e. estimations of the frequency exceedance of particular intensities);
- Analysis of the impact of hazards on the plant (i.e. loads associated with the hazard);
- Analysis of the plant response (i.e. fragilities);
- Level 1 PSA models and data, etc., for the plant."

6.1.2. Results

The responses to the questionnaire sent by the participating countries are summarised in Annex VII. Some general insights from the response are listed below.

6.1.2.1. Generic list of potential external hazards

It was not possible to identify a common generic list used in the external hazard initial identification process. All countries have used some references in search of external risks. This initial identification is reported in the final safety analysis reports (FSAR). References used by Canada and Romania included, among others, the following references: IAEA Specific Safety Guide No. SSG-3 (Ref. [1]); NUREG/CR-2300 (Ref. [10]); NUREG/CR-4839 [11]; IAEA Safety Series No. 50-SG-S9 [12][5], NUREG/CR-1407 [13] and NUREG/CR-5042 [14].

[5] This publication is superseded by IAEA Specific Safety Guide No. SSG-35, but the reference to it is kept here, since it was actually used by Canada and Romania.

6.1.2.2. Methodology to identify site specific external hazards

Regarding the methodology to identify site specific external hazards, most of the countries surveyed mentioned studies of climatological, hydrology, and seismology related with site specific risk assessment; Romania also mentioned plant/site specific walkdowns as an identification methodology. Human activities in the region are taken into account if they can potentially cause some hazard. Operational experience was utilized in these studies.

6.1.2.3. Considering potential combined external hazards

Most of the countries are not considering explicitly combined events (systematic use of generic lists could avoid this weakness). However, specific cases of consequential events foreseen for site characteristics are listed, for example, low water levels in the heat sink due to dam rupture as a consequence of an earthquake, reported by Argentina. Besides consequential events, Canada and Romania consider coincidental and correlated events.

6.1.2.4. Screening criteria (qualitative and/or quantitative)

With reference to the screening criteria (and based on the responses provided by Argentina, Canada and Romania), both probabilistic and deterministic criteria are used. The screening distance value is used and it is checked whether a particular event is outside the screening distance value. Also, if the event annual frequency is below the screening frequency level then the event is screened out. Deterministic assessment is evaluated if the hazard has an impact on the ability to control, cool and contain. Furthermore, if the event has an evolution slow enough to permit handling by operating procedures, the event can be screened out.

6.1.2.5. Specific parameters used for bounding analysis

Bounding analysis is usually based on a certain range of a specific parameter. Specific parameters were used for bounding analysis for seismic, high wind, external flood hazards. Some bounding criteria included the following: horizontal peak acceleration and uniform hazard spectrum for earthquake events; and wind velocity for extreme winds and tornados. In general, a review level is defined on the basis of current regulations, guidelines, and statistical data available. The methodology considers the type of hazard, its intensity and the frequency of occurrence. In some cases, the CDF is mentioned as a screening criterion.

6.1.2.6. Human-induced hazards

Most of the countries considered these events, although the screening methodology is different from one case to another. In some cases, the probabilistic criteria are used (Canada), in other cases like in Argentina, the probabilistic criteria are replaced by distance considerations to screen out human-induced fires. In Romania, human-induced hazards were screened out if they cannot cause core damage, except for aircraft crash that was screened out based on the calculated CDF.

6.1.2.7. Specific external events and adopted approaches

In reference to earthquakes, most of the countries have developed detailed analysis, some countries have developed seismic PSA, others have adopted SMA methodologies (EPRI approach or PSA based SMA). Canada also mentioned seismically induced fires and floods studies developed with a specific PSA methodology. In reference to high winds/tornados, quantitative approaches are used to assess the impact of these events. Canada reported that the quantitative PSA approach is used.

In reference to external floods, most of the countries adopted quantitative methods tending to screen out these events through the design conditions. Also, a qualitative analysis is mentioned to exclude this type of external hazard in one specific case (Argentina). Flood safety margin is adopted in China, and flood PSA, following IAEA Safety Standards Series No. 50-P-7, Treatment of External Hazards in Probabilistic Safety Assessment for Nuclear Power Plants[6], is reported by India.

Other external events for which specific analysis has been developed are the low level in ultimate heat sink after an earthquake (reported by Argentina); extreme ambient temperatures (reported by Canada for quantitative screening); sandstorms and tsunamis (both reported by Pakistan although the approach was not reported).

6.2. OBSERVATIONS

External events are a concern not only for siting evaluation in initial licensing process, but also during the lifetime of NPPs. Most of the Member States have recently developed studies to assess their impact in safety, although the approaches and level of detail vary in each case. The following general conclusions can be made.

6.2.1. Use of generic lists of potential external hazards

While the identification of external events that affect nuclear facilities corresponds to the particular characteristics of the site where they are located, use of updated generic lists during the periodic safety review (PSR) is to be considered. Some examples are provided in Refs [1] and [8]. This has been identified as a good practice by the CPWG, and Member States need to continue consulting relevant updated or new publications in the field.

6.2.2. Considering potential combined external hazards

Accounting for potential threat of this type of combined external events, re-evaluation of the possibility of such events may be considered during PSR updates.

6.2.3. Screening methodology

Methodologies and screening criteria are periodically reviewed to ensure that they are in line with current knowledge.

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Treatment of External Hazards in Probabilistic Safety Assessment for Nuclear Power Plants: A Safety Practice, Safety Series No. 50-P-7, IAEA, Vienna (1995) no longer valid.

APPENDIX I. STATUS OF CANDU PSA

I.1. LEVEL 1 PSA

The information provided in this section on the status of CANDU level 1 PSA applies to screened-in hazards and includes the use of alternative methods as allowed by the applicable regulatory requirements or use of bounding assessments.

	ARGENTINA	CANADA	CHINA	INDIA	REPUBLIC OF KOREA	PAKISTAN	ROMANIA
INTERNAL EVENTS							
Full power	Y[a]	Y	Y	Y	Y	Y	Y
Low power	Y	N[b]	Y	N	Y	N	N/A[c]
Shutdown	Y	Y	Y	Y	Y	N	Y
INTERNAL FIRES							
Full power	Y	Y	N	Y	Y	Y	Y
Low power	N	N	N	N	Y	N	N/A
Shutdown	N	bounded	N	Y	Y	N	Y
INTERNAL FLOODS							
Full power	N	Y	N	Y	Y	N	Y
Low power	N	N	N	N	Y	N	N/A
Shutdown	N	bounded	N	N	Y	N	Y
EXTERNAL FLOODS							
Full power	N/A	N/A	N	Y	N/A	N/A	N
Low power	N/A	N	N	N	N/A	N	N/A
Shutdown	N/A	N	N	N	N/A	N	N
SEISMIC							
Full power	Y (SMA)	Y (SMA and SPSA)	Y (SMA)	Y	Y (SMA/SPSA)	SMA IN PROGRESS	Y (SPSA)
Low power	N	N	N	N	Y (SMA/SPSA)	N	N/A
Shutdown	N	Bounded	N	N	Y (SMA/SPSA)	N	Y
HIGH WINDS							
Full power	Y	Y	N	N/A	N/A	N	N
Low power	N	N	N	N/A	N/A	N	N/A
Shutdown	N	bounded	N	N/A	N/A	N	N
OTHER (SPECIFY)							
Spent fuel bay (qualitative assessment)	Y	Y	Y	Y	Y	N	Y
Seismically induced dam failure	Y (qualitative assessment)	N/A	N/A	N	N/A	N/A	N/A

[a] Y – yes

[b] N – not considered

[c] N/A – screened out

I.2. LEVEL 2 PSA

The information provided in this section on the status of CANDU level 2 PSA applies to screened-in hazards and includes the use of alternative methods as allowed by the applicable regulatory requirements or use of bounding assessments.

	ARGENTINA	CANADA	CHINA	INDIA	REPUBLIC OF KOREA	PAKISTAN	ROMANIA
INTERNAL EVENTS							
Full power	Y[a]	Y	On-Going	Y	Y	N	Y
Low power	N[b]	N	N	N	Y	N	N/A[c]
Shutdown	N	Y	N	N	Y	N	Y
INTERNAL FIRES							
Full power	N	Y	N	N	Y	N	Y
Low power	N	N	N	N	N	N	N/A
Shutdown	N	bounded	N	N	N	N	Y
INTERNAL FLOODS							
Full power	N	Y	N	N	Y	N	Y
Low power	N	N	N	N	N	N	N/A
Shutdown	N	bounded	N	N	N	N	Y
EXTERNAL FLOODS							
Full power	N	N/A	N	N	N/A	N	N
Low power	N	N	N	N	N/A	N	N/A
Shutdown	N	N	N	N	N/A	N	N
SEISMIC							
Full power	N	Y	N	N	Y	N	Y
Low power	N	N	N	N	N	N	N/A
Shutdown	N	bounded	N	N	N	N	Y
HIGH WINDS							
Full power	N	Y	N	N	N/A	N	N
Low power	N	N	N	N	N/A	N	N/A
Shutdown	N	bounded	N	N	N/A	N	N
OTHER							
Spent fuel bay (qualitative assessment)	N	N	Y	N	N	N	Y

[a] Y – yes

[b] N – not considered

[c] N/A – screened out

I.3. LEVEL 3 PSA

ARGENTINA	CANADA	CHINA	INDIA	REPUBLIC OF KOREA	PAKISTAN	ROMANIA
N	Y (not up to date) *	N	N	N	N	N

* In the early days of PSA development in Canada (in the 1980s–1990s), a limited scope of level 3 PSAs were conducted on a voluntary basis. More recent PSAs are conducted for level 1 and level 2 in accordance with CNSC regulatory documents S-294 and REGDOC 2.4.2.

APPENDIX II. LIST OF CPWG TASKS

Appendix II lists all tasks implemented by the CPWG in relation to level 1 PSA for CANDU-type reactors and summarized in this publication.

TASK#	TITLE OF THE TASK	CYCLE 1: 06.2010–02.2011	CYCLE 2: 03.2011–12.2012	CYCLE 3: 07.2013–07.2014	CYCLE 4: 07.2014–07.2015	CYCLE 5: 07.2015–07.2016	CYCLE 6: 07.2016–07.2017	CYCLE 7: 10.2017–10.2018
2010-01	Develop a common acceptable definition of core damage and large (early) release frequency	X						
2010-02	Further input on classification and description of LOCAs, rationale for group of LOCA IEs and LOCA IE frequencies	X	X	X				
2010-03	Develop framework for generic database for use in CANDU-type PSA	X	X					
2010-04	CCF data collection	X						
2010-05	Comparison of different standards used by CANDU-type countries	X						
2010-06	Definition of mission time	X						
2010-07	Definition of success criteria for all mitigating and support systems in CANDU 6 PSA for all accident scenarios	X	X	X				
2010-08	Treatment of zero occurrences initiating events	X						
2010-09	Probabilistic risk measures for different PSA applications (e.g. 95th percentile Vs. Mean). C6 and NUREG/CR-1860	X	X					
2011-01	Comparison of L(E)RF definitions		X	X				
2011-02	Identify specific/generic modifications, if any to reduce the overall risk - comparison (Small LOCA ET structure, description of ET headers, success criteria etc.)		X	X				
2011-03	Assessment of applicability to SSG-3 & SSG-4		X	X				
2011-04	Internal guidelines for regulatory review of PSA (level 1, internal events, full power)		X					
2011-05	External events (screening methodology, fire, seismic, flooding, etc.)		X	X				
2011-06	Fukushima – lessons learned		X	X				
2013-01	Development of CANDU-type specific level 1 PSA examples for consideration in SSG-3			X	X	X	X	
2014-01	Development of examples for consideration in Safety Reports Series no. 25 for guidelines on regulatory review of level 1 PSA				X	X	X	X
2014-02	Development of CANDU-type specific level 2 PSA examples for consideration in SSG-4				X	X	X	X
2014-03	Identification of regulatory requirements and regulatory review of NPP PSAs				X			
2014-04	Risk Insights from level 1 PSA				X	X	X	
2015-01	Development of examples for consideration in Safety Reports Series no. 25 for guidelines on regulatory review of level 2 PSA					X	X	X

TASK#	TITLE OF THE TASK	CYCLE 1: 06.2010–02.2011	CYCLE 2: 03.2011–12.2012	CYCLE 3: 07.2013–07.2014	CYCLE 4: 07.2014–07.2015	CYCLE 5: 07.2015–07.2016	CYCLE 6: 07.2016–07.2017	CYCLE 7: 10.2017–10.2018
2015-02	Identification of level 2 PSA tasks					X	X	X
2015-03	Level 1 PSA – low power and shutdown PSA					X	X	X
2016-01	Identification of sections for consideration in the IAEA Safety Reports Series No. 25 for guidelines on regulatory review of level 2 PSA						X	X
2016-02	2016-02.1 Plant damage state definitions and the strategy of grouping level 1 sequences 2016-02.2 Containment failure modes and containment structural performance analysis methods 2016-02.3 Structure of accident progression event tree (APET) or Containment Event Tree (CET) 2016-02.4 Definitions and major nodal questions (major phenomena of severe accident progression)						X	X
2016-03	2016-03.1. Level 2 PSA end states definition and release category definitions 2016-03.2. Human reliability analysis, including the use of emergency operating procedure (EOP) and severe accident management program (SAMG) in the level 2 PSA 2016-03.3. The use and application of level 2 PSA results and/or insights						X	X
2016-04	Comparison of regulatory review process of PSA (required input from India)						X	
2016-05	Treatment of zero occurrence initiating events						X	
2016-06	SBLOCA						X	
2017-01	Computer codes used for the simulation of the accident progression, fission products behaviour in the heat transport system and in the containment							X
2017-02	Strategy of quantification of APET or CET end states							X
2017-03	Hydrogen and other explosive or flammable gases production during the severe accident progression							X
2017-04	Uncertainty/ sensitivity/ importance analysis methods for accident progression simulation and for release category quantification							X
2017-05	Off-site consequence analysis							X
2017-06	Lessons learned from previous CPWG meetings and the development of the TECDOC on CANDU-type level 1 PSAs							X
2017-07	PSA applications (selection of a few applications to be harmonized for level 1 and level 2)							X

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ANNEX I. REGULATORY REQUIREMENTS AND REVIEW OF PSA

I-1. BACKGROUND

Task 2014-03 was identified in the third CPWG meeting. A questionnaire was prepared to collect information from Member States regarding the regulatory requirements related to PSA, its review processes and the computer codes used for PSA analysis. In general, the regulatory requirements and review processes of different countries may vary from each other based on their legislation process, public involvement, and economic conditions. Important topics from the questionnaire were regulatory requirements related to PSA submission, scope of PSA, PSA acceptance criteria and the new requirements after Fukushima. This annex summarizes the regulatory requirements and review processes in countries operating CANDU-type reactors.

Moreover, regarding the use of computer codes for PSA, a number of verified and validated computer codes and software packages are currently available for performing a PSA. Typically, an integrated software package is used in the level 1 PSA analyses for the development and storage of system models, accident sequence models, failure data, and accident sequence quantification. Additionally, other computer codes may be used for the development of success criteria, but these are not discussed in this annex.

This annex summarizes the status of regulatory requirements and review of PSA in the Member States and briefly shows the responses from all Member States in the form of a comparison table developed for each question.

I-2. SUMMARY RESULTS OF THE SURVEY

I-2.1. Question 1: Is there any regulatory requirement regarding submission of PSA?

Almost all the countries require level 1 and level 2 PSA submissions, except for Pakistan, which only requires level 1 PSA as part of the licensing process.

MEMBER STATE	RESPONSE
Argentina	Level 1 and level 2 PSA. Additionally, some level 3 PSA focused on obtaining individual effective doses corresponding to the different accident sequences resulting in releases to environment.
Canada	Level 1 and 2 PSA for full power, shutdown and other states where the reactor is expected to operate for extended periods of time that are not covered by the full power and shutdown PSAs and for internal and external hazards.
China	Level 1 and level 2 PSA for full power, shutdown and low power internal events
Republic of Korea	New legislations (2015) for severe accidents and PSA are: 1. For an existing plant, the licensee needs to submit following updated PSA (by 2019): a. Full power /level 1, 2 / internal, external events b. Low power Shutdown / level 1 / internal, external events 2. The licensee is required to submit following reports for a new plant: a. Full power /level 1, 2/ internal and external events at preliminary safety analysis report stage. b. Full power /level 1, 2, 3 / internal, external events and, low power shutdown / level 1, 2/ internal, external events at final safety analysis report stage.
Pakistan	Level 1 PSA for internal at power events, low power and shutdown, internal and external hazards (fire, flooding, seismic, etc.)
Romania	Level 1 and level 2 PSA
India	For operating NPPs, level 1 PSA is mandatory, and level 2 PSA is desirable. For new NPPS, level 1 and level 2 PSA are regulatory requirements.

I-2.2. Question 2: What is the target core damage frequency value for acceptance of the PSA?

i. Argentina does not define any target CDF value for acceptance of PSA.
ii. Canada, China and the Republic of Korea have same target CDF values for operating and new NPPs respectively.
iii. India and Pakistan have defined only one CDF target value for all NPPs. Similarly, Romania also has single target CDF value, but is higher than India and Pakistan.

MEMBER STATE	RESPONSE
Argentina	There are no formally established criteria for CDF and large early release frequency (LERF). Criteria related to dose limit is defined in the form of a graph in standard AR 3.1.3 Ref. [I-1].
Canada	New NPPs: CDF < 1E-5/yr Existing NPPs: The explicit safety goals for existing plants are not given in the regulatory documents. The acceptable safety goals are: CDF < 1E-4/yr
China	Operating NPPs: CDF < 1.0E-4/yr New NPPs: CDF < 1.0E-5/yr
Republic of Korea	As per requirement of KINS/RS-16.0 (Rev. 2) Ref. [I-2] CDF for operating plants < 1.0E-4/yr CDF for new plants (after Shinkori Unit 3) < 1.0E-5/yr
Pakistan	CDF ≤ 1.0E-5/yr
Romania	CDF < 1.0E-4/yr
India	CDF < 1.0E-5/yr

I-2.3. Question 3: Is there any regulatory requirement related to PSA at construction stage, such as in the preliminary safety analysis report (PSAR) stage?

i. Most countries require level 1 and level 2 PSA submissions at construction stage.
ii. Argentina and Pakistan require preliminary results of the PSA and design PSA (full power internal events) respectively.
iii. China requires the submission of spent fuel bay PSA.

