JRC Scientific and Technical Reports

Risk Informed Support of Decision Making in Nuclear Power Plant Emergency Zoning

Generic Framework towards Harmonising NPP Emergency Planning Practices

Jozef Kubanyi, Ricardo Bolado Lavin, Dan Serbanescu, Bela Toth, Heinz Wilkening

EUR 23280 EN - 2008

JRC
EUROPEAN COMMISSION

Institute for Energy

The Institute for Energy provides scientific and technical support for the conception, development, implementation and monitoring of community policies related to energy. Special emphasis is given to the security of energy supply and to sustainable and safe energy production.

European Commission
Joint Research Centre
Institute for Energy

Contact information
Address: P.O. Box 2, 1755 ZG Petten, the Netherlands
E-mail: jozef.kubanyi@jrc.nl; heinz.wilkening@jrc.nl
Tel.: +31 224 56 5376; +31 224 56 5368
Fax: +31 224 56 5641

http://ie.jrc.ec.europa.eu/
http://www.jrc.ec.europa.eu/

JRC 43311

EUR 23280 EN
ISSN 1018-5593

Luxembourg: Office for Official Publications of the European Communities

Printed in the Netherlands

Risk Informed Support of Decision Making in Nuclear Power Plant Emergency Zoning

Generic Framework towards Harmonising NPP Emergency Planning Practices

Jozef Kubanyi, Ricardo Bolado Lavin, Dan Serbanescu,
Bela Toth, Heinz Wilkening

European Commission, DG JRC
Institute for Energy

January 2008

EUR 23280 EN

Preface

The phenomenology of severe accidents of nuclear power plants (NPP) is very complex. The improved understanding of them is a result of the benefit from research and development activities in severe accident phenomenology. Owing to the state-of-the-art understanding and increased characterisation of NPP severe accidents, overall management of them could be – and also should be - analysed as an integrated complex process. The interrelationship of NPP emergency operating procedures, safety and risk assessments, severe accident management guidelines, and emergency off-site actions should be planned and organized to minimize the consequences of such accidents. A deterministic approach, coupled with both probabilistic safety assessment (PSA) technology and PSA results can play significant roles in the development of relevant nuclear utility, regulatory and all stakeholders' policies.

This report describes the background, objectives and current state of a corresponding activity within JRC-IE's Analysis and Management of Nuclear Accidents (AMA) Action on probabilistic safety / risk assessment methodologies and practices for risk-informed decision making approach (RIDM) applied to NPP emergency zoning within FP7. It provides a systematic overview and generic framework of the essential aspects of RIDM in NPP emergency zoning as a contribution to possible future harmonisation of strategic planning practices in this area. The issue is challenging, because this approach is interdisciplinary by nature, based on integration of PSA technology, severe accident phenomenology, and radiological protection.

This activity is expected to complement - in terms of probabilistic aspects - current JRC-IE activities on traditional deterministic safety / risk assessment of NPPs.

TABLE OF CONTENTS

LIST OF ABBREVIATIONS

ABWR	Advanced Boiling Water Reactor
AC	Alternating Currant
APET	Accident Progression Event Tree
ARC	Advanced Reactor Concept
ASTEC	Accident Source Term Evaluation Code
AGR	Advanced Gas Reactor
ALARA	As Low As Reasonably Achievable
ALARP	Low As Reasonably Practicable
ALWR	Advanced Light Water Reactor
APWR	Advanced Pressurised Water Reactor
ASME	American Society of Mechanical Engineers
BDBA	Beyond Design Basis Accident
BWR	Boiling Water Reactor
CCDF	Complementary Cumulative Distribution Function
CDF	Core Damage Frequency; Cumulative Distribution Function
CET	Containment Event Tree
CRAC	Calculation of Reactor Accident Consequences
DBA	Design Basis Accident
DCF	Dose Conversion Factor
DCH	Direct Containment Heating
DEPZ	Detailed Emergency Planning Zone
DG	Directorate-General
EC	European Commission
ECCS	Emergency Core Cooling System
ECD	Effective Committed Dose
EDF	Electricité de France
EIA	Environmental Impact Assessment
EJ	Expert Judgement
EP	Emergency Planning
EPR	European Pressurised water Reactor
EPRI	Electric Power Research Institute
EPZ	Emergency Planning Zone
ERP	Emergency Response Planning
EU	European Union
EZ	Emergency Zone, Emergency Zoning
EUR	European Utility Requirements (Group)
FAO	Food and Agriculture Organization (of the United Nations - UN)
FAST	Fourier Amplitude Sensitivity Test
FRPZ	Food Restriction Planning Zone
GFR	Gas-cooled Fast Reactor
GIL	Generic Intervention Level
GRS	Gesellschaft für Anlagen und Reaktorsicherheit
HDMR	High Dimensional Model Representation
HSE	Health and Safety Executive

I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IE	Initiating Event
IE (of JRC)	Institute for Energy
ILO	International Labour Organization
INSAG	International Nuclear Safety Advisory Group
IPE	Individual Plant Examination
IPSN	Institut de Protection et de Sûreté Nucleaire
IRF	Iodine Release Fraction
IRIS	International Reactor Innovative and Secure
IRSN	Institut de Radioprotection at de Sûreté Nucléaire
ISLOCA	Interfacing Systems LOCA
JRC	Joint Research Centre
L	Level
LER	Large Early Release
LERF	Large Early Release Frequency
LFR	Lead-cooled Fast Reactor
LFW	Loss of steam generator Feed Water
LHS	Latin Hypercube Sampling
LOCA	Loss Of Coolant Accident
LOOP	Loss Of Offsite Power
LPI	Low Pressure Injection (of ECCS)
LPZ	Longer term Protective action Zone
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Program
MCCI	Molten-Core Concrete Interaction
MDB	Material Data Bank
MSLB	Main Steam Line Break
MSR	Molten Sault Reactor
NEA	Nuclear Energy Agency
NII	Her Majesty's Nuclear Installations Inspectorate
NPP	Nuclear Power Plant
NRWG	Nuclear Regulators Working Group
OCHA	Office for the Co-ordination of Humanitarian Affairs (of the UN)
OECD	Organisation for Economic Co-operation and Development
PAHO	Pan American Health Organization
PAG	Protective Action Guide
PAR	Passive Autocatalytic Recombiners
PAZ	Precautionary Action Zone
PCC	Partial Correlation Coefficient
PDF	Probability Density Function
PDS	Plant Damage State
PIUS	Process Inherent Ultimate Safe
P-G	Pasquill-Gifford (weather stability category)
PORV	Power (Pilot) Operated Relief Valve

PRA	Probabilistic Risk Assessment
PRCC	Partial Rank Correlation Coefficient
PSA	Probabilistic Safety Assessment
PWR	Pressurised Water Reactor
RC	Release Category
RCS	Reactor Coolant System
R&D	Research and Development
RIDM	Risk Informed Decision-Making
RPV	Reactor Pressure Vessel
RSS	Reactor Safety Study (WASH-1400)
RSWG	Reactor Safety Working Group
RV	Relief Valve
Ry	Reactor.year
SA	Severe Accident; Sensitivity Analysis
SAR	Safety Analysis Report
SAMG	Severe Accident Management Guidelines
SARNET	Severe Accident Research NETwork
SBO	Station Black-Out
SCTP	Source Term Code Package
SCWR	Supercritical Water-cooled Reactor
SFR	Sodium-cooled Fast Reactor
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SFR	Sodium-cooled Fast Reactor
SNL	Sandia National Laboratories
SRC	Standardised Regression Coefficient
SRRC	Standardised Rank Regression Coefficient
STC	Source Term Category
S&U	Sensitivity and Uncertainty (analysis)
TECDOC	Technical Document (of the IAEA)
TMI	Three Mile Island
UK	United Kingdom
UPZ	Urgent Protective (action planning) Zone
USNRC	United States Nuclear Regulatory Commission
VHTR	Very High Temperature Reactor
V&V	Verification and validation
WHO	World Health Organization
WWER	Water-Water Energy Reactor

EXECUTIVE SUMMARY

At the Institute for Energy (IE) of the Joint Research Centre (JRC) of the European Commission (EC), Petten, The Netherlands, an activity on probabilistic safety / risk assessment methodologies and their applications is in progress within the framework of the JRC FP7 Action Nr. 52101 "Analysis and Management of Nuclear Accidents" (AMA).

The objective of this activity is to contribute to the common, state-of-the-art understanding of the methodologies and their applications in order to improve the confidence in them, and to help decrease unnecessary conservatism for more effective ensuring the continued safety of nuclear power plants (NPP).

Level 2 (L2) and level 3 (L3) of the current, advanced PSA technology and their results can, in principle, be used to estimate the offsite consequences of beyond design basis accidents (BDBA) and severe accidents of NPP. They could provide an acceptable basis for implementation of risk informed support in decision making (RIDM) processes, related to NPP emergency planning measures, especially to defining emergency zones round NPPs.

Considerable experience has been gained during the past years regarding severe accident risk assessment and mitigation, mainly in the USA. Prediction of environmental impacts of severe accidents (in the form of probability-weighted consequences) was performed for all NPPs and the risk reduction potential was identified using severe accident mitigation alternatives. These activities are usually performed mainly within the license renewal process of the plants. In addition, the future risk is calculated for the extended lifetimes.

