ASAMPSA_E SEVENTH FRAMEWORK PROGRAMME	Advanced Safety Assessment Methodologies: extended PSA	EURATOM

"NUCLEAR FISSION"

Safety of Existing Nuclear Installations

Contract 605001

Summary report of already published guidance on L2 PSA for external hazards, shutdown states, spent fuel storage

Reference ASAMPSA_E

Technical report ASAMPSA_E / WP40 / D40.2 / 2014-08

D02ARV-01-050-776_A_FIN

Gerben Dirksen, Estelle SAUVAGE

AREVA-NP

Period covered: from 01/07/2013 to 31/12/2015	Actual submission date: 31/12/2014	
Start date of ASAMPSA_E: 01/07/2013	Duration: 36 months	
WP No: 40	Lead topical coordinator : Horst Loeffler	His organization name : GRS

Project co-funded by the European Commission Within the Seventh Framework Programme (2013-2016)		
Dissemination Level		
PU	Public	No
RE	Restricted to a group specified by the partners of the ASAMPSA_E project	Yes
CO	Confidential, only for partners of the ASAMPSA_E project	Yes

ASAMPSA 2 Quality Assurance page

Partners responsible of the document :	**AREVA, GRS, IRSN**
Nature of document	Technical report
Reference(s)	Technical report ASAMPSA_E/ WP40 / D40.2 / 2014-05 D02-ARV-01-050-776_A_FIN
Title	Summary report of already published guidance on L2 PSA for external hazards, shutdown states, spent fuel storage
Author(s)	Gerben Dirksen, Estelle Sauvage (AREVA-NP)
Delivery date	31/12/2014
Topical area	PSA Level 2
For Journal & Conf. papers	No

Summary :

This document, drafted during the first phase of the ASAMPSA_E project proposes a review of existing publications related to L2 PSA and relevant for the extended PSA approach developed in the ASAMPSA_E project.

The review covers a period until mid of July 2014.

Visa grid			
	Main author(s) :	**Verification**	**Approval (Coordinator)**
Name (s)	Gerben Dirksen Estelle Sauvage	Horst Löffler	E. Raimond
Date			
Signature			

MODIFICATIONS OF THE DOCUMENT

Version	Date	Authors	Pages or paragraphs modified	Description or comments
A	31/12/2013	Gerben Dirksen Estelle Sauvage (AREVA-NP)	N/A	Initial draft version sent to participants for review and comments
B	15/12/2014	Gerben Dirksen Estelle Sauvage (AREVA-NP)	All	This version includes comments and additional inputs by the WP40 partners.

LIST OF DIFFUSION

European Commission (scientific officer)

Name	First name	Organization
Passalacqua	Roberto	EC

ASAMPSA_E Project management group (PMG)

Name	First name	Organization	
Raimond	Emmanuel	IRSN	Project coordinator
Guigueno	Yves	IRSN	WP10 coordinator
Decker	Kurt	Vienna University	WP21 coordinator
Klug	Joakim	LRC	WP22 coordinator
Wielenberg	Andreas	GRS	WP30 coordinator
Loeffler	Horst	GRS	WP40 coordinator

REPRESENTATIVES OF ASAMPSA_E PARTNERS

Name	First name	Organization
Grindon	Liz	AMEC NNC
Mustoe	Julian	AMEC NNC
Cordoliani	Vincent	AREVA
Dirksen	Gerben	AREVA
Godefroy	Florian	AREVA

Name	First name	Organization
Kollasko	Heiko	AREVA
Michaud	Laurent	AREVA
Sauvage	Estelle	AREVA
Hasnaoui	Chiheb	AREXIS
Hurel	François	AREXIS

Name	First name	Organization
Schirrer	Raphael	AREXIS
De Gelder	Pieter	Bel V
Gryffroy	Dries	Bel V
Jacques	Véronique	Bel V
Van Rompuy	Thibaut	Bel V
Cazzoli	Errico	CCA
Vitázková	Jirina	CCA
Hugon	Michel	EC
Passalacqua	Roberto	EC
Banchieri	Yvonnick	EDF
Benzoni	Stéphane	EDF
Bernadara	Pietro	EDF
Bonnevialle	Anne-Marie	EDF
Brac	Pascal	EDF
Coulon	Vincent	EDF
Gallois	Marie	EDF
Henssien	Benjamin	EDF
Hibti	Mohamed	EDF
Jan	Philippe	EDF
Lopez	Julien	EDF
Nonclercq	Philippe	EDF
Panato	Eddy	EDF
Parey	Sylvie	EDF
Romanet	François	EDF
Rychkov	Valentin	EDF
Vasseur	Dominique	EDF
Burgazzi	Luciano	ENEA
Hultqvist	Göran	FKA
Karlsson	Anders	FKA
Ljungbjörk	Julia	FKA
Pihl	Joel	FKA
Loeffler	Horst	GRS
Mildenberger	Oliver	GRS
Sperbeck	Silvio	GRS
Tuerschmann	Michael	GRS
Wielenberg	Andreas	GRS
Benitez	Francisco Jose	IEC
Del Barrio	Miguel A.	IEC
Serrano	Cesar	IEC
		IEC

Name	First name	Organization
Apostol	Minodora	INR
Nitoi	Mirela	INR
Groudev	Pavlin	INRNE
Stefanova	Antoaneta	INRNE
Armingaud	François	IRSN
Bardet	Lise	IRSN
Baumont	David	IRSN
Bonnet	Jean-Michel	IRSN
Bonneville	Hervé	IRSN
Clement	Christophe	IRSN
Corenwinder	François	IRSN
Denis	Jean	IRSN
Duflot	Nicolas	IRSN
Duluc	Claire-Marie	IRSN
Dupuy	Patricia	IRSN
Durin	Thomas	IRSN
Georgescu	Gabriel	IRSN
Guigueno	Yves	IRSN
Guimier	Laurent	IRSN
Lanore	Jeanne-Marie	IRSN
Laurent	Bruno	IRSN
Ménage	Frédéric	IRSN
Pichereau	Frederique	IRSN
Rahni	Nadia	IRSN
Raimond	Emmanuel	IRSN
Rebour	Vincent	IRSN
Sotti	Oona	IRSN
Volkanovski	Andrija	JSI
Alzbutas	Robertas	LEI
Matuzas	Vaidas	LEI
Rimkevicius	Sigitas	LEI
Häggström	Anna	LRC
Klug	Joakim	LRC
Knochenhauer	Michael	LRC
Kumar	Manorma	LRC
Olsson	Anders	LRC
Borysiewicz	Mieczyslaw	NCBJ
Kowal	Karol	NCBJ
Potempski	Slawomir	NCBJ
La Rovere	Stephano	NIER

Name	First name	Organization
Vestrucci	Paolo	NIER
Brinkman	Hans (Johannes L.)	NRG
Kahia	Sinda	NRG
Bareith	Attila	NUBIKI
Lajtha	Gabor	NUBIKI
Siklossy	Tamas	NUBIKI
Morandi	Sonia	RSE
Dybach	Oleksiy	SSTC
Gorpinchenko	Oleg	SSTC
Claus	Etienne	TRACTEBEL
Dejardin	Philippe	TRACTEBEL
Grondal	Corentin	TRACTEBEL
Mitaille	Stanislas	TRACTEBEL
Oury	Laurence	TRACTEBEL
Zeynab	Umidova	TRACTEBEL

Name	First name	Organization
Bogdanov	Dimitar	TUS
Ivanov	Ivan	TUS
	Kaleychev	TUS
Hladky	Milan	UJV
Holy	Jaroslav	UJV
Hustak	Stanislav	UJV
Jaros	Milan	UJV
Kolar	Ladislav	UJV
Kubicek	Jan	UJV
Mlady	Ondrej	UJV
Decker	Kurt	UNIVIE
Halada	Peter	VUJE
Prochaska	Jan	VUJE
Stojka	Tibor	VUJE

REPRESENTATIVE OF ASSOCIATED PARTNERS (External Experts Advisory Board (EEAB))

Name	First name	Company
Hirata	Kazuta	JANSI
Hashimoto	Kazunori	JANSI
Inagaki	Masakatsu	JANSI
Yamanana	Yasunori	TEPCO
Coyne	Kevin	US-NRC
González	Michelle M.	US-NRC

CONTRIBUTING ASAMPSA-E PARTNERS

The following table provides the list of the ASAMPSA_E partners involved in the development of this document.

Table 1 : ASAMPSA-E Partners involved in this document

1	Institute for Radiological Protection and Nuclear Safety	IRSN	France
2	Gesellschaft für Anlagen- und Reaktorsicherheit mbH	GRS	Germany
3	Ricerca sul Sistema Energetico	RSE S.p.A.	Italy
4	Lloyd's Regsiter Consulting	LRC	Sweden
5	Nuclear Research Institute Rez pl	UJV	Czech Rep.
6	Cazzoli Consulting	CCA	Switzerland
7	Italian National Agency for New Technologies, Energy and the Sustainable Economic Development	ENEA	Italy
8	IBERDROLA Ingeniería y Construcción S.A.U	IEC	Spain
9	Electricité de France	EDF	France
10	Lietuvos energetikos institutas (Lithuanian Energy Institute)	LEI	Lithuania
11	NUBIKI	NUBIKI	Hungary
12	AREVA NP SAS France	AREVA NP SAS	France
13	NCBJ Institute	NCBJ	Poland
14	State Scientific and Technical Center for Nuclear and Radiation Safety”	SSTC	Ukraine
15	TRACTEBEL ENGINEERING S.A.	TRACTEBEL	Belgium
16	Institut Jozef Stefan	JSI	Slovenia
17	Institute of nuclear research and nuclear energy - Bulgarian Academia of science	INRNE	Bulgaria
18	Regia Autonoma Pentru Activatati Nucleare Droberta Tr. Severin RA Suc	INR	Romania
19	Technical University of Sofia - Research and Development Sector	TUS	Bulgaria
20	AREXIS S.A.R.L.	AREXIS	France

SUMMARY

This report (deliverable D40.2 of the project ASAMPSA_E) proposes a review of the existing guidance with relevance to ASAMPSA_E PSA Level 2 topics (external hazds, shutdown states, spent fuel pool).

As a complement of this task, the deliverable D40.2 tries to identify any potential missing guidance for the development of an extended PSA level 2, and any sources of knowledge beyond existing guidance which might help generating extended PSA level 2.

Based on this approach the last section provides a summary compilation which identifies possibilities for completing existing guidelines (especially the guidance developed in the previous ASAMPSA2 project) and/or creating new guidelines for extended PSA Level 2.

CONTENT

LIST OF TABLES

GLOSSARY

The glossary does not include the trigrams and abbreviations for company and agencies.

ASAMPSA-E	Advanced Safety Assessment: Extended PSA
BWR	Boiling Water Reactor
CCF	Common Cause Failure
CDES	Core Damage End States
CDF	Core Damage Frequency
CET	Containment Event Tree
CHLA	Candidate High Level Action
DBA	Design Basis Accident
DCH	Direct Containment Heating
DID	Defence In Depth
FV	Fussel Vessely
HRA	Human Reliability Analysis
I&C	Instrumentation & Control
IRIDM	Integrated Risk Informed Decision Making
ISLOCA	Interfacing System Loss Of Coolant Accident
LERF	Large Early Release Frequency
LRF	Large Release Frequency
LWR	Light Water Reactors
MCCI	Molten Core Concrete Interaction
NPP	Nuclear Power Plant
PDS	Plant Damage States
PHWR	Pressurized Heavy Water Reactors
PRA	Probabilistic Risk Assessment

PSA	Probabilistic Safety Analysis
PWR	Pressurized Water Reactor
RAW	Risk Achievement Worth
RBMK	Reaktor Bolshoy Moshchnosti Kanalnyi
RPV	Reactor Pressure Vessel
SAMP	Severe Accident Management Program
SBO	Station BlackOut
SFP	Spent Fuel Pool
SGTR	Steam Generator Tube Rupture
SSC	Structures, Systems and Components
TMI	Three Miles' Island
TSC	Technical Support Center
VVER	Vodo-Vodianoï Energuetitcheski Reaktor

1 INTRODUCTION

The Fukushima Daiichi Nuclear Power Plant (NPP) accident in Japan resulted from the combination of two correlated extreme external events (earthquake and tsunami). The consequences (flooding in particular) went beyond what was initially considered in the design of the NPP.
Historically based on a deterministic approach the design of the NPP can be supported by more widely using a probabilistic approach that complements the deterministic approach for beyond design and severe accident accidents.

The Advanced Safety Assessment: Extended PSA (ASAMPSA_E) project aims at identifying and providing the best practice guidelines to identify, prevent and manage the situations such as the Fukushima Daiichi accidents, and also other sequences as results of extreme external events and hazards, including combinations with internal hazards, using Probabilistic Safety Assessment (PSA) Levels 1 & 2.

Within the issues reviewed and discussed during the ASAMPSA_E project the PSA Level 2 deals with topics linked to core damage issues and beyond. Within the past years the severe accident research and development have seen an increased interest among organizations and safety authorities. PSA Level 2 guidelines have been developed worldwide to support the development of the PSA Level 2 in the nuclear industry but harmonization of practices remains a topic of interest in the European context (see also the previous ASAMPSA2 project [1]).

This document aims at summarizing existing guidelines and other international and national documents on the subject judged of interest by the ASAMPSA_E consortium. The list proposed is up-to-date in July 2014, and does not include documents issued after this date. For PSA Level 2 topics where no guidelines exist, this document discusses the need for an extension of the guidelines.

2 SUMMARY ON PUBLISHED GUIDES

For each document judged of interest by the ASAMPSA-E participants in the area of PSA Level 2, a screening review is performed. Table 2 provides the detailed list of document covered by the report. This list is mainly based on work previously perfomed by the ETSON expert group on PSA [2].

Besides the title and date of issue, the entity, group or company issuing each document and the scope of the document are first detailed. Then the technical features of each document are discussed, together with its applicability in terms of PSA Level 2 development or review.

The present list in sections 2.1 through 2.4 is rather voluminous. In order to provide a comprehensive overview specifically targeting ASAMPSA_E issues, section 2.5 summarizes the documents with the perspective of the ASAMPSA_E objectives.

Table 2 : List of Reviewed Documents

IAEA 50-P-8 **Erreur ! Source du renvoi introuvable.**	1995 updated by IAEA SSG-4	Procedures for Conducting Probabilistic Safety Assessment of Nuclear Power Plants (Level 2)
IAEA NG-G 2.15 [4]	1995, updated 2014/15	Severe Accident Management Programmes
IAEA Safety Reports Series N° 25 [5]	2002	Review of Probabilistic Safety Assessments by Regulatory Bodies
IAEA Safety Report Series N°32 [6]	2004	Implementation of Accident Management Programmes in Nuclear Power Plants
IAEA SSG-4 [8]	2010	Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants
IAEA-TECDOC-724 [9]	1993	Probabilistic safety assessment for seismic events
IAEA-TECDOC-801 [10]	1995	Development of safety principles for the design of future nuclear power plants
IAEA-TECDOC-905 [11]	1995	Approaches to the safety of future nuclear power plants, Report of a Technical Committee meeting held in Vienna, 29 May - 2 June
IAEA-TECDOC-986 [12]	1997	Implementation of defence in depth for next generation light water reactors
IAEA-TECDOC-1144 [14]	2000	Probabilistic safety assessments of nuclear power plants for low power and shutdown modes
IAEA-TECDOC-1200 [15]	2001	Applications of Probabilistic Safety Assessment (PSA) for nuclear power plants
IAEA-TECDOC-1229 [16]	2001	Regulatory Review of Probabilistic Safety Assessment (PSA) Level 2
IAEA-TECDOC-1487 [17]	2006	Advance nuclear plant design options to cope with external events
IAEA-TECDOC-1570 [18]	2007	Proposal for a Technology-Neutral Safety Approach for New Reactor Designs

IAEA INSAG-10 [19]	1996	Defence in Depth in Nuclear. A report by the International Nuclear Safety Advisory Group , .
IAEA INSAG-12 [20]	1999	Basic Safety Principles for Nuclear Power Plants 75-INSAG-3 Rev. 1; INSAG-12
IAEA INSAG-25 [21]	2011	A framework for an Integrated Risk Informed Decision Making Process
IAEA Nuclear Energy Series, No. NP-T-2.2 [22]	2009	Design Features to Achieve Defense in Depth in Small and Medium Sized Reactors
IAEA Safety Reports Series No. 46 [23]	2005	Assessment of Defence in Depth for Nuclear Power Plants
IAEA Procedings of Conference [24]	1998	Safety of Radiation Sources and Security of Radioactive Materials, Dijon, France, 14-18 September 1998
IAEA Procedings of Conference [25]	2005	Research Reactor Utilization, Safety, Decommissioning, Fuel and Waste Management, Santigo, Chile, 10-14 November 2003
IAEA Procedings of Conference [26]	2012	Protection against Extreme Earthquakes and Tsunamis in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant I/ International Experts Meeting, IAEA, Vienna, Austria, 4-7 September 2012
OCDE/GD(97)198 [27]	1998	Level 2 PSA Methodology and Severe Accident Management
OCDE/NEA/CSNI/R(2009)4[28]	2009	Probabilistic Safety Analysis (PSA) of Other External Events than Earthquake
EC ASAMPSA2 [1]	2010	EC - Best-Practices guidelines for L2 PSA development and applications (ASAMPSA2)
Bulgarian Safety Guide PP-6/2010 [29]	2010	Use of PSA to Support the Safety Management of Nuclear Power Plants, Bulgarian Nuclear Regulatory Agency
Canadian Nuclear Safety Commission REGDOC-2.4.2 [30]	2014	Probabilistic safety Assessment (PSA) for Nuclear Power Plants
Finland STUK YVL 2.8 [31]	2003	Probabilistic Safety Analysis in Safety Management of Nuclear Power Plants
Finland STUK YVL A.7 [32]	2013	Nuclear Power Plant Probabilistic Risk Analysis and Risk Management
Germany BfS Daten (D) BfS-SCHR-38/05 [33]	2005	Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke
Switzerland ENSI-A05/e [35]	2009	Probabilistic Safety Analysis (PSA): Quality and Scope, Guideline for Swiss Nuclear Installations
Switzerland ENSI-A06/e [36]	2009	Probabilistic Safety Analysis (PSA): Applications, Guideline for Swiss Nuclear Installations
United Arab Emirat FANR RG 003 [37]	2005	Regulatory Guide, Probabilistic Risk Assessment: Scope, Quality and Applications
United Arab Emirat FANR-RI-019 [38]	2010	2011 FANR Review instruction (PRA & Severe Accident Analysis
US ASME/ANS RA-Sa-2009 [39]	2009	Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications
US EPRI-101869 [40]	1992	Severe Accident Management Guidance Technical Basis Report

US EPRI 3002000498 [41]	2013	Spent Fuel Pool Risk Assessment Integration Framework (Mark I and II BWRs) and Pilot Plant Application
US EPRI ML12307A202 [42]	2012	Joint Nuclear Regulatory Commission and Electric Power Research Institute Workshop on the Treatment of Probabilistic Risk Assessment Uncertainties (Draft version)
US NRC RG 1.200 - Rev. 2 [43]	2009	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities
US NRC NUREG-1738 [45]	2001	Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants
US NRC NUREG/CR-6451 [48]	1997	A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants
US NRC NUREG/CR-4982 [47]	1987	Severe accidents in spent fuel pools in support of generic safety
US NRC NUREG/CR-7110 [49]	2012	State-of-the-Art Reactor Consequence Analyses Project Volume 1: Peach Bottom Integrated Analysis Volume 2 : Surry Integrated Analysis
US Wash-1400 [50]	1975	The Reactor Safety Study

The following subchapters summarize the content of each document.