MEMBER STATE	RESPONSE
Argentina	For the construction stage, at least the preliminary results of the PSA are required. The level of detail needs to follow the current practice described in international guidelines such as Ref. [I-3].
Canada	Canada has PSA requirements for the construction stage outlined in RD/GD-369 Section 7.7 Ref. [I-4], which indicates that the PSA needs to meet the expectations of S-294. The scope of the PSA is level 1 and 2 PSA for full power and shutdown states for internal and external events.
China	The requirement for PSA at the construction stage is same as HAF102 Code Ref. [I-5] and HAF103 Code (Rev. 2004) Ref. [I-6]. The scope of PSA, level of detail and soft model are not detailed in these regulatory documents. However, these detailed requirements are used by the Chinese regulatory body the National Nuclear Safety Administration (NNSA) during the PSAR stage. The current practices are as the following: • Level 1 PSA (full power and shutdown/ low power, internal event); • Level 2 PSA; • Spent fuel bay PSA.
Republic of Korea	Full power / level 1 and 2 / for internal events and internal & external hazards
Pakistan	PSA of full power internal initiating events
Romania	Level 1 and 2 PSAs are required at the design stage, as a minimum, in order to demonstrate the fulfilment of the quantitative objectives. In chapter 15 of Safety Analysis Report (PSAR/FSAR) the Romania National Commission for Nuclear Activities requires both deterministic and probabilistic analyses. Annex IV of NSN-02 Ref. [I-7] requires: Chapter 15 includes information regarding PSAs, as: - Methodology, computer codes and guides used to develop PSA; - Levels 1 and 2 PSAs performed and documented in accord with requirements of CNCAN norms; - A program for PSA updating in the stages of construction and design. Level 1 PSA contains at least: a) A general description of the plant; b) Identification, description and grouping of the initiating events; c) Definition of the success criteria for the operation of the systems and for the actions of operators considered in analysis; d) Modelling of accident sequences; e) Reliability analyses of the systems with safety functions; f) Human reliability analysis; g) Analysis of the data used in estimation of the core damage frequency; h) Common cause failure analysis; i) Uncertainty and sensitivity analysis; j) Quantification and interpretation of the results. Level 2 PSA contains at least: a) Establishing of the interface with level 1 PSA; b) Core damage analysis and the accident progress in time; c) Containment behaviour analysis and estimation of frequencies associated with accident sequences that lead to the loss of containment functions; d) Source term analysis; e) Reliability analyses for the systems with a role in mitigation of severe accidents; f) Uncertainty and sensitivity analysis; g) Presentation and interpretation of the results. Depending on the area of PSA results application, criteria used in decision making will be documented by the licensee and submitted to CNCAN for approval.
India	For NPPs under design/construction (PSAR stage), level 1 and 2 PSAs are regulatory requirements. As a minimum requirement, the plant have to carry out a level 1 PSA for internal and external events, as applicable to the plant. Shutdown and low power PSAs have to also be performed to obtain risk insights from these plant states.

I-2.4. Question 4: What is the requirement of PSA submission before issuance of fuel load permit (final safety analysis report stage)?

i. The Republic of Korea requires detailed PSA submissions i.e. level 1 and 2 for full power, low power and shut down internal and external events. Additionally, the Republic of Korea requires a level 3 PSA for full power internal events and external hazards.
ii. Romania requires submission of level 1, level 2 and limited level 3 PSAs.
iii. Argentina and Pakistan require only level 1 PSA.
iv. China requires spent fuel bay PSA instead of level 3 PSA. However, Canada and India do not have explicit requirement for this stage as per regulation.

MEMBER STATE	RESPONSE
Argentina	For commissioning stage, final results of PSA are required. The level of detail needs to follow the current practice described in international guidelines such as Ref. [I-3]. It is required to consider the equilibrium core inventory at full power operation conditions. Before issuance of fuel permit, a full power PSA is required (reactor core is considered at 100% of power), taking into account appropriate deterministic studies.
Canada	Canada has the requirements related to PSA submissions, but not explicitly stated in the regulatory document.
China	Same as the PSAR stage i.e. level 1 PSA, level 2 PSA and spent fuel bay PSA.
Republic of Korea	Full power / level 1, 2, 3 / for internal events and internal & external hazards. Low power shutdown / level 1, 2 / for internal events and internal & external hazards.
Pakistan	Full scope level 1 PSA, which includes an assessment of internal initiating events in full power operating conditions, low power and shutdown modes for internal and external hazards such as fire, flood earthquakes, etc.
Romania	Romania requires the submission of level 1, level 2 and limited level 3 PSAs in order to demonstrate the compliance with the safety goals established at the boundary of the exclusion zone.
India	There is no specific requirement related to PSA submission before the fuel load permit.

I-2.5. Question 5: Is there any regulatory requirement related to PSA at periodic safety review stage?

All countries require the review of the PSA at the PSR stage except for Romania. However, Romania requires the licensee to maintain a living PSA process for continuous update of PSA model.

MEMBER STATE	RESPONSE
Argentina	PSA is mandatory and included in the Operational License (OL) of NPPs. After every 10 years, it is necessary to review all the documentation included in the OL. Particularly, the information related with PSA is reviewed, taking into account that it is an item included as a safety factor in applicable documentation like, periodic safety review of nuclear power plants (Ref. [I-8]). Following this publication, review and upgrade of PSA is focused on: • Increasing amount of detail of the model; • Upgrade considering design and procedure modifications; • Application of study to decision making.
Canada	Canada does have requirement to review PSA in the safety factor report.
China	There is no detailed requirement about the scope or level of PSA; however, the following elements are needed for reviewing: • Existing PSA and its assumptions; • Updating of PSA to reflect the current plant status; • Postulated initiating events (for the existing PSA and a comparable list for a modern nuclear power plant); • Analytical methods and computer codes used in the existing PSA and comparable methods for a modern nuclear power plant, including validation; • Guidelines for PSA of operator action, common cause events, cross-link effects, redundancy and diversity; • Consistency of the accident management program for beyond design basis accidents with PSA results.
Republic of Korea	The licensee needs to submit PSA in PSR. The existing PSA needs to be updated before submission.
Pakistan	An updated full scope level 1 PSA is an important element of the periodic safety review. Keeping in mind the previous submissions, the level of detail and scope of PSA is finalized after discussion with the licensee as per current practice.
Romania	In accordance with the national regulation the licensee in Romania is required to maintain to implement a living PSA process for continuous update of the PSA model.
India	PSA is required to be revised / updated with plant specific operational experience as part of PSR. Level 1 PSA is revised during the PSR stage (every 10 years) and ARA (every 5 years).

I-2.6. Question 6: Is there any regulatory requirement related to level 2 PSA or any target value related to large early release frequency?

All countries except Argentina and Romania have defined regulatory targets for level 2 PSA. However, Argentina and Romania have defined some dose criteria.

MEMBER STATE	RESPONSE
Argentina	The regulatory standard AR 3.1.3 Ref. [I-1] is a probabilistic criterion to be applicable during the licensing process for new NPPs. For level 2 PSA, this criterion is focused on individual doses due to some hypothetical accidental sequences. In this assessment, it is not evaluated any case as a contributor to LERF (i.e. as radioactivity released). In general, all the sequences fulfil the criteria in the plot frequency vs. dose.
Canada	**For New NPPs** LRF <1E-6/yr ; small release frequency < 1E-5/yr **For Existing NPPs:** The explicit safety goals for existing plants are not given in the regulatory documents. However, acceptable safety goals are: LRF <1E-5/yr
China	There is no detailed requirement related to level 2 PSA. The target value is as follows: • LERF < 1.0E-5/yr (for the operating NPPs); • LERF < 1.0E-6 /yr (for the new NPPs).
Republic of Korea	In KINS/RS-16.0 (Rev. 2) Ref. [I-2]: • LERF for operating plants < 1.0E-5/yr; • LERF for new plants (after Shinkori unit 3) < 1.0E-6/yr.
Pakistan	Large early release frequencies ≤ 1.0E-6/yr.
Romania	There are no specific requirements related to LERF, but quantitative safety objectives are defined in NSN-2 Ref. [I-7] norms, which require limits for doses to population for each event class (classes are established based on initiating events or event sequences frequencies of occurrence).
India	Target for LERF < 1.0E-6/Rx/yr.

I-2.7. Question 7: What are the new regulatory requirements related to the PSA after the Fukushima Daiichi accident?

i. The common regulatory requirements related to PSA after Fukushima Daiichi accident in most of the countries are:
   a. The inclusion of spent fuel bay in the assessment;
   b. Updating of existing studies to include extreme external events.
ii. Argentina formalized a stress test and set regulatory requirements based on this test.
iii. India has introduced deterministic reviews like 'stress test' and required modifications for all NPPs (design stage as well as operating).
iv. Canada has also set the requirement of inclusion of multiunit station impact and consideration of hazard combinations.
v. Pakistan has asked the licensee to submit the results of probabilistic seismic hazard analysis for some NPPs and seismic PSA for new NPPs.

MEMBER STATE	RESPONSE
Argentina	As it is presented in Ref. [I-9] the Argentine Regulatory Authority formalized the stress test sending a regulatory requirement to the licensee of Embalse NPP. This requirement consists of a reassessment of the NPP safety margins assuming the occurrence of a sequential loss of the lines of defence in depth caused by extreme initiating events and includes: • The design basis and licensing basis compliance review. • The extreme initiating events conceivable at the NPP site. • The loss of safety functions caused for each one of the extreme initiating events considered; • Arrangement / disposal of structures, systems and components (SSCs) belonging to safety systems to assure they can continue fulfilling the corresponding safety function. • The severe accident management program corresponding to each one of the extreme initiating events considered. • The long term evolution of the severe accidents and the recovery capability of both the power supply and the water supply until a stable plant condition are reached. This is to identify the most adequate recovery strategies and the components that are available for each of the corresponding strategy implementation. • Spent fuel storage management strategy and spent fuel storage systems design and performance. • Prevention, recovery and mitigation measures: automatic and operator actions for abnormal conditions; severe accident management and emergencies. Although most of the studies used are deterministic, knowledge of the nuclear facility provided by the PSA was used in all cases.
Canada	The post Fukushima requirements include: • Inclusion of multiunit station impacts; • Inclusion of other radioactive sources in the analysis; • Consideration of hazard combinations.
China	Following are the requirements related to the PSA after the Fukushima incident: • Level 2 PSA; • Spent fuel bay PSA.
Republic of Korea	In June 2015, the Nuclear Safety Act was amended to enhance the regulatory framework on severe accidents. And the regulatory body in the Republic of Korea revised the rules, regulatory standards and regulatory guides for severe accidents and PSA. The main changes were as follows: • PSA will be submitted by the law, not administratively. • All existing plants' PSA needs to be updated and submitted until June 2019. • PSA scope, which needs to be submitted, is specified based on regulatory standard of KINS. • As a QHO (quantitative health objective), quantitative target is included in the notice of Nuclear Safety and Security Commission (NSSC). The risk to an individual in the vicinity of NPP need not to exceed all the risk from other reason. • For considering environment, quantitative environment objectives (QEO) quantitative target are included in Notice of NSSC (i.e. the sum of frequencies of all event sequences that can lead to release to the environment of more than 100 TBq of Cs-137 needs to be less than 1.0E-6/yr).

MEMBER STATE	RESPONSE
Pakistan	PNRA required the licensee to submit the 'Fukushima Response Action Plan' with short, medium and long term targets. The requirements related to PSA are: • Re-assessment of natural hazards (i.e. seismic, flood, tsunami, harsh environment, wind, tornado); • Probabilistic seismic hazard analysis; • Seismic PSA.
Romania	No new specific requirements have been issued formally by CNCAN regarding PSA, but CNCAN developed the National Action Plan based on conclusions and recommendations of the stress tests. This plan contained an activity regarding PSA updating, in order to include events at the spent fuel bay of Cernavoda NPP.
India	No new specific requirements related to the PSA have been issued. However, deterministic reviews including the 'stress test' and required modifications have been introduced for all NPPs (design stage as well as operating). PSA studies with respect to external events, low power and shut down are taken on priority.

I-2.8. Question 8: How many person-hours required for the review of PSA?

All the countries require different time periods for PSA review ranging from 500–15 000 person-hours. The average time required for PSA review is about 7000 person-hours.

MEMBER STATE	RESPONSE
Argentina	2000 to 3000 person-hours for internal at power events PSA
Canada	1000 to 15 000 person-hours
China	Approximately 2500 person-hours
Republic of Korea	The person-hours required for the regulatory review of each PSA has not been exactly assessed. However, it is estimated that 500–2500 person-hours are required for each review process. Recently, there have been 2–5 review processes are ongoing recently.
Pakistan	5000–6000 person-hours for internal at power events PSA
Romania	Approximately 4000 person-hours
India	Approximately 10 000–15 000 person-hours

I-2.9. Question 9: Does the regulatory body request the utility to submit an electronic copy of the PSA for review purpose?

The regulatory bodies of all the countries request the submission of an electronic copy of the PSA for review purpose.

MEMBER STATE	RESPONSE
Argentina	Usually an electronic copy is requested to be used during the review work
Canada	Due to the large size of the PSA reports, they are submitted electronically to the regulator
China	Yes
Republic of Korea	Yes
Pakistan	Yes
Romania	Yes
India	Yes

I-2.10. Question 10: Is there any regulatory guidance provided to the licensee regarding submission of PSA?

i. Argentina and Romania do not provide any specific regulatory guidance to the licensee regarding submission of PSA. They ask for the PSA submission as per international reference publications such as IAEA Safety Series No. 50-P-4 Ref. [I-10].
ii. In Canada, the PSA guides are developed by the licensees and are submitted to the regulator for acceptance and use in the development of PSA. Regulator does not issue any regulatory guide.
iii. All other countries provide their own regulatory guides to the licensees for PSA submissions.

MEMBER STATE	RESPONSE
Argentina	Argentina does not have its own guide to carry out the PSA study, but always requests the use of applicable international references. For example, Refs [I-10] and [I-11].
Canada	The format and content of the PSA documentation is included as part of the PSA methodology, which has to be submitted for acceptance by CNSC staff prior to the conduct of the PSA.
China	Yes, a standard PSA report format and content (level 1 internal event PSA) is issued (HAF. J0088).
Republic of Korea	KINS Regulatory Standards: • KINS/RS-N16.0(rev2), Probabilistic Safety Assessment Ref. [I-2]. KINS Regulatory Guides: • KINS/RG-N16.03, Level 1 Internal PSA Ref. [I-12]; • KINS/RG-N16.04, Level 1 External PSA Ref. [I-13]; • KINS/RG-N16.05, Level 2 PSA Ref. [I-14]; • KINS/RG-N16.06, Level 3 PSA Ref. [I-15]; • KINS/RG-N16.07, General approach in risk informed decisions on plant-specific changes to the licensing basis Ref. [I-16]; • KINS/RG-N16.08, An approach for plant-specific, risk informed decision making: technical specification Ref. [I-17]. KINS Safety Review Guides: • KSRG 19.1, Probabilistic Safety Assessment (revision 4) Ref. [I-18].
Pakistan	Probabilistic safety assessment of nuclear power plant level 1 (PNRA-RG-911.01) Ref. [I-19].
Romania	No, except for the requirements of the minimum content of level 1 and level 2 PSAs.
India	Yes, regulatory guidance is available regarding submission of PSA, including format and contents.

I-2.11. Question 11: Does any internal working procedure exist for the review of PSA?

All the countries use their own internal working procedures for the review of PSA except for Argentina and Pakistan. Argentina and Pakistan only use applicable international standards such as Refs [I-3], [I-10] and [I-20], etc. for the same purpose.

MEMBER STATE	RESPONSE
Argentina	Although we do not have a specific procedure for the review task, normally we use the applicable references from the IAEA.
Canada	CNSC staff developed internal review procedures based on IAEA publications: IAEA-TECDOC-1135 Ref. [I-21] for level 1 and IAEA-TECDOC-1229 Ref. [I-22] for level 2 PSA. CNSC staff also uses international best practices, such as the ASME PRA Standard Ref. [I-23] and other publications listed in the accepted methodology.
China	Yes, Chinese Nuclear Energy Association has issued a working procedure for review of PSA.
Republic of Korea	Yes. The KINS uses KSRG Ch.19 (KINS Safety Review Guides Ch. 19-1 Review guideline for PSA) Ref. [I-18] for regulatory review of PSA.
Pakistan	There is no specific procedure for the review of PSA. Generally, the applicable international standards such as Ref. [I-3] or Safety Reports Series No. 25 Ref. [I-20] are used for this task.
Romania	Yes, CNCAN developed and applied the internal procedures for level 1 and level 2 PSA review, which also contain guidance in annexes.
India	Yes. It is as per AERB guide SG-G-10 Ref. [I-23].

I-2.12. Question 12: What are the PSA applications for the regulatory use?

Some examples of the PSA applications for the regulatory use in different countries are:

i. Verify the effect over the total risk (for example observing shift in parameter CDF) due to some permanent or temporary design modifications.
ii. Event analyses, change in maintenance/test interval, incremental risk analysis, etc.
iii. Use of PSA as a tool in daily risk control and management.
iv. Check severe accident vulnerabilities of NPP and confirm that the quantitative risk level of the NPP is sufficiently low enough to meet QHO (or surrogate CDF, LERF) and QEO.
v. Verify that a balanced design has been achieved that no particular feature or postulated initiating events makes a disproportionately large or significantly uncertain contribution to the overall risk.
vi. Assessment of the adequacy of plant emergency procedures.
vii. Verify that small deviations in plant parameters that could give rise to severely abnormal plant behaviour (cliff edge effects) will be prevented.
viii. Assessments of the probabilities of occurrence of severe core damage states and assessments of the risks of major off-site releases necessitating a short term off-site response, particularly for releases associated with early containment failure, etc.
ix. For risk informed regulatory decision making in review of new designs, system design upgrade/modification, and revision of technical specifications, etc.

MEMBER STATE	RESPONSE
Argentina	The standard for licensing new plants requires level 1, 2 and 3 PSA studies (restrained only to the estimation of doses over exposed population). In the case of plants in operation, level 1 PSA is a mandatory part of the license. Also, PSA is used to assess the risk changes (for example, changes in CDF) due to some permanent or temporary design modification
Canada	Some applications include event analyses, change in maintenance/test interval, incremental risk analysis, etc
China	In 2010, the technical policy of the PSA application in nuclear safety was issued to encourage the utilities to make the PSA an important and useful tool in daily risk control and management
Republic of Korea	Based on the regulatory review process for the PSA, staff checks all accident vulnerabilities of the NPP and confirms that the quantitative risk level of the NPP is sufficiently low enough to meet QHO (or surrogate core damage frequency (CDF), large early release frequency (LERF)) and QEO
Pakistan	The PSA applications for the regulatory use in PNRA are: • To verify that a balanced design has been achieved such that no particular feature or postulated initiating events makes a disproportionately large or significantly uncertain contribution to the overall risk, and that the first two levels of defence in depth bear the primary burden of ensuring nuclear safety; • Assessment of the adequacy of plant emergency procedures; • To verify that small deviations in plant parameters that could give rise to severely abnormal plant behaviour (cliff edge effects) will be prevented; • Assessments of the probabilities of occurrence of severe core damage states and assessments of the risks of major off-site releases necessitating a short term off-site response, particularly for releases associated with early containment failure, etc.
Romania	CNCAN uses PSA results of Cernavoda NPP for licensing of the two units in operation and for decision making regarding plant modifications.
India	PSA results are applied for risk informed regulatory decision making in review of new designs, system design upgrade/modification, revision of technical specification etc.

I-2.13. Question 13: How are the PSA insights used in regulatory decision making?