Based on the information obtained during the first stage of the project within FP6, several significant differences have been found in the definitions of emergency planning zones (EPZ) of the NPPs in different countries within the EU and beyond. The current approach to emergency planning is, in general, traditionally deterministic, when usually a reference accident is defined to be used as a basis for drawing up corresponding emergency plans.

This report provides a supplement to the report EUR 21580 EN [1], produced by JRC-IE in 2005 on this subject, and aimed at benchmarking and harmonising strategic planning practices for NPP emergency zoning. In EU Member States, the practical application of L2 PSA results for the emergency management has been limited and, in fact, little risk based information has been used. In the course of that project, only the Czech Republic, Slovakia and the UK informed that L2 PSA results were used in some way as an input to emergency arrangements. The UK is the only EU Member State, which investigated how L2 PSA outcomes could be used in a systematic way for emergency planning purposes.

Since then, more L2 PSA studies have been initiated, completed or updated, for the variety of NPP designs being in exploitation in EU Member States. Besides, now there is a better understanding of complex severe accident phenomena and their modelling. The improved understanding is a result of the benefit from research and development activities in severe accident phenomenology.

This report describes the background, the objectives and current state of the corresponding JRC-IE activities in the field of probabilistic safety / risk assessment methodologies and practices with emphasis on the RIDM approach applied to NPP emergency zoning. This approach is interdisciplinary; it is based on integration of PSA technology, severe accident phenomenology, and radiological protection. It provides a systematic overview of the essential aspects of RIDM in NPP emergency zoning and contributes to the possible future harmonisation of strategic planning practices in this area. Owing to the current, advanced level of PSA technology, which is already mature enough, state-of-the-art understanding and increased characterisation of NPP severe accidents, overall management of them could be - and should be - analysed as an integrated complex process. The interrelationship of NPP emergency operating procedures, safety and risk assessments, severe accident management guidelines, and emergency off-site actions should be planned and organised to minimize the consequences of such accidents. A traditional deterministic approach, coupled with both PSA technology and PSA results, can play significant role in the development of relevant utility, regulatory and all stakeholders' policies.

The benefits from this project are:

1) A better understanding of important issues in PSA technology applications to risk informed supporting of NPP emergency zoning in relation to emergency management,
2) A better knowledge on the actual use of various current approaches and methods in the area, and
3) Information on the efforts undertaken by utilities, regulatory authorities and other stakeholders to explore possibilities and means of using probabilistic approaches for this topic.

The overall aim of this report is to establish a generic framework towards possible future harmonisation of NPP emergency planning practices in EU Member States. The resulting knowledge should help regulatory authorities, civil protection institutions, European institutions such as EC services, and various PSA users and developers to get a clear picture on the relevance of the issue, the consistency of current approaches and on related research and development needs.

1. INTRODUCTION

Emergency planning zones (EPZs) around a NPP are, in general, defined on a deterministic basis. They help to elaborate a strategy for protective actions during an emergency. The exact size and shape of each EPZ is a result of detailed analysis, which includes consideration of geographical features and demographic information specific at each site.

Predetermined protective action plans are set up for EPZs; they are designed to avoid or reduce doses from potential ingestion of radioactive materials. These actions include sheltering, evacuation, and use of stabile iodine-isotope tablets in the short term (first two weeks after the severe accident); then food bans, population relocation and decontamination in the longer term.

There are various differences in EU Member States in the way how emergency plans have been drawn up and how EPZs have been defined. Usually simplified deterministic approaches are used.

Based on the state-of-the-art developments and achievements in application of PSA technology, the original hypothesis that initiated this project is that PSA technology is currently already mature enough to support defining NPP emergency zones based on risk informed decision making approach. This resulted in the long term objective of this JRC project to agree - together with the developers and owners of this information - on a harmonised “template” to publish corresponding results to different stakeholders, (including the public), at a European level.

This report provides a supplement to the report EUR 21580 EN [1], produced by JRC-IE in 2005 on the same subject, and aimed at benchmarking and harmonising strategic planning practices for NPP emergency zoning. Since then, there have been more probabilistic safety assessment studies of level 2 (L2 PSA) developed, completed, updated, and elaborated in EU Member States for variety of NPP designs. Besides, there is now a better understanding of severe accident phenomena and modelling, which is extremely complex. The improved understanding of them is a result of the benefit from research and development activities in severe accident phenomenology.

Owing to the state-of-the-art understanding and increased characterisation of NPP severe accidents, there would be a better chance of using L2 and L3 PSA results for risk informed support of decision making (RIDM) in NPP emergency zoning. A traditional deterministic approach, coupled with both advanced, matured PSA technology and PSA results can play significant role in the development of relevant nuclear utility, regulatory and all stakeholders’ policies. The overall aim of this report is to establish a generic framework towards possible future harmonising of NPP emergency planning practices in EU Member States.

2. BACKROUND AND RELEVANCE OF THE ISSUE

The 1995-2000 activity programs of the Nuclear Regulators Working Group (NRWG) and the Reactor Safety Working Group (RSWG) of the EC were carried out within the framework of the 1975 and 1992 resolutions of the Council of Ministers on the technological problems of nuclear safety [1].

The 1975 resolution called for *"... progressive harmonisation of safety requirements and criteria in order to provide for an equivalent and satisfactory degree of protection of the population and of the environment against the risk of radiation resulting from nuclear activities …*" The 1995 Consensus Document on the safety of European Light Water Reactors (LWR) noted that *"... harmonisation begins with the identification of convergences and the assessment of divergences based on synthesis studies resulting from an intensive exchange of information of the actual practices in the different Member States".*

In 1993, the EC established a contract with a Consortium of European Technical Support Organisations (TSOs) in order to arrive at common views on technical safety issues related to large evolutionary Pressurised Water Reactors (PWR) in Europe, which could be ready for operation during the next decades. The TSOs involved were: AVN (Belgium) (Technical project leader), former AEA Technology (United Kingdom), ANPA (Italy), CIEMAT (Spain), GRS (Germany) and IPSN (France). The general objective of the European TSO Study Project on Development of a Common Safety Approach in the EU for Large Evolutionary Pressurised Water Reactors [2] was to develop, through a collaboration of EU TSOs, a common safety approach to issues related to large evolutionary PWRs in Europe. The TSO study represented an important step forward in the development of a common approach of the TSOs to the safety of advanced evolutionary PWRs. This goal was mainly achieved by an in-depth analysis of the **key safety issues**, taking into account new developments in the national technical safety objectives.

After careful considerations, and on the basis of the survey of advanced PWR concepts in preparation for the consolidated analysis, a list of 12 key issues was finally prepared and selected for in-depth analysis. These selected key issues, listed below (those key issues of the list, which are in close relation to the report in hand, are printed in **bold**), were judged to have the greatest safety significance:

- **Use of PSA in design and licensing**;
- **Reduced environmental source term and emergency plan**;

---

[1] http://europa.eu.int/comm/energy/nuclear/safety/index_en.htm

- Identification of postulated initiating events (PIEs) and associated acceptance criteria;
- Instrumentation and control systems important for safety (hardware and software aspects);
- System architecture;
- Passive systems behaviour;
- Practical elimination of core melt in shutdown states with open containment;
- Practical elimination of high pressure core melt;
- Practical elimination of core melt with containment bypass;
- **Practical elimination of large early releases resulting from containment failure;**
- Mitigation of low pressure core melt and vessel melt-through;
- Identification of severe accidents: methodology and acceptance criteria.

For all the key issues considered in the European TSO Study, conclusions have been developed covering the state of knowledge, safety approaches, and the approaches taken in selected reactor designs. In addition, TSO group positions have been formulated regarding the development of a **common approach** for each key safety issue, highlighting any studies still to be done in order to reach the required common understanding and consensus. These common positions formed the major achievement of the TSO study project. Areas in which further work was felt to be needed include:

- **PSA methods and use;**
- **In-containment source term and radiological releases;**
- Application of the Single Failure Criterion (SFC) and maintenance; consideration;
- Reliability of passive systems;
- Containment by-pass;
- Hydrogen risk, no occurrence of deflagration to detonation transition;
- Strategies for corium coolability;

- Demonstration of practical elimination of selected sequences;

- Qualification of systems for severe accidents.

In summary, an important step forward has been made in the development of a common safety approach of the TSOs. This was mainly achieved by an in-depth analysis of the key safety issues. The above lists of key issues and of areas for further work clearly indicate that risk informed support of emergency zoning for NPPs and potential future harmonisation of strategic planning practices are of high relevance.

A further argument for moving towards more risk informed approaches comes from the common practices in another high-risk industrial sector, the chemical process industry: Although in the process industry the probabilistic approach to risk assessment is certainly less complete and consistent as compared to the nuclear industry, risk informed results are nevertheless used in many countries for land use planning (risk / emergency zoning) purposes. Land use planning is a legal requirement in the EU under the so-called Seveso II Directive ("Directive 96/82/EC on the control of major-accident hazards")[2] and risk informed methods are encouraged in the practical implementation of the Directive.