2.1 IAEA SAFETY STANDARDS AND TECHNICAL DOCUMENTS

ID **Date of Issue**	**IAEA Safety Series N°50-P-8** **Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 2)** **Initial publication 1995**
Issuer / Scope of document	Provides guidance to perform or manage a PSA 2.
	First issued in 1995. Modified to include the lessons learned from the Fukushima Daiichi accident.
	Covers all subjects from PSA 1-PSA 2 interface (grouping of sequences), to source term, including sequences and containment analysis. Includes the analysis of results.
	Aim to promote a standardized framework, terminology and form of documentation for the results of Severe Accident Management Programs (SAMP).
	Applicable for all Light Water Reactors (LWR) (e.g. Pressurized Water Reactor (PWR), Boiling Water Reactor (BWR) and Vodo-Vodianoï Energuetitcheski Reaktor (VVER)) and Pressurized Heavy Water Reactors (PHWR). Basic philosophy and approach anticipated to remain valid for other reactors such as Reaktor Bolshoy Moshchnosti Kanalnyi (RBMK).
Features of document	It includes a detailed plan to develop a PSA 2 (i.e., structure and major procedural steps required in a PSA 2 development).
	Provide information on the severe accident code and deterministic analysis required to perform a PSA 2.
	Limited information regarding the treatment of uncertainties and new NPP PSA 2 (i.e., NPP including the severe accident in their design).
	Oriented to Plant Damage States (PDS) and not to Containment Damage End State (CDES) approach.
Applicability	As a PSA developer it is mandatory to follow the general structure proposed by this guideline, as it is generic enough to provide the required structure and content of a PSA 2.
	In addition, as a PSA developer it is of interest for plant specific PSA 2 development to use the advices provided.

ID **Date of Issue**	**IAEA Safety Guide NS-G-2.15** **Severe Accident management Plan (SAMP) for Nuclear Power Plants** **Draft, 2014**
Issuer / Scope of document	Provides guidance to develop severe accident management plan for the utilities, and assistance for the regulatory bodies in carrying out reviews of the SAMP produced by utilities.
	First issued in 2009. Modified to include the lessons learned from the Fukushima daiichi accident.
	Aim to cover the prevention and/or the mitigation of design extension conditions for beyond design basis accidents and severe accidents.
	Aim to cover events or combination of deficiencies not considered in the design basis, including external events.
	Aim to cover all operating conditions for both reactor and spent fuel pool (SFP).
	Aim to promote a standardized framework, terminology and form of documentation for the results of SAMP.
	Applicable for all LWRs (e.g. PWR, BWR and VVER) and PHWRs. Basic philosophy and approach anticipated to remain valid for other reactors such as RBMK.
Features of document	Comprehensive covering all parts of SAMP including the support studies, the guideline development, the verification and validation aspects : • provides which steps should be taken in setting up an accident management program, from the conceptual stage down to a complete set of instructions - procedures and guidelines • provides guidance on compliance with the regulatory aspects of the Safety Requirements on: Safety Assessment and Verification for Nuclear Facilities (GSR Part 4, 2009) in particular with requirement 13; Safety of Nuclear Power Plants: Design (SSR-2/1, 2012); and Safety of Nuclear Power Plants: Commissioning and Operation, (SSR-2/2, 2011).
	Covers interactions of SAMP with PSA Level 1 and 2 (not exhaustive).
	In particular provide guidance on drills or exercises.
Applicability	As PSA developer it provides mandatory requirements regarding interactions of the generic and/or plant specific probabilistic approach and severe accident plan.
	As PSA reviewer it can be of interest to highlight the potential inconsistencies between the PSA and the SAMP.

ID **Date of Issue**	**IAEA Safety Reports Series No. 25** **Review of Probabilistic Safety Assessments by Regulatory Bodies** **2002**
Issuer / Scope of document	Provides guidance to assist regulatory bodies in carrying out reviews of the PSAs produced by utilities.
	Intended to assist technical experts managing or performing PSA reviews
	Aim to promote a standardized framework, terminology and form of documentation for the results of PSA reviews.
Features of document	Comprehensive covering all parts of PSA (level 1, 2 and 3)
	Can act as the basic reference for a review guide. For specific issues, e.g. level 2, digital Instrumentation & Control (I&C), Human Reliability Analysis (HRA), external events etc, other method oriented documents needed for deeper guidance
	Content of previous TECDOC-1135 [13] and TECDOC-1229 [16] mostly within Safety Report Series N° 25, 2002
Applicability	As PSA reviewer: PSA to be reviewed must not violate statements of the document.
	Prepared 2002, overall applicability but requires a check of validity of the details.
	No guidance on risk criteria.

ID **Date of Issue**	**IAEA Safety Reports Series No. 32** **Implementation of Accident Management Programmes in Nuclear Power Plants** **2004**
Issuer / Scope of document	This document was issued by International Atomic Energy Agency in 2004. Document includes the description of the elements which should be addressed by the team responsible for the preparation, development and implementation of a plant specific accident management programme at a nuclear power plant. The report covers formation of the team, selection of accident management strategies, safety analyses required, evaluation of the performance of plant systems, development of accident management procedures and guidelines, staffing and qualification of accident management personnel, and training needs.
Features of document	This report provides a description of the elements to be addressed by the team responsible for developing and implementing a plant specific Accident Management Programme of a nuclear power plant. The report focuses on severe accident management guidelines and covers both internal and external events. The report concentrates on full power operational states and is limited to conditions under which a certain amount of control over the main power plant functions still exists. No large scale disruption or destruction of the NPP is assumed. The report considers primarily existing plants, i.e. plants which are either in operation or under construction. This document provides description of the basic features of the Accident Management Programme including its objectives, preventive and mitigation features, accident progression and degrees of severity, assessment of vulnerabilities and capabilities, accident management strategies and phases of the Programme. The probabilistic safety assessment is enlisted in Section 2.4 of the report as approach for identification of those vulnerabilities of the plant which are likely to cause challenges to the safety functions. The report describes application of the probabilistic safety assessment for development of the severe accident management guidelines in Appendix VIII. The Level 1 and Level 2 probabilistic safety assessments identify the core damage and core melt phenomena that are relevant for the particular plant or group of plants. The insights gained from probabilistic safety assessments are used for definition and selection of the candidate strategies to mitigate the relevant accident scenarios. The probabilistic safety assessment may serve to find the conditions both to enter the severe accident management guidelines domain and to leave the emergency operating procedure domain as the only or the dominant accident management tool. The quantitative information to support the technical support centre (or related group) can also be obtained from PSA. The Level 3 probabilistic safety assessment gives estimates of the source term and its external consequences that may serve as an upper estimate of the potential release in the environment after the accident.
Information depth	This document describes in detail all prerequisites and steps necessary for the preparation and development of the Accident Management Programme and implementation in the given nuclear power plant.
Applicability	This document is applicable in the nuclear power plant operators for development or upgrade of Accident Management Programme. As probabilistic safety assessment reviewer it can be used to assess level and quality of the application of the probabilistic safety assessment in other activities in the plant including Accident Management Programme As probabilistic safety assessment producer no particular applicability was identified.

ID **Date of Issue**	**IAEA Specific Safety Guide No SSG-4** **Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants** **2010**
Scope of document	Explain state of the art in PSA level 2 methodology
	SSG-4 is based on full power and internal initiators. Some general remarks are provided for low power / shutdown operation and for accidents initiated by internal / external hazards.
	Suggest project structure and resources in managing or performing PSA level 2
	Provide overview on uses of PSA level 2
Features of document	Comprehensive explanation of PSA Level 2 methods, representing state-of-the-art 2010.
	Little practical information about data / models
	Comprehensive listing of potential PSA level 2 applications
	Review of PSA level 2 is not an issue of the document
Applicability	As PSA producer: SSG-4 is useful as general guideline, not useful for selecting data/models
	As PSA reviewer: SSG-4 can be used as a standard for general features of the PSA (management, documentation …). SSG-4 is not suitable for technical evaluation of a PSA.

ID **Date of Issue**	**IAEA-TECDOC-724** **Probabilistic safety assessment for seismic events** **1993**
Issuer / Scope of document	This document is issued by the IAEA in 1993.
	It describes the general methodology of seismic PSA applicable to all types of reactors.
	It includes the description of guidance and insight to those who are considering starting a seismic PSA giving an overall picture of the seismic PSA and attempting to bridge the gap between an internal event PSA and a seismic PSA.
Features of document	This document provides description of the major steps for accomplishing a seismic PSA including development of a seismic hazard curve, structure and component seismic response determination, assignment of structure and component fragility, random failure data development, event/fault tree construction and solution, risk quantification incorporating results of the previous steps.
	The document considers a Level 1 PSA plus containment performance analysis.
	The document covers such aspects of the seismic PSA: the frequency of occurrence of ground motion, the seismic accident sequence initiators, the fragility analysis of safety related items, the capability of systems to mitigate accidents from seismic events and the integration of these aspects which might lead to a core damage.
	The document describes the probabilistic seismic hazard analysis. The aim of the analysis is to develop hazard curves which characterize the seismic exposure of a given site to the so-called primary effects (vibratory ground motion). The analysis is based on historical earthquake reports and instrumental records, as well as the geology of the region, including physical evidence of past seismicity.
	The document overviews secondary seismic effects. The purpose of this is to give some recommendations for identifying secondary effects which may appear as consequences of earthquakes.
	The document describes the methods of developing fragility curves that have been used in over 30 seismic PRAs to date. There are not described the details of the techniques, only their main features are presented along with guidance on which methods are usually appropriate for generic classes of equipment. The focus is on response from the free field ground motions up through the equipment, dominant failure modes and capacity. Also is presented a discussion of the use of generic fragility descriptions, expert judgement and earthquake experience to focus the detailed analytical activity on the most important and most vulnerable structures and equipment.
	This document does not consider some aspects which may have significant contribution to overall risk: - increased probability of human error subsequent to the occurrence of a destructive earthquake; - significant probability of damage to lifelines and other infrastructures which may have been planned for use in the context of emergency planning and evacuation; - increased probability of delayed response to the nuclear accident (by authorities and the public) due to the interference of another catastrophic event.
Information depth	The document at general level describes methods and approaches to perform seismic PSA
	The document at general level presents the results of a Level 1 seismic PSA and compares them to results from internal events and other external events.
Applicability	As PSA producer: the document tries to help the reader by providing some references for further information on current practices and insights obtained by conducting seismic PSAs but is not intended to be an extensive manual or handbook.
	As PSA reviewer: the document is useful as guidance for performing the review, and may draw the reviewer's attention to some critical points of the seismic PSA.

ID **Date of Issue**	**IAEA-TECDOC-801** **Development of safety principles for the design of future nuclear power plants** **1995**
Scope of document	Report of the IAEA
	The objective of the task was to review the methods for risk analysis of off-site external events other
Features of document	The document is obsolete in some extent after the event of the Fukushima disaster. The proposed safety principles shall, be extended further towards site evaluation, human survival and behaviour in case of severe accidents.
Applicability	As PSA producer: some international experience information regarding the external hazards PSA.
	As PSA reviewer: international examples of external event PSA development and applications.

ID **Date of Issue**	**IAEA-TECDOC-905** **Approaches to the safety of future nuclear power plants - Report of a Technical Committee meeting held in Vienna** **1995**
Scope of document	Report of the IAEA
Features of document	An appropriate set of external hazards is addressed explicitly in the design. The specific external sequences selected are determined on the basis of risk importance. External hazards, especially the natural ones, are very site specific, but customizing each future plant to the specific hazards appropriate to a specific site is not practical if standardization is to be achieved. Therefore, the design process achieves an optimum balance between standardization and site specific hazard protection. This is usually accomplished using a "site envelope" approach that requires the standard design to be protected against those external hazards most probable for a large number of potential NPP sites, and allows site specific treatment of those hazards unique to a smaller number of potential sites for that design.
Applicability	As PSA producer: some international experience information regarding the external hazards PSA. More detailed historical study on site shall be made. Modelling of scenarios related to L2 events must be developed.
	As PSA reviewer: Assessment of historical data for sites shall be extended not to a limited set of decades back, but to historical records and evidences for at least a century back.

ID **Date of Issue**	**IAEA-TECDOC-986** **Implementation of Defense in Depth for Next Generation Light Water Reactors** **1997**
	Report of the IAEA
Scope of document	The substantial innovation proposed in TECDOC-801 includes the explicit consideration of severe accidents in the design of future NPPs, together with the minimization of off-site effects in the event of a severe accident. This report, together with other factors discussed in the following, suggested the need for continuing discussion of several aspects of the safety approach to future NPPs, including further work on the concept of defence in depth. The importance of defence in depth as a fundamental strategy to achieving safety has been reaffirmed several times and is not under discussion. INSAG has recently prepared a report, INSAG-10, Defence in Depth in Nuclear Safety, which suggested a more structured interpretation of the concept of defence in depth, compared with the traditional meaning as outlined in INSAG-3. INSAG-10 presents defence in depth in general terms, with only a small part (Section 5) devoted to future reactors. The present report, specifically focused on future reactors, builds on and is consistent with INSAG-10, which was available in draft form during the preparation of this report. The report was developed before Fukushima.
Information depth	For future reactors, the new safety principles, the emergence of new technologies, the indications from operating experience and variations in safety trends and expectations for future plants from country to country all indicate the importance of improving the current guidance on how the defence in depth concept will be implemented. This report therefore includes a discussion of the balance between prevention and mitigation, and how efforts to achieve a higher standard of safety for future plants will be distributed among the five levels of defence in depth.
Applicability	As PSA producer: some international experience information regarding the external hazards PSA.
	As PSA reviewer: international examples of PSA development and applications.

ID **Date of Issue**	**IAEA TECDOC-1144** **Probabilistic Safety Assessments of Nuclear Power Plants for Low Power and Shutdown Modes** **2000**
Scope of document	The scope of the document covers all important areas of low power and shutdown PSA, in general.
	Important comments/rules regarding management and organization of low power and shutdown modes PRA project are part of document scope.
	Internal and external hazards, heavy load drops and accidents involving other sources of radioactive materials are briefly addressed in the document.
	Some general, but useful rules how to apply LPSD PRA results in a compact manner in the processes of NPP engineering support are presented in the document.
	Examples of typical sets of plant operational states and initiating events for some (most common) reactor types are given in Annexes.
Features of document	The document is structured two ways - into nine sections and three Annexes - and into 31 Tasks covering complete organization and performance of the low power and shutdown modes PRA project.
	The level of detail (and methodology completeness) of the document varies over the individual sections. Some PRA topics are not addressed enough in the document to use it as self consistent means for planning or verification of the analysis.
	In many cases, the document provides useful recommendations to some specific points of low power and shutdown modes PRA, including some very current issues identified in practice, in concrete low power and shutdown modes PRA projects.
	In some cases, the document provides comparison with full power PRA methodology and points out the differences between full power PRA and low power and shutdown modes PRA, which are important to be considered.
Applicability	TECDOC-1144 may not be used as complete low power and shutdown modes Probabilistic Risk Assessment (PRA) review guidance, but is may be very useful to read it before the review.
	Human reliability analysis is not covered sufficiently in TECDOC-1144 and should be addressed by means of some other specific methodology sources.