All the countries are using PSA insights for regulatory decision making in various ways, for example:

i. Argentina has extended the testing periods of some safety systems from 12 to 18 months.
ii. Canada utilizes PSA results and insights for licensing, identification of generic safety issues, identification of plant modification opportunities, identification of safety important systems for reliability program, training of inspectors, etc.
iii. China's process of using PSA insights is in progress. Only some PSA pilot applications such as AOT and STI extension are expected to be approved by NNSA.
iv. The Republic of Korea has improved the safety level in licensing process of new NPPs.
v. Pakistan, Romania and India use PSA insights to verify the compliance with regulatory targets, decision making in plant modifications and relicensing of operating plants during the PSR.

MEMBER STATE	RESPONSE
Argentina	The Embalse NPP PSA results were used to extend the testing period for some safety systems from 12 to 18 months.
Canada	In Canada, PSA results and insights are used for licensing, identification of generic safety issues, identification of plant modification opportunities, identification of safety important systems for the reliability program, training of inspectors, etc.
China	The risk informed regulatory framework is still underway. Only some PSA pilot applications such as Allotted Outage Time (AOT) and Surveillance Test Intervals (STI) extension are expected to be approved by NNSA.
Republic of Korea	PSA Insights are used to determine the safety improvement level in the licensing process for a new NPP. If the licensee submits a risk informed application (i.e. risk informed technical specifications amendments), staff need to review the PSA insights to justify that change.
Pakistan	PSA insights are primarily used for verification of compliance with regulatory targets for regulatory decision making
Romania	PSA insights are used in regulatory decision making for plant modifications.
India	PSA insights are used for regulatory decision making for review of new designs, system design upgrade/modification, and revision of TS. PSA insights are used also for licensing during PSAR stage and relicensing of operating plants during the periodic safety review.

I-2.14. Question 14: Which computer code has been used in the performance of level 1 PSA? List any requirement related to validation and verification of codes?

i. China, Argentina Pakistan and India are using Risk Spectrum for level 1 PSA. Additionally, China also uses CAFTA for CANDU PSA.
ii. Canada uses CAFTA, FTREX, ACUBE and FRANX and Romania uses CAFTA, ETA, PRA Quant, One4All, QRECOVER, FORTE.
iii. In the Republic of Korea, the regulatory body uses AIMS code (Quantification engine: FTREX) and the utility uses SAREX code (Quantification engine: FORTE) for level 1 PSA.

MEMBER STATE	RESPONSE
Argentina	The code used is Risk Spectrum.
Canada	The computer codes used include CAFTA, FTREX, ACUBE, and FRANX. Validation and verification of the codes is done by EPRI. In addition, new code versions are checked by vendors prior to use.
China	CAFTA is used for the CANDU PSA model, and Risk Spectrum is used for other PWR PSA model.
Republic of Korea	The regulatory body uses AIMS (Quantification engine: FTREX) code and the utility uses SAREX code (quantification engine: FORTE) for level 1 PSA. The AIMS is supported by KAERI and vendor of SAREX is KEPCO E&C (Korea Electric Power Co. Engineering & Construction).
Pakistan	Risk Spectrum PSA Professional.
Romania	The following codes have been used in the developing of PSA level 1 for Cernavoda NPP: CAFTA, ETA, PRA Quant, One4All, QRECOVER, FORTE. CNCAN norms NMC-12 require that all computer codes used for design and analysis of nuclear installation to be verified and validated.
India	Risk Spectrum software is used for level 1. The regulatory requirement is brought in AERB safety guides.

I-2.15. Question 15: Which publications are used for PSA review?

i. Following international publications are used for review of PSA:
   a. IAEA SSG-3;
   b. IAEA Safety Series No. 50-P-4;
   c. IAEA Safety Series No. 50-P-10;
   d. IAEA Safety Reports Series No. 25;
   e. IAEA INSAG-12;
   f. IAEA-TECDOC-1135;
   g. ASME/ANS RA-SA-2009;
   h. NUREG/CR-581-CCF;
   i. NUREG/CR-4772-HEP;
   j. IAEA-TECDOC-478;
   k. IAEA-TECDOC-592;
   l. NUREG/CR-0492.

ii. Romania uses their own procedure based on a selection of standards and guides referenced in each procedure.

MEMBER STATE	RESPONSE
Argentina	IAEA Safety Reports Series No. 25 Ref. [I-20] and IAEA Specific Safety Guide No. SSG-3 Ref. [I-3]
Canada	IAEA TECDOC-1135 Ref. [I-21] for level 1 PSA and TECDOC-1229 Ref. [I-22] for level 2 PSA
China	IAEA Safety Series No. 50-P-4 Ref. [I-10], IAEA Safety Reports Series No. 25 Ref. [I-20] and ASME/ANS RA-Sa-2009 Ref. [I-23]
Republic of Korea	IAEA SSG-25 Periodic Safety Review of Nuclear Power Plants Ref. [I-8] IAEA INSAG-12 Basic Safety Principle for Nuclear Power Plants Ref. [I-24]
Pakistan	IAEA Safety Reports Series No. 25 Ref. [I-20], IAEA SSG-3 Ref. [I-3]
Romania	For review of PSA, CNCAN has used its own procedures based on a selection of standards and guides referenced in each procedure.
India	1. IAEA Safety Reports Series No. 25 Ref. [I-20]. 2. IAEA Specific Safety Guide No. SSG-3 Ref. [I-3]. 3. IAEA Safety Series No. 50-P-4 Ref. [I-10]. 4. ASME/ANS RA-SA-2009 Ref. [I-23]. 5. NUREG/CR-5801 Ref. [I-25]. 6. NUREG/CR-4772 Ref. [I-26]. 7. IAEA-TECDOC-478 Ref. [I-27]. 8. IAEA-TECDOC-592 Ref. [I-28]. 9. NUREG/CR -0492 Ref. [I-29]. 10. IAEA-50-P-10 Ref. [I-30].

I-2.16. Question 16: How many review phases are required for the regulatory body to complete the PSA review?

i. For most countries, the PSA review is performed in multiple stages. China divides the PSA review work according to PSA elements (i.e. IE Analysis, ET Analysis, FT Analysis, HRA).
ii. Argentina's regulatory review process is different, such that it includes the preliminary review of the report during its development process and the final review is made by the regulatory body after receiving final PSA report.
iii. The PSA review is carried out on a sample basis in most of the countries except for China, Pakistan, Romania and India who perform a complete review of licensees' PSAs.

MEMBER STATE	RESPONSE
Argentina	Usually the review is completed using the methodology, during PSA development and sometimes preliminary reports. Finally, the review is made over the final PSA version submitted to the regulatory body
Canada	The review is carried out in following two phases: • Stage 1 review: qualitative review which aims to demonstrate that the PSA follows the accepted methodology • Stage 2 Review: detailed review which consists of spot review of dominant accident sequences and system fault trees to ensure adequacy of different PSA tasks, and PSA results
China	There is normally no clear allocated review phase. Generally, the review work is subdivided to initiating event, event tree, human reliability analysis and accident sequence quantification For a formal NNSA review, the review work is comprehensive. A sample basis is applicable for peer review
Republic of Korea	We had 2 or more Requests for Additional Information phases for regulatory review for PSAs that was submitted by 'Severe Accident Policy' statement. For new reactors, PSA review is a part of licensing review. Thus, many of the request for additional information phases follow the licensing schedule. Recently, new reactor licensing needs three or four requests for additional information phases. In the review process of Wolsong #1 continued operation, it takes 3 years and 4 times of request for addition information. Staff reviews all technical elements of PSAs. Similar to all other regulatory reviews, each technical element is reviewed on a sample basis.
Pakistan	The PSA review is carried out in 3 phases as similar to the review of other chapters of safety analysis report: • Phase-I: review of format and contents; • Phase-II: detailed review; • Phase-III: international experience feedback. The complete details provided in PSA reports are reviewed and an independent analysis is also performed for some NNPs PSAs
Romania	The review of level 1 PSA for Cernavoda NPP Unit 1 has been performed by CNCAN staff with the help of external consultants in 2 stages: internal events (in about 5 workers for 3 months) and external events (including seismic events, fire and flooding: this has been done in around 5 workers for 2 months). The first review of level 1 PSA of Cernavoda NPP was completed, but an independent analysis has not been performed.
India	The PSA review is completed in two phases: phase 1 is an initial review of the submitted PSA reports and phase 2 consists of reviewing the compliance report and updated results based on review comments in phase 1. The total person-hours required to complete the PSA review is approximately 10 000 to 15 000. Phase 1 of the review is performed by the experts team constituted by AERB. Phase 2 of the review is performed by the expert committee with members from regulatory board, TSO, utilities and independent PSA experts. The content subject to phase 2 review is the methodology, plant response modelling, reliability analyses, HRA and the final results (minimum cutset, pie charts, etc.).

I-2.17. Question 17: Is the PSA review performed by regulatory body or by technical support organization, consultant?

In Argentina, China, the Republic of Korea and Romania, the PSA review is performed by regulatory bodies, whereas in Pakistan, the same is done by TSO. In Canada, the review is performed by regulatory body and supported by TSOs, and in India, the regulatory review is performed by the expert committee with members from AERB, BARC (TSO), IGCAR, utilities and independent PSA Experts.

MEMBER STATE	RESPONSE
Argentina	Normally TSO consultancy is used as a complementary opinion to the regulatory body's review work.
Canada	The PSA review is performed by the regulatory body and supported by TSOs.
China	For the license application of PSAR or FSAR stage for a constructing NPP, the regulatory body will perform the review. For other conditions such as updating the existing PSA model, a TSO/consultant review is preferred.
Republic of Korea	PSA review is performed by regulatory body (NSSC) and KINS. KINS conducts all PSA reviews, regulatory reviews and inspections.
Pakistan	PNRA has its own TSO called the Centre for Nuclear Safety. All regulatory reviews are performed by CNS.
Romania	The review of Cernavoda NPP level 1 PSA was performed by the CNCAN staff with the support of international experts.
India	The regulatory review is performed by AERB expert committee with members from AERB, BARC (TSO), IGCAR, utilities and independent PSA experts.

I-2.18. Question 18: What corrective actions are performed against dominant MCS? Please give plant specific examples, if any.

Following are few examples of the corrective actions performed against dominant MCS in different countries:

i. Design improvements;
ii. Identification of needs to modify systems, components, time testing period and/or procedures;
iii. Reduction of conservatism by additional analysis;
iv. Modification of EOPs;
v. Strengthening and optimizing the training of EOPs for operators to familiarize with a certain accident scenario;
vi. Improvements in seismic qualifications of systems/components important for safety.

MEMBER STATE	RESPONSE
Argentina	Typically, accident sequences with important contributions to CDF were used to select design improvements. Additionally, the assessment of MCS dominant contributions has allowed for the identification of needs to modify systems, components, test intervals and/or procedures. For example, in case of Embalse NPP some specific human action was identified as an important contribution during very small LOCA sequences. Consequently, a specific design modification will be introduced during life extension project of the NPP (implementation of an automatic triggering) to reduce the contribution to CDF.
Canada	For dominant MCS, either analytical improvements are made to remove any conservatism, or improvements are made at the plant to reduce the risk. The level of effort depends on the amount of risk presented by the cutset.
China	Most corrective actions are mainly about the strengthening and optimizing the training of EOP for operators to familiarize with a certain accident scenario, which was newly found out by PSA analysis.
Republic of Korea	There are no regulatory counter measures to dominant MCSs. But licensee derives the safety improvement items on voluntary basis and implements it reflected in the update process of PSA. For example, in case of loss of instrument air (LOIA), one of safety improvement items is the operator training enhancement to open MSSV in emergency operating procedures.
Pakistan	Some corrective actions have been taken for different NPPs in Pakistan based on dominant MCS. Some examples are: i. For Karachi Nuclear Power Plant (KANUPP) 1, the dominant MCS revealed that the failure of one valve (MH-MV7) of emergency injection system (IJW) could cause failure of the whole system. Based on this information, design modifications were carried out in IJW to add redundant valves and injection paths; ii. For Chasma Nuclear Power Plant (CHASNUPP) 1, the failure of secondary side cooling from intact steam generator in steam generator tube rupture was one of the dominant MCS. Keeping in view of this, the EOP of SGTR was revised and the provision to use ruptured SG for secondary side cooling was added (on failure of intact SG) to reduce the risk associated with SGTR.
Romania	The following corrective actions are performed against dominant MCS at Cernavoda NPP: • Introduction of automatic initiation of low pressure ECC at unit 1; • Introduction of sustained low pressure signal for small LOCA at unit 1; • Increase the seismic qualification of batteries at both units.
India	Dominant MCS are reviewed further for identification of refinements that can be done analytically to reduce their contribution. Conservatism is reduced by additional deterministic analysis for success criteria for dominant contributors. The success criteria for ECCS have been redefined separately for small break LOCA (SBLOCA) and large break LOCA.

I-3. CONCLUSIONS

Based on a review of responses received from the Member States on the questionnaire related to identification of regulatory requirements and regulatory review of NPP PSAs the followings can be concluded:

- Most of the countries require level 1 and 2 PSA submissions except for Pakistan, which only requires level 1 PSA as part of the licensing process. Moreover, the Republic of Korea also requires a level 3 for full power internal and external events.
- Argentina does not define any target CDF value for acceptance of PSA. However, Canada, China and the Republic of Korea have the same target CDF values for already operating and new NPPs (i.e. CDF < 1.0E-4/yr and CDF < 1.0E-5/yr respectively).
- Pakistan and India have only one defined CDF value for all NPPs (CDF < 1.0E-5/yr). Similarly, Romania also has single target CDF value (CDF < 1.0E-4).
- All countries require the review of PSA at PSR stage except for Romania. However, Romania requires the licensee to maintain a living PSA process for continuous update of the PSA model.
- All countries except for Argentina and Romania have defined regulatory targets for level 2 PSA. However, Argentina and Romania have some defined dose criteria.
- Common PSA-related regulatory requirements for changes after Fukushima are:
    a. The inclusion of spent fuel bay in the assessment;
    b. Updating of existing studies to include extreme external events.
- Argentina and Romania do not provide any specific regulatory guidance to the licensee regarding submission of PSA. All other countries provide their own regulatory guides to the licensees for PSA submissions.
- China, Argentina and Pakistan use Risk Spectrum for level 1 PSA. Additionally, China also uses CAFTA for CANDU PSA. However, Canada uses CAFTA, FTREX, ACUBE, FRANX and Romania uses CAFTA, ETA, PRA Quant, One4All, QRECOVER, FORTE. Moreover, in the Republic of Korea, the regulatory body uses AIMS code and the utility uses SAREX code for level 1 PSA.
- The results of the survey show that most of the countries operating CANDU-type reactors are using IAEA guidance for their regulatory review process i.e. mainly Refs [I-3] and [I-22]. However, some countries also use TECDOC-1135 Ref. [I-19], IAEA Safety Series No. 50-P-4 Ref. [I-8], IAEA INSAG-12 Ref. [I-24], TECDOC-1229 Ref. [I-20] (for level 2 PSA), and ASME/ANS RA-SA-2009 Ref. [I-21].
- For most countries, the PSA review is performed in 2 or more stages. However, the regulatory review process of Argentina is different, which includes the preliminary on-line review of the report during its development process and after final review is made by the regulatory body after receiving final PSA report. Moreover, the PSA review is carried out on sample basis in most of the countries except for China, India Pakistan and Romania who perform a complete review.
- In Argentina, China, the Republic of Korea and Romania, the PSA review is performed by regulatory bodies, whereas in Pakistan, the PSA review is done by the TSO. In Canada, the review is performed by regulatory body and supported by TSOs, and in India, regulatory review is performed by the expert committee with members from AERB, BARC (TSO), IGCAR, Utilities and independent PSA Experts.

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ANNEX II. SMALL BREAK LOCA

II-1. POSITION PAPER ON SMALL LOCA QUESTIONNAIRE

The objective of task 2016-06 was to identify the possible attributes to the variations observed in the earlier evaluation of PSA results of the CANDU Senior Regulators Group (CSRG) Member States for loss of coolant accident (LOCA) events. The small break LOCA (SBLOCA) event tree was selected for detailed comparison to identify such attributes. A set of questions was prepared to get the elaborate picture of the methodology for quantification of SBLOCA accident sequences.

This Annex highlights the insights obtained following evaluation of Member States' responses.

II-1.1. Question 1: What is the definition of SBLOCA as used in Member States? What break sizes are considered? What physical locations are included as SBLOCA?

- There exists a consensus on break size considered by all countries for the SBLOCA category. The marginal variation in the break range may be due to the difference in the capacity of the primary pressuring pumps or the way the emergency core cooling system (ECCS) is initiated (manually or automatically).
- Separate initiating events covering PT-calandria tube are considered.

MEMBER STATE	RESPONSE	
	BREAK SIZE	LOCATION OF BREAK
Argentina	0.3% to 5% of RIH	Primary heat transport (PHT) headers, PHTS feeders, end fittings, and pressure tube (PT) rupture
Canada*	Point Lepreau: 0.3% to 5% of RIH	Point Lepreau: PHTS headers, PHTS feeders, PT rupture, end fittings, top of pressurizer, FM connections. Pickering B: For Pickering B PRA study, the LOCA initiating event category includes all ruptures that cannot be isolated in the heat transport (HT) system pressure boundary outside the reactor core and boilers.
China	0.3% to 5% of RIH	PHTS headers, PHTS feeders, PT rupture
India	0.5% to 5% of RIH	PHTS headers, PHTS feeders, end fittings, gland seal of PHTS pumps
Romania	0.8% to 2.5% of RIH	PHTS headers, PHTS feeders, others
Republic of Korea	0.16% to 5% of RIH	PHTS headers, PHTS feeders, end fittings, PT rupture, fail to open of pressurizer safety valve

* There is some degree of variability in definition of SBLOCA among various Canadian CANDU stations.

II-1.2. Question 2: Brief description of deterministic analysis for SBLOCA in header

All Member States similarly consider deterministic analyses for SBLOCA. The deterministic safety analysis is carried out with and without ECCS. The fuel failures are not predicted during the transient if ECCS works. The decay heat is removed through ECCS in the broken loop and through thermo-syphoning in the intact loop. In case of ECC impairment, a limited number of fuel failures are expected; fuel cooling is ensured by PT/CT thermal contact and transfer of decay heat to the moderator system. The estimated radiological consequences (i.e. public doses) are below the reference dose limits.