Owing to the state-of-the-art understanding and increased characterisation of NPP severe accidents as well as advanced understanding of PSA technology, overall management of NPP severe accidents could be – and also should be - analysed as an integrated complex process. The interrelationship of NPP emergency operating procedures, safety and risk assessments, severe accident management guidelines, and emergency off-site actions should be planned and organized to minimize the consequences of such accidents. This approach might be a contribution to ensure the continued safety of NPPs and to improve effectiveness of regulatory practices in EU Member States.

[2] http://europa.eu.int/comm/environment/seveso/index.htm

3. OBJECTIVES OF THE PROJECT

The current approach to NPP emergency planning is, in general, traditionally deterministic, when usually a reference accident is defined to be used as a basis for drawing up corresponding emergency plans.

On the other hand, Level 2 (L2) and level 3 (L3) of the current, advanced PSA technology and their results can, in principle, be used to estimate the offsite consequences of beyond design basis accidents (BDBA) and severe accidents of NPP. They could provide an acceptable basis for implementation of risk informed support in decision making (RIDM) processes, related to NPP emergency planning measures, especially to defining emergency zones round NPPs.

This report provides a supplement to the report EUR 21580 EN [1], produced by JRC-IE in 2005 on this subject, and aimed at benchmarking and harmonising strategic planning practices for NPP emergency zoning. In EU Member States, the practical application of L2 PSA results for the emergency management has been limited and, in fact, little risk based information has been used. In the course of that project, only the Czech Republic, Slovakia and the UK informed about some cases where L2 PSA results were used in a way as an input to emergency arrangements. The UK is the only Member State of the EU, which has been carrying out research to consider how L2 PSA outcomes could be used in a systematic way for emergency planning purposes.

The **general objective** of this project is to contribute to the common, state-of the-art understanding of the methodologies and their applications in order to improve the confidence in them, and to help decrease unnecessary conservatism for more effective ensuring the continued safety of NPPs.

The **more detailed objectives** are:

- To provide a systematic overview and generic framework of the relevant aspects of risk informed decision making (RIDM) in NPP emergency zoning as a contribution to possible future harmonisation of strategic planning practices in this area, based on integration of essentials of PSA technology, severe accident phenomenology, and radiological protection.
- To document the current status together with some concrete and specific examples of NPP risk informed emergency zoning practice.
- To complement - in terms of probabilistic aspects - current JRC-IE activities on traditional deterministic safety / risk assessment of NPPs.

4. EMERGENCY PLANNING ZONES AND MEASURES

4.1 IAEA Reference Documents Basic Information

This section provides a brief summary of the International Atomic Energy Agency (IAEA) guidance on emergency planning zones, actions and intervention criteria. The information has been extracted from IAEA document EPR-METHOD 2003 [3] and IAEA Safety Standards Series GS-R-2 [4]. This provides an international perspective on the pertinent issues related to the emergency zoning (EZ) requirements.

The IAEA Requirements [4], jointly sponsored by Food and Agriculture Organization of the United Nations (FAO), IAEA, International Labour Organization (ILO), OECD Nuclear Energy Agency (NEA), Pan American Health Organization (PAHO), United Nations Office for the Co-ordination of Humanitarian Affairs (OCHA), and World Health Organization (WHO) are specified for five threat categories. For the purpose of this report, threat category I is relevant: it applies to facilities, such as Nuclear Power Plants (NPP), for which on-site events, including very low probability events are postulated that could give rise to severe deterministic effects[3] off the site, or for which such events have occurred in similar facilities. The on-site events involve an atmospheric or liquid release of radioactive material or external exposure that originates from a site location.

The IAEA document [4] establishes numerous requirements related to generic areas: on the site (on-site) and off the site (off-site). In addition, the document [4] establishes requirements for two off-site emergency zones: the precautionary action zone (**PAZ**) and urgent protective action planning zone (**UPZ**) [5]. Facilities in threat category I and II (facilities in threat category II are such as some type of research reactors) warrant extensive on and off-site emergency preparedness arrangements. In addition, threat category V is considered further, as it applies for activities not normally involving sources of ionizing radiation, but which yield products with a significant likelihood of becoming contaminated as a result of events at facilities in threat category I (or II), including such facilities in neighbouring countries.

On-site area
This is the area under control of the operator or the first responder[4]. It is surrounding the facility within the security perimeter, fence, which is under

[3] Doses in excess of those for which intervention is expected to be undertaken under any circumstances. Deterministic effect is a health effect of radiation for which generally a threshold level of dose exists above which the severity of the effect is greater for a higher dose. Such an effect is described as a 'severe deterministic effect' if it is fatal or life threatening or results in a permanent injury that reduces quality of life [3, 4, 5].
[4] The first members of an emergency service to respond at the scene of an emergency [3, 4, 5].

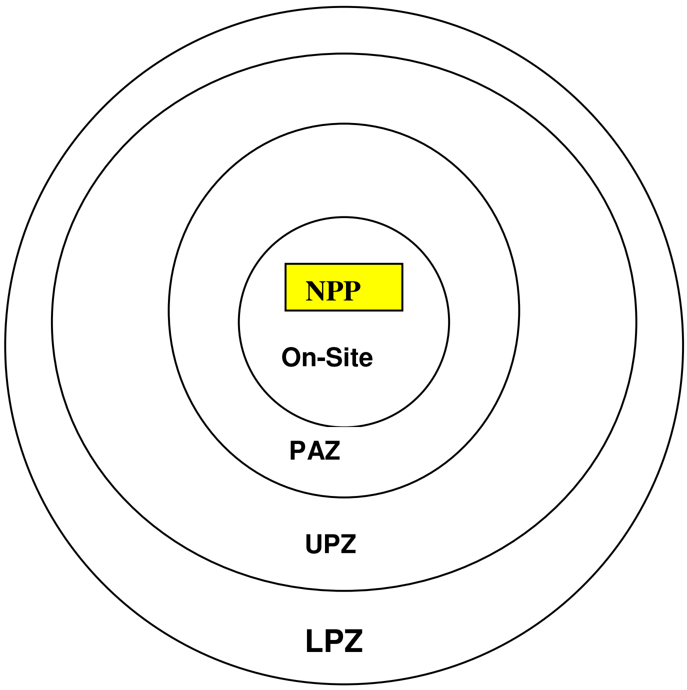
immediate control of the facility operator.

Off-site area
This is the area beyond that which is under the control of the facility operator or first responders.

The document [4] requires that for facilities in threat category I or II, arrangements shall be made for effectively making and implementing decisions on urgent protective actions to be taken off the site within:

(a) a PAZ, for facilities in threat category I, within which arrangements shall be made with the goal of taking precautionary urgent protective action, before a release of radioactive material occurs or shortly after a release of radioactive material begins, on the basis of conditions at the facility (such as the emergency classification) in order to reduce substantially the risk of severe deterministic effects.

(b) an UPZ, for facilities in threat category I or II, within which arrangements shall be made for urgent protective action to be taken promptly, in



On-Site: Internal zone, under control of NPP operator
PAZ: Precautionary Action Zone
UPZ: Urgent Protective action planning Zone
LPZ: Long-term Protective Zone (Food Restriction Planning Zone-**FRPZ**)

Fig. 4.1. NPP Emergency planning zones.

accordance either with international or national standards, in order to avert dose off the site.

The PAZ and UPZ should be roughly circular areas around the facility, their boundaries should be defined, where appropriate, by local landmarks (e.g. roads or rivers) to allow easy identification during a response as illustrated in Fig. 4.1. It is important to note that the zones should not stop at national borders. The size of the PAZ and the UPZ should be consistent with the guidance provided in Appendix II of [5].

In addition to PAZ and UPZ, there is also a Food Restriction Planning Zone (**FRPZ**), which is more often called Longer-term Protective action Zone (**LPZ**). This is an area around the facility where preparations for effective implementation of protective actions to reduce the long term dose, i.e. the risk of stochastic health effects[5] from deposition and ingestion of locally grown food, should be developed in advance. The longer term protective action zone will of course include the PAZ and the UPZ and extend to a further radius. On the bases of severe accident studies, the United States Nuclear Regulatory Commission (USNRC) for instance has adopted this zone of 80 km (50 miles), however, it might be much larger, up to a couple of hundreds of kilometres.

4.2 Emergency Planning Measures

Concerning urgent protective action, it is the “action in the event of an emergency which must be taken promptly (normally within hours) in order to be effective, and the effectiveness of which will be markedly reduced if it is delayed. The most commonly considered urgent protective actions in nuclear or radiological emergency are evacuation, decontamination of individuals, sheltering, respiratory protection, iodine prophylaxis and restriction of the consumption of potentially contaminated foodstuffs” [4]. The urgent protective actions are, effectively, the radiological exposure protective options, which represent the consequence mitigation part. The main options, i. e emergency planning (EP) actions for preventing and limiting exposures are generally known as following:

Evacuation. The best strategy for preventing serious exposures, if feasible, is to evacuate people from the area before the radioactive materials arrive.

Sheltering. Placing barriers between the radioactive materials and people is effective for some releases. The most commonly available and suitable barrier is a building, the walls and roof of which attenuate to some extent the gamma radiation. The heavier the construction, the more effective the shielding;

[5] A radiation induced health effect, the probability of occurrence of which is greater for a higher radiation dose and the severity of which (if it occurs) is independent of dose. Stochastic effects may be somatic effects or hereditary effects, and generally occur without a threshold level of dose. Examples include thyroid cancer and leukaemia [3, 4, 5]

basements are particularly advantageous locations.