ID **Date of Issue**	**IAEA TECDOC-1200** **Applications of probabilistic safety assessment (PSA) for nuclear power plants** **2001**
Scope of document	Addresses the PSA application process, outlines the general requirements for PSA tools and provides a discussion on PSA aspects such as PSA level, scope and level of detail.
	Discusses the technical aspects of individual applications, in particular the design related PSA applications, the PSA applications that are related to the plant operations and the PSA applications used to support the mitigation and management of incidents and accidental situations.
	Discusses the regulatory perspective on the use of PSA, and points out the main issues and regulatory concerns.
	Discusses the establishment of numerical criteria for use in decision making using PSAs.
Features of document	The document is based on the premise that the use of PSA can provide useful information for the decision maker.
	The document compiles information on a comprehensive set of PSA applications in the areas of NPP design, operation, and accident mitigation and management.
	The document emphasise Living PSA as prerequisite and has focused on those PSA applications that have been reported extensively in the literature.
	The document is intended to provide an overview of current PSA applications, but it is actually outdated.
Information depth	The document is just intended to provide an overview of current PSA applications.
	The technical aspects of individual applications and the list of them are not intended to be complete in its coverage of uses of PSA.
	The discussion regarding numerical criteria concentrates on general issues, but gives specific examples to illustrate application of those principles.
Applicability	PSA producer and investigator: the document may be useful as PSA tasks and application overview; it may not be useful for performing PSA.
	PSA reviewer: the document may be useful as overview of regulatory perspectives on the use of PSA; it may not be useful as source for specific criteria definition and their guidance for practical application.

ID **Date of Issue**	**IAEA-TECDOC-1229** **Regulatory Review of Probabilistic Safety Assessment (PSA) Level 2** **2001**
Scope of document	Guidance how the regulatory authority should carry out the review
	Guidance on the technical issues that need to be addressed in carrying out the review of PSA Level 2
	Guidance on the review of the PSA quality assurance
	Scope is limited to full power and internal initiators
Features of document	Most parts of TECDOC-1229 are very general.
	TECDOC-1229 points out some potential critical points of PSA which should be scrutinized. This is helpful for reviewers.
	The document identifies various potential problems in a PSA, but it generally fails to give support how to address these issues correctly either as a PSA producer or as a PSA reviewer.
	Content of TECDOC-1229 in principle contained in more recent document IAEA safety Series Report 25, 2002
Applicability	TECDOC-1229 can help setting up and running a review team.
	TECDOC-1229 may draw the reviewer's attention to some critical points of the PSA.
	TECDOC-1229 does not provide guidance on how to document / report the review.

ID **Date of Issue**	**IAEA-TECDOC-1487** **Advanced nuclear plant design options to cope with external events** **2006**
Scope of document	IAEA in the report intends to define design options for protection from external event impacts in NPPs with evolutionary and innovative reactors.
	No limitations were set on the scope of external events, which include human induced events and various natural events. Likewise, there were no limitations on specific types of evolutionary or innovative reactors within the NPP projects to be addressed. Most of the NPPs addressed were with reactors of evolutionary type.
	The report includes introduction, 5 dedicated sections on selected topics, conclusions and suggestions for further work, 2 appendices and 8 annexes (including papers on on elaboration and application of the methodology to assess external hazards and uncertainties and issues of protection from external events for certain NPPs).
Features of document	The objective of the report is to present the state-of-the-art in design approaches for the protection of NPPs with evolutionary and innovative reactors from external event impacts, as well as to assist the designers of advanced NPPs in the definition of a consistent strategy of design and siting evaluation in relation to extreme external events.
	Through direct cooperation with the designers of advanced NPPs, the document intends to define, collate and present the state-of-the art in design features and approaches used to protect plants from external event impacts, making a focus on NPPs with evolutionary and, when possible, innovative designs.
	The document reflects best practices achieved in IAEA member states, to provide a technical and information background to assist designers of advanced NPPs in defining a consistent strategy regarding selected design and site evaluation issues in relation to extreme external events.
	The document is intended to bring to the attention of designers of advanced NPPs the recently updated IAEA safety guides and other publications on issues of plant protection from external event impacts; to collect comments on their applicability to NPPs with evolutionary and innovative reactors; to identify safety and technological issues and proposals for their resolution; and to outline future challenges and potential contribution of the IAEA.
Information depth	The document presents a summary of twelve responses to the questionnaire, which requested designers of advanced NPPs to identify, for their respective designs, the scope of accidental mean or median external events considered in the design.
	The report also addresses safety requirements for siting of NPPs with advanced reactors. The topics addressed include hazard types and combinations and relevant return periods.
	The report presents and analyzes the design features and approaches used in 14 advanced NPPs, with respect to protection from both external and internal events.
	The document also addresses the issues of component qualification, including special testing, mock-ups, fragility evaluations, and special requirements.
Applicability	PSA producer and investigator: the document may be useful mainly in the evolutionary or innovative NPP design phase; it may not be useful for performing PSA of operating NPPs, expecialy if focus is not on external events.
	PSA reviewer: the document may be useful as overview of best practicies on the design options to cope with external events; it may not be useful as source for specific criteria definition and their guidance for practical application.

ID **Date of Issue**	**IAEA-TECDOC-1570** **Proposal for a Technology-Neutral Safety Approach for New Reactor Designs** **2007**
Scope of document	Report of IAEA
Features of document	Many states are considering an expansion of their nuclear power generation programmes. Many of the technologies and concepts are new and innovative. The current design and licensing rules are applicable to mostly large water reactors and there are no accepted rules in place for design, safety assessment and licensing for new innovative nuclear power plants.
	This TECDOC proposes a (new) safety approach and a methodology to generate technology neutral (i.e. independent of reactor technology) safety requirements and a “safe design” for advanced and innovative reactors. The experience gained in decades of design and licensing, combined with the development of risk-based concepts, has provided insights that will form the basis for new safety rules and requirements. Many lessons learned acknowledge the importance of such concepts as safety goals and defence in depth and the benefits of integrating risk insights early in an iterative design process. A new safety approach will incorporate many of the new developments in these concepts. For example, the probabilistic elements of defence in depth will help define the cumulative provisions to compensate for uncertainty and incompleteness of our knowledge of accident initiation and progression
	In the document are covered some issues related to site, site specific events: Site Related Characteristics I.3.17. In determining the design basis of a nuclear power plant, various interactions between the plant and the environment, including such factors as population, meteorology, hydrology, geology and seismology, shall be taken into account. The availability of off-site services upon which the safety of the plant and protection of the public may depend, such as the electricity supply and fire fighting services, shall also be taken into account. I.3.18. Projects for nuclear power plants to be sited in tropical, polar, arid or volcanic areas shall be assessed with a view to identifying special design features which may be necessary as a result of the characteristics of the site.
Information depth	In part “External events” - I.3.15. Natural external events which shall be considered include those which have been identified in site characterization, such as earthquakes, floods, high winds, tornadoes, tsunami (tidal waves) and extreme meteorological conditions. Human induced external events that shall be considered include those that have been identified in site characterization and for which design bases have been derived. The list of these events shall be reassessed for completeness at an early stage of the design process.
Applicability	As PSA producer: some international experience information regarding the external hazards PSA.
	As PSA reviewer: International experience to be considered in review..
Comment	The reports were developed before Fukushima, but after the tsumami which occurred in Indonesia in 2004.

<table>
<tr><td>ID
Date of Issue</td><td>Defence in Depth in Nuclear Safety INSAG-10
International Nuclear Safety Advisory Group
1996</td></tr>
<tr><td rowspan="2">Scope of document</td><td>Report of the IAEA</td></tr>
<tr><td>This report addresses the Three Mile Island (TMI) accident in the United States of America in 1979 and Chernobyl in 1986.</td></tr>
<tr><td rowspan="3">Features of document</td><td>The report is structured as follows:
— Section 1 summarizes the historical development of safety concepts, focusing on defence in depth;
— Section 2 discusses the concept of defence in depth hi terms of objectives, strategy, physical barriers and levels of protection;
— Section 3 describes the implementation of defence in depth and illustrates how its various elements interrelate;
— Section 4 indicates how defence in depth can be enhanced for the nuclear power plants that are currently operating;
— Section 5 proposes a development of defence in depth which could be applied systematically for plants to be built in the future.</td></tr>
<tr><td>The concept of defence in depth was therefore gradually refined to constitute an increasingly effective approach combining both prevention of a wide range of postulated incidents and accidents and mitigation of their consequences. Incidents and accidents were postulated on the basis of single initiating events selected according to the order of magnitude of then" frequency, estimated from general industrial experience.</td></tr>
<tr><td>On basic stage, the concept of defence in depth generally included three levels:
— conservative design, providing margins between the operating conditions foreseen (covering normal operation as well as postulated incidents and accidents) and the failure conditions of equipment;
— control of operation, including response to abnormal operation or to any indication of system failure, by the use of control, limiting and protection systems to prevent the evolution of such occurrences into postulated incidents and accidents;
— engineered safety features, to control postulated incidents or accidents in order to prevent them from progressing to severe accidents or to mitigate their consequences, as appropriate.
The concept of defence in depth was further refined to include consideration of external hazards, quality assurance, automation, monitoring and diagnostic tools. Furthermore, additional severe accidents were considered in studies and probabilistic safety analyses</td></tr>
<tr><td rowspan="2">Applicability</td><td>As PSA producer: some international experience information regarding the external hazards PSA. More detailed study on external events shall be made in order to identify probability for share in L2 event probability.</td></tr>
<tr><td>As PSA reviewer: After the Fukushima accident “Level 0” shall be applied - for feasibility of the site selection. With wrong site selection and mitigated risk of disastrous impact on the site all levels of defense in depth as currently stated may be insufficient and useless.</td></tr>
</table>

ID **Date of Issue**	**Basic Safety Principles for Nuclear Power Plants** **75-INSAG-3 Rev. 1, INSAG-12** **1999**
Scope of document	Report of the IAEA
	The present report is a revision of the original 75-INSAG-3 which was issued in 1988 to provide a statement of the objectives and principles of safe design and operation for electricity generating nuclear power plants. This revision was prepared in order to bring the text up to date with improvements in the safety of operating nuclear power plants as well as to identify principles to be applied for future plants. It presents INSAG's understanding of the principles underlying the best current safety policies and practices of the nuclear power industry.
Features of document	In point 181 is mentioned the importance of seismic event, and other site specific factors: "Of the extreme external hazards, seismic events receive special attention owing to the extent to which they can jeopardize safety. A nuclear power plant is protected against earthquakes in two ways: by siting it away from areas of active faulting and related potential problems such as susceptibility to soil liquefaction or landslides; and by designing the physical barriers and the safety systems contributing to the defence in depth of the plant to bear the vibratory loads associated with the most severe earthquake that could be expected to occur in its vicinity, on the basis of historical input and tectonic evidence. This is termed the design basis earthquake. Seismic design of plant structures, components and systems is carried out using response function methods, making use of a frequency spectrum for the design basis earthquake that is appropriate to the site. Seismic design takes account of soil-structure interaction, the potential amplification and modification of seismic motion by the plant structures, and interaction between components, systems and structures. The design ensures that the failure of non-safety-related equipment in an earthquake would not affect the performance of safety equipment." More detailed study on external events is imposed after 2011.
Applicability	As PSA producer: some international experience information regarding the external hazards PSA. General information regarding design of nuclear power plants.
	As PSA reviewer: After the Fukushima accident "Level 0" shall be applied - for feasibility of the site selection. With wrong site selection and mitigated risk of disastrous impact on the site all levels of defense in depth as currently stated may be insufficient and useless.

ID **Date of Issue**	**IAEA-INSAG-25** **A Framework for an Integrated Risk Informed Decision Making Process** **2011**
Scope of document	The document/report presents a framework that is termed Integrated Risk Informed Decision Making' (IRIDM).
	This report identifies the framework, principles and key elements for IRIDM. It describes the interrelationship between the key elements, and the integration of their inputs. The need for documentation, communication and follow-up on the implementation of the decisions, including performance monitoring and corrective action, is emphasized.
	The IRIDM process, involving several key factors, brings transparency to complex decisions and its added value is explained in this report. These factors may be weighed differently to reflect their relative importance to the situation under consideration.
Features of document	While the details of IRIDM methods may change with better understanding of the subject, the framework presented in this report is expected to apply for the foreseeable future.
	This report is intended to promote a common understanding among the international nuclear community (designers, suppliers, constructors, licensees, support organizations and regulators) of how the concept of risk can be used in making safety decisions relating to nuclear installations.
	Although this report is focused on the use of IRIDM in the context of NPPs, including their fuel handling and storage systems, it can be equally applied with appropriate adjustments to other nuclear facilities and activities as well as to non-nuclear applications.
Information depth	This report focuses only on key IRIDM aspects and considerations that bear on their application which should be taken into account in order to arrive at sound risk informed decisions.
	This report just describes the foundations of an integrated decision making process.
	The application of IRIDM process allows decisions to be reviewed and, if appropriate, reconsidered to reach a robust conclusion.
Applicability	Producer and investigator of PSA or PSA applications: the document may be useful as general IRIDM framework overview; it may not be useful for performing PSA and even PSA applications.
	Reviewer of PSA applications: the document may be useful as it reflects the overview of IRIDM concepts in relation to PSA applications; it may not be useful as source for PSA review and not be useful as guidance for review of PSA applications.

ID **Date of Issue**	**IAEA Nuclear Energy Series, No. NP-T-2.2 ,** **Design Features to Achieve defense in depth in Small and Medium Sized Reactors2009**	
Scope of document	Report of the IAEA	
	In the report are presented some safety features for Small/medium size reactors that may be applicable to some extent for larger installations: Table 3 (page 16). Design features of pressurized water small and medium sized reactor concepts contributing to level 3 of defence in depth.	
Information depth	In particular some examples for external impact are given.	
Applicability	As PSA producer: - actual and conservative data shall be used	
	As PSA reviewer: some international experience information regarding the external hazards PSA.	

ID **Date of Issue**	**IAEA Safety Reports Series No. 46** **Assessment of Defence in Depth for Nuclear Power Plants** **2005**
Scope of document	Report of the IAEA
	The objective of the task was to review the methods for risk analysis of off-site external events other
Features of document	To ensure the safety of plants by avoiding the failure of barriers against the release of radioactive material and by mitigating the consequences of their failure, the following fundamental safety functions have to be performed in operational states, during and following Design Basis Accidents (DBA) and, to the extent practicable, in, during and following the considered plant conditions beyond the DBA [4]: (1) Control of reactivity; (2) Removal of heat from the core; (3) Confinement of radioactive materials and control of operational discharges, as well as limitation of accidental releases. Henceforth in the present report, fundamental safety function (2) 'removal of heat from the core', as mentioned in Ref. [4], will be replaced by the more general fundamental safety function 'removal of heat from the fuel' to cover also the fuel removed from the core but that is still on the site of the plant and is a potential radioactive source.
	The fundamental safety functions are essential for defence in depth and as a measure of the appropriate implementation of defence in depth through the various provisions for the design and operation of the plant, as indicated by the underlying relevant safety principles. The aim of the defence in depth provisions is to protect the barriers and to mitigate the consequences if the barriers against the release of radioactive material are damaged.
	The identification of what can have an impact on the performance of an fundamental safety function as well as of the variety of options that exist for avoiding this impact for each level of defence is an essential task in the development of the framework for making an inventory of the defence in depth capabilities of a plant. For developing the framework, concepts are presented.
	In the report different aspects of specification of provisions that prevent mechanisms or combinations of mechanisms from occurring that might challenge the performance of the fundamental safety functions and safety functions.
Applicability	As PSA producer: some international experience information regarding the external hazards PSA.
	As PSA reviewer: international experience to be taken into account in review
Comment	The report was developed before Fukushima.

ID **Date of Issue**	**Research Reactor Utilization, Safety, Decommissioning, Fuel and Waste Management** **2005**
Scope of document	Reports of experts in different areas
	Experimental reactor security, safety, decommissioning, fuel and waste management and etc. issues.
Features of document	In the articles collected in the proceedings of the conference are indicated, discussed and evaluated different aspects of experimental reactor security.
	There is considered the potential to challenge plant safety.
	In different papers are presented results of calculations for shares of contribution of different initiating events and resulting Core Damage Frequency (CDF).
Information depth	The typical set constituted all credible accidents that are: —Loss of normal electric power; —Insertion of excess reactivity; —Loss of flow; —Loss of heat sink; —Loss of coolant from the primary cooling system; —Loss of coolant from the reactor and service pools cooling system; —Loss of heavy water; —Erroneous handling or failure of equipment or components; —Special internal events; —Reactor utilization malfunctions; —Spurious triggering of safety systems; —External events; —Human errors.
	Dynamic human interactions. In the present PSA, the actuation of the safety systems is automatic, and no credit is taken for manual actions. Therefore, the third class of dependent failures, which corresponds to dynamic human interactions (e.g. inability to act due to operator error in response to the event), is not relevant because it does not contribute to the failure of safety systems. Where credible operator actions that could jeopardize safety system functions were identified, they were included as basic events in the fault tree models. Furthermore, conservative assumptions were made regarding plant operations (e.g. that the operator prematurely shuts down the primary cooling system pumps, following a trip).
Applicability	As PSA producer: Could be used: 1st - the approaches applied in studies in different countries which are presented in the reports of the participants. 2nd - the analysis different constructions of reactors and systems to prevent CDF, and also of the eventual emergency scenarios.
	As PSA reviewer: The articles focused on Defence In Depth (DID) principles.
Comment	The report was developed before Fukushima.