MEMBER STATE	RESPONSE
Argentina	Two groups of SBLOCA are considered in the analysis, namely S2 and S3. The most important characteristics of the expected NPP behaviour after S2 postulated SBLOCA are: • Pressurizer level drops, the inventory is added through pressure and inventory control (PIC) pumps • Reactor trips on through RPS-1 on high reactor building pressure or through RPS-2 on low PHTS pressure • Loop isolation and ECCS injection takes place • Steam generators provide cooling in both broken and intact loop The expected NPP behaviour after S3 postulated SBLOCA are similar to the previous S2 case. The ECCS actuation is manual for S3 as the automatic action is avoided by design for the leakages considered in this group.
Canada	In case of SBLOCA, reactor power is controlled automatically by reactor regulating system with the use of liquid zone control units and absorber rods. Crash cool down is achieved by opening the main steam safety valves (MSSV). ECCS injection and loop isolation takes place when the PHT pressure reaches the design set points. Following injection, the ECC would quickly refill the broken loop. Cooling in the intact loop would be maintained by forced circulation with heat removal by the boilers. In case of ECC impairment, a limited number of fuel failures are expected; fuel cooling is ensured by PT/CT thermal contact and transfer of decay heat to the moderator system. Containment of radioactivity is provided by the containment system, which closes the containment isolations valves in the early stage of the accident when high pressure in RB is detected.
China	Two types of analyses are done: LOCA with ECCS available: The post-trip thermal-hydraulic analysis demonstrates that systematic fuel failure does not occur for the complete range of small breaks, that is breaks smaller than the largest feeder. The channel integrity is assured during the whole transient. Radionuclide releases to the environment and doses to the individual of the critical age group and population are bounded by the inlet feeder break for which the resulting public doses are below the reference dose limits. LOCA with ECCS impairment: Fuel and pressure tubes in the broken loop can heat up as the loop inventory is depleted. Under the most limiting heat up conditions, the release of radionuclides from failed fuel into containment and subsequently to the environment is such that the doses to the public are below the reference dose limits. Hydrogen can also be generated from the reaction between steam and Zircaloy in fuel sheaths and pressure tubes, but under the limiting conditions for hydrogen production, concentrations in containment remain below unacceptable levels. Pressure tubes may heat up and contact calandria tubes by ballooning or sagging; however, there is adequate heat removal to the moderator such that channels remain intact. There are no fuel failures for small breaks with failure of heat transport system loop isolation, since both loops are refilled with ECC, in a similar manner as for the case with loop isolation available.
India	To study the system behaviour under SBLOCA, a wide range of break sizes is considered in the analysis. In short term system response, the effect of reactor trip, crash cool down, pressurizer and both loop isolation, along with initial history of system depressurization, coolant flow through core, void/sheath temperature are analysed (with and without Class-IV power supply). Since the decay heat is being removed through ECCS in broken loop and through thermo-syphoning in intact loop, gross failure of fuel cladding is not expected. However, fuel sheath failure in broken feeder channel is expected. Radiological consequences are analysed in both scenarios. It is found that the consequences are well within the bounding situation of large brake LOCA.

MEMBER STATE	RESPONSE
Romania	The deterministic analyses of SBLOCA event considered trip coverage, thermal-hydraulics, and single channel, fuel, containment and dose calculation analyses. SBLOCA event with subsequent failures (i.e. loss of containment functions, loss of ECC functions as singular subsequent events or combined with loss of electrical power) are separate analyses. No fuel failures represent the success criteria for trip coverage and circuit / single channel and fuel analyses. However, dose calculations are performed because of the limited inventory (tritium and limited remnant iodine and noble gases inventory) from the discharged heavy water inside containment.
Republic of Korea	If the SBLOCA accident occurs, reactor power is controlled automatically by the reactor regulating system and the reactor trips because of pressurizer low level and PHT low pressure. After the ECCS injection actuation signal occurs, crash cool down is achieved by opening the MSSVs. ECCS injection and loop isolation takes place when the PHT pressure reaches the design set points. Following injection, the ECC would quickly refill the broken loop. Cooling in the intact loop would be maintained by forced circulation with heat removal by the boilers. In case of ECC impairment, a limited number of fuel failures are expected; fuel cooling is ensured by PT/CT thermal contact and transfer of decay heat to the moderator system. Containment of radioactivity is provided by the containment system, which closes the containment isolation valves in the early stage of the accident when high pressure in RB is detected.

II-1.3. Question 3: How has the frequency of SBLOCA been calculated? What uncertainty distribution has been considered for the initiating event frequency and the parameters?

- The methodology adopted for frequency estimation of SBLOCA in Member States is different. However, lognormal distribution is used for uncertainty analysis.
- In case of Canada and the Republic of Korea, generic values were used with Bayesian update.
- A notable difference was the mean value of the SBLOCA frequency. This could be one of the possible attributes for the wide variations observed in the earlier evaluation in the SBLOCA break size definition.

MEMBER STATE	RESPONSE
Argentina	The frequencies for S2 and S3 were estimated referencing the values included in TTR221 and safety design matrix N° 3 and N° 7 for similar ranges of breaks. Frequency S2 5E-3/yr Frequency S3 5E-3/yr Frequency SB 2E-4/yr (generic CANDU PSA – reference analysis)
Canada	Single Unit NPP: The prior LOCA 2.5 event frequency is derived from one event found in the COG OPEX database. Prior: 3.12E-3 per reactor-year, EF: 4.0, Experience: 379.509 years Posterior: 2.90E-3 occurrences per year, EF: 4.0, Experience: No failures in 22.72 years (Point Lepreau) Pickering NPPs: Prior: industry-wide experience in the USA and Canada Posterior: estimated with Pickering B NGS experience LOCA1 = 1.3E-2 with uncertainty factor 2.42 LOCA2A/2B = 2.16E-3 with uncertainty factor 5.61
China	Since the occurrence of SBLOCA is zero in CANDU operating experience, the chi-square approximation is applied. The equivalent full power days are determined based on the CANDU Owners Group (COG)-CANDU performance annual report. The frequency of SBLOCA is 4.82E-4 and the uncertainty distribution is lognormal
India	Plant specific LOCA frequencies are not estimated. The generic LOCA frequency given in CANDU-6 PSA (based on Canadian operating experience) is used in the PSA. The SBLOCA (in Header) frequency is taken as 2.0E-3/yr. The total frequency of SBLOCA is 6.6E-3/yr. Lognormal distribution is assumed for uncertainty analysis
Romania	The frequency of SBLOCA is calculated based on the EPRI Pipe segments method starting from plant technological flow sheets. The uncertainties associated are based on the same method. The SBLOCA frequency is 6.5E-4 events/yr and uncertainty distribution is lognormal

<table>
<tr><th>MEMBER STATE</th><th>RESPONSE</th></tr>
<tr><td>Republic of Korea</td><td>Because the occurrence of SBLOCA is zero in the Republic of Korea and CANDU operating experience, we cannot estimate for plant-specific LOCA frequencies. We used the generic CANDU-2002 SBLOCA data and analyse it using Bayesian update, considering operating time Ref. [II-1].
<table>
<tr><th>SBLOCA IE FREQUENCY</th><th>IE GROUPING</th><th>IE GROUPING DESCRIPTION</th><th colspan="2">PRIOR INPUT (GENERIC CANDU-2002)</th><th colspan="2">LIKELIHOOD INPUT (W1234 SPECIFIC DATA)</th><th>POSTERIOR</th></tr>
<tr><td rowspan="9">1.97E-2</td><td>IE-SL</td><td>Small LOCA</td><td>2.00E-3</td><td>10</td><td>0</td><td>54.7</td><td>1.44E-3</td></tr>
<tr><td>IE-FBIO</td><td>Feeder break - inlet/outlet guillotine break</td><td>2.00E-3</td><td>10</td><td>0</td><td>54.7</td><td>1.44E-3</td></tr>
<tr><td>IE-EFB1</td><td>End-fitting break outside annulus gas system</td><td>1.00E-3</td><td>10</td><td>0</td><td>54.7</td><td>8.14E-3</td></tr>
<tr><td>IE-PRLB</td><td>Break in piping upstream of pressurizer relief valves/steam bleed valves</td><td>2.33E-4</td><td>3</td><td>0</td><td>54.7</td><td>2.31E-4</td></tr>
<tr><td>IE-GSC</td><td>Loss of gland seal cooling to all HT pumps</td><td>1.38E-5</td><td>3</td><td>0</td><td>54.7</td><td>1.38E-5</td></tr>
<tr><td>IE-PTR</td><td>Pressure tube rupture</td><td>8.46E-3</td><td>10</td><td>0</td><td>54.7</td><td>3.96E-3</td></tr>
<tr><td>IE-FMEFF</td><td>Fuelling machine induced end fitting failure</td><td>8.80E-4</td><td>3</td><td>0</td><td>54.7</td><td>8.57E-4</td></tr>
<tr><td>IE-HPCL</td><td>Heat transport pressure control failure - low</td><td>1.05E-2</td><td>2</td><td>0</td><td>54.7</td><td>9.50E-3</td></tr>
<tr><td>IE-CFB</td><td>Channel flow blockage</td><td>2.00E-3</td><td>10</td><td>0</td><td>54.7</td><td>1.44E-3</td></tr>
</table></td></tr>
</table>

II-1.4. Question 4 (a): Event progression from deterministic analyses indicating the sequence of actuation of the safety systems

The event progression is similar in PSA studies of all Member States with the following observations.

- The reactor is tripped by the process parameters (i.e. PHT low pressure or pressurizer low level). In case that reactor power increases, the neutronic trip parameters are also providing trip coverage.
- Once ECC injection is successfully initiated, the broken loop is refilled, and fuel sheath temperature is reduced. The steam generators and the break itself ensure broken loop cooling. The cooling of the intact loop is ensured by thermo-syphoning.

MEMBER STATE	RESPONSE
Argentina	The success criteria of various safety systems required during the accident response are provided in other questions. The accident progression scenario is not elaborated as required in the question
Canada	In case of SBLOCA, reactor power is controlled automatically by the reactor regulating system with the use of liquid zone control units and absorber rods. Crash cool down is achieved by opening the MSSVs. ECCS injection and loop isolation takes place when the PHT pressure reaches the design set points Following injection, the ECC would quickly refill the broken loop. Cooling in the intact loop would be maintained by forced circulation with heat removal by the boilers In case of ECC impairment, a limited number of fuel failures are expected; fuel cooling is ensured by PT/CT thermal contact and transfer of decay heat to the moderator system Containment of radioactivity is provided by the containment system, which closes the containment isolations valves in the early stage of the accident when high pressure in RB is detected
China	Gradual loss of inventory from the primary circuit would cause depressurization and reduction in the overall circuit flow. First indication of the event in the control room is D_2O storage tank low level for small HTS leaks and high RB pressure for 2.5% RIH with a discharge rate of 450 kg/s. There is a simultaneous fall in the pressurizer level from time zero of the event as the break discharge exceeds capacity of the D_2O feed pumps and pressurizer heaters Once the reactor is tripped, ECCS is injected and containment is isolated. To reduce the pressure rise in the containment, dousing is also actuated. Two primary loops are isolated from the purification system, pressurizer and heavy water feed and bleed circuit The automatic high-pressure ECC injection takes place to broken loop and crash cool down is initiated in SGs. The coolant recovered from the sump is cooled by either of the ECC recovery heat exchangers and re-injected into the primary circuit by the ECC pumps to provide long term makeup
India	Following the SBLOCA, the PHT depressurizes slowly, pressure control system plays a major role, and neutronics do not have a significant role. The reactor trip is expected to occur by process signals. The pump room high pressure signal initiates crash cool down and light water injection and ECCS recirculation. If crash cool down fails, the auto fast cool down can also be used to depressurize the PHT because of reduction of coolant flow during the event, thereby increasing in coolant voiding. In the absence of crash cooling, the voiding becomes persistent before the ECCS comes into operation PHT pressure and the pressurizer get isolated. Simultaneously, both PHT loops along with the feed and bleed system get isolated Intact loop: The inventory in this unbroken loop is maintained at more than 85% of initial inventory. The cooling of the intact loop is achieved by thermo-syphoning in the steam generators Broken loop: At 40 Kg/cm2 (g) PHT pressure, along with the 'conditional signal' (i.e. RB pressure high), ECCS injection starts from the light water accumulators. The ECCS re-circulation pumps also get started following opening of pump suction valves. The injection of the 'cold' water would quench the voids and limits the sheath temperature rise. When PHT system depressurizes below the shut off head of re-circulation pumps, the re-circulation mode gets established through heat exchangers and provides long term PHT cooling along with thermo-syphoning in the steam generators

MEMBER STATE	RESPONSE
Romania	Following the break initiation, reactor-regulating system maintains the reactor power constant. The reactor is tripped by the process parameters (i.e. low pressure or pressurizer low level). In case that reactor power increases, the neutronic trip parameters are also providing trip coverage. Continuing inventory discharge, larger than the feed pumps capacity; determine the pressure reduction and loop isolation actuation, steam generators crash cool down and ECC injection. Once ECC injection is successfully initiated, the broken loop is refilled, and fuel sheath temperature is reduced. The steam generators and the break itself ensure broken loop cooling. The cooling of the intact loop is ensured by thermo-syphoning. The deterministic analyses show no fuel failures. Depending of the initial discharge, the containment pressure may increase to the level the containment is isolated and dousing is initiated. Dousing spay may cycle few times until dousing inventory is depleted, or the energy discharged through the break is not enough to increase the containment pressure
Republic of Korea	If the SBLOCA accident occurs, reactor power is controlled automatically by the reactor regulating system and reactor trips by pressurizer low level and PHT low pressure. After ECCS injection, the actuation signal occurs Crash cool down is achieved by opening the MSSVs. ECCS injection and loop isolation takes place when the PHT pressure reaches the design set points. Following injection, the ECC would quickly refill the broken loop. Cooling in the intact loop would be maintained by forced circulation with heat removal by the boilers. In case of ECC impairment, a limited number of fuel failures are expected; fuel cooling is ensured by PT/CT thermal contact and transfer of decay heat to the moderator system. Containment of radioactivity is provided by the containment system which closes the containment isolations valves in the early stage of the accident when high pressure in reactor building is detected

II-1.5. Question 4 (b): Detailed/schematic/modified event tree for SBLOCA with listing the first few dominant cutsets for the function events modelled in event tree.

The event progression is similar for PSA studies of all Member States.

- Similarities are observed in the function events and the accident progression considered in the accident sequence analysis in the responses submitted by India, China and the Republic of Korea. Some of the differences were observed in the event tree submitted in the response by Argentina:
    - The function events on 'loop isolation' (LI) and 'crash cool down' (CC) are not considered in event tree submitted by Argentina;
    - Two separate ETs are developed considering the availability/Unavailability of class-IV supply in Argentina;
    - Operator action is considered at the beginning of the event tree and not at individual functional event level in Argentina.
    - Nomenclature used for the function events in event tree are different than the event trees submitted by other Member States.
- Canada and Romania did not provide event tree structure in the response.
- The Republic of Korea did not consider the function events on FFWS in event tree.

EVENT TREE HEADINGS	EVENT TREE HEADINGS DESCRIPTION	ARGENTINA	CHINA	INDIA	REPUBLIC OF KOREA
RPS (K22R)	Reactor power scratch (Trip)	✓	✓	✓	✓
LI	Loop isolation	-	✓	✓	✓
CC	Crash cooldown	-	✓	✓	✓
ECC-D (D1D4/D1D3)	ECC demanded	✓	✓	✓	✓
ECC-LT (D2D4/ D2D3)	ECC long term	✓	✓	✓	✓
FWS (MPA11/MA1)	Feed water supply to steam generators	✓	✓	✓	✓
SDC-O (H1S1/H1S2)	Operator initiates shutdown cooling	✓	✓	-	-
SDC (P7/P4)	Shutdown cooling operation	✓	✓	-	-
EWS-O (FFWS-O) (H1S1/H1S2)	Operator initiates emergency water supply to steam generators	✓	✓	✓	✓
EWS (FFWS) (ME3/ME4)	Emergency water supply operation	✓	✓	✓	
MHS (MSL1/MSL2/MSL3/MSL4)	Moderator acting as a heat sink	✓	✓	✓	✓

II-1.6. Question 5 (a): Sequences considered for quantification of CDF in PSA with listing the first few dominant cutsets for the accident sequences considered

The wide variations are observed in the MCS lists of the SBLOCA event tree results submitted by Member States. Two notable observations are:

- Human actions for ECCS and moderator system are dominant for Argentina.
- Simultaneous failure of Class-III and Class-IV power supply contributes to the MCS list submitted by China. CCFs are manually modelled.

MEMBER STATE	RESPONSE
Argentina	Accident sequence does not involve SBLOCA initiating event, but involves human errors in actuating ECCS and moderator system
Canada	SBLOCA (LOCA2.5), sequences leading to severe core damage (PDS-02) are considered for quantification SBLOCAs (LOCA1, LOCA2A, LOCA2B) sequences are considered for quantification of severe core damage (FDC2)
China	Accident sequence involving SBLOCA initiating event and simultaneous failure of Class IV and Class III power supplies Accident sequence involving SBLOCA initiating event and common cause failures (CCFs) of ECCS pumps (manual CCF)
India	Accident sequence involving SBLOCA initiating event and CCFs of ECCS MVs Accident sequence involving SBLOCA initiating event and CCFs of ECCS pumps
Romania	All core damage sequences have been considered for quantification. The dominant cut-sets are: • Accident sequence involving SBLOCA initiating event and failure of ECC low pressure stage initiation • Accident sequence involving SBLOCA initiating event and failure to provide cooling to ECC low pressure heat exchanger
Republic of Korea	SBLOCA (NO3), sequences leading to severe core damage are considered for quantification. Accident sequence involving SBLOCA initiating event and simultaneous failure of feedwater supply working (MFW/AFW) and emergency water supply to the SGs

II-1.7. Question 5(b): How were common cause failures modelled? Which models and the parameters were used for the modelling?

- There is some variation regarding the methods used for treatment of CCF in PSA. However, these are acceptable approaches as per the ASME PRA standard Ref. [II-2].
- China indicated that CCFs are manually modelled and UPM method is used for CCF analysis. This is different from the approaches mentioned by other Member States.
- Romania indicated that the β-factor CCF method is used for groups of more than 6 components. The α-factor CCF method is used for groups up to 6 components, that is contrarily to NUREG/CR-5801 Ref. [II-3] that specifies that needs to be used for up to groups of maximum 4 components.

MEMBER STATE	RESPONSE
Argentina	For CCF models, the method of multiple Greek letters was used, and the publication INEL 94/0064 Ref. [II-4]. CCF parameter estimation was used for quantification
Canada	Overall CCF: Beta Factor Main CCF contributors recalculated using US NRC α-factor method Ref. [II-5]
China	Unified partial method (UPM) is used to analyse the CCF. The CCF events are manually modelled in the fault tree
India	Common cause failure (CCF) analysis is performed for the redundant components or trains of systems. The α-factor model is used for CCF analysis. The generic parameters as given in NUREG/CR-5801 Ref. [II-3] are used in the CCF analysis
Romania	α-factor method for groups of less than 6 components β-factor method for groups of more than 6 components
Republic of Korea	α-factor method for groups of all components

II-1.8. Question 5(c): Were human errors modelled? What methodology was used for the same?

There exists a consensus on the use of HRA methods among the Member States.

MEMBER STATE	RESPONSE
Argentina	• For diagnosis human error (S3 case), the curves of the HCR (P 5,6 104) • For failures previous to initiating event, ASEP (accident sequence evaluation program)- THERP (technique for human error rate prediction) was used
Canada	• Pickering NPPs: THERP • Single unit: ASEP-THERP
China	• ASEP method is used for pre-accident human error • HCR + THERP method are used for the post-accident human error events
India	• THERP is used for pre-initiator human error probability estimation • Human cognitive reliability (HCR) is used for the post-initiator human error probability estimation
Romania	• THERP method was used for pre-initiator human failure events • HCR + THERP method was used for post-initiator human actions
Republic of Korea	• K-HRA (similar to ASEP-THERP)

II-1.9. Question 6: What success criteria were used for safety function events considered in event trees?