Respiratory protection. Breathing through any of a variety of materials – facemasks, tissues, towels, or other cloth – offers significant protection against the inhalation of particles.

Relocation. If large amounts of radioactivity persist in the area, sheltering is not a sufficient protective measure, and people must be moved from the area until it is decontaminated.

Potassium iodide (KI) prophylaxis. Iodine uptake by the body can be blocked by the ingestion of stable iodine prior to, or immediately after, exposure. If taken properly, potassium iodide will help reduce the dose of radiation to the thyroid gland from radioactive iodine, and reduce the risk of thyroid cancer.

Decontamination of people. Apart from removing people from the vicinity of radioactivity or using barriers, it is, in some situations, desirable to remove radioactive materials from the immediate vicinity of people. Decontamination includes removing contaminated clothing and washing off external contamination.

Decontamination of land and buildings. This is not generally considered an emergency response; however, it is important to remember that the significant off-site economic costs of a major accident will be for attempted decontamination and for property that is unusable because it cannot be sufficiently decontaminated.

Protection of the food chain. Ingestion of contaminated food and water can account for nearly half of the aggregate population's exposure to radioactivity. Food-chain interventions are thus crucial to emergency response efforts directed toward delayed health effects.

Medical treatment. Finally, there is a need for medical efforts to alleviate consequences. Medical care entails screening and follow-up capabilities and the possibility of deploying a significant medical infrastructure.

The most recent document IAEA [5] from Feb. 2005 specifies more precisely the urgent protective actions and countermeasures should include the following:

- isolation of a contaminated area or radioactive source and prevention of inadvertent ingestion;
- evacuation;
- sheltering;
- respiratory protection and protection of skin and eyes;
- decontamination of individuals;
- stable iodine prophylaxis;

- protection of the food supply and prevention of the consumption of significantly contaminated foodstuffs and water;
- managing the medical response;
- protection of international trade[6].

It should be mentioned, that in some countries including the following ones mentioned further, term “emergency planning zone (EPZ)” is often used instead of the above term UPZ. The differences in basic terminology have historical reasons, since the traditional terms were usually based on USNRC NUREG documents, particularly essential NUREG-0654 Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants (1980) [6], which is still in force. Effective EP measures have to be based on a comprehensive threat assessment, which considers the most likely accidents, as well as also addressing less probable (including the least likely) but more severe (including the most severe) accidents. The level of EP measures has also to reflect the health risk, which takes into account both the possible consequences of accidents and their likelihood as well.

4.2.1 Emergency Planning Measures in Some Countries

EP measures can be defined as the measures, which enable individuals, bodies and authorities to organise rapid and effective emergency responses. Protective actions against nuclear emergencies cover measures to limit the exposure of the

Possible Exposure Rate	Prior to Release	Post-Start of Release	
	Emergency Phase		Post-Emergency Phase
	Early Phase	**Intermediate Phase**	**Late Phase Emergency Plans**
	Impossible Exposure	Possible High Exposure	Possible Low Exposure
	Precautionary Protective Measures: (Based on prediction) Evacuation (in PAZ) Sheltering (in UPZ) Iodine Prophylaxis	**Urgent Protective Measures:** (Based on environmental monitoring) Evacuation (beyond PAZ) Sheltering (in UPZ) Iodine Prophylaxis	**Longer Term Protective Measures and Recovery Operations:** (Based on environmental monitoring) Temporary Relocation (in LPZ) Permanent Resettlement (in LPZ) Foodstuff Restrictions (in LPZ)
	Hours	Hours / Days	Weeks / Months / Years
	Time After Start of Accident		

Fig. 4.2. Exposure rate against time and some EP essentials.

6 This item is not relevant to this report and is mentioned here only due to completeness

public to radioactive contamination through external exposure, inhalation, and ingestion. The objectives of these measures are to prevent deterministic effects, i.e. early mortality, and to reduce stochastic effects, e.g. thyroid cancer and leukaemia. Figure 4.2 plots exposure rate against time and covers some essentials on emergency planning. This provides an overview of the EP basic terminology and which protective actions against release of radioactive materials are appropriate at which stage of a severe accident[7] [10].

The following definitions of the emergency planning zones (EPZ) (roughly circular with specified radius) are given in some particular countries; this is updated information given in [1, 7]:

Belgium: The general EPZs are associated with the following protective actions: evacuation (10 km), sheltering (10 km), stable iodine intake (20 km) and food chain (whole country). The size of these zones has been defined taking into account a rough (presumably largely deterministic) estimation of the associated risks.

China: The zones as follows are applied for NPP Tianwan (PWR): internal zone 3-5 km, outer zone 7-10 km and ingestion exposure pathway zone 20 km [8].

Czech Republic: The predetermined evacuation of people is performed within 5 km internal zone around Temelín NPP and within 10 km internal zone around Dukovany NPP. The emergency planning zone is a territory of 20 km around Dukovany NPP and 13 km around Temelín NPP. The predetermined actions are sheltering and taking iodine tablets. The difference between the EPZ for Temelín NPP and for Dukovany NPP is due to different population densities, meteorological and evacuation conditions.

Finland: Rescue service plan (by rescue service authorities) for emergency preparedness zone (20 km); advance iodine pellets and quick actions (sheltering, evacuation) for 5 km zone.

France: Around each NPP there are two zones defined. The emergency planning zone of 5 km radius around a nuclear power plant is the zone where evacuation is pre-planned and prepared in detail. The emergency planning zone of 10 km radius around a NPP is the zone where sheltering is pre-planned. Stable iodine tablets have been previously distributed in France to the population within a radius of 10 km around a NPP. The emergency planning zones of 5 km and 10 km radii around a NPP provide reasonable assurance that the doses to the population in the short term would be below the different intervention levels[8] for a spectrum of accidents and radionuclide releases, in particular for most core melt accidents. Another important consideration is that 5 and 10 km are

---

[7] An accident that involves extensive core damage and fission product release into the reactor vessel and containment, with potential release to the environment [28].
[8] The level of avertable dose at which a specific protective action is taken in an emergency or a situation of chronic exposure [3, 4, 5].

practicable distances for planning in France. It is also recognized that protective actions could be extended beyond 10 km if conditions warrant. Much more time would be available for emergency response beyond these distances. Concerning intervention levels, sheltering is recommended when the projected effective dose exceeds 10 mSv, whereas evacuation is recommended when this dose exceeds 50 mSv. The intake of stable iodine is recommended when the thyroid committed equivalent dose by inhalation exceeds 100 mSv for most sensitive population [9].

Hungary: There are three planning zones: the smallest in radius of 3 km is the "precautionary protective action-planning zone" in which the measures are introduced without delay. This zone is surrounded by 30 km circle within which the "urgent protective action planning zone" can be found; and then the largest zone of 80 km is located. That is the "long term protective action planning zone". Concerning the latter two zones, specific laws determine the intervention levels.

Japan: The EPZ is about 8 to 10 km for the facilities of commercial plants and research reactors with power levels greater than 50 MWt. The standard of EPZ is the zone whose boundary (distance from the nuclear facilities) is defined so as to keep less than the lower limit of radiation exposure at the boundary, 10 mSv to whole body dose and 100 mSv to thyroid with sufficient margins supposing hypothetical accidents that cannot happen technically. Outside this range, there is no necessity of emergency actions such as sheltering and evacuation.

The Netherlands: The various zones for direct measures are defined geographically as follows: 1) Evacuation zone circle with a radius of 5 km, 2) Iodine prophylaxis circle with a radius of 10 km, 3) Sheltering zone: circle with a radius of 20 km. The measures in cases of nuclear emergencies are coordinated at the national level.

Slovakia: The EPZ is defined in relation to the maximum size of any radiation emergency that can be reasonably foreseen. The hazard area represents a circle with the centre in the nuclear facility and radius 30 km for Bohunice site, and 20 km for Mochovce site. In case that the boundary demarcating the hazard area interferes with an inhabited area, the whole inhabited area is considered as a hazard area. The difference in the EPZ for Bohunice NPP and Mochovce NPP is due to different population density, meteorology and evacuation conditions.

South Africa: In an effort to develop effective emergency plans, the utility (Koeberg Nuclear Power Station, Eskom Generation) implemented a programme to derive the risk to the public and to use these risk insights to aid the optimisation of the emergency planning actions, zones and response times. The emergency plans are based on a comprehensive threat assessment which takes into account the most likely accidents while also addressing the less probable but more severe events. The level of emergency preparedness also reflects the health risk, which considers both possible consequences of accidents and their likelihood. International guidance as IAEA documents [3, 4] were followed in developing the emergency response requirements [10]. Internal zone of 5 km and outer zone of 16 km have been established on the basis of severe accidents reference scenarios analyses.

Spain: The definition is included in the Basic Nuclear Emergency Plan and it is common to all NPPs. These zones are predefined in function of the distance at the nuclear site (concentric zones) and of the wind direction (sector zones). The required different actions depend on each zone and the emergency situation. This is related to the emergency category, established in the Internal Emergency Plan and according to the Final Safety Assessment Report.