ID **Date of Issue**	**Protection against Extreme Earthquakes and Tsunamis in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant** **Proceeding of Conference** **2012**
Scope of document	Report of the IAEA on International Experts Meeting hold from 4–7 September 2012, Vienna, Austria. The report was developed before Fukushima.
	This report endorses the application of both probabilistic and deterministic methods to address such hazards. INSAG fully endorses this approach, as it can enable a deeper understanding of uncertainties, of cliff edge effects and of risk. We are hopeful that the recent INSAG report entitled A Framework for an IRIDM process1 can help in this effort.
Features of document	The recent improvements in numerical tsunami modelling are recognized for application in assessing the associated hazards at nuclear installation sites. In general, the modelling is more widely applied to earthquake generated tsunamis and, partly, to landslide generated tsunamis. There are no remarkable modelling applications to volcano generated tsunamis. The key aspect that is vital for accurate tsunami modelling is proper determination of the source mechanisms that generate the phenomena. The generation and coastal amplification of associated phenomena such as (i) seiches due to forcing of continuous energy input to the basins, (ii) amplification and resulting resonance oscillations inside the semi-enclosed basins or (iii) long waves resulting from large scale atmospheric pressure differences in the region should also be taken into account in the assessment of external flooding. Hazard analysis by tsunami numerical modelling for nuclear power plants should also cover these associated phenomena. The highest resolution of bathymetric and topographic data, covering land use plans at the site and in site vicinity areas and including all morphological details, is essential for high quality tsunami modelling applications. Modern tsunami hazard evaluations following current guidance are based on numerical simulations for deterministic scenarios, and the key issue is the proper characterization of the potential tsunami genic sources. Conservative assumptions on the sources are to be used, but aleatory variability of the tsunami wave parameters for a given source scenario is not usually addressed. As seen in ground motion hazard estimates, the aleatory variability can have a large effect on both deterministic and probabilistic evaluations and the computed tsunami waves will not be bounding values.
	The international experts meeting has stated The large uncertainties associated with the parameters involved in tsunami hazard assessment, particularly the characteristics of the event that may generate the tsunami; —The uncertainties associated with the potential inundation levels at different locations on a nuclear power plant site due to the plant layout; —The difficulties in incorporating effective tsunami protection measures for operating nuclear power plants; —The intolerance of a number of Structures, Systems and Components (SSC) to increased flooding levels, for example, flood related cliff edge effects.
	In the document are indicated key issues of safety against tsunami and topics for furthers studies.
Information depth	Systematized information from top-level experts. The aspects of safety and critical events related to tsunami are well summarised.
Applicability	As PSA producer: international experience information regarding the external hazards PSA.
	As PSA reviewer: This report in the light of the Fukushima disaster to be considered in reviews.

2.2 OECD / NEA / CSNI DOCUMENTS

ID **Date of Issue**	**OCDE/GD(97)198** **Level 2 PSA Methodology and Severe Accident Management** **1998**
Issuer / Scope of document	This document was issued by the Organisation for Economic Co-operation and Development in 1997. The document was prepared by the Committee on Nuclear Regulatory Activities Working Group on Inspection Practices. The document includes the review and evaluation of the Level 2 PSA results and methodologies with respect to the plant type specific and generic insights. The document examines approaches and practices for using PSA results in the regulatory context and for supporting SAMP. The report is based on the information contained in PSA procedure guides, review guides and regulatory guides for the use of PSA results in IRIDM and plant specific probabilistic safety analyses.
Features of document	This report presents the results of the review of Level 2 PSA methodologies and practices and investigation how can support SAMP, i.e. the development, implementation, training and optimisation of accident management strategies and measures. The presented material reflects the state of art of Level 2 PSA in 1996. The state of application, results and insights from Level 2 PSA are presented and summarised in Section 2 of the document. The main severe accident phenomena and modelling issues are discussed in Section 3. The report presents approaches and practices in the area of accident management with respect to investigations and evaluations that should be performed in Level 2 probabilistic safety assessment in Section 4. The presentation of the available Level 2 PSA methodologies is given in Section 5. The document presents in Section 6 aspects important to quantification, including the use of expert judgement and the proper treatment of uncertainties with examples of use of PSA results and insights in the context of risk informed decision making presented in Section 7.
Information depth	The overall scope of the report included the review of the Level 2 PSA methodologies and practices and to investigate how Level 2 PSA can support SAMP, i.e. the development, implementation, training and optimisation of accident management strategies and measures. The presented material, for the most part, reflects the state-of-the-art in 1996.
Applicability	As PSA producer the report is applicable for identification of the state-of-the-art methods available for performing Level 2 PSA and severe accident/source term uncertainty analyses. As PSA reviewer the document is applicable within regulatory process for review of Level 2 probabilistic safety assessment. The document can be used for evaluation/implementation of severe accident management strategies.

ID **Date of Issue**	**NEA/CSNI/R(2009)4** **Probabilistic safety analysis (PSA) of other external events than earthquake** **May 2009**
Scope of document	Report of the Working Group on Risk Assessment (WGRisk) task on the state-of-practice regarding non-seismic external events (started 2007- ended 2009).
	The objective of the task was to review the methods for risk analysis of off-site external events other than earthquake as well as the results and the insights developed in these analyses in order to present a basis for advances in the area.
Features of document	Compilation of information provided by countries.
	Regulatory Requirements and Status of external event PSA.
	Definition of external event PSA scope.
	Analysis Methods.
	Results and Practical Applications.
Information depth	General information is provided.
Applicability	As PSA producer: some international experience information regarding the external hazards PSA.
	As PSA reviewer: international examples of external event PSA development and applications.
Comment	The report was developed before Fukushima.

2.3 EU TECHNICAL DOCUMENTS

ID **Date of Issue**	**ASAMPSA2** **EC Best-Practices Guidelines for L2PSA Development and Applications**
Scope of document	Best practice guideline for the performance and review of L2 PSA for internal initiating events, full range of initial states
	Provides some general views on the management of a L2 PSA, the existing background in many countries or international organisation
	Discusses the link between L2 PSA results, the approaches to present the results and their final application
	Provides recommendations regarding specific methodologies to be used in a L2 PSA (Level1/Level 2 PSA interface, accident progression event tree, release categories, human reliability analysis, etc)
	Provides recommendations on analysis that have to be performed to support a L2PSA (physical phenomena, system behaviour, source term assessment)
	Provides some views on the applicability of existing L2 PSA approaches for BWR and PWR to four Gen IV concepts
Features of document	Document elaborated by organizations having different responsibilities in nuclear safety activities (utilities, Technical Support Centers (TSC), safety authorities, research organizations, designer, …)
	Detailed description of a set of acceptable existing solutions to perform a L2 PSA (not a step by step procedure)
	Detailed and comprehensive description, with many recent references
	Illustrated by examples of applications from the different organizations involved
	Applicability for both "limited-scope" and "extended-scope" studies
Information depth	Detailed and comprehensive description
	Illustrated by examples of applications
Applicability	As PSA producer: useful and up-to-date document to develop L2 PSA
	As PSA reviewer: useful and up-to-date document to review L2 PSA
	Comment: the guideline does not provide a single solution for each issue of a L2 PSA but is good starting point for most of them.

2.4 NATIONAL DOCUMENTS

ID **Date of Issue**	**Use of PSA to Support the Safety Management of Nuclear Power Plants** **Safety Guide PP - 6/2010** **Bulgarian Nuclear Regularation Agency** **2010**
Scope of document	The PSA scope, reviewed in this Guide, includes the stages of NPP design and operation, respectively the different NPP operational states (full power, low power and shutdown state) and all potential initiating events and hazards, such as: a) Internal initiating events caused by random failures of components and human error, b) Internal hazards (for example, internal fires and floods, flying objects) and c) External hazards of natural character (for example, earthquakes, strong winds, tornados, external floods) as well as caused by human activities (for example, falling airplanes, accidents at nearby industrial plants).
Features of document	This Guide includes different elements which should be reviewed when using PSA to support the NPP safety management, meaning the necessary NPP PSA characteristics as well as its use based on international good practices.
Applicability	Covers the general approach for PSA. This safety guide is based on different references, inc. IAEA. Consequently it may be used in addition to these guides.

ID **Date of Issue**	**Canadian Nuclear Safety Commission REGDOC-2.4.2** **Probabilistic safety Assessment (PSA) for Nuclear Power Plants** **May 2014**
Scope of document	Regulatory Standard that sets out high level requirements for PSA Level 1 and Level 2 for licensees
	Supersedes the previous version of the same title that was identified as S-294. REGDOC-2.4.2 includes amendments to reflect lessons learned from the Fukushima nuclear event of March 2011, and to address findings from the CNSC Fukushima Task Force Report, as applicable to S-294.
	Refers to IAEA SSG-3 [7]and SSG-4 [8] for conducting quality PSA
Features of document	Requires that internal and external events be covered by the PSA
	Examples of internal and external hazards are provided
	Some general guidance is provided on various aspects of the PSA
	Provides guidance on public disclosure.
Applicability	At the time of publication, the CNSC was reviewing the methodology for developing multi-unit PSA to evaluate the site integrated risk. The CNSC will establish the safety goals for site-wide PSA.
	No practical guidance for PSA developers. More useful for PSA reviewers.

ID **Date of Issue**	**YVL 2.8** **Probabilistic Safety Analysis in Safety Management of Nuclear Power Plants** **2003**
Issuer / Scope of document	This document is issued by the STUK (Finnish regulatory authority). It includes requirements on PSA for design, construction and operation of NPP.
Features of document	The PSA Level 2 is addressed in chapter 4.3 of the YVL 2.8. YVL 2.8 contains a definition of large release of 1E+14 Bq of Cs-137. YVL 2.8 details a list of subjects which need to be covered in Level 2 PSA. The scope of Level 2 PSA is limited to the core. The document is kept very general and the requirements for Level 2 PSA are written only as bullet list.
	YVL 2.8 details a list of subjects which need to be covered in Level 2 PSA. The scope of Level 2 PSA is limited to the core.
Information depth	Only requirements, no depth of information about the requirement.
Applicability	As PSA producer: All points in the YVL 2.8 must be addressed in the PSA. As PSA reviewer: The YVL 2.8 can be used as checklist only, not as guidance.

ID **Date of Issue**	**YVL A.7** **Nuclear power plant probabilistic risk analysis and risk management** **draft, 2013**
Issuer / Scope of document	This document is issued by the STUK (Finnish regulatory authority). It includes requirements on PSA for design, construction and operation of NPP.
Features of document	The PSA Level 2 is addressed in chapter 4 of the YVL A7. YVL A.7 contains a definition of large release of 1E+14 Bq of Cs-137. YVL A.7 details a list of subjects which need to be covered in Level 2 PSA. In principle there is no limit of scope. In particular, the fuel pool also has to be analyzed. The document is kept very general and the requirements for Level 2 PSA are written only as bullet list.
Information depth	Only requirements, no depth of information about the requirement.
Applicability	As PSA producer: All points in the YVL A.7 must be addressed in the PSA. As PSA reviewer: The YVL A.7 can be used as checklist only, not as guidance.

ID **Date of Issue**	**BfS-SCHR-37/05, 2005: Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke (Methodenband); [33]** **BfS-SCHR-38/05, 2005: Daten zur probabilistischen Sicherheitsanalyse für Kernkraftwerke (Datenband) [34]**
Issuer / Scope of document	This document is issued by the German radiation protection agency (Bundesamt für Strahlenschutz). It includes guidelines to PSA and is binding for PSA for German nuclear power plants. The first document (Methodenband) contains the methodology to be used in the PSA. The second document (Datenband) contains specific data which may be used, such as Common Cause Failure (CCF) probabilities, branch probabilities for phenomena, etc.
Features of document	These documents provide general information about general PSA methodology, also including internal and external hazards. The PSA Level 2 is addressed in chapter 5 of the Methodenband and Chapter 7 of the Datenband. It is limited however to internal events during power operation. The Methodenband provides a suggestion to define a large release as more than 1E+16 Bq for Iodine and Cesium and an early release as a release that is too fast to allow full-scale evacuation measures. A time span of 10 hours is suggested. The Methodenband provides a list of phenomena that have to be addressed in Level 2 PSA, whereas the Datenband suggests possible methodology and also provides generic branch probabilities for specific phenomena. The documents specify in great detail how to define plant damage state and release categories. Based on this, there is a demand on the use of deterministic calculations for Level 2 PSA.
Information depth	Not only general level information but also specific information about branch probabilities and hazard methodology (PSAL1 only).
Applicability	As PSA producer: It can be used as a guideline which may provide input data, especially the Datenband document. As PSA reviewer: German PSA are reviewed based on compliance with the Methodenband and Datenband documents. Minor update of the documents is planned in 2014.

ID **Date of Issue**	**Guideline for Swiss Nuclear Installations - ENSI-A05/e** **Probabilistic Safety Analysis (PSA) : Quality and Scope** **2009**
Issuer / Scope of document	This guideline was issued by the Swiss Federal Nuclear Safety Inspectorate (ENSI) in 2009. The document defines the scope requirements regarding the plant-specific L1 and L2 PSAs for both internal and external events and covering all operating modes of the NPPs. Technical requirements for L1 PSA cover component reliability data analysis, human reliability analysis, internal events, internal and external plant hazards, and quantification / presentation of results. Regarding technical requirements for L2 PSA, the report covers definition and quantification of plant damage states, containment performance, containment loads, severe accident progression, source term analysis, and quantification / presentation of results. The guideline indicates that a specific PSA shall be performed for the spent fuel pool, which follows the same requirements as set forth for NPPs.
Features of document	<u>Technical requirements for L2 PSA</u> : This part is less detailed than the L1 PSA part. In particular, no information is given concerning HRA issue. The report begins by describing the plant damage state definition step (note that the document underlines that frequency uncertainty for each PDS must be derived from the L1 PSA). The containment performance assessment part indicates that the fragility containment curve must be pressure and temperature dependent. In addition to loading conditions typically considered for power states (Direct Containment Heating (DCH), Molten Concrete Corium Interaction (MCCI) …), it is mentioned some specific severe accident phenomena relevant for low-power and shutdown states (e.g., air ingression to the fuel assembly or potential for increased oxidation and zirconium fire). Concerning the source term analysis, the document underlines that a source term must be calculated including both the magnitude and the timing of radiological release.
Information depth	This document describes technical requirements both for NPP L1 and L2 full-scope PSAs (more information is given for L1 PSA). In particular, external hazards are described only in terms of L1 PSA (it is implicitly assumed that external hazards are considered in L2 PSA through the L1/L2 interface).
Applicability	For external hazards that are mentioned in the guide, the document can be used to assess the corresponding PDS frequencies. For spent fuel pool, no particular applicability to level 2 PSA was identified.

ID **Date of Issue**	**Guideline for Swiss Nuclear Installations - ENSI-A06/e** **Probabilistic Safety Analysis (PSA) : Applications** **2009**
Issuer / Scope of document	This guideline was issued by the Swiss Federal Nuclear Safety Inspectorate (ENSI) in 2009. The document formalizes the requirements for the application of probabilistic safety analysis for nuclear power plant. It presents the general principles, the requirements for maintenance and upgrade of the PSA, as well as the minimum required scope of PSA applications.
Features of document	The document presents the principles to maintain and upgrade a plant-specific PSA. A complete revision must be at least carried out in the course of the periodic safety review (changes in the PSA model must be carried out according to a procedure that ensures that the PSA model represents the current sate of the plant). The document indicates also that the impact of the plant modifications not yet incorporated in the PSA model must be quantitatively estimated and summarize at least (contents of the list are specified in the document). The document lists also those PSA applications which must be carried out as a minimum requirement : o probabilistic evaluation of the safety level (risk measure and criteria are given for existing operating plants), o evaluation of the balance of the risk contributors (this part concerns implementation of measures to reduce the risk), o probabilistic evaluation of the technical specifications (in particular, in terms of component maintenance and changes of technical specifications), o probabilistic evaluation of changes to structures and systems (applied to all PSA-relevant structural or system-related plant modifications), o risk significance of components (the Fussel-Vessely (FV) and Risk Achievement Worth (RAW) importance measures are used), o probabilistic evaluation of operational experience e.g. in case of methodological changes having significant impact on CDF (a detailed procedure is given).
Information depth	This document describes all steps necessary for maintenance and upgrade PSA for a given nuclear power plant. The risk measure and evaluation criteria to be applied are defined for these PSA applications.
Applicability	This document is applicable for maintenance and upgrade of level 1 and level 2 PSA.

ID **Date of Issue**	**FANR RG 003** **Regulatory Guide. Probabilistic Risk Assessment: Scope, Quality and Applications, Version 0** **2005**
Scope of document	The objective of this guide is to provide guidance for implementation of the requirements in FANR-REG-05, Regulation for the Application of PRA at Nuclear Facilities.
	This guide applies to the conduct of a PRA for application to nuclear facility siting, design, construction and operation. It addresses PRA scope, quality, application, maintenance and documentation. This guide is written for application to a LWR.
	Full scope level 1 and 2 PRA required, and a number of PRA applications.
Features of document	Refers to U.S.NRC regulatory guides and IAEA PSA guides.
Applicability	Example of regulatory requirements for PRA.