The success criteria used by all member states are similar. Some variations in success criteria of ECCS and moderator system were observed for India, which is attributed to the design configurations of ECC pumps and HXs.

SAFETY FUNCTION EVENT CONSIDERED IN EVENT TREE	ARGENTINA	INDIA	ROMANIA	REPUBLIC OF KOREA
Reactor protection system	RPS-1: 26 rods out of 28 rods RPS-2: 5 out of 6 tubes	RPS-1: 26 rods out of 28 rods RPS-2: 5 out of 6 tubes	Shutdown system (SDS)#1: 26/28 rods SDS#2: 5/6 injection nozzles	SDS#1: 26 /28 rods SDS#2: 5/6 injection nozzles
Emergency core cooling system	1 out of 2 ECCS pumps Success criteria for ECCS heat exchangers (HXs) is not clear	1 out of 4 ECCS pumps and 1 out of 3 ECCS HXs	1 out of 2 ECC pumps/ 1 ECC HX + recirculating service water RSW/ recirculating cooling water RCW	1 out of 2 ECC pumps/ 1 ECC HX + RSW/RCW
Feed water system	Boiler Feed Pumps (BFPs): 1 out of 3 pumps Auxiliary Boiler Feed Pumps (ABFPs): 1 out of 2 pumps	BFPs: 1 out of 3 pumps ABFPs: 1 out of 2 pumps	1/3 Main feedwater pump (MFWP) or auxiliary feedwater pump (AFWP)	1/3 MFWP or AFWP
Emergency water system/ fire fighting water system	1 emergency water supply (EWS) pump	1 firefighting water system (FFWS) pump	1 EWS pump	1 EWS pump
Moderator system	1 pump, 1 out of 2 HXs with low pressure ECCI, failure 2 out of 2 HXs with high/medium pressure ECCI failure	1 pump and 1 HX	1 main pump/ 2 pony motors + 2 HXs + RCW/ RSW	1 main pump/ 2 pony motors + 2 HXs + RCW/ RSW

II-1.10. Question 7: What was the contribution of SBLOCA to overall CDF?

Wide variations are observed in the SBLOCA contribution to overall CDF among the Member States. SBLOCA contribution for China is very low.

MEMBER STATE	RESPONSE
Argentina	The contribution of S3 initiating event to CDF value is about 50 %. For all the other SBLOCA cases the results obtained are indicating contributions less than 1 % However, preliminary results considering the design modifications that will be implemented next year during the life extension process of the Embalse NPP show the following updated contributions: S3 contributes to 7.3 % of CDF and S2 contributes to 1.4 % of CDF
Canada	Based on RAW >2 and FV > 0.005 importance indices, SBLOCA initiating events are not risk significant contributors to SCDF Single unit (2008 Submission): The accident sequences, following a SBLOCA are equivalent to 2.5 % RIH break (LOCA2.5) initiating event, contributing 1.49 E-07 events per year to the SCDF
China	The contribution of SBLOCA is 0.4 % (3.29E-08 events/yr) compared to the total SCDF of 8.395E-06
India	SBLOCA contribution to overall CDF ~ 9%
Romania	~ 2.5 % of total CDF SBLOCA contribution is 1.71 E-07 events/yr while CDF value is 6.91E-06 events/yr
Republic of Korea	The SBLOCA contribution is 12.693 % (3.11 E-06 events/ year) of the total CDF (2.45 E-05 events/year)

II-2. CONCLUSIONS

The following conclusions were identified based on the review of responses from Member States regarding the SBLOCA questionnaire:

- There exists a consensus on break size considered by all countries under SBLOCA category.
- The methodology adopted for frequency estimation of SBLOCA in Member States is different. These methods are statistically equivalent and are 'acceptable'.
- The event progression is similar in PSA studies of all Member States. However, wide variations are observed in the MCS lists of the SBLOCA event tree results submitted by Member States.
- SBLOCA contribution to overall CDF range from 0.4% to 12.7% among Member States.

Variations observed in the minimum cutset list and SBLOCA contributions to overall CDF is expected considering the difference in initiating event frequency, the fault tree models, component failure data, treatment of CCF and HRA. Harmonization is achieved in the definition of SBLOCA, methodology for frequency estimation and accident sequence modelling. Countries operating CANDU-type reactors continue to share experience and information regarding SBLOCA.

REFERENCES TO ANNEX II

[II-1] NUCLEAR REGULATORY COMMISSION, Generic CANDU Probabilistic Safety Assessment – Reference Analysis (91-03660-AR-002) Rev. 0, Washington, DC (2002).

[II-2] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, Standard for Level 1/ Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME RA-Sa-2009, Washington, DC (2009).

[II-3] NUCLEAR REGULATORY COMMISSION, Procedures for Analysis of Common Cause Failure in Safety Analysis, NUREG/CR-5801, Washington, DC (1993).

[II-4] IDAHO NATIONAL ENGINEERING LABORATORY, Common-Cause Failure Data Collection and Analysis System, Vol.6, INEL-94/0064, Idaho Natl Lab., ID (1995).

[II-5] NUCLEAR REGULATORY COMMISSION, Guidelines on Modeling Common Cause Failures in Probabilistic Risk Assessment, NUREG/CR-5485, Washington, DC (1998).

ANNEX III. TREATMENT OF ZERO OCCURRENCE INITIATING EVENTS

This Annex provides the results from the CPWG survey on the treatment of zero occurrence initiating events.

III-1. CHINA

For some initiating events from the Third Qinshan Nuclear Power Plant (NPP) PSA (TQNPP PSA), there are zero occurrences in CANDU-type operating experience. Initiating events with zero occurrences require fault tree analysis or special quantification techniques, such as the chi-square approximation or piping failure frequency calculations. The methodology is presented below:

- For supporting system (cooling water, electrical power, instrument air), fault tree analysis is used.
- For piping failure related initiating events, piping failure rate data (OH, No.86296 report Ref. [III-1]) and piping length data are used for the initiating events frequency calculation.
- For other initiating events, fault tree analysis needs to be used first. If it is not possible, CANDU operating experience is applied using the chi-square approximation.
- The final step is to compare with PWR initiating events frequency using NUREG/CR-5750 Ref. [III-2] and other CANDU plant data. If it is necessary, expert judgment will be applied to determine the final frequency.

III-2. CANADA

A zero-occurrence initiating event approach was developed to ensure that the proposed methodology and the results based on its application will meet the expectations of the CNSC and ensure it reflects Canadian best practices for CANDU 6 (C6).

- For the cases when the C6 generic operating experience is zero and there are one or more occurrences in the generic data (Canadian stations, other than C6-generic), a Bayesian approach is used to update the prior distribution using the posterior zero occurrence.
- For the cases where is both the C6-generic operating experience and generic data for the analysed period, the generic data is extended.

The Bayesian technique is used to calculate initiating events frequencies when the initiating events are rare events (their occurrences are zero or less than 10 per year). The frequency is calculated by combining generic data with specific C6 generic data using the Bayesian approach.

III-3. ARGENTINA

Various approaches were used for initiating event frequency estimation of zero occurrences. Descriptions of the different approaches used for events with zero occurrences are:

- In some cases, conservatively, one occurrence in twice the lifetime of the plant was considered (e.g. spurious opening of the liquid relief valve, uncontrolled depressurization of heat transport system). Nevertheless, each case was compared with available data from other CANDU reactors.

- For other cases, there was consideration from data obtained from reference publications (Generic CANDU PSA, CERNAVODA PSA, etc.) (i.e. the frequency for total loss of HTS pump flow was taken from Generic CANDU PSA) Ref. [III-3].

- For some cases, fault trees were developed and quantified. Some examples include loss of class IV power supply, total loss of service water, loss of high pressure service water, loss of instrument air.

Piping failure data was taken from Canadian Safety Design Matrices (1982 and 1983) and it also provided the pipe lengths in terms of equivalent diameters. This criterion is also used in other reference publications Ref. [III-1]).

Other piping failure data were taken from NUREG/CR-4407 Ref. [III-4], where the failure rates are calculated for a typical PWR taken as a mean from representative reactors. NUREG/CR-4407 divides failure rates by systems. A failure rate is suggested for each system or group of systems and piping susceptible to a particular break size is considered. This approach was used in PSA developed for the Atucha I NPP. NUREG/CR-4407 Ref. [III-4] was primarily used for:

- Steam line failures from PWR data, where the failure rate λ = 1.6E-3/(yr × L-reactor), where L-reactor is the medium length of the steam lines in PWR, which is a value that has to be multiplied by CNE steam line pipes;

- For service water piping failures, λ = 1E-4/yr;

- In addition, global data for λ = 5E-4/yr is provided (for piping failures that implies a system failure including all water systems other than PHTS, volume control system and feedwater system).

III-4. INDIA

The following table summarizes the method of zero occurrence initiating events used.

NO	DESCRIPTION	METHOD ADOPTED
1.	Zero failure occurrences for components	Bayesian update with consideration of the operating period
2.	Zero occurrences for LOCA group of Initiators	Generic values Ref. [III-5]
3.	a) Zero occurrences for transient group of initiators	Chi-squared approximation
	b) Zero occurrences for transient group of initiators caused by failure of process systems for which detailed fault trees are developed	Fault tree method

III-5. PAKISTAN

For the Karachi Nuclear Plant (KANUPP):

a) For support systems initiating events (i.e. total loss of instrument air), fault trees were developed and quantified.

b) For initiating events involving a pipe break, (i.e. large LOCA or steam line break), NUREG/CR-5750 Ref. [III-2] and NUREG/CR-6928 Ref. [III-6] were used. If the initiating event frequency was available in only one of the above NUREG publications, then the initiating event frequency was updated with plant evidence (zero occurrences in plant life) using a Bayesian approach. If initiating event frequency of the particular event was available in both NUREGs then a conservative value was selected and updated with plant evidence as indicated above.

III-6. ROMANIA

Non-informative prior distributions are a class of prior distributions that minimize the relative importance of the prior distribution in generating a posterior estimation. Non-informative prior distribution is used when little or no generic prior information is available.

Where the experience for some or all the events in a group is zero occurrences, it was considered inappropriate to estimate the frequency of each event independently. This would have resulted in the total frequency of the category becoming a function of the number of individual events chosen. Therefore, in such cases the frequency of those events was considered to be much smaller than the frequencies of the rest events of group and did not contribute to the frequency of group.

The use of chi-squared distribution permits the estimation of frequency where zero failures and no prior distribution have been observed. The value of interest to be calculated in this case is:

$$\lambda_{0.5} = \frac{\chi^2(0.5, 2f+1)}{2T}$$

The best estimate or the point estimate value of λ has been interpreted to be the mean of the lognormal uncertainty distribution.

$\lambda_{mean} = \lambda_{0.5} * e^{\frac{\sigma^2}{2}}$, where $\sigma = \frac{\ln(EF)}{1.645}$, and 1.645 is the 95th percentile of the standard normal distribution function.

The numerical value of zero occurrences in 404.6 operational reactor years of CANDU experience is 1.92E-03/y.

III-6.1. Uncertainty in non-informative EF calculations

When non-informative and 0 failures have been observed, the uncertainty value is:

$$EF = \frac{\lambda_{0.95}}{\lambda_{0.5}}.$$

The numerical result of this uncertainty is 13.2. In accordance with other studies, Cernavoda NPP considered the maximum uncertainty factor for the case of 0 occurrences is 10.

REFERENCES TO ANNEX III

[III-1] ONTARIO HYDRO, Component Reliability Data for CANDU Nuclear Stations Design and Construction Branch, Report No. 86 296, Ontario (1986).

[III-2] NUCLEAR REGULATORY COMMISSION, Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995, NUREG/CR-5750, Washington, DC (1999).

[III-3] ATOMIC ENERGY OF CANADA LIMITED, Generic CANDU PSA Methodology, Ontario (2001).

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[III-5] ATOMIC ENERGY OF CANADA LIMITED, Probabilistic Safety Assessment Methodology, ACR 108-03660-AB-001, Ontario (2003).

[III-6] NUCLEAR REGULATORY COMMISSION, Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants, NUREG/CR-6928, Washington, DC (2007).

ANNEX IV. SUCCESS CRITERIA

IV-1. INTRODUCTION

Two questionnaires were prepared in order to assess and identify the similarities in success criteria between Member States. The first questionnaire was about front line systems and their associated success criteria. In addition, the questionnaire also covered safety functions and initiating events for which the respective criteria are applied. For illustration purposes, the detailed information provided by Canada is presented in Table IV-1. Following the completion of the first questionnaire, a supplementary investigation was completed regarding the differences highlighted from the responses of the Member States. This was the basis of the second questionnaire.

The second questionnaire was prepared in order to gather further information regarding differences in the success criteria observed in the first questionnaire. The following information was requested for the second questionnaire:

- Number and type of valves for boiler crash cooldown function (required to open in order to ensure primary heat transport system (PHTS) heat sink);
- Number of headers required to be supplied with water by emergency core coolant injection (required in order to ensure coolant inventory in case of loss of coolant accident type of events);
- Number of required ECC heat exchangers (required in order to ensure decay power removal after LOCAs);
- Number of steam generators per loop required to be supplied with water (required to ensure decay power removal);
- Number of moderator heat exchangers and/or pumps required to ensure functionality of the moderator as an ultimate heat sink and/or moderator temperature control functionality required and/or cover gas functionality required;
- Digital control computer (DCC) operation (justification for the way it was considered in the PSA study).

Each question has been detailed in a table format in order to cover all the initiating events. China, Canada and India provided updated answers for the success criteria questionnaire.

IV-2. OBSERVATIONS AND NOTES

IV-2.1. Topic 1: Number and type of valves required for boiler crash cooldown function

Differences originated from the design (valves capacity itself) or from refined safety analyses.

MEMBER STATE	LARGE LOCA*	SMALL LOCA	VERY SMALL LOCA / LEAKS	TRANSIENTS
Argentina (C6)	-	10/16 main steam safety valves (MSSVs)	10/16 MSSVs	5/16 MSSVs
Canada (C6)	-	8/16 MSSVs**	10/16 MSSVs	1/16 MSSVs
Canada (non-C6)	-	7/16 MSSVs	7/16 MSSVs	7/16 MSSVs or 4/4 ASDVs or 4/8 CSDVs
China (C6)	-	8/16 MSSVs	8/16 MSSVs	8/16 MSSVs
India	-	7+3 ASDVs	7+3 ASDVs	7+3 ASDVs
Republic of Korea (C6)	-	7/16 MSSVs	7/16 MSSVs	7/16 MSSVs or 12/12 CSDVs
Romania (C6)	-	10/16 MSSVs	10/16 MSSVs	8/16 MSSVs

* Break is sufficiently large to depressurize PHTS in case of large LOCAs.

** Applicable for in-core LOCAs and multiple steam generator tube rupture (SGTR).

IV-2.2. Topic 2: Number of headers required to be supplied with water by ECC for inventory control for different events, separated for each loop

Differences were based on the design CANDU 6 (C6) versus the Enhanced CANDU 6 and potentially the interpretation for India). The results are based on different or refined safety analyses for C6 in support of the less restrictive success criteria (refer to the in-core LOCA). BL is for broken loop and IL is respectively for Intact Loop.

MEMBER STATE	LARGE LOCA		SMALL LOCA		VERY SMALL LOCA/LEAKS	
	BL	IL	BL	IL	BL	IL
Argentina (C6)	4/4	1/4	1/4	1/4	1/4	1/4
Canada (C6)	4/4	1/ 4	2/4*	1/4	1/4	1/4
Canada (non-C6)	4/4	2/4**	4/4	0/4	4/4	0/4
China (C6)	4/4	1/4	4/4	1/4	4/4	1/4
India	All	All	All	All	All	All
Republic of Korea (C6)	4/4	1/4	4/4	1/4	4/4	1/4
Romania (C6)	4/4	1/4	4/4	1/4	4/4	1/4

* Applicable also for in-core LOCA.
** Applicable for in-core LOCA and multiple SGTR.

IV-2.3. Topic 3: Number of ECC heat exchangers required for low pressure ECC injection

This is similar for C6 plants. Supplementary deterministic safety analyses demonstrate emergency water supply being available for ECC heat exchanger cooling in a special case (in-core LOCA).

- Number of ECC HXs varies between plants.
- Requirements: one HX for C6 and two HXs for India.
- Cooling is provided from raw service water (RSW) for most of C6, whereas for Canada, cooling is provided by RSW or emergency water supply (EWS) for in-core LOCAs. Similar analysis is in progress in Romania to evaluate and justify the success criteria.

IV-2.4. Topic 4: Number of steam generators per loop required to ensure heat sink

Number of steam generators per loop required to ensure heat sink are determined by the conditions required for the thermo-syphoning process. The number of steam generators per loop is determined based on specific safety analyses assumptions and results.

MEMBER STATE	LARGE LOCA		SMALL LOCA		VERY SMALL LOCA/ LEAKS		TRANSIENTS	
	BL	IL	BL	IL	BL	IL	CL IV	NO CL IV
Argentina (C6)	0	1	1	1	1	1	1	2
Canada (C6)	0	2	2	2	2	2	1	2
Canada (non-C6)	2	2	2	2	2*	2*	1	1
China (C6)	2	2	2	2	2	2	2	2
India	2	2	2	2	2	2	2	2
Republic of Korea (C6)	0	2	2	2	2	2	2	2**
Romania (C6)	0	2	2	2	2	2	2	2

* 1 SG is required for SGTR or leaks.
** Exceptions for specific feedwater events and loss of moderator cover gas deuterium control.

IV-2.5. Topic 5: Number of moderator heat exchangers / pumps required

Number of moderator heat exchangers / pumps required determined by the specific deterministic safety analyses.

- All events:
  - 1 pump + 1 HX Argentina, India;
- All external LOCAs (except In-core LOCA):
  - 1 main pump + 1 HX Canada (class IV power available);
- All external LOCAs (except In-core LOCA):
  - 1 main pump started at 1000 seconds + 1 HX Canada (class IV power un-available);
- All external LOCAs:
  - 1 main pump + 2 HX China, the Republic of Korea, Romania;
- Moderator Temperature Control functionality:
  - Available Canada, China, the Republic of Korea, Romania (for 3 hours for temperature control valve control - Canada);
- Cover gas functionality required:
  - Available all (for 2 hours Canada);
- DCC Operation:
  - Control programs have been considered in the PSA models to represent the plant operation. Program failures due to hardware or software failures were included. In addition, DCC failure is considered as a separate initiating event (Argentina, Canada, China, the Republic of Korea and Romania). Differences related to the crediting of the reactor control functions (setback and step back) have been identified.

IV-3. CONCLUSIONS

With the exception of differences in plant designs, differences in deterministic safety analyses (supplementary analysis in support of PSA) are considered the basis for the observed variability in the success criteria among similar plants.

Supplementary sensitivity analysis cases or refined analyses could be conducted to harmonize the success criteria between plants.