Switzerland: Current zoning around NPP consists of an inner zone of 3-5 km, where a dose > 1 Sv is possible, and outer zone of 20 km where there is no acute threatening of life. Since 1998 there are legal provisions in force, requiring protection of the public for expected doses > 1mSv. Selection of scenarios for emergency planning have been based on both deterministic as well as probabilistic approach [11].

UK: For each nuclear licensed site in the UK there is a defined zone round the site – the Detailed Emergency Planning Zone (DEPZ) within which the arrangements to protect the public are planned in detail. The boundary of this zone is defined in relation to the maximum size of any radiation emergency that can be reasonably foreseen and ranges from 1 to 5 km. It is also recognised that radiation emergencies could occur that would have consequences beyond the DEPZ. The nature of the response required is more difficult to predict and will depend on a number of factors such as the characteristics of the release that has occurred and the prevailing weather conditions. To deal with this, there is a requirement that the emergency plans incorporate arrangements for "extendibility" beyond the DEPZ.

USA: To facilitate a preplanned strategy for protective actions during an emergency, there are two EPZs around each NPP. First, the plume exposure pathway EPZ has a radius of about 10 miles (16 km) from the reactor. Predetermined protection actions include sheltering, evacuation, and the use of potassium iodide where appropriate. Second, the ingestion exposure pathway EPZ. It has a radius of about 50 miles (80 km) from the reactor. Predetermined protection actions include a ban of contaminated food and water.

4.3 Suggested Emergency Planning Zones and Radius Sizes

For threat category I, i. e. for NPPs, IAEA document [3] in its Appendix 5 provides suggestions for the approximate radius of the EP zones and food restriction planning radius as given in the following Table 4.1. The radii were selected based on calculations performed using RASCAL 3.0 computer code[9]

9 RASCAL 3.0, Description of Model and Methods, NUREG-1741, USNRC, Washington DC, 2001.

[12]. The calculations assumed average meteorological conditions, no rain, ground level release; 48 hours of exposure to ground shine, and calculates the centralized dose to a person outside for 48 hours. The suggested sizes for the PAZ were based on expert judgment considering the following:

Facilities	PAZ radius	UPZ radius	FRPZ radius
Reactors > 1000MW (th)	3 – 5 km	25 km	300 km
Reactors > 100 - 1000MW (th)	0.5 -3 km	5 – 25 km	50 - 300 km

Table 4.1. Suggested Emergency Zones and Radius Sizes for NPPs.

(1) Urgent protective actions taken before or shortly after a release within this radius will prevent doses above the early death thresholds for the vast majority of severe emergencies postulated for these facilities.

(2) Urgent protective actions taken before or shortly after a release within this radius will avert doses above the urgent protective action generic intervention level[10] (GIL) for the majority of emergencies postulated for the facility.

(3) Dose rates that could have been fatal within a few hours were observed at these distances during the Chernobyl accident.

(4) The maximum reasonable radius for the PAZ is assumed to be 5 km because:

a) except for the most severe emergencies, it is the limit to which early deaths are postulated [13];

b) it provides about a factor of ten reduction in dose compared to the dose on the site;

c) it is very unlikely that urgent protective actions will be warranted at a significant distance beyond this radial distance;

d) it is considered the practical limit of the distance to which substantial sheltering or evacuation can be promptly implemented before or shortly after a release; and

e) implementing precautionary urgent protective actions to a larger radius may reduce the effectiveness of the action for the people near the site,

[10] The level of avertable dose at which a specific protective action is taken in an emergency or situation of chronic exposure

who are at the greatest risk.

The suggested sizes for the UPZ are also based on expert judgment considering the following:

(1) These are the radial distances to which the reference NUREG-1150 [13] suggests that monitoring to locate and evacuate hot spots (deposition) within hours/days may be warranted in order to significantly reduce the risk of early deaths for the worst emergencies postulated for power reactors.

(2) At these radial distances there is a factor of approximately 10 reduction in concentration (and thus risk) from a release compared to the concentration at the PAZ boundary.

(3) This distance provides a substantial base for expansion of response efforts.

(4) 25 km is assumed to be the practical limit for the radial distance within which to conduct monitoring and implement appropriate urgent protective actions within a few hours or days. Attempting to conduct initial monitoring to a larger radius may reduce the effectiveness of the protective actions for the people near the site, who are at the greatest risk.

(5) For average meteorological (dilution) conditions, beyond this radius, for most postulated severe emergencies, the total effective dose for an individual would not exceed the urgent protective action GILs for evacuation.

As far as long term protective zone FRPZ is concerned, in general, protective actions such as relocation, food restriction and agricultural countermeasures are based on expert judgement considering the following:

(1) Detectable excess stochastic effects (cancers) are very unlikely beyond this distance.

(2) Detailed planning within this distance provides a substantial basis for expansion of response efforts.

(3) Food restrictions were warranted to about 300 km following the Chernobyl accident in order to prevent detectable excess thyroid cancers among children [3].

It should be mentioned here so called Environmental Impact Assessment (EIA), that is an assessment of the likely influence a project may have on the environment. EIA is a procedure that ensures that the environmental implications of decisions are taken into account before the decisions are made and can be very briefly mentioned as the process of identifying, predicting, evaluating and mitigating the biophysical, social, and other relevant effects of development proposals prior to major decisions being taken and commitments made. The purpose of the assessment is to ensure that decision-makers consider environmental impacts before deciding whether to proceed with new projects.

The EIA directive[11] (Directive 2001/42/EC) was first introduced in 1985 and was amended in 1997 and 2003. However, it has little practical relevance to the issue, as there is no background technical guideline or similar on how to evaluate zones in an EU-wide manner. Nevertheless, the issue of zoning is more and more mentioned in some current EIA studies for NPPs under operation, e.g. Temelin in Czech Republic, Mochovce in Slovakia, or under construction (Belene in Bulgaria).

[11] It requires EU Member States to achieve a particular result without dictating the means of achieving that result.

5. SOME WAYS OF DEFINING NPP EZ

5.1 Background considerations

The main characteristics of NPP accidents are some general features of accidents, i. e. Design Basis Accidents[12] (DBA), Beyond Design Basis Accidents[13] (BDBA), severe accidents (SA), fission product characteristics, meteorological considerations, exposure pathways, adverse health effects, and avoiding adverse health effects. Overall considerations of them then result in essential concerns: (1) the sources of radiation from postulated accidents, (2) the potential pathways of radiation to the environment, and (3) the possible health effects of exposure to such accidental releases [14].

Frequency of occurrence [1/reactor year]	Terminology used	Acceptance criteria
$10^{-2} – 1$ Expected in the life of the plant	Anticipated Operational Occurrence	No additional fuel damage
$10^{-4} – 10^{-2}$ Chance greater than 1% over the life of the plant	DBA	No radiological impact at all or no radiological impact outside exclusion area
$10^{-6} – 10^{-4}$ Chance lower than 1% over the life of the plant	BDBA	Radiological consequence outside exclusion area within limits
< 10^{-6} ***Very unlikely to occur***	***Severe accidents***	***Emergency response needed***

Table 5.1. Grouping of NPP event occurrences.

Grouping of NPP events including accidents by frequency of their occurrence differs in different countries. One of the possible subdivisions is given in Table

[12] DBA: plant design covers them and plant engineered safety features will cope with them.
[13] BDBA: plant design does not cover them and plant engineered safety features would not cope with them.

5.1 [15, 16]. The probabilistic values given in the table are illustrative. They are to be considered more qualitatively than quantitatively [16].

Usually, there is a close interrelation between probability of occurrence and acceptance criteria of the results of safety analysis. BDBAs and severe accidents are typically treated separately in accident analysis, although some initiating events are the same. The results help to determine measures to prevent severe accidents and to mitigate the radiological consequences. Accident management and emergency response measures are necessary if all the barriers against radioactive releases are significantly degraded in a BDBA. For severe accidents, containment and/or confinement typically remains as the only barrier to limit accidental releases. The measures to restore and maintain the safety functions under such conditions include the use of:

- Alternative, or diverse systems, procedures and methods (e.g. in-vessel melt retention), including the use of non-safety-grade equipment;
- External equipment for temporary replacement of a standard component;
- Off-site emergency measures (limitations on food consumption, sheltering and evacuation).

The internationally accepted common approach to radiological consequences of nuclear accidents is that the recommendations given in International Commission on Radiological Protection (ICRP) Publication No. 63 "Principles for Intervention for Protection of the Public in a Radiological Emergency" [17] and the IAEA Safety Series No. 115 "International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources" [18] can be regarded as widely agreed references concerning the initiation of protective actions. The ICRP recommendations provide relatively high intervention levels for the "nearly always justified" protective actions (e.g. 50 mSv for sheltering, 500 mSv for evacuation), while it provides a rather wide range of values for the so-called "optimised values" (50-500 mSv for evacuation). The above cited IAEA document [18] provides relatively low "generic optimised intervention levels" (for example 10 mSv of avertable dose for sheltering for a period of no more than 2 days, 50 mSv of avertable dose for evacuation for a period of no more than 1 week). These levels may be lowered or increased based on local conditions as, for example, population density, adequate transportation, weather conditions, etc. It seems to be consensus that new NPPs should be designed in a way taking into account the ICRP document [17] as well as other relevant documents, such as the IAEA standard document [18] and, besides, of course the well known As Low As Reasonably Achievable (ALARA) approach.