ID **Date of Issue**	**FANR RI 019** **2011 FANR Review instruction (PRA & Severe Accident Analysis), FANR** **2010**
Scope of document	Internal document how to review PRA submissions. The review of PRA and severe accident analysis covers the following main areas: * PRA scope, level of detail and technical adequacy * PRA quality, documentation and life-cycle update The extent to which PRA results have been integrated with deterministic analysis and used to identify and reduce risks during both the design as well as the operation phases. The extent to which PRA results compare to FANR safety goals in terms of CDF and Large Release Frequency (LRF) The extent of balance between preventive and mitigation-type features of severe accident management.
Features of document	Preliminary document, rather short. For detailed guidance on the referenced topic it is recommended that the reviewer utilise the following sections of USNRC NUREG-800 (Standard Review Plan - SRP): 19.0 Risk assessment and severe accident evaluation 19.1 Determining technical adequacy of PRA results
Applicability	Low value – mostly referring to other documents.

ID **Date of Issue**	**ASME/ANS RA-Sa-2009** **Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications** **2009**
Scope of document	Establishes technical requirements for Level 1 PRA of internal and external hazards for LWRs (covering internal events, internal flood, internal fire, seismic events, high wind, external flood, and other external hazards).
	Establishes also technical requirements for a limited Level 2 PRA (sufficient to evaluate Large Early Release Frequency (LERF)).
	At-power states are considered. Low power and shutdown states are not included (to be done in a future update of the Standard).
	"High-level requirements" as well as "supporting requirements" (i.e. more detailed technical requirements) are provided.
	These requirements apply to PRAs used for risk-informed decision-making related to design and operation of operating LWRs. They may be used for power plants under design or construction and for advanced LWRs, but then revised or additional requirements may be needed.
Features of document	The "High-level requirements" are rather general, but the so-called "supporting requirements" are often very detailed technical requirements.
	Focus on what has to be included (i.e., modelled and documented) in a PRA, in a very detailed manner. Many hints on how this can be done, but no further details on models and data.
	The requirements for internal events are developed in much more detail, considering that many of these requirements are fundamental for any hazard group. Hence, the requirements for internal and external hazards are limited to the requirements that are more specific for each hazard
	Distinction between 3 types of PRA capable of supporting PRA applications (so-called Capability Categories I, II and III), characterized by increasing level of detail and depth of the analysis.
	Most useful for experienced PRA practitioners (developers or reviewers), but not useful for beginners or for training in PRA techniques.
Applicability by TSO	As PSA producer: can take much benefit from the "supporting requirements" given in this standard; particularly useful when an internal or external peer review of the PRA is foreseen.
	As PSA reviewer: Standard can be used to guide the review of a PRA (model and documentation), in particular when the PRA is used in risk-informed decision making.

ID **Date of Issue**	**EPRI TBR TR-101869-V2** **Severe Accident Management Guidance Technical Basis Report : Volumes 1 and 2** **December 1992 (updated in October 2012)**
Scope of document	Comprehensive assessment of the possible effects that could result if specific actions are taken following core damage.
	Aim to provide a technical basis for the development of severe accident management guidelines by the individual owners groups.
Features of document	This report is organized into two volumes. Volume 1 defines the reactor coolant system (RCS), spent fuel pool (SFP), and containment damage conditions that could be relevant for severe accidents, identify the CLHAs, and summarize the effects that could result from each CHLA. Volume 2 is composed of appendices, each of which describes the physical behavior for one type of phenomenon relevant to severe accidents. These appendices also include the technical bases for calculation aids that can be used to estimate the core, RCS, and containment response if an action is taken.
	Warning: limited use regarding the PSA 2 Human Reliability Analysis as this CHLA approach is not shared by all severe accident management guidelines worldwide, and as the report stay generic in nature.
	EPRI developed the reference information and candidate high-level actions (CHLAs) used to support the development of severe-accident management guidelines (SAMGs) and published that information in 1992 as the EPRI report Severe Accident Management Guidance Technical Basis Report: Volumes 1 and 2 (TR-101869). This information and the CHLAs have been updated to account for the initial lessons learned from the Fukushima Dai-ichi accidents that occurred in March 2011. This update also reflects enhanced understanding of severe- accident behavior gained from research and analyses conducted in the 20 years since the original report was published.
	Provide discussion on the key phenomenological uncertainties.
Applicability by TSO	As PSA producer or reviewer: can take much benefit from the information included in the volume 2 for the phenomenological evaluation of the PSA 2 analysis.

ID **Date of Issue**	**US EPRI 3002000498** **Spent Fuel Pool Risk Assessment Integration Framework (Mark I and II BWRs) and Pilot Plant Application** **2013**
Issuer / Scope of document	This document is issued by the US EPRI.
	It describes the generic framework and methodology for conducting a SFP-Reactor PRA.
	This report summarizes the application of the generic framework on a pilot demonstration plant, a typical BWR Mark I. The pilot study addresses potential interactions between accident progressions in the reactor pressure vessel with events in the SFP.
	The objectives of this document include the following: - Understand the risks associated with SFPs. - Evaluate potential interactions between the reactor and SFP accident progression and risks. - Demonstrate plant-specific implementation for a selected Mark I BWR. - Provide guidance and insights for SFP risk evaluations to nuclear utilities. - Provide a stepping-off point for similar development for PWRs and BWR Mark III plants.
Features of document	The SFP-Reactor PRA framework presented in the document encompasses the following modelling aspects of postulated severe accidents: - Initiators - Common failure modes - Consequential failure modes - Integrated event tree structure - Success criteria - System fault trees - Operator actions and the applicable performance shaping factors - Appropriate end states (Level 1 and Level 2) - Appropriate LERF definition
	This document covers the following: - Full power internal events hazards - External events hazards (seismic) - Low power/shutdown hazards - Level 1 and Level 2 PRA for the above hazards (e.g., frequency of fuel damage and radionuclide release from the SFP)
	The document provides the evaluation of SFP risk due to (1) direct initiators affecting SFP cooling or inventory control, plus (2) initiators that cause core damage and containment challenges inducing loss of SFP inventory.
	The document also describes a plant-specific pilot demonstration for a typical Mark I BWR providing insight on the dominant sequences and their relative importance. The document includes results from sensitivity studies, along with safety insights that can be gained from the analysis.
Information depth	The document includes the considerations of selecting and optimizing the number of POSs for SFP.
	The document includes a list of initiating events, developed based on the engineering judgment derived from the evaluation.

	The document provides the Reactor PRA Level 1 Event Tree structures used to support the SFP-Reactor PRA model. The Reactor event tree structures include the following: - Level 1 Full power internal events PRA event trees - Level 1 Seismic PRA event trees - Level 1 Shutdown PRA event trees
	The document provides the model framework for the Level 2 Containment Event Trees (CET) and SFP event trees: - SFP event trees for accident progression and radionuclide release - Radionuclide release from the Reactor Pressure vessel (RPV) (Level 2 CETs) - Definition of LERF
	The document includes analysis of the risk profile associated with operation of the SFP and the reactor for key severe accident phenomena.
	The document addresses the human reliability analysis for shutdown and spent fuel pool risk assessment integration.
Applicability	As PSA producer: the report provides quite detailed methodology of PSA for SFP and mutual interaction between SFP and reactor core, supported with the practical examples of its implementation, and may be useful for PSA producer. This document is indented to the BWR type reactor but methodological aspects are invariant and may be used for PWR as well.
	As PSA reviewer: the report may be used as a basis for review of the SFP PSA and interaction with reactor core

<table>
<tr><td>ID
Date of Issue</td><td>ML12307A202
Joint Nuclear Regulatory Commission and Electric Power Research Institute Workshop on the Treatment of Probabilistic Risk Assessment Uncertainties (Draft version)
2012</td></tr>
<tr><td rowspan="2">Issuer / Scope of document</td><td>This document is issued by U.S. Nuclear Regulatory Commission (NRC)</td></tr>
<tr><td>It includes the description of the Workshop which took place in Rockville, MD, on February 29 – March 1, 2012. Its purpose was to bring together experts to gain a better understanding of the sources of uncertainty, how they are manifested in Probabilistic Risk Assessments (PRAs), and their potential significance to the PRA model and results for internal fires, seismic events, low power and shutdown conditions, and for the Level 2 portion of PRAs</td></tr>
<tr><td rowspan="4">Features of document</td><td>This document provides general information about sources of uncertainties associated with risk assessments for internal fires, seismic events, low power and shutdown conditions, and for the Level 2 portion of PRAs</td></tr>
<tr><td>The PSA Level 2 is addressed in chapter 6.</td></tr>
<tr><td>This document provides some general safety criteria such as, each topic discussed was assigned in a subjective significance ranking of:
- HIGH = The uncertainty has a moderate to high impact on the conclusions and risk insights.
- MEDIUM = The uncertainty has a small to moderate impact on the conclusions and risk insights.
- LOW = The uncertainty has a negligible to small impact on the conclusions and risk Insights.</td></tr>
<tr><td>This document provides description of the most important uncertainty sources treated at the Workshop, giving the following information: 1) a description of the issues or sources of uncertainty, 2) how the issues are manifested in the PRA, 3) a discussion of how the issues are relevant to the base PRA, application, or both, if the issues are applicable to new, existing, or advanced reactors, and the significance ranking (HIGH, MEDIUM, or LOW) for that issue as related to the Standard or draft Standard technical element, and 4) a discussion of potential research and development work which may be needed to resolve the issues or uncertainties.</td></tr>
</table>

	This document describes each uncertainty discussed and was categorized as model uncertainty, completeness uncertainty, level of detail uncertainty, or parameter uncertainty. The model uncertainty is related to an issue for which no consensus approach or model exists and where the choice of approach or model is known to have an effect on the PRA model (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, and introduction of a new initiating event). A model uncertainty results from a lack of knowledge of how SSC behave under the conditions arising during the development of an accident. The completeness uncertainty is caused by the limitations in the scope of the model, such as whether all applicable physical phenomena have been adequately represented, and/or all accident scenarios that could significantly affect the determination of risk have been identified. The level of detail generally refers to the level to which a system is modelled (e.g., function level, train level, component level), the extent to which systems are included in the success criteria (e.g., safety systems and non-safety systems), the degree to which events or sequences are subsumed, the extent to which phenomena are included in the challenges to the plant in the Level 2 analysis, and the extent to which operator actions are considered (e.g., accident management strategies). The parameter uncertainty is the uncertainty in the values of the parameters of a model and is typically represented by a probabilistic distribution. Examples of parameters that could be uncertain include initiating event frequencies, component failure rates and probabilities, and human error probabilities that are used in the quantification of the accident sequence frequencies.
Information depth	This document goes in depth in PSA uncertainties making a detailed description, showing its manifestation in the PRA and evaluating its relevance. The total number of individual uncertainty issues was: 59 issues for internal fire, 22 issues for seismic events, 22 issues for low power and shutdown, and 30 issues for Level 2.
Applicability	As PSA producer: Through the document the producer can know the most relevant uncertainties and see at which part of PRA affects, giving in some cases a possible solution, improving PRA. It appears for seismic events and Level 2 that model uncertainty was the predominant source of uncertainty, while internal fire and low power and shutdown had a more even spread among the various sources of uncertainty.
	As PSA reviewer: this document can be used as a guide to be informed about uncertainties existing and see for the most relevant if they are treated correctly in PRA.

ID **Date of Issue**	**US NRC - RG 1.200, Rev. 2** **An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities** **2009**
Scope of document	Approach for assessing the technical adequacy of PRA applied for risk-informed activities and regulatory decision-making for LWRs.
	Guidance on scope and main technical elements of a full-scope Level 1 and Level 2 PRA.
	Summaries of technical characteristics and attributes of each Level 1 and Level 2 technical element.
	Provides such guidance for internal events, internal flood, internal fire, seismic events, high wind, external flood, and other external hazards.
	5 For more detailed technical requirements and assessments, this SG refers to Consensus PRA Standards (ASME/ANS RA-Sa-2009 Standard in particular) and to the associated peer review process.
Features of document	Regulatory Position 1 is not a comprehensive guide but a concise description of all technical elements (and their main characteristics) as expected for a "technically acceptable PRA".
	No practical information on models and data.
	For more detailed process and technical requirements, this RG refers to "Consensus PRA Standards", in particular the ASME/ANS Standard for PRA Level 1 and limited Level 2 (LERF); Standards that are under development (for low power and shutdown states and for Level 2 PRA) are not yet indicated.
	Concise description of a peer review process (in particular based on the Standard ASME/ANS RA-Sa-2009).
Applicability	As PSA producer: can take benefit from the peer review process using this general guideline in combination with the ASME/ANS Standard.
	As PSA reviewer: this RG can be used to check general PRA features ; more inspiration for a review can be found in the ASME/ANS Standard which is endorsed by this RG.

ID **Date of Issue**	**US NRC NUREG-1738** **Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants** **2001**
Issuer / Scope of document	This document is issued by US NRC.
	This report documents a study of SFP accident risk at decommissioning nuclear power plants.
Features of document	The document is divided into three main parts. The first part (Section 2) is a summary of the thermal hydraulic analysis performed for SFPs at decommissioning plants. The second part (Section 3) discusses how the principles of risk informed regulation are addressed by proposed changes. The third part (Section 4) discusses the implications of the study for decommissioning regulatory requirements.
	The document studies a risk change at decommissioning plants depending on the strictness of the offsite emergency planning.
	The document analyses the relevance of a risk-informed examination of both the deterministic and probabilistic aspects of decommissioning to decisions on regulatory requirements for emergency preparedness, security, and insurance.

	The document provides insights for the design and operation of SFP cooling and inventory makeup systems and practices performance during off-normal conditions. and procedures necessary to ensure high levels of operator.
Information depth	The document describes in detail a modelling approach of a typical decommissioning plant with design assumptions and industry commitments; the thermal-hydraulic analyses performed to evaluate the behaviour of spent fuel stored in the spent fuel pool at decommissioning plants; the risk assessment of spent fuel pool accidents; the consequence calculations; and the sensitivity study and implications for decommissioning regulatory requirements.
	Typical outputs of thermal-hydraulic analyses are comprehensively summarized in specific annex.
Applicability	As PSA producer: the document is focused on SFP at decommissioning stage, however the sections with methodology of SFP PSA may be useful for PSA studies of operating units.
	As PSA reviewer: the document may be useful as guidance for performing the review of SFP PSA and for decision making process.

ID **Date of Issue**	**NUREG/CR-6451 BNL-NUREG-52498** **A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants** **August 1997**
Issuer / Scope of document	This document is issued by US NRC.
	This document presents a regulatory assessment for generic BWR and PWR plants that have permanently ceased operation in support of NRC rulemaking activities in this area.
	The objective of this document is to provide recommendations for those operationally based regulations that could be partially or totally removed for permanently shutdown plants without impacting public health and safety.
Features of document	The document focuses on investigation of spent fuel storage configurations for older PWR and BWR designs.
	This document has defined four spent fuel configurations which encompass all of the anticipated spent fuel characteristics and storage modes following permanent shutdown.
	The document identifies a list of candidate regulations which may not be applicable. The list of candidate regulations was identified from a screening of 10 CFR Parts 0 to 199. The continued applicability of each regulation was assessed within the context of each spent fuel storage configuration and the results of the consequence analyses.
	The document also identifies a list of regulations that may to be partially applicable to the permanently defueled facility.
Information depth	The identified four spent fuel configuration within the document are: 1. "Hot Fuel in the Spent Fuel Pool" – encompasses the period commencing immediately after the offload of the core to a point in time when the decay heat of the hottest assemblies is low enough such that no substantial zircaloy oxidation takes place and the fuel cladding will remain intact 2. "Cold Fuel in the Spent Fuel Pool" - the fuel can be stored on a long-term basis in the spent fuel pool, while the rest of the plant is in safe storage or decontaminated 3. Fuel moved to Spent Fuel Storage Installation (alternatively to configuration 2) - this would allow complete decommissioning of the plant and closure of the Part 50 license 4. This configuration assumes the plant Part 50 license remains in effect only because the plant has not been fully decontaminated and cannot be released for unrestricted public access.
	The set of regulations that are designed to protect the public against full power and/or design basis accidents are no longer applicable and can be deleted for all spent fuel storage configurations of the permanently shutdown plant. These regulations include combustible gas control (50.44), fracture prevention measures (50.60, 50.61), and ATWS requirements (50.62).
	Other regulations, although based on the operating plant, may continue to be partially applicable to the permanently defueled facility. This group of requirements includes the Technical Specifications (50.36, 36b), the fire protection program (50.48) and Quality Assurance (50.54(a) and Part 50 Appendix B).
	The requirements for emergency preparedness (50.47, 50.54(q) and (t), and Part 50 Appendix E), onsite property damage insurance (50.54(w)) and offsite liability insurance (Part 140), were evaluated using the accident consequence analysis.
Applicability	As PSA producer: the document is relevant for safety analyses of spent fuel pools of permanently closed plants
	As PSA reviewer: limited applicability