TABLE IV-1. EXAMPLES OF THE SUCCESS CRITERIA USED IN LEVEL 1 PSA FOR FULL POWER AS RECEIVED FROM CANADA

FRONT LINE SYSTEM		SAFETY FUNCTION	SUCCESS CRITERIA	INITIATING EVENTS	COMMENTS
Shutdown system #1		Subcriticality	26 out of 28 shut-off-rods	All	
Shutdown system #2		Subcriticality	5 out of 6 injection nozzles	All	
Loop isolation		Primary heat transport system inventory	All valves automatically closed at 5.52 MPa	Large loss of coolant accident; Pipe break upstream of pressurizer relief /steam bleed valves; Multiple tube ruptures in any RCW HX;	Not considered for other LOCA events
Boiler crash cooldown		PHTS pressure control	8 out of 16 MSSV's	Small LOCAs; In-core LOCAs	Not considered for large LOCA
			10 out of 16 MSSV's automatically open or Manual action within 15 minutes	Very small LOCA; Single steam generator tube rupture; Multiple single steam generator tube rupture;	
			10 out of 16 MSSV's open manual action	leak inside containment; leak outside containment;	
Emergency core cooling	High pressure & medium pressure	Primary heat transport system (PHTS) inventory control	High & medium pressure stages initiated automatically ECC flow to all headers in broken loop and one header in intact loop	Large LOCA	
		Primary heat transport system (PHTS) inventory control	High & medium pressure stages initiated automatically ECC flow to two headers in broken loop and one header in the intact loop	All small LOCAs (including containment bypass), in core LOCA	
			High & medium pressure stages initiated automatically ECC flow to one header in broken loop and one header in intact loop	Very small LOCA	
	Low pressure (LP)		Low pressure stage initiated automatically ECC flow to all headers in broken loop and one header in intact loop, service water cooling available to ECC HX	Large LOCA	
			Low pressure stage initiated automatically ECC flow to two headers in the broken loop and one header in the intact loop, service water cooling available to ECC EWS	All small LOCAs (except for the containment bypass), in core LOCA	
			Not required for very small LOCA		

TABLE IV-1. EXAMPLES OF THE SUCCESS CRITERIA USED IN LEVEL 1 PSA FOR FULL POWER AS RECEIVED FROM CANADA (cont.)

FRONT LINE SYSTEM	SAFETY FUNCTION	SUCCESS CRITERIA	INITIATING EVENTS	COMMENTS
Emergency water system	Heat sink	Success criteria: • EWS provided to one boiler per loop • PV7 has to be manually selected opened & maintained opened. PV41 is closed for long term pumped EWS • One out of two EWS pumps is credited to start and run	All transients	
Emergency water system	Heat sink	Success criteria: • EWS provided to both boilers of the intact loop • PV7 has to be manually selected opened and maintained open. PV41 is closed for the long term pumped EWS • One out of two EWS pumps is credited to start and run	Large LOCA events	
		Success criteria: • EWS provided to all 4 boilers • PV7 has to be manually selected opened and maintained opened. PV41 is closed for long term pumped EWS • One out of two EWS pumps is credited to start and run	All other LOCA events	
Boiler make-up water system	Heat sink	• One MSSV opened in at least one boiler per loop • PV7 & PV41 credited to open manually • Flow provided to one boiler per loop	All transients	
PHT Pressure Relief valves	PHT pressure control	• 1 out of 4 Liquid Relief Valves 3332-PV3, PV4, PV12 or PV13 opens on logic	Liquid relief valves spurious open; loss of feedwater, DCC, loss of instrument air, loss of service water, loss of class IV, partial loss of heat transport system pumped flow	

TABLE IV-1. EXAMPLES OF THE SUCCESS CRITERIA USED IN LEVEL 1 PSA FOR FULL POWER AS RECEIVED FROM CANADA (cont.)

FRONT LINE SYSTEM	SAFETY FUNCTION	SUCCESS CRITERIA	INITIATING EVENTS	COMMENTS
Shutdown cooling system	Heat sink	• One out of 2 PHT pumps per loop or 2 out of 2 SDC pumps for the first 6 hours, then 1 out 2 SDC pumps • 2 out of 2 Heat Exchangers • Service water available to SDCS, HX, PHT pumps and SDCS pumps	All transients except for loss of Class IV, PHT pump spurious trip or loss of service water system	
		• 2 out of 2 SDC pumps for the first 6 hours, then 1 out of 2 SDC pumps • 2 out of 2 heat exchangers • Service water available to SDCS HX and SCDS pumps	Spurious trip	
Pressure inventory control	PHTS inventory and pressure control	• 1 out of 2 D_2O feed pumps available • 1 out of 2 F/M D_2O Supply pump • 1 out of 2 feed valves LCV11/ LCV12 control automatically • 1 out of 2 bleed valve LCV14/ LCV15 control automatically • 1 out of 2 feed isolating valves 33310-MV13 or MV22 to remain open • 1 out of 2 bleed isolating valves 33350- MV3 or MV4 to remain open • D_2O supply from 33330-TK1 • Make-up to TK1 from D_2O supply or • Make-up from D_2O recovery	Very small LOCAs, leakages, single steam generator tube rupture	

TABLE IV-1. EXAMPLES OF THE SUCCESS CRITERIA USED IN LEVEL 1 PSA FOR FULL POWER AS RECEIVED FROM CANADA (cont.)

FRONT LINE SYSTEM	SAFETY FUNCTION	SUCCESS CRITERIA	INITIATING EVENTS	COMMENTS
Pressure inventory control	PHTS inventory and pressure control	• 1 out of 2 D_2O feed pumps available • 1 out of 2 F/M D_2O supply pump • 1 out of 2 feed valves LCV11/ LCV12 control automatically • 1 out of 2 bleed valve LCV14/ LCV15 control automatically • 1 out of 2 feed isolating valves 33310-MV13 or MV22 to remain open • 1 out of 2 bleed isolating valves 33350- MV3 or MV4 to remain open • D_2O supply from 33330-TK1	All transients except for loss of service water	
		• 1 out of 2 D_2O feed pumps available • 1 out of 2 F/M D_2O supply pump • 1 out of 2 feed valves LCV11/ LCV12 control automatically • 1 out of 2 feed isolating valves 33310-MV13 or MV22 to remain open • D_2O supply from 33330-TK1 • 3331-PV25 close automatically on high HX2 temperature or manually or 3335-MV3 and MV4 close automatically on high HX2 temperature or manually	Loss of service water	
Degasser condenser	PHT inventory and pressure control	• PV16 or LCV8 and LCV15 closed automatically on high temperature signal • Degasser flow valves close automatically on high D/C pressure signal • Degasser spray valve close automatically • RV11 and RV21 to remain closed • Bleed valve to be manually closed	Liquid relief valves spurious open; Loss of feedwater, DCC, Loss of instrument air, loss of service water, loss of class IV, partial loss of heat transport system pumped flow	

TABLE IV-1. EXAMPLES OF THE SUCCESS CRITERIA USED IN LEVEL 1 PSA FOR FULL POWER AS RECEIVED FROM CANADA (cont.)

FRONT LINE SYSTEM	SAFETY FUNCTION	SUCCESS CRITERIA	INITIATING EVENTS	COMMENTS
Feedwater system	Heat sink	• Feedwater to 1 boiler per loop • 1 out of 3 main Feedwater pumps or 1 out of 2 auxiliary feedwater pumps • Feedwater supply from DA storage tank or reserve feedwater tank • Make-up provide via main or aux. condensate • Automatic flow control via small feedwater LCVs • Main pumps and standby pumps cooled by RCW or backup cooling	All transients except for loss of Class IV	
		• Feedwater to 1 boiler per loop • 1 out of 2 auxiliary feedwater pumps • Feedwater supply from DA storage tank or reserve feedwater tank • Make-up provide via aux. condensate • Automatic flow control via small feedwater LCVs • Auxiliary pumps cooled by RCW or backup cooling	Loss of Class IV	
Main moderator system	Heat sink	• 1 out of 2 main moderator pump with standby starting from automatic low delta-P signal • Cooling from 1 out of 2 moderator heat exchanger • Temperature control valve controls via DCC for first 3 hours (IA isolated after 3 hours) • Moderator crash cooling	All external core LOCA with Class IV available	Moderator cannot be credited after in core LOCA

TABLE IV-1. EXAMPLES OF THE SUCCESS CRITERIA USED IN LEVEL 1 PSA FOR FULL POWER AS RECEIVED FROM CANADA (cont.)

FRONT LINE SYSTEM	SAFETY FUNCTION	SUCCESS CRITERIA	INITIATING EVENTS	COMMENTS
Feedwater system	Heat sink	• Feedwater to both boilers of the intact loop • 1 out of 3 main Feedwater pumps or 1 out of 2 auxiliary feedwater pumps • Feedwater supply from DA storage tank or reserve feedwater tank • Make-up provide via main or aux. condensate • Automatic flow control via small feedwater LCVs • Main pumps and standby pumps cooled by RCW or backup cooling	Large LOCA	
		• Feedwater to all 4 boilers • 1 out of 3 main feedwater pumps or 1 out of 2 auxiliary feedwater pumps • Feedwater supply from DA storage tank or reserve feedwater tank • Make-up provide via main or aux. condensate • Automatic flow control via small feedwater LCVs • Main pumps and standby pumps cooled by RCW or backup cooling	All other LOCA events	
Boiler pressure control	Secondary side pressure control	Cooldown: 3-out-of-10 CSDVs controlled via BPC or 4-out-of-16 MSSVs opened manually via MCR or secondary control area hand switch (one MSSV per boiler)	All transients except for loss of instrument air, loss of Class IV and dual computer failure	
		Cooldown: 4-out-of-16 MSSVs opened manually via MCR or secondary control area hand switch (one MSSV per boiler)	Loss of instrument air, loss of Class IV and dual computer failure	

TABLE IV-1. EXAMPLES OF THE SUCCESS CRITERIA USED IN LEVEL 1 PSA FOR FULL POWER AS RECEIVED FROM CANADA (cont.)

FRONT LINE SYSTEM	SAFETY FUNCTION	SUCCESS CRITERIA	INITIATING EVENTS	COMMENTS
Boiler pressure control	Secondary side pressure control	Pressure relief: 3-out-of-10 CSDVs or 4-out-of-16 MSSVs (one MSSV per boiler) opened at their respective boiler pressure relief setpoint only	All transients except for loss of instrument air, loss of Class IV, loss of condenser vacuum and dual computer failure	
		Pressure relief: 4-out-of-16 MSSVs (one MSSV per boiler) opened at their respective boiler pressure relief setpoint only	Loss of instrument air and dual computer failure	
Service water system	Support system, ensuring cooling to HX and pumps	• RCW flow provided by 1 out of 4 pumps with 2 previously running • RSW flow provided by 1 out of 4 pumps with conditioning of winter or summer mode • 1 control valve assures adequate RCW system pressure control • 2 of 5 service water system HXs, are available to remove the loads heat after an initiating event • 1 of 3 traveling screens is available to wash fish/debris • 1 of 3 traveling screen pumps is available to assure adequate screens wash flow • 1 of 2 RSW bearing cooling water pumps is available	All transients except loss of instrument air or loss of Class IV	
Instrument air system	Support system	• 1 out 3 compressors available • 1 out 2 dryers available • Cooling from service water or back-up cooling	All events	

TABLE IV-1. EXAMPLES OF THE SUCCESS CRITERIA USED IN LEVEL 1 PSA FOR FULL POWER AS RECEIVED FROM CANADA (cont.)

FRONT LINE SYSTEM	SAFETY FUNCTION	SUCCESS CRITERIA	INITIATING EVENTS	COMMENTS
Service water system	Support system, ensuring cooling to HX and pumps	• 1 out of 4 RCW pumps with successful Load Shedding OR 2 out of 4 RCW pumps if load shedding fails • RCW flow provided by 1 out of 4 pumps with 1 control valve assures adequate RCW system pressure control • 2 of 5 service water system HXs, are available to remove the loads heat after an initiating event • 1 of 2 traveling screens is available to wash fish/debris • 1 of 3 traveling screen pumps is available to assure adequate screens wash flow • 1 of 2 RSW bearing cooling water pumps is available	Loss of Class IV	
		• RCW flow provided by 2 out of 4 pumps • RSW flow provided by 1 out of 4 pumps • 1 control valve ensures adequate RCW pressure control • 2 out of 5 service water system HXs, are available to remove the loads after an initiating event • 1 out of 2 traveling screens is available to wash fish/debris • 1 of 3 traveling screen pumps is available to assure adequate screens wash flow • 1 of 2 RSW bearing cooling water pumps is available	LOCA or loss of instrument air	
Class IV power	Support system	Provide power to either odd or even buses	All events except for loss of Class IV	
Class III power	Support system	Provide power to either odd or even buses	All events	
Class II power	Support system	Provide power to either odd or even buses	All events	
Class I power	Support system	Provide power to either odd or even buses	All events except for the loss of Class I power	
EPS	Support system	1 out of 2 EPS diesel provides		

ANNEX V. MISSION TIME

V-1. BACKGROUND

For many types of non-CANDU reactors Ref. [V-1], **standard (or default)** mission times of 24 hours or 48 hours are often used. Standard mission time is the default assumed for most PSA systems.

V-1.1. Trend summary

1) Argentina, China, the Republic of Korea and Pakistan use a standard mission time of 24 hours. However, the Canadian utilities adopt a mission time of 72 hours.
2) A smaller mission time (< 24 hours) was reflected for pumps.
3) The definition of mission time for level 2 PSA appears to be at an early stage of development.

V-1.2. Conclusion

It would be worthwhile to consider whether 72 hours (as opposed to 24 hours) is a general Canadian PSA trend or whether the difference is a manifestation of inherent design differences between the single and multiunit CANDU designs.

V-2. COMPILATION OF PARTICIPANT RESPONSES

V-2.1. Question 1: What mission times are used in the level 1 PSA models (short, standard or long term)?

MEMBER STATE	RESPONSE
Argentina	A standard mission time of 24 hours was considered in Embalse Level 1 PSA. In some cases, mission time is very short and failure to operate is disregarded (i.e. mission time for high pressure emergency core coolant (ECC) in large loss of coolant accident was disregarded).
Canada	The standard mission time of 72 hours was selected for most systems. The selected mission time was considered to be a conservative value, since it was judged to allow enough time for suitable action to alter the course of an accident sequence.
China	A mission time of 24 hours for accident sequence quantification was used for the Third Qinshan Nuclear Power Plant (TQNPP) PSA.
Republic of Korea	Only one standard mission time defined as 24 hours is used at the latest Wolsong CANDU PSAs
Pakistan	The Karachi Nuclear Power Complex (KANUPP) is a single unit plant and generally a standard mission time of 24 hours is used. It was evaluated that after 24 hours stable condition is reached. In the event 24 hours is not accurate, a larger mission time needs to be used. This situation did not occur in KANUPP PSA.

V-2.2. Question 2: What CANDU systems fall into the various mission time categories? Provide any available justification.

MEMBER STATE	RESPONSE
Argentina	Some exceptions considered were shutdown cooling (SDC) system 2 out of 2 pumps and 2 out of 2 heat exchangers (HXs) for 6 hours; 1 out of 2 pumps and 1 out of 2 HX for 18 hours.
Canada	The exceptions were the emergency coolant injection system, moderator system, airlocks and transfer chambers, the hydrogen ignition system, t pressure relief valve (PRV) system and the emergency filtered air discharge system (EFADS) for which a 30 day mission time (long term) was chosen. This long term mission time was chosen because core cooling and containment integrity may need to be assured for that length of time without the possibility of repair. For the Bruce A PRA, several short term mission times were used: 24 hours for standby generators, 36 hours for the qualified power supply generators, and some short phase operation of emergency coolant injection. The systems used for power reduction (including shutdown) are considered to have extremely short, zero mission times.
Republic of Korea	The standard mission time is used for all the components when a running failure mode is defined. Although it is not explicitly described, justifications can be made based on repairing actions, repairing time and system redundancy.
Pakistan	For all front line and support systems modelled in level 1 PSA, a single standard mission time of 24 hours is used. However, for specific situations or components, short mission times have been used. For example, multiple starts are modelled for diesel generator fuel oil pump, which needs to start and run for multiple cycles for makeup of day tank during the DGs mission time of 24 hours. The diesel generator fuel oil pump starts on low level of day tank and trip on high level after running 10 minutes.

V-2.3. Question 3: Would a level 2 analysis introduce a different mission time for some systems that use the standard mission time in level 1 PSA?

MEMBER STATE	RESPONSE
Argentina	In future development of level 2 PSA, a more specific definition of mission time will be considered
Republic of Korea	No different mission time is used at the level 2 PSA analysis
Pakistan	Level2 PSA analysis is not performed for KANUPP Plant

REFERENCES TO ANNEX V

[V-1] INTERNATIONAL ATOMIC ENERGY AGENCY, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-3, IAEA, Vienna (2010).

ANNEX VI. RISK INSIGHTS AND USE OF PSA

VI-1. DESIGN/PLANT CONFIGURATION

VI-1.1. NPP updates / plant modification / configuration control

MEMBER STATE	PRESENT THE RISK / IMPROVEMENT	OBSERVATIONS
Argentina	The emergency core cooling system design was upgraded/ improved to reduce the percent contribution to CDF from 50%– 6%	The design modification was completed to improve the availability of the ECC system. Human action was initially required for very small LOCAs but was improved by automation
	The PSA was used to assess the operating conditions with regard to failures of inverters	Analysis concludes the possibility to keep the plant in operation
Canada	Some plant improvements were identified from external hazards risk assessments, such as to tie down of emergency mitigating equipment to withstand high winds	
China	1) The PSA model is being used to assess the risk of scheduled and unscheduled maintenance work during the full power 2) Shutdown low power (SDLP) PSA model is used to assess the risk of specific configuration during outage. The Third Qinshan Nuclear Power Plant (TQNPP) requires that 3 of the 4 power supplies (standby diesel generator 1 & 2, station service transformer and main output transformer/ unit service transformer) is available during outage. TQNPP is using the SDLP PSA model to assess the possibility that only one offsite power supply and one train of standby diesel generators are required availability of Mode 5b. However, we do not make the final decision by now	
Republic of Korea	Containment integrity was improved due to a post-Fukushima action of CFVS (containment filtered vent system) installation 1) LCF frequency was reduced by ~98% 2) Total CF frequency was reduced by ~ 16%	
	Korea Hydro and Nuclear Power uses the following risk monitoring systems to continuously monitor plant configuration: 1) RIMS (risk monitoring system) for full power operation 2) ORION (outage risk indicator of NPPs for low power shutdown operation 3) SPV (single point vulnerability) monitoring system to prevent the transients induced by inadvertent maintenance or test of a specified single component	
	Improvement of the automatic signals for ECCS from an additional installation of sustained heat transport system low pressure Internal CDF was reduced by ~93%	

MEMBER STATE	PRESENT THE RISK / IMPROVEMENT	OBSERVATIONS
India	PSA results and insights are used for the following purposes: • Regulatory risk informed decision making. For example, PSA results are used for the review of new designs, system design upgrades/modifications, and revision of technical specification • Licensing during the preliminary safety analysis report (PSAR) stage and relicensing of operating plants during the periodic safety review (PSR) • Determination / optimization of the level of redundancy for critical components (i.e. ECCS for new designs) • Detailed deterministic analysis performed to determine realistic success criteria • Surveillance test interval (STI) optimization for some sensors in reactor protection system (RPS) for new designs	
Pakistan	Areas showing a high contribution to risk were analysed and, in some cases, re-evaluated by deterministic/thermal hydraulic groups to remove unnecessary conservatism to reduce the CDF All plant modifications require change approvals (CAs). These change approvals contain a check list, including PSA group evaluation prior to finalization. The PSA results ultimately determine whether the CA can be implemented	
Romania	Initially, the transfer of MP ECC to LP ECC required manual action. The transfer was automated in order to reduce the risk associated to inadequate operator response Initially, there was a manual conditioning signal for ECC initiation for some small LOCAs. There was the implementation of a supplementary conditional signal for ECC initiation	
	Response to technical operability evaluations.	PSA results are used during the technical operability evaluations

VI-2. OPERATION

VI-2.1. Evaluation of safety issues and identification of plant vulnerabilities

MEMBER STATE	PRESENT THE RISK / IMPROVEMENT	OBSERVATIONS
Argentina	The PSA identified a weakness such that it is impossible to check the status of the check valves V96/97/76/77 after the action of ECC. Valves 3432 PV /33/34/47/48 can be operated as an alternative and are controlled from MCR.	Valves 3432 PV /33/34/47/48 can be operated from MCR through test circuits
Canada	Level 2 PSA analysis provides insights on critical containment components that may benefit from restoration of power in a severe accident. In addition, many plant vulnerabilities were identified and improved using external hazards risk assessments based on level 1 and level 2 PSA.	