In general, there are three basic ways of defining NPP EPZs, based on probabilistic criteria in sense of probabilistic targets, or use of NPP PSA results. The essential basis for NPP EPZ considerations is a source term - the fractions defining the portion of the radionuclide inventory in the nuclear reactor at the start of an accident that is released to the environment. The initial elevation, i.e. height, energy, and timing of the release are also included in the source term. More generally, a source term is a specific type of release characteristic of a reactor family and a representative of a type of accident, i.e. in general, a mode of containment failure following complete core meltdown [19]. It is taken into consideration to define appropriate corrective actions for the protection of populations under these extreme emergency conditions.

Basically, there are three source terms as follows in decreasing order of seriousness and covering, by definition, a certain number of possible scenarios:

- Source term ST1 corresponds to early containment failure a few hours after onset of the accident;
- Source term ST2 corresponds to direct release to the atmosphere following loss of containment integrity one or several days after accident initiation;
- Source term ST3 corresponds to indirect, delayed release to the atmosphere, through paths enabling a certain amount of fission products to be retained.

The Reactor Safety Study (RSS) under the references WASH -1400 and NUREG 75-014 as the first example of probabilistic risk assessment (PRA), or probabilistic safety assessment (PSA), giving figures for the probable accident impact on the population, is still the basis of NPP severe accident studies. The containment failure mode classification as in WASH 1400 is still used and comprises six main modes:

- α: steam explosion in the reactor pressure vessel (RPV) or reactor pit, including loss of containment integrity in the short term;
- β: initial, or fast-induced lack of integrity;
- γ: hydrogen explosion;
- δ: slow overpressurization;
- ε: basemat melt-through by the corium;
- V: bypass the containment using outgoing pipes (this mode does not directly concern the behaviour of the containment).

Modes α, β, and γ without prevention and mitigation provisions could lead to ST1 type release, mode δ could lead to ST2 type release and mode ε could lead to ST3 type release [19].

The publication of the WASH - 1400 as a first major PSA technology application, subsequent conducted NPP PSA studies in the USA and later on definitive, comprehensive “compendium” NUREG 1150 [13] reference document on guidance for all PSA users had a tremendous impact on the thinking of nuclear safety experts. Two major insights from WASH-1400 were [20]:

- Prior thinking was that (no quantified) frequency of severe core damage was extremely low and the consequences of such damage would be catastrophic. The WASH 1400 calculated a core damage frequency (CDF) in the order of 10^{-4} to 10^{-5} per reactor-year, a much higher number than anticipated, and showed that the consequences would not always be catastrophic.

- A significant failure path for radioactivity release that bypasses the containment building was identified. Traditional safety analysis methods had failed to do so.

However, unlike the RSS, most core melts are not expected to result in large off-site consequences. The small fraction of accidents that might lead to large off-site consequences generally involve either an early failure of the containment in relation to the time of core melt, or, a containment bypass. For other containment response modes, the retention properties of the containment are substantial. Analyses have shown that both natural and engineered retention mechanisms can significantly reduce the inventory of radionuclides available for release if enough time is available for those mechanisms to act. Therefore, source terms are strongly affected by whether or not the containment fails, and, if it fails, by the time and mode of the failure. The following global insights about off-site consequences have also been identified [21]:

- Estimated risks of the early fatalities and injuries are very sensitive to source term magnitudes, the timing of releases and assumptions about the effectiveness of emergency plans;

- Estimates of early health effects differ greatly from one site to another, but site to site differences are substantially less for latent cancers;

- Airborne pathways are much more important than liquid pathways.

5.2 The Use of Reference Source Term

The establishment of emergency planning zones requires the definition of technical basis, in particular the choice of reference source terms. They are defined based on a deterministic approach. The use of reference source term and the follow-up framework for defining NPP EPZs is practised in e. g. France [9], to some extent also in the Netherlands.

The emergency plans must be able to respond effectively to accidents liable to occur at a NPP. This implies the definition of technical bases, i.e. the adoption of one or more accident scenarios encompassing the possible consequences, with a view to determining the nature and extent of the remedial means required. This task is difficult, since cases of real significant accidents are extremely rare, with the result being that a conservative theoretical approach is usually adopted to estimate the source terms (i.e. the quantities of radioactive materials released), calculate dispersion in the environment and finally assess the radiological impact.

The possibility of severe accidents with reactor core meltdown, as the result of the combination of more or less complex failures cannot be fully excluded and it is the duty of the administration to prepare off-site emergency plans, specific to each nuclear installation, for the protection of the neighbouring population under such very unlikely, but serious circumstances. Since the Three Mile Island (TMI) and Chernobyl accidents numerous studies have been conducted to have a better knowledge of the risks and consequences of severe accidents on a light water reactor (LWR). Determining the timing, quantity and composition of the radionuclide mixture that might be released from the reactor core and from the containment is generally considered the most difficult aspect of determining potential offsite consequences of the accident. It is clear that the potential releases are strongly dependent on accident conditions and that those conditions cannot be delineated accurately in advance.

The reference source terms in France are defined according to a deterministic approach [9]. Concerning the consequences of severe BDBAs, they can be assessed according to the behaviour of the containment after reactor core meltdown. The containment is one of the most important barriers in protecting the public. There have been increasing efforts for almost 40 years to determine NPP severe accident risks on a plant-specific basis. The first comprehensive plant-specific examination of risk was the above mentioned RSS WASH-1400. The risk values calculated in RSS were later updated in NUREG-0773 The Development of Severe Reactor Accident Source Terms: 1957-1981 and used in USNRC final environmental impact statement, published after 1980, as well as in the Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants (NUREG-1437) [14]. Later, more complex and more intensive plant-specific risk studies were developed, both by USNRC and the industry. The most important USNRC studies of severe accident consequences are found in the NUREG-1150 analyses [13], as already mentioned.

On the basis of the adaptation of the above documents to the French NPP units, three general classes of severe accidents with core meltdown were distinguished as a basis for designing the French severe accident policy and making operational decisions [9]:

- Accidents resulting in “early” failure of the containment, represented by source term ST1.

- Accidents resulting in “delayed” failure of the containment, at least 24 hours after the beginning of the accident, without filtration of the corresponding releases. These accidents are represented by source term ST2.

- Accidents resulting in “delayed” failure of the containment, at least 24 hours after the beginning of the accident, with releases in a way ensuring some filtration. These accidents are represented by source term ST3.

The three basic classes of source terms are summarized in Table 5.2 as percentages of the radioactive inventory released from the reactor core.

In all cases, the consequences of the releases are dominated in the short term by iodine and in the long term by caesium. Furthermore, there is, in orders of magnitude, a factor of roughly 10 between source terms ST1 and ST2 and a factor of 10 between source terms ST2 and ST3. As a comparison, the Chernobyl accident releases, which amounted of 20 to 50 % of iodines and caesiums, are close to the ST1 source term.

Source term	ST1	ST2	ST3
Noble gases	80	75	75
Organic iodine	0.6	0.55	0.55
Inorganic iodine	60	2.7	0.3
Caesium	40	5.5	0.35
Strontium	5	0.6	0.04

Table 5.2. Basic classes of source term [%].

The source term ST1, resulting from a total and “early” failure of the containment, could result from phenomena like steam explosion or hydrogen detonation. It is considered that such failure of the containment can be excluded due to the characteristics of the large drywell containment used in France. The accidents corresponding to source term ST1 are considered as improbable.

Other improvements have been brought to French nuclear power units with a view to reducing ST2 type releases to ST3 type releases as in, for example, the implementation of an “ultimate” procedure to improve the containment function, including the possibility of releases through a sand bed filter completed by a metallic pre-filter [9].

These evaluations explain why source term ST3 was finally adopted as the “maximum conceivable release” for French nuclear power units in operation or under construction. It must be clearly understood that the source term ST3 does not correspond to a particular scenario but is a reasonable envelope of the releases of various scenarios. It covers a set of possible scenarios and is not related to a precise accident scenario [9].

The source term ST3 is used as a technical basis for emergency plans for protection of the civil population, for determining the response of the utility and the public authorities concerned. The source term consequence assessment has a direct impact on the establishment of EP zones, protective actions, resources, procedures and the requirements for a preparedness organization. Emergency plans in France are designed to cope, as far as possible, with the consequences of a ST3 type release.

The radiological consequences of a ST3 type release and their evolution in time were assessed for both the adult and the one-year-old child, in the vicinity of the NPP (from 1 to 20 kilometres). For calculation of the atmospheric dispersion and deposition of the radioactive substances, the Doury atmospheric diffusion model[14] was used, with the hypothesis of a 5 $m.s^{-1}$ wind velocity, normal diffusion conditions and absence of rainfall. These atmospheric conditions are average conditions for the French nuclear power plants.

Nuclide groups	Noble gases	Iodine	Caesium / Strontium	Tellurium
Total activity released (Bq)	~ 5×10^{18}	~ 8×10^{16}	~ 2.5×10^{15}	~ 1.5×10^{16}

Table 5.3. Total activity released within 48 hours for 900 MW(e) PWR NPP.