ID **Date of Issue**	**NUREG/CR-4982 BNL-NUREG-52093** **Severe accidents in spent fuel pools in support of generic safety - Issue 82** **1987**
Issuer / Scope of document	This document is issued by US NRC
	This document provides an assessment of the likelihood and consequences of a severe accident in a spent fuel storage pool
	The objective of this document is to introduce an assessment of the potential risk from possible accidents in spent fuel pools
Features of document	This document identifies potential mechanisms and conditions for failure of the spent fuel, and the subsequent release of the fission products
	The document considers two older PWR and BWR spent fuel storage pool designs based on a preliminary screening study which tried to identify vulnerabilities
	The document presents conditions which could lead to failure of the spent fuel Zircaloy cladding as a result of cladding rupture or as a result of a self-sustaining oxidation reaction. Propagation of a cladding fire to older stored fuel assemblies is evaluated. Spent fuel pool fission product inventory is estimated and the releases and consequences for the various cladding failure scenarios are provided.
	The document identifies the uncertainties in the risk estimate and areas where additional evaluations are needed to reduce uncertainty
Information depth	The document considers three factors that had not been included in earlier risk assessments: 1. Spent fuel is currently being stored rather than shipped for reprocessing or repository disposal, resulting in much larger inventories of spent assemblies in reactor fuel basins than had previously been anticipated; 2. In order to accommodate the larger inventory, high density racking is necessary; 3. A theoretical model suggested the possibility of Zircaloy fire, propagating from assembly to assembly in the event of complete drainage of water from the pool.
	The document considers both internal and external accident initiating events, including: - pool heat up due to loss of cooling water circulation capability, - structural failure of pool due to seismic events or missiles, - partial drain down of pool due to pneumatic seal failure, - structural failure of pool due to a heavy load drop
	The document provides a evaluation of the calculation results obtained by computer code SFUEL (SFUELIW) developed at Sandia National Laboratories and their applicability to beyond design-basis accidents in spent fuel pools
	The document presents data on the potential for releases of radio nuclides under various cladding failure scenarios and compares the projected releases with releases associated with severe core accident sequences
	In this document risk profiles are developed in terms of person-rem population doses for several accident sequences
	The document considers measures that might mitigate pool draining and/or Zircaloy fire propagation
Applicability	As PSA producer: the document is quite old but still may be used for SFP PSA in part of SFP severe accident phenomena and their consequences evaluation
	As PSA reviewer: limited applicability

ID **Date of Issue**	**U.S.NRC** **NUREG/CR-7110, State-of-the-Art Reactor Consequence Analyses Project** **Volume 1: Peach Bottom Integrated Analysis** **Volume 2 : Surry Integrated Analysis** **January 2012**
Issuer / Scope of document	These documents are included in SOARCA (State-of-the-Art Reactor Consequence Analysis) project, issued by U.S.NRC. The documents focused on providing a realistic evaluation of accident progression, source term, and offsite consequences for Peach Bottom Atomic Power Station, BWR design with Mark-I Containment, and Surry Power Station, PWR design with a large dry (subatmospheric) containment. SOARCA is an updated analysis incorporated the wealth of accumulated research and used more detailed, integrated, and best estimate modeling than past analysis. Also consider all mitigative measures, contributing to a more realistic evaluation.
Features of document	The SOARCA project evaluates plant improvements and changes not reflected in earlier NRC publications such as NUREG/CR-2239 [46]"Technical Guidance for Siting Criteria Development", NUREG-1150 [44] "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" and WASH-1400 [50]"Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants". SOARCA includes system improvements, improvements in training and emergency procedures, offsite emergency response, and security-related improvements, as well as plant changes such as power uprates and higher core burnup.
Information depth	SOARCA project identified two major groups of accident scenarios for analysis from the results of existing PRA. The first group common to both Peach Bottom and Surry includes short-term and long-term Station BlackOut (SBO). Both types of SBOs were analyzed as if they were initiated by a seismic event. SOARCA's second severe accident scenario group, which was identified for Surry only, is the containment bypass scenario. Two containment bypass scenarios were identified and analyzed. The first bypass scenario is a variant of the short-term SBO scenario, involving a thermally-induced Steam Generator Tube Rupture (SGTR). The second bypass scenario involves an Interfacing Systems Loss Of Coolant (ISLOCA) accident caused by an unisolated rupture of low-head safety injection piping outside containment. The project narrowed its approach by using an accident sequence's possibility of damaging reactor fuel, or CDF, as a surrogate for risk. SOARCA's analyses were performed with two computer codes, MELCOR for accident progression and the MELCOR Accident Consequence Code System, Version 2 (MACCS2) for offsite consequences. The analysis of offsite consequences in SOARCA incorporates the improved modeling capability reflected in the MELCOR and MACCS2 code as well as detailed site-specific public evacuation models. For scenarios that release radioactive material to the environment, MACCS2 uses site-specific weather data to predict the downwind concentration of material in the plume and the resulting population exposures and health effects. SOARCA modeled several types of mitigation measures. To assess its benefits and to provide a basis for comparison to the past analyses of unmitigated severe accident scenarios, the SOARCA project analyses the selected scenarios twice: first assuming that the event proceeds unmitigated, and then assuming that mitigation is successful. An appendix to this report compares and contrasts the SOARCA study and the Fukushima accident based on currently available information for the following topics: (1) operation of the reactor core isolation cooling system, (2) hydrogen release and combustion, (3) 48-hour truncation of releases in SOARCA, (4) multiunit risk, and (5) spent fuel pool risk.
Applicability	SOARCA results, while specific to Peach Bottom and Surry, may be generally applicable to plants with similar designs for development or upgrade of PRAs. Additional work would be needed to confirm this, however, since differences exist in plant-specific designs, procedures, and emergency response characteristics.

ID **Date of Issue**	**WASH 1400 (NUREG-75/014)** **Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants** **October 1975**
Scope of document	First risk assessment study performed with PSA tools (Event Trees / Fault Trees).
	The methodology used in WASH 1400 form the basis of the PSA/PRA developed today for NPPs or other complex systems.
Features of document	Detailed document quantifying intermediate (core damage) and final (to the public) consequences of an accident.
	Study developed by using simplified and conservative assumptions.
	The uncertainties on the results are rather high.
	An extensive comparison with other non nuclear risk was performed.
Applicability	WASH-1400 was replaced by NUREG-1150 / 1991 (Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants).
	Some data and insights of WASH 1400 may still be applicable even for modern PSA.

2.5 SUMMARY OF PUBLISHED GUIDES IN THE LIGHT OF ASAMPSA_E

The previous sections provide a rather comprehensive list of guides in the field of L2PSA. However, within the ASAMPSA_E project, external hazards, shutdown states, and spent fuel storages are of particular interest.

The following documents mentioned in the sections above address at least one of these topics:

Title	ASAMPSA-E topics covered					Comments
	Multiple site PSA	All sources of fission product releases (reactor, spent fuel pool, storages, etc…)	All operating states	All possible initiating events (induced multiple events or combination)	Post Fukushima modifications	
IAEA 50-P-8	X	X	X	X		Replaced by SSG-4. Includes PSA Level 1, 2, 3. Covers also some aspects from the analysis of the external hazards.
IAEA NG-G 2.15		X	X	X	X	Includes preparation and development of accident management programs, procedures and guidelines.
IAEA Safety Reports Series N° 25			X			General topics. Concerns PSA Level 2 - PSA project management and organization and application.
IAEA Safety Report Series N°32	X					General topics. Concerns mainly procedure for seismic PSA and seismic hazard analysis.
IAEA SSG-4		core only	X	X		Includes Level 2 PSA - organization, quality, accident progression analysis. Also includes WWER specific issues.
IAEA-TECDOC-724			X	X		Includes accident sequence modelling, and briefly external hazards.
IAEA-TECDOC-801	X			X , but not all		
IAEA-TECDOC-905	X			X , but not all		
IAEA-TECDOC-986	X		X	X , but not all		

IAEA-TECDOC-1144			X			Concerns mainly conducting the review of the level 2 PSA.
IAEA-TECDOC-1200				X		General topics. Includes defence in depth comination of internal and external sequences.
IAEA-TECDOC-1229			X			Includes procedure to achieve quality in PSA applications. Limited to full power and internal initiators.
IAEA-TECDOC-1487			X			Concerns integrated risk informed decision making process management. Notes deficiencies in external event PSA.
IAEA-TECDOC-1570	X		X	X , but not all		
IAEA INSAG-10			X			
IAEA INSAG-12				X , but not all		
IAEA INSAG-25		X	X	X		General topics. Includes preparation and development of accident management programs, procedures and guidelines.
IAEA Nuclear Energy Series, No. NP-T-2.2				X , but not all		
Safety Reports Series No. 46				X , but not all		
OCDE/GD(97)198						Technical aspects universally applicable.
OCDE/NEA/CSNI/R(2009)4				X , but not all		
EC ASAMPSA2		X	X	X	partially	Technical aspects universally applicable.
Bulgarian Nuclear Regulatory Agency Safety Guide PP-6/2010			X			Aims to cover intial events and hazards sush as external hazards of natural character.
Canadian Nuclear Safety Commission REGDOC-2.4.2	X	X	X	X	X	Regulatory standard with high-level requirements for conducting PSA taking into account Fukushima lessons.
Finland STUK YVL 2.8						high level requirements
Finland STUK YVL A.7	X	X	X		X	high level requirements
Germany BfS Daten (D) BfS-SCHR-38/05 and BfS-SCHR-37/05			X (PSA Level 1)	partially		Technical aspects universally applicable.

Switzerland ENSI-A05/e		Technical aspects universally applicable.				
Switzerland ENSI-A06/e		X	X	X		
United Arab Emirat FANR RG 003						high level requirements
United Arab Emirat FANR-RI-019						mostly referring to other documents
US EPRI TBR TR-101869-V2					X	Comprehensive assessment of the possible effects that could result if specific actions are taken following core damage.
US EPRI 3002000498		spent fuel pool	X	X	X	
US EPRI ML12307A202			X	X	X	
US NRC RG 1.200 - Rev. 2						
US NRC NUREG 1150		spent fuel pool				Technical aspects universally applicable.
US NRC NUREG-1738		spent fuel pool				focus on decommissioning
US NRC NUREG/CR-6451		spent fuel pool				focus on permanently shut down plants
US NRC NUREG/CR-4982					X	partly focus on SAMG
US Wash-1400						Historical report (technically outdated).

The post-Fukushima feedbacks and technical information are only covered by a limited number of documents and international references. In addition the PSA Level 2 issues are most of the time covered as an extension of the PSA Level 1 issues. The ASAMPSA-E PSA Level 2 if focused on the post-Fukushima and severe accident issues will be of great value for the international community.

3 SUMMARY OF MATERIAL OTHER THAN PUBLISHED GUIDES

3.1 EXAMPLES OF RECENT POST FUKUSHIMA DAIICHI ACCIDENT DEVELOPMENTS

In the wake of the Fukushima disaster, many activities have begun which could be of relevance for ASAMPSA_E. One example is the stress tests. Another example is the potential updates in the PSA post Fukushima Daiichi accident.

3.1.1 BULGARIA

The Fukushima Daiichi accident raised a lot of questions concerning the design, operation and safety of NPPs and prompted the authorities and the management of Kozloduy NPP in Bulgaria, as well as the countries all over the world, to join the world-wide efforts in relation to nuclear reliability and safety.

In the first place it could be underlined that the principles of the Act on the Safe Use of Nuclear Energy 2002 in Bulgaria determine the need to use and to apply of the lessons learned from nuclear accidents, which applies also in the case of the Fukushima accident.

A process of planning and implementation of improvement measures was initiated on the basis of Fukushima lessons learned and results from the stress tests performed on European NPPs, incl. Kozloduy NPP.

Part of the main improvements, some of them linked with PSA L1 and L2, included in the “Program for Implementation of Recommendations Following the Stress Tests Carried Out on Nuclear Facilities at Kozloduy NPP plc” could be listed:

- Construction of a new emergency management system, outside the Kozloduy NPP;
- Development of technical means for direct water supply to the steam generators and also to the spent fuel storage facility;
- Closing the ionizing chamber channels in the walls of the reactor cavity;
- Implementation of the symptom based emergency operating procedures for the shutdown states with open reactor;
- Installation of additional hydrogen recombiners in the containment;
- Installation of additional wide range temperature sensors for monitoring of the reactor vessel;
- Study of the options for localizing the molten core in case of a severe accident;
- Reconsideration and improvement of the SAMGs;
- Development a separated Spent fuel pool SAMG;
- Improving on-site and off-site emergency plans, taking into account difficulties in accessing the emergency control rooms, providing alternative routes for evacuation, transport of fuels and materials, access of the staff, etc.

On the other hand the National Report of Bulgaria concerning the “European Stress Tests for NPPs” identifies modifications for further enhancements as possible measures to increase robustness of Kozloduy NPP as for instance:

- Development of measures for prevention of water intake in the plant drainage network in case of valley flooding;
- Modernization of the draining and sewage systems of the plant;
- Development of an emergency procedure for personnel actions in case of wall ruptures of waterpower dams of Danube (Jelezni Vrata 1 and 2);
- Investigation of possibilities to protect the equipment of bank pumping stations of extremely high external flooding.

The extreme weather conditions and the combinations with other hazard events still need to be considered. In this regard, the Bulgarian Nuclear regulatory authorities requested the plant to perform a consolidated review of extreme weather hazards in correspondence with IAEA requirements and relevant guidance.

The Kozloduy NPP is in compliance with the licensing and Bulgarian national regulations on nuclear energy and radiation safety and deterministic as well as the probabilistic assessment studies have been developed for all operational units in order to confirm the design basis and the defence-in-depth, but the existing PSA for 5th and 6th units of Kozloduy NPP does not include external flooding or extreme weather, that was determinative in the case of Fukushima accident, consequently it should be included in the next PSA updates. For instance an approach of reassessment of the seismic hazard is made and should continue in the future.

The consequences from the Fukushima accident lead to the conclusion that all operational modes should be taken into account in PSA, as well all postulated events as severe weather conditions (a combination of extreme weather conditions), fire, flooding and seismic events, etc.

3.1.2 CANADA

Following the events at the Fukushima Dai-ichi nuclear power plant, the Canadian Nuclear Safety Commission established a Fukushima Task Force in April 2011 to review licensees’ responses to the CNSC order to re-examine the safety cases of their nuclear power plants, with the objective of reviewing the capability of Nuclear Power Plants (NPPs) to withstand conditions similar to those that triggered the Fukushima accident. Specifically, the CNSC Task Force examined the response of NPPs to external events of higher magnitude than previously considered. Based on the post-Fukushima review, the CNSC Task Force confirms that the Canadian NPPs are robust and have a strong design relying on multiple layers of defence. The design ensures that there will be no impact on the public from external events that are regarded as credible. The design also offers protection against more severe external events that are much less likely to occur. Nevertheless, the Task Force made 13 recommendations to further enhance the safety of nuclear power plants in Canada. One recommendation is specific for the external hazards. Licensees should conduct more comprehensive assessments of site-specific external hazards to demonstrate that:

- the considerations of magnitudes of design-basis and beyond-design-basis external hazards are consistent with current best international practices;

- the consequences of events triggered by external hazards are within applicable limits. Such assessments should be updated periodically to reflect gained knowledge and modern requirements.

The general approach to address this CNSC action is mainly based on the main steps as expressed in the CNSC Regulatory Document S-294 (Supeseded by REGDOC 2.4.2; See section 2.4). Note that for the three types of external hazards such as seismic, external floods and high winds, they are not supposed to be screened out from detailed analysis. The licensees developed different methodologies and the CNSC staff has accepted some of them.

3.1.3 CZECH REPUBLIC

After the Fukushima accident, stress tests were performed for both Czech NPPs - Dukovany and Temelin. Results of the stress tests considerably accelerated implementation of modifications which were planned in order to get approval for operation extention. At Dukovany NPP many changes have been performed, the most important related to PSA-2 are: 1. Installation of hydrogen recombiners, 2. Modifications related to in-vessel retention, 3. Mobile diesel generator, 4. new ventilating towers used for containment heat removal. New SAMGs covering the modifications have been developed as well. Probabilistic assessment of these modifications is in process and should by finished till the end of 2014.

Basic (without external events) PSA-2 for SFP of Dukovany NPP was performed in 2013 and the results show only very limited contribution to the total LERF. The main reason is that the time windows of releases from SFP are typically very long and do not fall into definition of “early” release.

Related to activities of Czech regulatory body, a new guideline for PSA-2 development and preparation has been issued, and new regulation, which will require PSA-2 development for each NPP, is in process.

3.1.4 FRANCE

Following stress tests analysis, three phases have been defined by EDF and are currently discussed with French Safety authorities to upgrade reactor safety according to Fukushima feedback:

Phase 1 (2012 – 2015): availability of connectors (water and electricity) to connect mobile equipment for situations beyond design failure such as cumulative total loss of power and total loss of cooling chain ; the mobile equipment will be transported and operated by a dedicated regional crisis team (FARN = Rapid Nuclear Task Force).

Phase 2 (2015 -2020): additional on site redundant equipment such as additional generator and additional water tank (with the opportunity to refill by local pumpage) will be available; a local crisis center will also be built on each nuclear site.

Phase 3 (starting 2019): preventing containment venting (with additional containment cooling system) or preventing basemat failure in order to face GEN 3 requirements, depending on industrial feasibility.

Concerning PSA development, the French Safety Authorities requests EDF to extend progressively the scope covered by L1 and L2 PSA. This request was formulated before the Fukushima accident (in relation with periodic safety reviews, cost-safety methodologies, long term operation perspective) and confirmed after this accident.

The safety benefit of the new equipment, as quantified by PSAs, will be firstly examined in the framework of of the 4th PSR of 900 MWe series.

3.1.5 GERMANY

The accidents at the Fukushima-Daiichi plant happened in the last phase of the development of the new German "Safety Requirement for NPP" [51], promulgated in early 2013. The lessons from the accident were incorporated into the new German regulation, leading to a number of changes especially related to the deterministic safety assessment approach. With regard to PSA, the role of PSA insights in the provision of evidence and regulatory decision making outside of the PSR was strengthened. However, no specific requirements on PSA were newly introduced because of the Fukushima-Daiichi accidents.