MEMBER STATE	PRESENT THE RISK / IMPROVEMENT	OBSERVATIONS
China	TQNPP has the backup recirculating service water system (BRCW), which is used as alternative heat sink to recirculating cooling water (RCW) during the outage. The SDLP PSA results showed that the risk is high when BRCW is used during the mode 5c, low level drained state (LLDS)	
Republic of Korea	The safety critical components identified based on risk reduction worth (RRW). Operators are educated to give more attention to these components in daily work-down 1. Common cause failure (CCF) of main steam safety valve (MSSV) failure to operate (FTO) 2. Instrument air (IA) system 3. Standby diesel generator failure to run 4. Auxiliary feedwater (AF) pump testing & maintenance (T&M) 5. Emergency service water (ESW) pump T&M	
India	The safety issues/events that occurred during operation of the plant is considered in PSA studies The operating experience of the nuclear power plant, feedback/significant event reports, and partial failures of equipment are included in the determination of initiating events of PSA studies and in subsequent analysis The operator actions leading to initiating events are also appropriately considered in PSA studies	
Pakistan	Safety issues with respect to the plant are discussed with PSA group and after discussion/ analysis. The decision using the PSA results as one of the major inputs Vulnerabilities were identified when using the PSA results, and modifications were completed through corrective actions	
Romania	The risk associated to different sequences has been acknowledged and plant modifications / improvements have been proposed for implementation	

VI-2.2. Plant maintenance and ranking of safety critical components

MEMBER STATE	PRESENT THE RISK / IMPROVEMENT	OBSERVATIONS
Argentina	PSA was not initially used in the maintenance plan. The maintenance plan uses the original criteria based on engineering design	
Canada	PSA has been used to update the ranking of components at the stations	
China	The Fussell-Vesely (FV) importance list is used to support the ranking of safety critical components	
India	PSA results are used for ranking the safety critical components. PSA studies identified important manual valves in front line and support systems. This helped in performing physical inspection before start-up and after annual shutdown to ensure that they are put back in the desired position after maintenance. Verification of the identified valves was included in the post maintenance check list	
Pakistan	A safety report is issued every year where targets are set for critical safety components. Each year the reliability of these critical safety components is evaluated using plant data (i.e. from maintenance events) and compared with the set targets. This enabled improved planning and maintenance scheduling	
Romania	Equipment out of service (EOOS) model is incorporated in the plant operation in order to evaluate the impact of plant maintenance activities	

VI-2.3. Technical specifications

MEMBER STATE	PRESENT THE RISK / IMPROVEMENT	OBSERVATIONS
Argentina	The information provided by the level 1 PSA and based on the minimum conditions for residual heat removal required chains was used	The TSs of critical components was evaluated. For example, they were evaluated for challenges of external events
Canada	Not applicable to OPG and Bruce Power	
China	The risk informed method was used to extend the test interval of high log N rate trip functional tests and verify the neutron overpower trip function from 7 days to 14 days	
India	PSA is being used to optimize TS with regard to allowed outage times (AOTs) and surveillance test intervals (STIs), to assure reliability in functioning of SSCs Some changes to TSs are based on PSA results, including the AOT for start-up transformer, diesel generator (DG) sets, ECCS motorized valves, and the secondary shutdown system (SSS) helium circuit bank Modification in STI for fire water system back up valves (moderator HX injection valves), and atmospheric steam discharge valves are supported by PSA studies	
Pakistan	Input from PSA was considered before any decision regarding revision of technical specifications. Some examples include the involvement of human actions, operating procedures, AOTs, and STIs	

VI-2.4. Human error probability

MEMBER STATE	PRESENT THE RISK / IMPROVEMENT	OBSERVATIONS
India	Dominant human actions are identified in crew training programs and simulator training Emergency operating procedures (EOPs) are improved for human actions identified in the PSA. For example, the following elements were improved: 1) Manual initiation of ECCS for lower range of small break loss of coolant accident (SBLOCA) scenarios 2) Manual initiation of secondary cool down in case of total loss of feed water	

VI-3. ACCIDENT MANAGEMENT

VI-3.1. Improve operator training

MEMBER STATE	PRESENT THE RISK / IMPROVEMENT	OBSERVATIONS
Argentina	Improvement of procedures because of possible lack of water supply to the steam generators in case of automatic opening of the valve PV41 to EWS. Also, changes POEAS 3 and 4	Instruction to close valve PV41 was not possible if it automatically opened. The operator has a procedure to close the valve manually
Canada	PSA is a part of operator training to provide them with an introduction of the purpose of PSA, its inputs, and how insights used. EOOS models are based on PRA are being incorporated in operations. PRA orientation training is implemented for the engineering groups	
Republic of Korea	1. Fail to operate standby diesel generator 2. Manual reactor trip 3. Recovery of class 4 (CL4) power	Enhanced training program of important actions from PSA results
India	Level 1 PSA results highlight human errors as one of the significant contributors to core damage. PSA results aided in the selection of accident scenarios for training Based on PSA studies, special emphasis is given on training and licensing programs for operators to create awareness about the consequences of rare events requiring prompt operator response One of the important conclusions from internal flood PSA is to train operators to develop an understanding of possible flooding scenarios, water flow tendencies and required actions to avoid accumulation of water Severe accident scenarios were included in classroom training for plant operators	
Pakistan	There are some instances where human actions are added the training program of plant operators	
Romania	The operational staff are trained for more diverse accident sequences during the plant emergency drills	

VI-3.2. Support accident management (improve EOPs/APOPs)

MEMBER STATE	PRESENT THE RISK / IMPROVEMENT	OBSERVATIONS
Argentina	It was found that in cases of earthquakes or station blackout the EWS system ventilation is not required	Tests were performed by operating pumps without room fans for 24 hours. It was verified that conditions remained constant on a warm day. It was concluded that there is no reason to modify the ventilation fan capacity
Canada	Improvements were made to EOPs for additional trigger points for emergency mitigating equipment deployment	
India	For severe accident management guidelines (SAMG) development, PSA results are used to identify all accident sequences with potential to lead to core damage. An exhaustive list was made available The SAMG indicated hook up arrangements, operator actions related to SAMG which are based on PSA studies PSA studies are used for the improvement of EOPs, abnormal plant operating procedures, and SAMGs In addition, level 2 PSA considering deterministic safety analysis (DSA) for SAMGs is performed	
Pakistan	PSA was one of the major inputs for the development of EOPs	
Romania	Supplementary specific APOPs have been developed for spent fuel bay events and for station blackout	

VI-3.3. Support emergency planning

MEMBER STATE	PRESENT THE RISK / IMPROVEMENT	OBSERVATIONS
Argentina	Not used on this area up to now	
Canada	Insights from severe accident analysis have been used for improvements to SAMGs, defining accident progression for drills and potential radiological impact	
India	PSA studies enabled identification of significant contributors for large and early releases	
Pakistan	PSA is not used for emergency planning yet	
Romania	The set of emergency drill scenarios was supplemented based on the set of PSA accident sequences	

VI-4. REGULATORY PROCESS

VI-4.1. Decision making

MEMBER STATE	PRESENT THE RISK / IMPROVEMENT	OBSERVATIONS
Argentina	Extended the period of operation between outages from 12 to 18 months	PSA results were used for risk informed decision making, because of changes in the frequency of periodic safety systems testing
Canada	Support the annual reliability program reporting (e.g. risk impact of observed impairments and missed tests). PSA can be used to evaluate any modifications in design, operation or operating procedures	
Republic of Korea	The PSA sensitivity study result of safety enhancement action items were considered in regulatory review process of continued operation (30 years to 40years) of Wolsong unit 1	
India	The AERB safety code AERB/NPP-PHWR/SC/D (Rev.1), Design of Pressurised Heavy Water Reactor Based Nuclear Power Plants Ref. [VI–1] elaborates about the probabilistic approach The deterministic approach is supplemented by probabilistic approach. PSA is done to prove that core damage frequency (CDF) and large early release frequency (LERF) is limited for a given NPP. Limits on CDF and LERF are specified as 10-5/reactor-year and 10-6/reactor-year respectively The AERB safety code, AERB/NPP/SC/O (Rev. 1), Nuclear Power Plant Operation Ref. [VI–2] elaborates the requirement of PSA As a minimum requirement, internal event plant-specific level 1 full power PSA is performed for all NPPs. For new NPPs, it is completed prior to first criticality and for NPPs in operation, it is updated and presented as a part of PSR The PSA is kept up to date during the plant lifetime taking into account design modifications, changes in operational practices and updated statistical data on initiating event frequencies and component reliability data obtained during the plant operation The AERB safety guide AERB/NPP-RR/SG/G-10, Regulatory review of level 1 probabilistic safety assessment for nuclear power plants and research reactors Ref. [VI–3] elaborates about review aspects of level 1 PSA. It is mainly applicable for review of level 1 PSA for NPPs and research reactors covering both internal and external events	
Pakistan	The PSA studies are widely used during the all-important stages of licensing/ authorization of nuclear power plants, including: a. Issuance of construction license b. Issuance of fuel load permit c. Revalidation of operating license (after every 10 years) d. Licensing beyond design life	
Romania	Evaluate the proposed design changes and improvements in terms of overall risk	

VI-4.2. Evaluation and rating of the initiating events

MEMBER STATE	PRESENT THE RISK / IMPROVEMENT	OBSERVATIONS
Argentina	Operational experience is analysed periodically to update the model	This item is covered by procedures for handling operational experience. Eventually, operational experience results are used to update PSA model. For instance, upgrading the frequencies of initiating events groups
Canada	PSA is used to evaluate impact of experienced initiators and the significance of operational events. It can also be used to identify precursors	
India	Significant event reports are prepared, and root cause analyses are carried out. Operating initiating events are rated based on their contribution to CDF (conditional core damage probability (CCDP) to the occurrence of the event)	
Pakistan	PSA is not used specifically in this area yet	
Romania	Used to identify the possible initiating event precursors from the plant operational transients / verify the initiating event frequency	

VI-4.3. Regulatory staff training

MEMBER STATE	PRESENT THE RISK / IMPROVEMENT	OBSERVATIONS
Argentina	PSA is used for specialists only	PSA is used by subject specialists but is not used for training of other staff members.
Canada	For inspection activities, the regulatory staff can be trained using information from the Licensee PSAs to identify important SSCs, initiating events and human actions	
India	'Engineering and science post-graduates are trained through highly specified course modules in the training school AERB has arrangements with various IITs (which are the premier academic institutes of international repute) under which highly qualified engineers are inducted via direct recruitment at the required level of expertise and experience All newly joined personnel undergo an AERB 'Orientation Course for Regulatory Processes' (OCRP), which elaborates all the functions and responsibilities of AERB along with the methodologies used. After this course, the officials undergo 'On-Job Training' involving various activities for further training. AERB recruited officers undergo the level–III training to qualify for the control engineer's license for operating NPPs. This training provides an opportunity for in-depth understanding of the SSCs employed for safety assurance in an NPP. This practice has been recognized internationally for the perspective it adds to the regulatory supervision and for the first-hand-experience of the operational practices and constraints of an NPP	

	Assignments with experienced officials enable exposure to the regulatory perspective and safety culture. They participate in safety review committees, deliberations of various working groups and expert groups and gradually get involved in providing assistance in regulatory activities	
Pakistan	The Pakistan Nuclear Regulatory Authority (PNRA) developed an independent regulator's level 1 full power PSA model for the training regulatory staff. PNRA is in the process of developing an independent level 2 PSA model for its 300 MWe NPPs. Moreover, regular training courses, fellowships and workshops are arranged with the help of IAEA for the regulatory staff	
Romania	Regulatory staff training includes: - On-site staff training for evaluation of the outage work planning activities - General regulatory staff training - Evaluation of operating procedures regarding the expected operator actions	

VI-4.4. Surveillance activities

MEMBER STATE	PRESENT THE RISK / IMPROVEMENT	OBSERVATIONS
Romania	Regulatory surveillance of the instantaneous risk induced by specific plant configurations during outage periods	On-site regulatory surveillance using EOOS tool to determine the instantaneous risk during outage periods and compliance with approved procedures by the Romania National Commission for Nuclear Activities

VI-4.5. Licensing process

MEMBER STATE	PRESENT THE RISK / IMPROVEMENT	OBSERVATIONS
Argentina	PSA is mandatory for the operating license for NPPs in operation. Operating licenses have a periodical safety review, which includes updating the PSA	In case of new plants, the Argentina regulatory framework has a standard requiring PSA levels 1, 2 and 3. This standard was established after starting the operation of Embalse NPP
Canada	PSA requirements for operating licences are outlined in REGDOC-1.1.3 Ref. [VI–4]. PSA requirements for the application of licence to construct are outlined in RD/GD-369 Ref. [VI–5]. In addition, PSA may also be used in support other licensing activities such as environment assessment, license to prepare site, licence to decommission, etc	
Romania	Assessment of the compliance with the safety goals required by regulations	

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ANNEX VII. FUKUSHIMA LESSONS LEARNED

This Annex contains information on lessons learned from the Fukushima Daiichi accident and their implications on PSA.

VII-1. RESULTS OF CPWG REGULATORY QUESTIONNAIRE

For task 2011-06, Canada sent out a questionnaire to all CPWG Member States on the topic of regulatory impacts as a result of the Fukushima event. A brief analysis of the questionnaire is given below:

VII-1.1. Question 1: Are there any implications or direct changes to the regulatory framework and processes in your country?

Responses to Question 1:

- Argentina is working on a proposal to amend/update the Regulatory Standard AR 3.10.1, Protection against earthquakes in Nuclear Power Plants Ref. [VII-1].
- For India, there are no direct changes to the regulatory framework and safety audits were performed.
- Canada is amending Class I Nuclear Facilities Regulations, the Radiation Protection Regulations, and few other specific regulatory documents.
- No change in the Pakistan regulatory framework.

VII-1.2. Question 2: Any good practice directly related to PSA?

Responses to Question 2:

- Argentina and Pakistan added PSA-based seismic margin assessment (SMA) and seismic PSA as a requirement.
- For India, no specific conclusions related to PSA are derived in light of the Fukushima event.
- Canada identified 2 action items to be addressed through Ref. [VII-2].

VII-1.3. Question 3: What are the safety goals, and have they changed as a result of the Fukushima Daiichi accident?

Responses to Question 3:

- No change in the existing safety goals for all the responders.

VII-2. RESPONSES TO EXTERNAL HAZARD QUESTIONNAIRE

An external hazard questionnaire was sent to CPWG Member States. Below are the questions asked and country-specific responses:

VII-2.1. Question 1: General approach adopted or used for the initial identification and screening of the external hazards (probabilistic and/or deterministic).

- Does the methodology consider any comparison with generic lists of potential external hazards? Indicate the documents used as references.
- How is made the identification of site-specific external hazards for the plants?
- Potential combined external hazards are considered in this kind of identification process?
- Which are the screening criteria used? Specify qualitative and/or quantitative criteria.

ISSUE OF INTEREST	ARGENTINA	CANADA	CHINA	INDIA	PAKISTAN	ROMANIA
Comparison with generic lists of potential external hazards	For external hazards, while not explicitly specified any comparison with referenced list, all related events with seismic hazards, high winds and external floods were analysed in the final safety analysis report (FSAR). For human induced hazards it was used a specific reference document provided by local firefighter organization	The hazard identification stage starts with consideration of existing literature that include list of external hazards. These include IAEA (Refs [VII-3, 4, 5]), NUREG (Ref. [VII-6, 7]) and ASME Appendix 6 A (Ref. [VII-8]) publications	The general approach used for the screening of external hazards considers the generic lists of potential external hazards.	AERB has prepared a PSA manual, providing guidelines for carrying out external event PSA. The publication suggests preparing an exhaustive list of all potential external events at NPPs. Based on the site characteristics, some of these events would be screened out from further analysis	PSAs for external hazards have not been performed for Karachi Nuclear Power Complex (KANUPP). Some of the external hazards were addressed in FSAR for KANUPP. A general approach is used in FSAR for screening of external hazards	The generic list includes some IAEA (Refs [VII-3, 9, 10]) and US NRC (Ref. [VII-6, 11]) external hazards publications
Identification method of site-specific external hazards for the plants	The external events identification is presented at final safety analysis report (FSAR) of Embalse NPP. This assessment is based in different studies covering different areas of interest including: • Climatological assessment of the site • Regional hydrology system • Regional and local geology and seismology • Human activities on the region that eventually could cause events • Specific environmental reports on events at the site related to meteorology and tornadoes, hydrological data and earthquake risk assessment Also, OPEX took into account, and administrative countermeasures were adopted for these events (intense rains, changes in land uses, and ecological changes in the lake)	Based on company-wide and site-specific documents such as station hazard analysis reports, abnormal incidents manuals, station safety reports and operational experience (OPEX)		Based on the site characteristics	On the basis of historic data of the region- and site-specific analysis was performed at design stage	The list of potential initiators was developed through plant / site specific walk downs, detailed regional investigations, geographic studies and consultation with relevant experts and their publications. This list was systematically reviewed as part of the screening analysis
Potential combined external hazards	Potential external hazards combinations were not explicitly and systematically considered up to now in the analysis. However, some cases were considered of interest such as the ability of the dam to withstand a design basis earthquake. This combination could cause extreme	Combinations of external hazards are considered. These consist of coincidental, consequential, and correlated events	Potential combined external hazards are not considered	The utility has completed a comprehensive external event PSA's for flood and seismic events for a representative NPP. In this study, potential for combined external hazards such as seismic-fire interactions, seismic-flood	No	Potential combinations of external hazards were considered. These consist of coincidental, consequential, and correlated events

ISSUE OF INTEREST	ARGENTINA	CANADA	CHINA	INDIA	PAKISTAN	ROMANIA
	downpipe of the lake, which is the main heat sink of the NPP. Also, as part of stress test for Embalse, the Nuclear Regulatory Authority asked for the analysis of Earthquake and flooding or low water (dam failures upstream and downstream). Combinations of external events and consequential failures (i.e. external event and prolonged grid loss) are considered			interactions, are not accounted for quantitatively		
Screening criteria used (qualitative and/or quantitative)	The screening criteria used are probabilistic in most of the cases. Identification is based on the event for which the likelihood is greater than some threshold value (screening frequency level). Examples of these are tornado and earthquake external initiators. For instance, the tornado design basis (DB) was estimated as 150 km/h wind velocity. This event has frequency about 6 E-05 /year for Embalse NPP siting. For earthquakes, the review level earthquake is an earthquake with recurrence exceedance of ten thousand years. In other cases, the criteria used is deterministic. For instance, external flooding was discarded because the dam spillway level is below the maximum water inlet level foreseen by the NPP facility. In other cases, screening distance value (SDV) is considered	Hazards are first assessed using qualitative screening criteria. If they cannot be screened out based on qualitative criteria, then the hazards are evaluated against quantitative criteria. Some qualitative criteria used include: • Consequences are within the plant design basis • Event has less severe consequence than other events screened out • Distance • Event is bounded by another event • Slow-developing event • Does not cause an initiating event and loss of a Safety System • Does not require actuation of front-line system Quantitative criteria are based on frequency of a hazard. The most common quantitative screening criteria is initiating event frequency with or without CCDP		Screening is based on the site characteristics	It was performed by the designer at the time of construction of plant.	Preliminary screening: a hazard can be excluded if one of different criteria, considering potential damage, occurrence frequency, distance to the plant or time of development, is met. Quantitative screening The following criteria are used to screen out an event: • The event cannot cause a core damage accident • The core damage frequency that is calculated using a quantitative bounding analysis has a mean value less than 10^{-6}/year and a median value less than 10^{-7}/year

VII-2.2. Question 2: External hazards that generally cannot be screened out (e.g. seismic, high winds, external floods or human induced hazards)

- Summarize the methodology used with the intention of reducing the list of external hazards subject to detailed analysis.
- Is bounding analysis based on certain range of specific parameter for each case (peak ground acceleration, wind velocity, water level)?
- For human induced hazards (fire spreading, explosions, chemical releases, aircraft crash, collisions of ships, etc.), describe the bounding analysis if applicable, for each case.