Table 5.3 presents the total activity expected to be released for the source term ST3 for a 900 MW(e) PWR unit used in France [9].

The results of the ST3 source term consequences calculations were compared to the intervention levels recommended in France by the Ministry of Health, to determine to which distance the respective countermeasures should be prepared. Respecting the recommended intervention levels implies the possibility of evacuating the population within a radius of 5 kilometres and of confining the population indoors within a radius of 10 kilometres around the NPP within less than 24 hours, which is in accordance with what is planned in emergency plans.

The doses assessment also showed that evacuation of the population within a

[14] E.g. Thielen, H., Martens, R., Schnadt, H., Maßmeyer, K.: Further development of the French-German dispersion model - SODAR pre-processor. International Journal of Environment and Pollution (IJEP), Vol. 14, No. 1/2/3/4/5/6, 2000.

radius of 2 kilometres around the NPP must not be delayed in the case of a ST3 type release, which reinforced the interest of the creation of a respond action stage in off-site emergency plans.

The thyroid committed equivalent dose is the most restrictive equivalent dose. The ST3 consequences could result in the decision, to be taken within 12 to 24 hours, to organise absorption of stable iodine within a radius of 10 kilometres and in the decision to be taken before 48 hours, to organise absorption of stable iodine within a radius of less than 20 kilometres from the plant. This result reinforced the decision taken in France to create stable iodine stocks in each particular region.

Despite a number of uncertainties in dose calculations (uncertainties on behaviour of the aerosols in the containment atmosphere, on atmospheric transport, on meteorological conditions…), the assessment of the short term consequences of a ST3 type release has allowed to check that the prepared provisions in emergency plans should be able to respond to accidents liable to occur at a NPP, even to low probability core-melt accidents [9].

The second basic way of defining NPP EPZ is the use of European Utility Requirements Group original method for defining criteria for limited impact in case of severe accident.

5.3 The Use of EUR Group Severe Accident Limited Impact Approach

The European Utility Requirements (EUR) Group was created in the early nineties by a small group of European Utilities participating in the US Advanced Light Water Reactor Program (ALWR). The objective of the group was to build upon the experience gained in the ALWR program to issue a set of requirements, in particular safety relevant ones, common to all members, in order to facilitate interaction with national Safety Authorities and allow standardized designs to be built in all participating countries. Safety relevant requirements, though reflecting a common analysis and understanding of some safety issues, were never contemplated as substitutes to national Safety Authority requirements [22].

One of the prerequisites identified for reaching this objective was to increase public confidence in nuclear. Excellent performance of current NPP fleets in all EU Member States was deemed a key element for increased support by the public, and best European practice in the field of design, operation and maintenance were adequately reflected in some EUR documents.

However, in particular in the wake of the Chernobyl accident, it was felt that demonstrating that the design of future plants to be built in Europe would more explicitly address issues related to severe accidents and their consequences than currently operating plants was likely to further increase acceptance of new nuclear units in Europe. Criteria for limited impact in case of severe accidents were defined, and special attention was devoted to issuing requirements related to containment capability for all situations contemplated for plant design.

Further to these requirements, a methodology was developed to demonstrate compliance with the above mentioned objectives. This methodology was translated in very simple requirements allowing both vendors and utilities to assess whether the containment system, as designed, had adequate capability. From a utility standpoint, this approach also had the advantage of providing significant flexibility in discussions with Safety Authorities on adequate public protection, as neither a specific site or meteorology, nor a specific reactor output, are addressed at the requirement level [22].

This approach, or philosophy concerns also one aspect of the conclusions of the EC DG JRC-IE/OECD NEA International Seminar on Emergency & Risk Zoning around NPPs, held in 2005 in Petten, Netherlands. One of the conclusions of the seminar was, that further considerations need to be given on how emergency planning and the EP zones would be defined for **future NPPs,** where the risk from the plant in terms of large off-site releases of radioactivity would be very much lower than for the current plants. This needs to be reconciled with the expectations of the Regulatory Authorities and the public. Consideration needs to be given on whether the moral obligation to provide an emergency plan would outweigh the technical conclusion that this would not be required. The trend is, of course, to improve the level of safety for future NPPs. This would significantly reduce the potential for severe accidents and releases of radioactive material from the plant to occur. In principle, this could be considered to significantly reduce, or perhaps eliminate, the need for emergency planning [23, 24].

Four goals have been defined to substantiate the notion of limited impact in case of severe accidents [22]:

1. No emergency protection action is needed beyond the site boundary, i.e. beyond 800m from the reactor. This means that the averted Effective Committed Dose (ECD)[15] over a period of one week (7 days) following accident initiation, will remain below 50 mSv, which is the generic intervention value reported in ICRP No. 63 [17] and which was adopted in the IAEA Basic Safety Standards No. 115 [18]. Practically speaking, evacuation and sheltering of people are not needed for such low values.

2. No delayed action is needed beyond 3 km from the reactor. This means that the averted ECD over a period of thirty consecutive days following

[15] The ECD for each internally deposited radionuclide is calculated by summing the products of the committed equivalent doses and appropriate tissue weighting factor values for all tissues irradiated. The committed equivalent dose or dose equivalent is the time integral of the equivalent dose rate in a specific tissue following intake of a radionuclide into the body. Dose rate is a quotient of dose and time, often indicated as mSv/h.

release termination remains below 30 mSv. This assures that the temporary relocation intervention level reported in the IAEA Basic Safety Standards No. 115 is not reached.

3. No long term action is required beyond site boundary. This means that the averted ECD over a period of fifty years following release termination remains below 100 mSv. Though this limit is lower than that recommended in the IAEA Basic Safety Standards No. 115 [18], it is considered consistent with the two above mentioned criteria by the EUR group.

4. Limited economical impact linked to the restriction of foodstuff consumption. This means allowing free trading of foodstuffs, provided a 5 mSv dose to individuals eating contaminated food for one year is not reached:

   - ✓ after one month following the end of the accident over a 30 km^2 area;
   - ✓ after one year following the end of the accident over a 10 km^2 area.

More clearly, this means that foodstuffs produced in areas of 30 km^2 and 10 km^2 surrounding the site could be marketed after one month and one year respectively.

As far as probabilistic criteria in sense of **probabilistic targets** are concerned, ambitious goals, some of them well in line with the international mainstream, e. g. INSAG 3 [25] are set as follows [22]:

- The cumulated frequency of sequences leading to a core melt (CDF) has to be kept below 10^{-5} per reactor year for all plant states, considering internal as well as external events.
- The cumulated frequency of sequences leading to unacceptable releases (i.e. in excess of maximum allowable releases for design extension conditions) has to be kept below 10^{-6} per reactor year.
- The cumulated frequency of sequences leading to early containment failure or to very large releases – large early release frequency - (LERF) has to be kept below 10^{-7} per reactor year.

These requirements are meant to reduce challenges to containment integrity through decreasing the probability of occurrence of potentially damaging sequences.

The same paper [22] refers to the **deterministic requirements**, which have to be met to increase confidence in containment coping capability for severe accidents:

- Probabilistic targets for early containment failure have to be met without reliance on overpressure protection equipment.

- Early containment failure has to be prevented by design (i. e. design provisions having adequate reliability for decreasing the probability of early containment failure to an acceptably low value must be implemented).

- At least one severe accident sequence must be addressed in the design of the containment, regardless of PSA results.

- It must be demonstrated that catastrophic failure of the reactor vessel bottom head cannot occur as a result of debris-water interaction.

- Direct containment heating must be prevented by design through:

    ✓ Providing a reliable depressurization system;

    ✓ Designing the reactor cavity such that debris transfer to the containment atmosphere is minimized in case of vessel failure; additionally, demonstrating absence of debris dissemination to the containment atmosphere if vessel failure occurs with Reactor Coolant System (RCS) pressure below 20 bars.

- The consequences of hydrogen generation must be accommodated through:

    ✓ Providing means for hydrogen control inside containment, which promote natural circulation and mixing of the containment atmosphere;

    ✓ Demonstrating that the probability of a global hydrogen detonation capable of endangering primary containment performance is sufficiently low to meet the probabilistic target for early containment failure;

    ✓ Enhancing hydrogen dilution in the containment atmosphere;

    ✓ Demonstrating that hydrogen concentration inside containment will remain below 10% (dry conditions), assuming 100% of the zirconium in the active part of the core has reacted with water, and taking credit for hydrogen mitigating devices;

    ✓ Demonstrating that the primary containment can withstand the loads resulting from slow hydrogen recombination cumulated with the global deflagration of the 10% average concentration (adiabatic, isochoric, complete combustion), and assessing that containment leak tightness can be maintained under such conditions;

  - ✓ Organising containment layout such that formation of local hydrogen pockets is prevented and the risk of transition from deflagration to detonation is minimized.

- The consequences of the presence of other noncondensables must be accommodated through:

  - ✓ Evaluating the quantity generated in case of severe accidents and verifying that the containment system can withstand the subsequent loads;

  - ✓ Providing a small, manually actuated, filtered purge capability to release noncondensables in a controlled manner.