Within the German (regulatory) framework, the guidelines for performing and assessing PSA level 1 and level 2 are developed by the "Facharbeitskreis Probabilistische Sicherheitsanalysen" (FAK), an advisory body to the federal regulator Bundesministerium für Umwelt, Naturschutz, Bau und Reaktorsicherheit (BMUB). The FAK is currently working on supplementary guidelines to the PSA guideline ("Leitfaden PSA") of the BMUB, issued in 2005 /GRS 3/. These supplements will give more detailed requirements on the scope and methods for PSA on a number of specific issues, e.g. hazards assessment. In particular, the following issues have been emphasized / introduced following the lessons learned:

- Applying a fixed analysis time of e.g. 24 h for PSA (level 1) and assuming that scenarios will be contained due to successful emergency measures if core damage does not happen before is no longer accepted. It has to be demonstrated that a controlled plant state has been reached that can be maintained for a prolonged period barring additional (probabilistic) failures.
- The reliability of the cooling of the spent fuel storage has to be included into the scope of the PSA.
- The scope of the PSA level 1 is extended to "fuel damage states", i.e. specifically including damages to fuel outside of the reactor core. This extension is in line with the German Safety Requirements, where PSA level 1 "core damage" frequency is defined to include all initiators and all operating states.
- The scope of PSA has to be extended to systematically screen and if necessary assess in detail combinations of initiating events. This is specifically requested for combinations of hazards (external as well as internal).
- For the probabilistic assessment of emergency operating procedures as well as severe accident management actions, the specific boundary conditions of the scenario (accessibility/operability of equipment, environment/high radiation areas, etc.) have to be taken into account.
- It has always been required that PSA level 2 takes into accout all relevant phenomena and scenarios. After Fukushima, Hydrogen issues outside of the containment have been mentioned explicitly both for releases by containment venting and for other hydrogen releases into the containment (containment failures).

In addition, GRS is performing research into specific issues for a PSA with an extended scope. One focus is on a systematic and efficient extension of detailed PSA level 1 assessments to internal and external hazards: Analysing the hazards and their combinations with respect to relevance and frequency (of exceedance); defining initiating events induced by each relevant hazard; extending the plant model to include the hazard of induced failures and unavailabilities of SSC. Another research issue is the extension of PSA level 2 analyses to low-power and shutdown operating states and severe accident scenarios in the SFP and after external initiating events.

Shortly after the Fukushima accident, the federal government ordered that 8 of the older NPPs in Germany had to be shut down immediately for three months in a so called moratorium. During this period, the reactor safety commission performed “stress tests” for all plants, taking into account extremely pessimistic assumptions about boundary conditions. The reactor safety commission concluded that there were a few weaknesses to be adessed, but did not identify reasons for shutting down plants. In parallel to the reactor safety commission, a so-called ethics commission was set up by the federal government as well. This commission concluded that renewable electric power production could and should be promoted to such extent that nuclear power is no longer needed within 10 years. The ethics commission voted for phasing out nuclear because according to their opinion less hazardous technologies are available. On the 6th of June, 2011 – less than three months after the accident – the federal parliament ruled in an act (version of this act as of 2014-09-12 see [52]) that the 8 older plants had to remain shut down permanently, and that the rest of the plants be shut down successively until 2022. It seems that politics and the German public will not alter this act in the future, and that phase-out of nuclear in Germany is definitive.

3.1.6 SLOVENIA

Following the accident at the Fukushima Daiichi nuclear power plant (NPP), the European Council requested that a comprehensive safety and risk assessment, in the light of preliminary lessons learned, be performed on all EU nuclear plants [53], [54] including Krško NPP in Slovenia.

The Slovenian National Action Plan [55] was prepared as a result of all activities executed in Slovenia in response to the nuclear accident in the Fukushima Daiichi NPP in March 2011. These activities include, but are not limited to, the implementation of the European Stress test process, implementation of June 2011 short-term improvements, review and analysis of possible long-term improvements based on which the Krško NPP’s Safety Upgrade Program (SUP) was prepared, review of several reports, reviews and analyses regarding the Fukushima lessons learned.

The core of the Slovenian National Action Plan and post-Fukushima improvements represents the planned Krško NPP’s Safety Upgrade Program (SUP), which was ordered, reviewed and approved by the Slovenian Nuclear Safety Administration (SNSA). It required from the plant to upgrade its systems, structures and components to enable coping with severe accidents after the plant lifetime was extended. After the Fukushima accident the SNSA ordered the plant to implement these measures in advance.

Additional systems, structures and components, which will be implemented within the SUP, will be designed and structured in accordance with the design extension conditions (DEC) requirements specific for the Krško NPP design and site location.

3.1.7 SPAIN

The NPPs in Spain shall be improved in response to the lessons of the Fukushima accident, according to the national assessments, the recommendations and suggestions of the European Stress Tests and the conclusions of the CNS process and other sources.

The assessment followed the structure proposed by ENSREG and covered all aspects specified in the ENSREG Action Plan. An important additional topic: potential inaccessibility of large areas at a NPP – which is at the interface between safety and security – was also addressed. This information is accessible on the regulator's website.

At each site with nuclear power plants a "Local information Committee" is established to inform at least annually the local authorities, NGOs, and the general public about relevant aspects concerning the operation and any other topic which could be considered of interest in respect to the nuclear installations.

The implementation of improvement measures is clearly scheduled in three steps: short term (until end of 2012), medium term (until the end of 2014) and long term (until the end of 2016). The timeframe to implement all the improvement measures until the end of 2016 is ambitious and commendable. Nevertheless some measures scheduled for long term are crucial ones, like filtered venting and installation of PARs.

3.1.8 SWEDEN

Following the severe accident at the Fukushima Daiichi nuclear power plant in 2011 and the EU stress tests completed in 2012, a Swedish national action plan [56] covering all Swedish nuclear power plants has been developed to implement lessons learned from the accident and to deal with the conclusions from the second extraordinary meeting [57] under the Convention on Nuclear Safety in 2012. The Swedish action plan mainly contains crosscutting and comprehensive measures and presents investigations whose aim is to determine and consider which technical and administrative measures are fit for purpose, how they shall be implemented and the appropriate time schedule for implementation. The measures listed in the Swedish national action plan [56], which consists of further analyses and investigations, are scheduled in three different categories, 2013, 2014 and 2015, corresponding to the year when the measures shall be completed. This categorization is based on an assessment of the urgency of the measures' implementation as well as the complexities of these measures.

In addition to the national action plan, a number of measures to increase the level of safety at Swedish nuclear power plants were implemented within a year after the accident at the Fukushima Dai-ichi nuclear power plant. These measures were mainly identified in connection with investigative work linked to the licensees' international forum, WANO, and in connection with the stress test assessments conducted by Swedish nuclear facilities [58]. A majority of the measures had been completed by the end of 2012. These measures are relatively straightforward measures, feasible to take in the short term to increase the likelihood of preventing a serious incident, while also reinforcing the work on severe accident management including emergency response organizations [59].

Below, a summary is provided of some of the Swedish actions taken, or to be taken, in the light of the Fukushima Dai-ichi nuclear power plant accident, and related to level 2 issues relevant for ASAMPSA_E.

PSA issues:

According to the safety regulations SSMFS 2008:1, all Swedish reactors have to be analysed with probabilistic methods to supplement the basic deterministic safety studies. All power reactors have to perform complete level-1 and level-2 PSA studies including all operating modes and all relevant internal and external hazards for the sites. Today, all power reactors have performed level 1 and level 2 studies. The level-1 studies have been updated continuously with regard to plant modifications. Work has been performed to fill gaps in the level-1 studies and to finalize studies for low power operation, area events and external hazards.

The basic PSA studies are expected to be updated every year taking into account the past year's plant modifications which have an impact on the PSA-result. In principle most licensees are moving towards practising a so-called "Living PSA". PSA results are also used routinely by the licensees to support decisions concerning significant modification of the designs, modification of operations, documentation and assessment of events.

As mentioned in earlier national reports, the numerical PSA figures are not regarded as a definitive and exact value of the actual risk level. There are no requirements related to numerical PSA results, although the licensees have such safety objectives. The studies should be sufficiently detailed, comprehensive and realistic to identify weaknesses in the designs and to be used to assess plant modifications, modifications of technical specifications and procedures as well as assessment of the risk significance of events.

Severe Accident Issues

The comprehensive risk and safety assessments demonstrate the importance of the consequence-mitigating systems, where the accident filters are key. In an accident situation where residual heat removal has failed and the reactor core is melting through the reactor vessel, the pressure in the containment will rise until valves to the accident filter open and relieve the pressure from the containment into the atmosphere. This filter has been designed so that a considerable proportion of the radioactive substances that may be present in the gases passing through the accident filters are captured, thus largely preventing ground contamination.

The accident filters were originally designed for 24 hours of operation without operator actions. As the lessons learned from the accident at Fukushima Dai-ichi have demonstrated that accident sequences can be prolonged and that it can be difficult in these situations to carry out manual actions within 24 hours, the licensees need to evaluate the accident filters in terms of long-term operation.

In Sweden, work has long been underway to develop the facilities for the purpose of preventing hydrogen explosions. It has nonetheless been established that the licensees have not conducted a detailed and thorough study of the risk of hydrogen leakage to the reactor building, which in fact did occur from the reactors of

Fukushima Dai-ichi. For this reason, the licensees must investigate these risks further. Above all, these investigations should focus on the risk of hydrogen accumulation in reactor buildings, as well as the need for additional monitoring to assist operators and other working staff. Beyond this, dealing with hydrogen over a long-term perspective needs to be taken into account.

Strategies for emergency response management are at the present time oriented at sequences where the consequence-mitigating systems protect containment integrity and thus prevent large and uncontrolled radiological discharges into the environment. Lessons learned from the accident at Fukushima Dai-ichi nevertheless indicate that pre-planned strategies are also needed covering accidents involving failure of the containment function and where considerable releases of radioactive materials are unavoidable.

When updating existing strategies for emergency response management, an in-depth analysis of the accident response organisation's structure and staffing also needs to be performed to ensure that it is capable of dealing with all situations, in particular situations where several reactors are affected simultaneously.

3.1.9 SWITZERLAND

The Swiss federal inspectorate ENSI has undertaken a program to implement international recommendations in the wake of the Fukushima accidents. The action plan has been detailed in 2012 in [60], which is yearly updated ([61] and [62]). A summary of the status of the implementation is also published and updated ([63]). With the premise that PSA analyses as contemplated in ASAMPSA_E have been conducted in Switzerland already for over 20 years, the action items listed below are of interest to ASAMPSA_E. The list is not exhaustive of the work requested by ENSI to operators (and internally to ENSI itself), and shows that even with more than 20 years experience in performing and reviewing complete PSAs there are areas which are most sensitive and which require more work, attention and quality assurance. The most important action items impacting PSA studies are highlighted:

- Investigate restoration of containment integrity in case of a total SBO at shutdown.
- **Increase safety margins in case of external hazards and combinations thereof, some analyses to be based on PSA results.**
- Revisit (review) seismic fragilities in view of earthquakes with return probability of 1E-4 or greater.
- Re-assess secondary events (flooding, fires ….) in case of earthquakes.
- Installation of mobile flood barriers for auxiliary buildings.
- **Regarding flooding, increase safety margins for BDBA events (i.e. with return period less than 1E-4), again, analysis to be based on PSA.**
- Investigate the potential of additional heat sinks, if current configuration is not sufficient to cope with Fukushima type events.
- **Investigate the potential for releases of hazardous substances other than radioactivity (may become a new PSA requirement).**
- **Review impact of instrumentation (and lack thereof) for the SFP.**

- Study the effect of ventilation during SBO (possible enhancements) for equipment operability.
- Backfit SFP cooling systems for protection against earthquake events.
- **Check whether resources and means to deliver coolant and provide cooling to the SFP are actually sufficiently available in case of earthquake with extensive damage to the SFP and induced SBO.**
- **Check adequacy of accident management procedures (see if the PSA assumptions are consistent, especially with respect to mobile equipment).**
- Re-investigate hydrogen hazards from SFP.
- **Investigate the potential for release of large quantities of radioactive contaminated water (may become a new PSA requirement).**
- **Promote the idea that the use of IAEA Safety Standards should be strengthened in regular peer review missions on the assessment of the regulatory framework and activities.**
- **Check on the safety culture of the operators.**
- **Check on independence of regulators from operators.**
- **Ensure cooperation with neighboring countries (prevention of spill-over of consequences of a severe accident to other countries).**
- **Review assumptions to determine whether coolant supply is guaranteed from alternate systems from diverse sources against external hazards and combinations thereof.**
- **Review design, operation, procedures and consequences of filtered containment venting systems.**
- **It must be insured that internationally harmonised assessment scales for nuclear safety are established at the highest level of safety.**

Summarizing, this elaborate program is aimed at developing strategies to avoid and deal with severe accidents and extreme natural hazards at a nuclear power plants.

3.2 PUBLICATIONS (IN SCIENTIFIC CONFERENCES AND OTHER)

The purpose of the following section is to propose a list of PSA Level 2 publications of particular interest for ASAMPSA_E. This list is not trying to be exhaustive and is not trying to cover all PSA Level 2 issues. The content of the technical material was proposed by the ASAMPSA-E participants but not reviewed by the scientific community. This will be updated and completed during the next phase of the ASAMPSA_E project. Many publications from research can be expected during the next few years (for example, ERMSAR 2015).

Table 3 : PSA Level 2 Publications of Interest for ASAMPSA_E

Title	Date	Author(s)	Reference and/or Company	ASAMPSA-E Issues	Conference (if applicable)
General Scope					
What level of robustness shall be expected for NPPs severe accident management provisions and how to demonstrate it? Some IRSN views based on recent experience (poster)	2014	Raimond E, Dubeuil M., Guigueno Y., Cenerino G., Menage F., Pichereau F.	IRSN	Severe accident management provisions	International Experts Meeting on Severe Accident Management in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant Vienna (Austria) (IAEA-CN-233)
Estimation of Nuclear Fuel Cycle Cost and Accident Risk Cost (Statement),	November 10, 2011	Japan Atomic Energy Commission	Japan Atomic Energy Commission	Risks, risk metrics	
Risk Targets in view of Fukushima: Myths and Facts.	Septembe r 2011	Vitázková J., Cazzoli E.:	Vitázková-Vitty Slovakia, Cazzoli Consulting Switzerland	Various issues related to nuclear safety, risks and current practices	Proceedings of International Nordic PSA Conference, project No. 01/004, 5-6 Johannesbergs Slott, Gottröra, Sweden, , collection of papers, Chapter 7

Identification of Research Areas in Response to the Fukushima Accident	January 2013	Report of the SNETP Fukushima Task Group, Chairman Jozef Misak	SNETP (Sustainable Nuclear Energy Technology Platform)	Phenomenology	
The principle of Defence-in-Depth in the perspective of Probabilistic Safety Analyses in wake of Fukushima,	June 2014	J. Vitázková, E. Cazzoli	Risk Analysis IX, Book series: WIT Press, ISSN 1743-3517, and http://library.witpress.com CCA	DiD	9th International Conference on Risk Analysis and Hazard Mitigation
IAEA International Fact Finding Expert Mission of the Nuclear Accident Following the Great East Japan Earthquake and Tsunami, Tokyo, Fukushima Dai-ichi NPP, Fukushima Dai-ini NPP and Tokai NPP, Japan, 24 May - 1 June 2011, Preliminary Summary	July 2011	IAEA	IAEA	DiD	IAEA
Focus areas for a Level 2 PSA that supports a site NPP risk analysis	2015	D.M. Helton US-NRC, M. Zavisca & M. Khatib-Rahbar, ERI	*Proceedings Of The European Safety And Reliability Conference, ESREL 2014* (2015-01-01) p. 1605-1610.	L2PSA scope	ESREL 2014

			ISBN: 9781-138026810		
EU stress test					
Peer review report Stress tests performed on European nuclear power plant	2012	EU and EC Board	ENSREG	EU Stress Test	Joint statement of ENSREG and the European Commission (presented at the June 2012 European Council)
Technical summary on the implementation of comprehensive risk and safety assessment of nuclear power plants in the EU (SWD/2012/287 final 2)	2012	Commission Staff	European Commission	EU Stress Test	
National Action Plants Workshop. Summary Report	2013	-	ENSREG	EU Stress Test	ENSREG National Action Plan Workshop (Brussels April-2013)
French stress tests					
IRSN Analysis of Post-Fukushima « Hardened Safety Core »: Use of PSA Insights	2013	LANORE J.M., CORENWINDER C., GEORGESCU G., GUIGUENO Y., HERVIOU K., LAVARENNE C. & RAIMOND E.	IRSN	Stress tests, Hardened Safety Core, Extreme external hazards, PSA	Probabilistic Safety Assessment and Management Topical Conference, Tokyo (Japan)
Phenomenological Evaluation					
SARNET Benchmark on VVER1000 Molten Core Concrete Interaction Reactor Test Cases	2012	R. Gencheva, A. Stefanova, P. Groudev, M. Cranga, D. Dimov, C. Spengler, I. Ivanov, J. Foit, P.		MCCI, Ex-Vessel, VVER 1000	5th European Review meeting on Severe Accident Research (ERMSAR-2012)