ISSUE OF INTEREST	ARGENTINA	CANADA	CHINA	INDIA	PAKISTAN	ROMANIA
Seismic hazards, high winds, external floods (if any of these are applicable): • Is bounding analysis based on certain range of a specific parameter for each case (peak ground acceleration, wind velocity, water level)?	In case of Embalse NPP, bounding analysis is based on horizontal peak acceleration and uniform hazard spectrum for earthquake events and wind velocity for extreme wind or tornadoes	Seismic hazards are outside the scope of methodology for other external hazards. Seismic events are considered in detail as part of separate PRA studies. The same applies to internal floods and fires. For external floods, the screening criteria are the same. However, the parameters are determined based on the location of the site and its susceptibility to external floods of various forms and sources. For high wind, tornados are considered bounding. A review level tornado needs to be defined, on the basis of current regulations, guidelines, and statistical data available. The methodology considers the types of hazard, the frequency of occurrence and intensity (i.e. of the wind / hurricane / tornado)		Specific parameters such as water level rise due to storm surge, precipitation and peak ground acceleration are considered for flood and seismic PSA studies	These hazards are already covered / discussed in FSAR. Moreover, after Fukushima some of them have been re-evaluated.	This response is also applicable to the quantitative screening criteria mentioned in the previous question. The following events were subject to bounding analysis: • Forest fires • External flooding • Extreme winds and tornadoes It was found that events could be screened out if they cannot cause core damage. Regarding water level hazards, comprehensive assessments have been performed for Cernavoda Site. The analysis was based on derivation of the hazard curve (intensity vs. frequency)
Human-induced hazards (fire spreading, explosions, releases of chemical materials from nearby units or facilities; aircraft crash, collisions of ships) • Describe bounding analysis, if applicable, for each case.	The initial screening identified potential cases like forest fire, fire affecting external grid and transmission lines and flammable fire due to traffic of flammable material on the road bordering the NPP. All these potential fire hazards were analysed based on distance and potential consequences, being disregarded.	The same screening criteria are used with the actual screening values being determined for each specific human-induced hazard.	The first PSR of Qinshan CANDU NPP was performed from 2013 to 2014. Another hazard analysis is performed in the first PSR.	Human-induced hazards are not analysed using probabilistic methods.	No potential sources of external fire and chemical releases are in the vicinity of KANUPP. An event like aircraft crash, etc., has already been re-assessed recently.	Human-induced hazards were screened out if they cannot cause core damage, except for aircraft crash that was screened out based on the following criterion: The core damage frequency that is calculated using a quantitative bounding analysis has a mean value less than 10-6/year and a median value less than 10-7/year

VII-2.3. Question 3: Specific list of external events considered in CANDU plants.

The intention is to identify events and adopted approaches for their analysis, with the aim of comparison between different plants/countries. The following information is of particular interest:

- Which methodologies are used for each external event analysis?
- Are specific methodologies included in the regulatory framework?
- Specify if secondary effects are considered in the approach adopted for each kind of external hazards (loss of coolant accidents, internal fire, flood, etc.).
- Include reference documentation describing methodology.
- Include (if applicable) results of the reassessment after Fukushima Daiichi accident.

ARGENTINA - EMBALSE NPP	
External event considered	Approach of analysis adopted (qualitative or quantitative, seismic PSA or SMA, etc.)
Extreme wind - Tornado	A new reassessment of tornadoes is currently under development (stress test) for refurbishment. Quantitative approach is used.
Earthquake	PSA based SMA for refurbishment of the plant
External flooding	Due to regulation by downstream dam, it is disregarded (qualitative approach).
Low water level	Consequential to earthquake, provisions are being analysed (quantitative approach).
CANADA (COMBINED)	
External event considered	Approach of analysis adopted (qualitative or quantitative, seismic PSA or SMA, etc.)
Earthquake	PSA based SMA – seismic PSA
Seismically induced fires and floods	Specific PSA methodology to be determined as part of post-Fukushima work based on EPRI (result of Fukushima)
Extreme ambient temperatures	Quantitative screening
High winds/tornados	Quantitative PSA
External flood	Quantitative screening
CHINA- QINSHAN CANDU NPP UNITS 1&2	
External event considered	Approach of analysis adopted (qualitative or quantitative, seismic PSA or SMA, etc.)
Seismic hazards	The EPRI SMA methodology was adopted for seismic margin assessment of Qinshan CANDU6 (C6) NPP. The overall objective of the EPRI SMA is to demonstrate that the plant has a seismic margin greater than the design basis earthquake (DBE). A review level earthquake with 0.3 g peak ground acceleration was selected for Qinshan EPRI SMA. It was observed that Qinshan has relatively good seismic design. The wide use of embedded parts rather than anchor bolts for the anchorage of most of the SSCs has increased the seismic ruggedness of the plant significantly. Secondary effects are not considered in SMA methodology.
External floods	Flood safety margin analysis is performed, and quantitative method was adopted. Probability is not considered. Heightening sea wall in-site was considered and performed in 2013. Secondary effects are not considered in this methodology.
INDIA	
At present, performing level 1 (Internal events, full power) is a mandatory requirement. However, performing external event PSA is desirable. Seismic PSA has been developed for a reference plant. Methodology given in Ref. [VII-3] is used for flood PSA studies.	
PAKISTAN - KARACHI NPP (KANUPP), CANDU 137MWE	
External event considered	Approach of analysis adopted (qualitative or quantitative, seismic PSA or SMA, etc.)
Earthquake	No responses received from Pakistan
CERNAVODA PSA	
External event considered	Approach of analysis adopted (qualitative or quantitative, seismic PSA or SMA, etc.)
Seismic events	Seismic PSA
Other external events mentioned above	Qualitative or quantitative screening

VII-3. FUKUSHIMA IMPLICATIONS ON PSA IN CANADA (REGULATORY ASPECT)

After the Fukushima Daiichi accident, the Fukushima Task Force Report (FTF) in INFO-0824 undertook a comprehensive review of the regulatory framework in Canada. They concluded that the Canadian regulatory framework is strong, comprehensive and effectively applied to the whole range of plant conditions including severe accidents. However, some actions were identified to improve reactor safety and defence-in-depth, emergency preparedness and, the Canadian nuclear regulatory framework.

Specifically, the Canadian Nuclear Safety Commission amended regulatory standard Ref. [VII-2], which was superseded by regulatory document REGDOC-2.4.2 in April 2014. The amendments are as given below:

a. **Objectives of the PSA**: These were newly added following the CNSC FTF recommendation which noted that Ref. [VII-2] does not spell out the purpose and the objectives for the conduct of the PSA. The added objectives are in accordance with those listed in the Ref. [VII-4].

b. **Consideration of other radioactive sources**: This was added as an amendment to the existing requirement directing the licensees to perform a level 1 and level 2 PSA for each NPP by explicitly stating that radioactive sources other than the reactor core, such as the irradiated fuel bay, are to be considered. The licensees may, with the agreement of persons authorized by the Commission, choose an alternate analysis method for assessment.

c. **Multiunit considerations**: This was added as an amendment to the existing requirement which directed the licensees to perform a level 1 and level 2 PSA by specifically stating that multiunit impacts are to be considered in the PSA.

d. **Inclusion of external events and their potential combinations**: This was added as an amendment to the existing requirement to include site specific initiating events (internal events, internal hazards, and external hazards) as well as the potential combinations of external hazards. The licensees may, with the agreement of persons authorized by the Commission, choose an alternate analysis method to conduct the assessment of internal hazards and external hazards.

e. **PSA update**: The PSA periodic update was changed from 3 to 5 years to align with safety report update submissions required by regulatory document REGDOC-3.1.1. PSA models are to be updated sooner if the facility undergoes major changes.

f. **Public disclosure**: Guidance was added following the public request for an increased disclosure of the PSA results and in accordance with licensees' public information programs established under RD/GD-99.3.

The CNSC FTF urged the licensees, through implementation of Ref. [VII-2] (in force at that time), to re-evaluate, using modern calculations and state-of-the-art methods, the site-specific magnitudes of each external hazard to which a site is susceptible and to evaluate if the current site-specific design protection is sufficient.

Re-evaluation of external hazards and the hazards screening analyses, through the implementation of Ref. [VII-2], was completed by the licensees as part of the CNSC FTF recommendations.

After the Fukushima Daiichi accident, there is an increasing interest from Canadian stakeholders regarding the risk posed by multiunit sites. Based on the Commission’s direction, the Canadian industry — through the CANDU Owners Group (COG) — has developed a

concept level whole-site PSA methodology, and a pilot project to perform a whole-site PSA was completed for the Pickering NPP before the end of 2017.

CNSC staff is actively engaged with industry and the international community for the development of a whole-site PSA methodology, and a CNSC working group for the development of site safety goals has been established. The Canadian industry held a workshop on multiunit PSA in January 2014 and the CNSC also hosted and organized the international workshop on multiunit PSA in November 2014.

VII-4. FUKUSHIMA IMPLICATIONS ON PSA IN CANADA (INDUSTRY ASPECT)

The key CNSC FTF recommendation that impacted the scope of PSA is the following:

> "A Level 1 and 2 PSA should be required to cover irradiated fuel bay events and multiunit considerations as well as plant wide internal fires, internal floods, seismic events and other external events", [VII-12].

Other FTF recommendations pertain to how PSA is to be performed and applied post-Fukushima. Eventually, all the FTF recommendations on PSA were mapped into REGDOC-2.4.2 which superseded Ref. [VII-2] as described in Section 5.1.1. Canadian licensees are now progressing on compliance with REGDOC-2.4.2.

At the time of the Fukushima Daiichi accident, Canadian utilities were already in various stages of developing PSAs to comply with Ref. [VII-2]. The industry took action to address the CNSC FTF recommendations wherever it was still possible in Ref. [VII-2] project phases. Notwithstanding this, the impact of Fukushima on recent Canadian industry PSA can be broadly classified into three categories:

1) Modelling of multiunit effect;
2) Scope of initiating events to be considered;
3) Other impacts on future PSA.

VII-4.1. Modelling of multiunit effect

The Fukushima event brought to the forefront the significant potential consequences of a common mode event (e.g. seismic) on stations with multiple units. All multiunit NPPs in Canada are currently in Ontario. These are summarized in the table below.

STATION	NO./ SIZE/ TYPE	COMMERCIAL
Pickering NGS-A	2 × 514 MWe	1971
Pickering NGS-B	4 × 514 MWe	1983
Bruce NGS-A	4 × 830 MWe	1977
Bruce NGS-B	4 × 850 MWe	1984
Darlington NGS	4 × 880 MWe	1990

The complexity of PSA modelling of multiunit sites arises because of a) shared or interconnected SSC; and, b) the physical interactions between units. There are many shared SSCs in a multiunit CANDU, examples are:

- Electrical systems;
- Emergency electrical systems;
- Emergency service water;
- Emergency coolant injection and recovery;

- Containment system;
- Shared powerhouse.

Pre-Fukushima PSA modelling already accounted for shared and interconnected systems to a significant degree. In general, a reference unit is modelled to be supported by other inter-unit systems and common systems. Internal initiating events that affect more than one unit (e.g. powerhouse steam line break) are explicitly accounted for at the system level. Fukushima's influence is primarily in increased awareness of the significance of inter-unit effects of shared and common systems, and the common effects of initiating events across units.

The second important multiunit aspect is the physical interactions across units brought about by the shared containment system and the environmental effect of an accident in one unit impacting neighbouring units. Although Ref. [VII-2] prior to the Fukushima event already recognized these aspects of multiunit PSAs, the multiunit nature of the Fukushima event created a higher expectation on the effort directed towards PSA consequence modelling of multiunit sites.

The impact of the Fukushima event on accident consequence modelling in PSAs was directed by actions in CNSC's integrated action plan which the nuclear utilities addressed as Fukushima action items. Among these were:

- Modelling of passive sources for boiler inventory make-up (e.g. deaerator);
- Modelling of emergency mitigating equipment as part of flexible response strategies;
- Increased emphasis on in-vessel retention phenomena;
- Assessing effect of on-site worker doses (habitability);
- Modelling possible enhanced containment heat sinks or venting strategies.

In addition, enhancements in the industry standard toolset severe accident analysis code for level 2 PSA consequence modelling have been pursued. Since the severe accident code used in Canada traditionally models severe accident consequence progression for one unit only, two innovative approaches were developed by industry. These are (a) the scaling approach and (b) the forcing function/injection approach.

The scaling approach involves scaling down features of containment in proportion to the number of units involved in the accident. The advantage of this approach is that it is relatively simple to implement. However, it implicitly assumes that the progression in each of the accident reactors is identical and occurs simultaneously. With this, it is not possible to study the effects of delayed or different stage of progression in one or more reactor units. Nonetheless it does provide a conservative means to assess multiunit consequences.

The forcing function or injection approach injects mass and fission products into multiple units in a standalone containment model. These forcing functions are defined based on energy and mass flow source terms generated by separate parallel severe accident analysis runs.

VII-4.2. Scope of initiating events to be considered

Reference [VII-2] mandated inclusion of a wide range of initiating events including internal and external hazards. Prior to Ref. [VII-2], industry PSAs modelled a wide range of internal initiating events.

The Fukushima event has increased awareness and emphasis on external hazards as initiating events in modelling plant risk. The impact on industry PSAs was driven by Fukushima action items that pertain to external hazards and can be summarized as:

a) Wide consideration of potential external hazards that undergo screening;
b) Consideration of external hazards combinations;
c) PSA modelling of unscreened external hazards, especially those that have potential multiunit impact (such as seismic, high winds and external flood).

VII-4.3. Other impacts on future PSA

A CANDU Owners Group (COG) Joint Project (JP4499) is being executed to address multiunit impacts on PSA. The Joint Project includes defining a hierarchical safety goal framework, site-based safety goals, and an approach to aggregate site risk to include all hazards and all units. Further severe accident code enhancements and development for future PSA use are being pursued by industry as part of the ongoing maintenance of the IST.

REGDOC-2.4.2 added a requirement to consider radioactive sources outside the reactor core, in particular, the irradiated fuel bay. The industry has proposed methodologies to assess these sources via means other than PSA with acceptance of the CNSC.

VII-5. FUKUSHIMA TASK FORCE IMPLEMENTATION

The Fukushima Daiichi accident has had a significant impact on both PSA regulatory requirements and the performance of PSA by utilities in Canada. All the FTF recommendations on PSA were mapped into CNSC REGDOC-2.4.2 which superseded Ref. [VII-2]. PSA scope changes identified through FTF recommendations have been (a) incorporated in PSA analysis to date as part of Ref. [VII-2] implementation; or, (b) completed or planned as separate supporting PSA assessments. The latter additions will support compliance with REGDOC-2.4.2 which licensees are now pursuing.

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ABBREVIATIONS

AERB	Atomic Energy Regulatory Board, India
ARN	Autoridad Regulatoria Nuclear, Argentina
CANDU	Canada Deuterium Uranium
CCF	common cause failure
CDF	core damage frequency
CNCAN	National Commission for Nuclear Activities Control, Romania
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
CPWG	CANDU PSA Working Group
CSDV	condenser steam discharge valve
CSRG	CANDU Senior Regulators Group
DBA	design basis accident
DCC	digital control computers
ECC(S)	emergency core cooling (system)
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
EPS	emergency power system
EWS	emergency water system
FDC	fuel damage category
FM	fuelling machine
FSAR	final safety analysis report
FTF	Fukushima Task Force
HRA	human reliability analysis
HTS	heat transport system
HX	heat exchanger
IE	initiating event
IGCAR	Indira Gandhi Centre for Atomic Research
INSAG	International Nuclear Safety Advisory Group
KANUPP	Karachi Nuclear Power Complex
KINS	Korea Institute of Nuclear Safety
KEPCO	Korea Electric Power Corporation
KHNP	Korea Hydro Nuclear Power
LCV	liquid control valve
LERF	large early release frequency
LOCA	loss of coolant accident
LRF	large release frequency
MCR	main control room
MCS	minimum cutset
MSL	main steam line
MSSV	main steam safety valve

NNSA	National Nuclear Safety Administration
NPP	nuclear power plant
NSSC	Nuclear Safety and Security Commission
NUREG	US Nuclear Regulatory Commission Regulation
OPEX	operating experience
PHTS	primary heat transport system
PNRA	Pakistan Nuclear Regulatory Authority
POS	plant operating state
PRA	probabilistic risk assessment
PSA	probabilistic safety analysis
PSAR	probabilistic safety analysis report
PSR	periodic safety review
RCW	recirculated cooling water
RIH	reactor inlet header
RSW	raw service water
SAMG	severe accident management guidelines
SBLOCA	small break loss of coolant accident
SCDF	severe core damage frequency
SDC	shutdown cooling system
SDV	steam discharge valve
SGTR	steam generator tube rupture
SMA	seismic margin assessment
SPSA	seismic probabilistic safety analysis
TCV	temperature control valve
TSO	technical support organization
US NRC	US Nuclear Regulatory Commission

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