		Kostka			
Evaluations of MCCI Risks for the Fukushima Events, Related IRSN R&D Strategy on Corium Retention and Coolability	2012	CRANGA M., CHEVALIER-JABET K., MARCHETTO C. & MUN C.	IRSN	Corium Retention and Coolability	International Meeting on Severe Accident Assessment and Management : Lessons Learned from Fukushima, ANS Winter Meeting, San Diego (USA)
Simulation of the Core Degradation Phase of the Fukushima Accidents using the ASTEC Code	2014	BONNEVILLE H. & LUCIANI A.	Nuclear Engineering and Design (in press)	Core-Degradation	
Investigation of some phenomena and parametrical studies on VVER1000 MCCI	2013	R. Gencheva, A. Stefanova, P. Groudev, M. Cranga, V. Tyrpekl, J. Duspiva, B. Kujal, G. Lele, B. Chatterjee		Ex-Vessel, MCCI	6th European Review meeting on Severe Accident Research (ERMSAR-2013)
Transposition of 2D Molten Corium-Concrete Interactions (MCCI) from Experiment to Reactor	2013	C. Spengler, A. Fargette, J. Foit, K. Agethen, M. Cranga		Ex-Vessel, MCCI	6th European Review meeting on Severe Accident Research (ERMSAR-2013)
Towards an European consensus on possible causes of MCCI ablation anisotropy in an oxidic pool	2013	M. Cranga, C. Spengler, K. Atkhen, A. Fargette, M. Fischer, J. Foit, R. Gencheva, E. Guyez, J. F. Haquet, C.		Ex-Vessel, MCCI	6th European Review meeting on Severe Accident Research (ERMSAR-2013)

		Journeau, B. Michel, C. Mun, P. Piluso, T. Sevon, B. Spindler			
The results of the ARTIST project – consequence analysis of a spontaneous SGTR	April 2013	Lind T., Güntay S., Dehbi A., Suckow D.	Proceedings Paper No. FA121	Source terms – retention phenomenology and uncertainties	ICAPP 2013 Jeju Island, Korea, April 14-18, 2013,
Severe Accident Supporting Analysis					
Verification of severe accident management strategies for VVER 1000 (V320) reactor,	2010	Chatterjee, B., Mukhopadhyay, D., Lele, H.G., Atanasova, B., Groudev, P.,		SAMG,	2nd nternational Conference on Reliability, Safety and Hazard, ICRESH-2010: Risk-Based Technology and Physics-of-Failure Methods; Mumbai; Pages 280-287
Environmental Impact Assessment of Nuclear Facility and Prevention of Accidents	2010	Ivanov I.	TUS	EIA, prevention, accidents	Proceedings of 2010 American Nuclear Society Winter Meeting and Nuclear Technology Expo, US, Las Vegas, 2010.
Severe accident management strategy verification for VVER-1000 (V320) reactor	2011	B. Chatterjee, D. Mukhopadhyay, H.G. Lele, B. Atanasova, Pavlin Groudev; et al.	NED, Volume 241, Issue 9, pages 3977 – 3984	SAMG	

ASTEC investigations of severe core damage behaviour of VVER-1000 in case of loss of coolant accident along with Station-Black-Out	2013	P Groudev, B Atanasova, B Chatterjee, H G Lele	NED, http://dx.doi.org/10.1016/j.nucengdes.2013.06.039	Core damage, fission product release, LOCA, SBO	
ATHLET-CD/COCOSYS Simulation of the Accidents in Units 2 and 3 of Fukushima Daiichi	August 2014	S. Band, M. Sonnen-kalb, GRS,		Fukushima accident evaluation	GRS contribution for the OECD/NEA BSAF Project, Phase-1
Source Term Calculation					
Fukushima Dai-ichi's Radioactive Source Term and Release in the Environment	2012	BRUNA G. B., ISNARD O., RAIMOND E., CORBIN D. & DENIS J.	IRSN	Source Term and Release	"One year after Fukushima: Rethinking the Future" Workshop, Bologna (Italy)
FUKUSHIMA ACCIDENT PROGRESSION, 1 Jahr nach Fukushima	March 2012	Steven C. Sholly	Wiener Umwelt Anwaltschaft, Wien, März 2012.	Accident progression and source terms	Symposium zum 1. Jahrestag der Reaktorkatastrophe,
Influence of the nodalisation/zoning by the ThAI – Iod 11 and Iod 12 tests analysis with ASTEC code applications	2013	Kaleychev P., I. Ivanov	TUS	THAI experiments, ASTEC code, Iodine chemistry	Bulgarian Nuclear Society International Conference NUCLEAR POWER FOR THE PEOPLE. 18-21 September 2013, Sunny Beach Base of the Council of Ministers, Bulgaria.
Human Reliability Analysis (HRA)					

Human reliability analysis for EPR NPP PSA level 2 (last findings)	2010	E. Sauvage P. Duncan-Whiteman	AREVA	Human Reliability Analysis for PSA Level 2	Tenth Conference on Probabilistic Safety Assessment and Management - PSAM 10, 7th -11th June 2010, USA
PSA Level 2 Results					
Estimate of Consequences from the Fukushima Disaster	2011	J.Vitázková , E. Cazzoli	Vitazkova-Vitty, Cazzoli Consulting	Phenomenolog y/releases	Proceedings of International Nordic PSA Conference, Project Nr. 01/004, 5-6 Sept. 2011, Johannesbergs Slott, Gottröra, Sweden
Spent Fuel Pool Risk Assessment Integration Framework (Mark I and II BWRs) and Pilot Plant Application	2013		EPRI		**EPRI 3002000498**
Risk Targets in view of Fukushima. Proceedings of International Conference Probabilistic Safety Assessment	**June 2012**	Vitazkova J., Cazzoli E	PSAM11 - ESREL 2012, Helsinki, Finland, 25-29 June 2012, ISBN: 978-1-62276-436-5.	Risk metrics	PSAM11 - ESREL 2012, Helsinki, Finland, 25-29 June 2012
Consequences of the Fukushima Accident	June 2012	Vitazkova J., Cazzoli E	Proceedings, ISBN: 978-1-62276-436-5.	Risk metrics	International Conference Probabilistic Safety Assessment PSAM11 - ESREL 2012, Helsinki, Finland 25-29 June 2012,

METHODOLOGY OF COMMON RISK TARGET ASSESSMENT AND QUANTIFICATION FOR SEVERE ACCIDENTS OF NUCLEAR POWER PLANTS BASED ON INES SCALE	May 2014	J. Vitázková	Ph.D. Thesis, Slovak University of Technology, Faculty of Electrical Engineering and Information Technology, Institute of Nuclear and Physical Engineering,	Risk metrics	
Risk Targets in view of Fukushima: Myths and Facts. Proceedings of	September 2011	Vitázková J., Cazzoli E.		Risk metrics	International Nordic PSA Conference, project No. 01/004, 5-6 Johannesbergs Slott, Gottröra, Sweden, September 2011, collection of papers, Chapter 7.

4 EVALUATION OF EXISTING MATERIAL

Following the work performed to summarize all existing information of particular interest for ASAMPSA_E, regarding the PSA Level 2, an evaluation of the above information is proposed for all the extended PSA issues in Table 4 below. The purpose is to identify any missing issues in the international guides.

The extended PSA Level 2 issues are:

- the methodology for developing event trees,
- multiple site PSA,
- the PSA for spent fuel storage,
- the low power / shutdown PSA,
- the fire and flooding after internal events,
- the human reliability analysis and the related severe accident management guidelines,
- the specific issues for accident evolution after external events.

Table 4 : Evaluation of Existing Material for Extended PSA Usage

Issue	Example for existing guidance for "ordinary" PSA	Existing guidance for extended PSA	Guidance need for extended PSA
Event tree methodology	IAEA-SSG-4 provides technique and application	None	Guidance needed how to transfer event tree method to extended L2PSA (E.g. to take in consideration recovery of failed equipment)
Multiple site PSA	none	none	guidance needed
Spent Fuel Pool PSA	US EPRI 3002000498 provides guidance for PSAL Level 1 for SFP	Some limited guidance is given for Level 2 PSA for SFP	More guidance needed for PSA Level 2 for SFP (e.g. open RPV, phenomenology for fuel melt in air environment , development of codes to support the spent fuel storage phenomenological evaluation)

Low power / Shutdown PSA	IAEA TECDOC-1144 provides guidance for PSA Level 1 for low power and shutdown states	US EPRI 3002000498 provides some guidance for PSA Level 2 for low power / shutdown PSA	More guidance needed for PSA Level 2 for low power / shutdown in particular for open RPV
Human reliability analysis	Several guidance for PSA level 1 human reliability is available	Human Reliability for PSA Level 2 is insufficiently covered by existing guidance.	More guidance needed for PSA Level 2 human reliability, especially with regard to dependency with L1PSA human reliability, confidence of personnel in measurements and availability of an increasing amount of personnel (e.g. crisis team or external support) and for specific SAMG action (e.g., containment isolation)
SAMG	Guidance for SAMG compilation exists (e.g. US EPRI-101869)	More guidance on how to incorporate SAMG into PSA Level 2 (see human reliability analysis column).	See human reliability analysis. More guidance needed on the feedback of PSA Level 2 results into SAMG.
Specific issues for accident evolution after external events	Detailed guidance is available for L1 PSA (e.g. IAEA-TECDOC-724)	More guidance on how to extend into PSA Level 2.	Guidance needed how to perform PSA Level 2 following external events (e.g., pratical guidances on screening of materials/components to recover after seisms, system reliability in particular for all systems that are used in PSA 2 but either started or prepared for utilisation as part of EOP, modelling of the system backup outside of unit in PSA Level 2).

5 LIST OF REFERENCES

[1] ASAMPSA2, EC - Best-Practices guidelines for L2 PSA development and applications (ASAMPSA2), 2010

[2] ETSON Expert group

[3] IAEA 50-P-8, Procedures for Conducting Probabilistic Safety Assessment of Nuclear Power Plants (Level 2), 1995

[4] IAEA NG-G 2.15, Severe Accident Management Programmes, 1995

[5] IAEA Safety Reports Series N° 25, Review of Probabilistic Safety Assessments by Regulatory Bodies, 2002

[6] IAEA Safety Report Series N°32, Implementation of Accident Management Programmes in Nuclear Power Plants, 2004

[7] IAEA SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants, 2010

[8] IAEA SSG-4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants, 2010

[9] IAEA-TECDOC-724, Probabilistic safety assessment for seismic events, 1993

[10] IAEA-TECDOC-801, Development of safety principles for the design of future nuclear power plants, 1995

[11] IAEA-TECDOC-905, Approaches to the safety of future nuclear power plants, Report of a Technical Committee meeting held in Vienna, 29 May - 2 June 1995

[12] IAEA-TECDOC-986, Implementation of defence in depth for next generation light water reactors, 1997

[13] IAEA-TECDOC-1135, Probabilistic safety assessments of nuclear power plants for low power and shutdown modes, 2000

[14] IAEA-TECDOC-1144, Applications of Probabilistic Safety Assessment (PSA) for nuclear power plants, 2001

[15] IAEA-TECDOC-1200, Applications of probabilistic safety assessment (PSA) for nuclear power plants, 2001

[16] IAEA-TECDOC-1229, Regulatory Review of Probabilistic Safety Assessment (PSA) Level 2, 2001

[17] IAEA-TECDOC-1487, Advance nuclear plant design options to cope with external events , 2006

[18] IAEA-TECDOC-1570, Proposal for a Technology-Neutral Safety Approach for New Reactor Designs, 2007

[19] IAEA INSAG-10, Defence in Depth in Nuclear. A report by the International Nuclear Safety Advisory Group

[20] IAEA INSAG-12, Basic Safety Principles for Nuclear Power Plants 75-INSAG-3 Rev. 1; INSAG-12. 1999

[21] IAEA INSAG-25, A framework for an Integrated Risk Informed Decision Making Process, 2011

[22] IAEA Nuclear Energy Series, No. NP-T-2.2, Design Features to Achieve Defense in Depth in Small and Medium Sized Reactors, 2009

[23] IAEA Safety Reports Series No. 46, Assessment of Defence in Depth for Nuclear Power Plants, 2005

[24] IAEA Procedings of Conference, Safety of Radiation Sources and Security of Radioactive Materials, Dijon, France, 14-18 September 1998

[25] IAEA Procedings of Conference, Research Reactor Utilization, Safety, Decommissioning, Fuel and Waste Management, Santigo, Chile, 10-14 November 2003

[26] IAEA Procedings of Conference, Protection against Extreme Earthquakes and Tsunamis in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant I/ International Experts Meeting, IAEA, Vienna, Austria, 4–7 September 2012

[27] OCDE/GD(97)198, Level 2 PSA Methodology and Severe Accident Management

[28] OCDE/NEA/CSNI/R(2009)4, Probabilistic Safety Analysis (PSA) of Other External Events than Earthquake

[29] Bulgarian Safety Guide PP-6/2010, Use of PSA to Support the Safety Management of Nuclear Power Plants, Bulgarian Nuclear Regulatory Agency, 2010

[30] Canadian Nuclear Safety Commission REGDOC-2.4.2, Probabilistic safety Assessment (PSA) for Nuclear Power Plants, 2014

[31] Finland STUK YVL 2.8, Probabilistic Safety Analysis in Safety Management of Nuclear Power Plants, 2003

[32] Finland STUK YVL A.7, Nuclear Power Plant Probabilistic Risk Analysis and Risk Management, 2013

[33] Germany BfS Daten (D) BfS-SCHR-37/05, Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, 2005

[34] Germany BfS Daten (D) BfS-SCHR-38/05, Daten zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, 2005

[35] Switzerland ENSI-A05/e, Probabilistic Safety Analysis (PSA): Quality and Scope, Guideline for Swiss Nuclear Installations, 2009

[36] Switzerland ENSI-A06/e, Probabilistic Safety Analysis (PSA): Applications, Guideline for Swiss Nuclear Installations, 2009

[37] United Arab Emirat FANR RG 003, United Arab Emirat FANR-RI-019, Regulatory Guide, Probabilistic Risk Assessment: Scope, Quality and Applications, 2005

[38] United Arab Emirat FANR-RI-019, 2011 FANR Review instruction (PRA & Severe Accident Analysis, 2010

[39] US ASME/ANS RA-Sa-2009, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, 2009

[40] US EPRI-101869, Severe Accident Management Guidance Technical Basis Report,1992

[41] US EPRI 3002000498, Spent Fuel Pool Risk Assessment Integration Framework (Mark I and II BWRs) and Pilot Plant Application, 2013

[42] US EPRI ML12307A202, Joint Nuclear Regulatory Commission and Electric Power Research Institute Workshop on the Treatment of Probabilistic Risk Assessment Uncertainties (Draft version), 2012

[43] US NRC RG 1.200 - Rev. 2, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, 2009

[44] US NRC NUREG 1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, 1990

[45] US NRC NUREG-1738, Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, 2001

[46] US NRC NUREC/CR-2239, Technical Guidance for Siting Criteria Development, 2012

[47] US NRC NUREG/CR-4982, Severe accidents in spent fuel pools in support of generic safety, 1987

[48] US NRC NUREG/CR-6451, A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants, 1997

[49] US NRC NUREG/CR-7110, State-of-the-Art Reactor Consequence Analyses Project Volume 1: Peach Bottom Integrated Analysis, Volume 2 : Surry Integrated Analysis, 2012

[50] US Wash-1400, The Reactor Safety Study, 1975

[51] http://regelwerk.grs.de/deutsches-regelwerk/index.html

[52] http://www.gesetze-im-internet.de/bundesrecht/atg/gesamt.pdf

[53] ENSREG. Stress tests performed on European nuclear power plants - Peer review report. 2012

[54] ENSREG. Stress tests performed on European nuclear power plants - EU "Stress tests" specifications. 2011

[55] SNSA. Slovenian Post-Fukushima National Action Plan. 2012

[56] SSM, Swedish action plan for NPP, Response to ENSREG's request, Dec 2012 (http://www.stralsakerhetsmyndigheten.se/Global/Pressmeddelanden/2012/%C3%85tg%C3%A4rdsplaner/swedish-action-plan.pdf)

[57] Ds 2012:18 Convention on nuclear safety 2012 extra ordinary meeting – The Swedish National Report, Regeringskansliet, Ministry of the Environment. (http://www.regeringen.se/content/1/c6/19/73/60/2cba4dce.pdf)

[58] SSM, Investigation of long-term safety in the Swedish nuclear power industry and measures owing to the accident at Fukushima Dai-ichi, 31-10-2012 (https://www.stralsakerhetsmyndigheten.se/Global/Pressmeddelanden/2012/Investigation-of-long-term-safety-eng.pdf)

[59] Sweden´s sixth national report under the Convention of Nuclear Safety, Ministry of the Environment, Ds 2013:56, Stockholm 2013. (https://www.riksdagen.se/sv/Dokument-Lagar/Utredningar/Departementsserien/Swedens-sixth-national-report_H1B456/?text=true)

[60] ENSI, Fukushima Action Plan 2012, ENSI-AN-7844, 28 February 2012, http://static.ensi.ch/1331720486/ensi_actionplan_2012_fukushima.pdf

[61] ENSI, Fukushima Action Plan 2013, ENSI-AN-8283, 28 February 2013, http://static.ensi.ch/1369139826/ensi_actionplan_fukushima_2013_eng.pdf

[62] ENSI, Fukushima Action Plan 2014, ENSI-AN-8711, 28 February 2014, http://static.ensi.ch/1393918895/ensi_2014_aktionsplan_fukushima_de.pdf

[63] ENSI, Implementation of post-Fukushima international recommendations, Switzerland, no reference yet, update 2014.