- The containment system must be designed to guarantee that decay heat can be removed through:

  - ✓ Demonstrating that corium debris will relocate in a coolable geometry (in-vessel or ex-vessel);

  - ✓ Providing a containment heat removal system designed to operate in case of conditions beyond the design basis;

  - ✓ Providing a system for timely flooding the reactor cavity when needed for ex-vessel debris retention.

- Devices contemplated for dealing with the consequences of severe accidents must have the capability of carrying out their functions under harsh environments expected for design extension conditions.

The approach described above [22] can be applied also for future NPPs with advanced reactor concepts (ARC), which are briefly mentioned further. There is general international consensus that for the various nuclear ARCs, which are currently under development, design and operational safety goals as well as modern review frameworks for risk assessment required for licensing related decision-making shall be developed.

The evolution of NPPs over the last approx 50 years is usually subdivided into four Generations [26]. The first prototype NPPs for (commercial) electricity production are classified as Generation I (e. g Shippingport, Dresden, Magnox), the currently operating ones as Generation II (LWR – PWR, BWR, WWER, graphite moderated RBMK). ARCs classified in Generation III are evolutionary improved versions of NPPs currently in operation and are in different stages of development and construction. These ARCs have typically

- A standardised design for each type to expedite licensing, reduced capital cost and reduced construction time;

- A simpler and more robust design, making them easier to operate and less vulnerable to operational disturbances;
- Higher availability and longer operating lifetime – typically 60 years;
- *Reduced possibility of core melt;*
- *Improved mitigation of severe accidents and minimal effect on the environment;*
- Higher burn-up to reduce fuel use and the amount of waste, burnable absorbers to extend fuel life.

The following ARCs are typical and most relevant:

- EPR (European Pressurized Reactor), Germany/France;
- System 80+ (PWR), USA;
- ABWR (Advanced Boiling Water Reactor), USA/Japan.

Evolutionary improved concepts – nowadays in the design phase – are classified as Generation III+. Typical and most relevant concepts are as follows:

- SWR-1000 (BWR), Germany/France;
- IRIS (International Reactor Innovative and Secure), PWR, IRIS Consortium, which is an international team of companies, laboratories and universities, coordinated by Westinghouse;
- AP600/AP1000 (PWR), USA;
- PIUS (Process Inherent Ultimate Safe), PWR, Sweden;
- WWER-640/1000 (PWR), Russia.

Finally, there are Generation IV concepts, or Generation IV nuclear energy systems comprising the nuclear reactor and its energy conversion systems, as well as the necessary facilities for the entire fuel cycle, from ore extraction to final waste disposal. The following six systems, listed alphabetically, were selected to Generation IV by the Generation IV International Forum (GIF)[16]:

- Gas-Cooled Fast Reactor System (GFR);
- Lead-Cooled Fast Reactor System (LFR);

[16] http://www.nei.org/filefolder/doe_gen_iv_diagrams.pdf

- Molten Sault Reactor System (MSR);
- Sodium-Cooled Fast Reactor System (SFR)
- Supercritical Water-Cooled Reactor System (SCWR);
- Very-High-Temperature Reactor System (VHTR).

The above probabilistic criteria in the sense of probabilistic safety targets, i.e. CDF < 10^{-5} per reactor year, LERF (100 TBq Cs-137) < 10^{-7} per reactor year have also been applied for the new OL3 unit EPR 1600 NPP in Finland. The Finnish Government has in February 2005 granted a construction licence for the Generation III concept 1600 MWe EPR to the utility TVO at the Olkiluoto site [27].

5.4 PSA Based Source Terms to Support EP Zones

Probabilistic safety assessment (PSA[17]), or synonymous with probabilistic risk assessment (PRA[18]) of a NPP provides a comprehensive, structured approach to identifying accident scenarios and deriving numerical estimates of the risk to members of the public from the operation of the plant. For the purpose of this report the term PSA is further used. Risk is viewed as the likelihood of specified undesired events occurring within a specified period or in specified circumstances arising from the realisation of a specified hazard (a physical situation with a potential for human injury, damage to property, damage to the environment or some combination of these) [21]. The risk may be expressed as either frequency (the expected number of specified events occurring in time unit, or a probability (the probability of a specified event following a prior event), depending on the circumstances. The risk in [28] is defined as probability and consequences of an event, as expressed by the “risk triplet” that is the answer to the following questions: 1) What can go wrong? 2) How likely is it? and 3) What are the consequences if it occurs?

For a NPP, the PSA proceeds as follows: enumeration of sequences of events that could produce a core melt; clarification of containment failure modes, their probabilities and timing; identification of quantity and chemical form of radioactivity released if the containment is breached; modelling of dispersion of radionuclides in the atmosphere; modelling of emergency response effectiveness involving sheltering, evacuation, and medical treatment; and dose-response modelling in estimating health effects on the population exposed [29].

The insights gained from the PSA are used along with those from the deterministic analysis in the decision making process on safety issues for the

17 The term used in IAEA documents

18 The term used in USNRC NUREG documents, as well as in a reference standard document [28].

plant. PSAs are normally performed at three levels (L) as follows [30, 31]:

- L1 PSA which starts from an initiating event or an internal or external hazard that challenges the safe operation of the plant and identifies the combinations of failures of the safety systems that can lead to core damage. This provides an estimate of the frequency of core damage and gives insights into the strengths and weaknesses of the safety systems and the emergency procedures provided to prevent core damage.

- L2 PSA which models the phenomena that could occur following the onset of core damage that have the potential to challenge the containment integrity and lead to a release of radioactive material to the environment. The analysis considers the effectiveness of the design and the severe accident management measures that can mitigate the effects of core damage, and provides an estimate of the frequency and magnitude of a release of radioactive material to the environment.

- L3 PSA which models the consequences of a release of radioactive material to the environment and provides an estimate of the public health and other societal risks such as the contamination of land or food.

L1 PSAs have now been carried out for most of the NPPs worldwide. However, in recent years, the emerging standard has been for L2 PSAs to be carried out for all types of NPPs. To date, L3 PSAs in EU Member States have been performed for only a few plants.

The L2 PSA provides a structured assessment of the possible accident sequences including their frequencies that could occur, following core damage. It provides insights into which of the phenomena that could arise have the greatest potential to lead to containment failure or bypass resulting in a release of radioactive material to the environment. The starting point for L2 PSA is the grouping of a large number of accident sequences, derived in L1 PSA, into a smaller number of plant damage states (PDS) in accordance with accident characteristics and containment response characteristics for various accident sequences. PDS group sequences that would be anticipated to have similar effects on containment response and fission product source terms. It is therefore important to identify those attributes of an accident progression that will influence either the containment response or the release of fission products to the environment.

PDSs can be grouped into two main classes: those in which radioactive materials are initially released to the containment, and those in which the containment is either bypassed or ineffective. Thus, the PDSs identify the containment status (e.g. intact and isolated, intact and not isolated, failed or bypassed and, for bypass[19], the type and size of the bypass (e.g. interfacing systems LOCA[20] - ISLOCA, steam generator tube rupture – STGR) [30]. Thus, the L2 PSA provides an integrated analysis that takes account of plant specific features to determine how the fault sequences that have occurred leading to core damage would progress to challenge the containment and lead to a release of radioactive material to the environment. The main subject related objectives of the L2 PSA [32] are to:

- Gain insights into how severe accidents progress and identify plant specific vulnerabilities;
- Determine how severe accidents challenge the containment and to identify major containment failure mechanisms;
- Estimate the quantities of radioactive material that would be released to the environment for different types of accident sequences;
- Determine the overall frequency of a large release of radioactivity to the environment;
- Evaluate the impacts of various uncertainties, including assumptions relating to phenomena, systems and modelling, on the magnitude and frequency of the release;
- Provide a basis for the identification of plant specific severe accident management measures and determine their effectiveness;
- Provide an input into the development of off-site emergency plans;
- Provide an input into the decision making model for the timely execution of the off-site emergency actions.

The accident progression analysis part of the L2 PSA models the progression of the accident from core damage to the challenges to the containment and the subsequent release of radioactive material for each of the PDSs. This is generally carried out by using an event tree approach – referred to as either containment event trees (CETs) or accident progression event trees (APETs). The APETs/CETs provide the conditional probability[21] that a containment failure[22] can be realized, given a PDS. Typical containment failure modes and

[19] Containment bypass is a direct or indirect flow path that may allow the release of radioactive material directly to the environment bypassing the containment [28]

[20] Interfacing systems LOCA (ISLOCA): a LOCA when a breach occurs in a system that interfaces with the RCS, where isolation between the breached system and RCS fails. An ISLOCA is usually characterized by the over-pressurization of low pressure system when subjected to RCS pressure and can result in containment bypass [28].

[21] Conditional probability is the probability of occurrence of event A, given that event C occurs.

[22] Containment failure is loss of integrity of the containment pressure boundary from a core damage accident that results in unacceptable leakages of radionuclides to the environment [28].

mechanisms are shown in Table 5.4 [30].

Mode of failure	Mechanism for failure
Direct bypass	ISLOCA; SGTR; Externally initiated
Isolation failure	System failure; Operating mode
Vapour explosion	Rapid pressurization; Blast loads; Missile generation
Overpressurization	Steam spike; Gradual boil-off; Incondensable gases; Direct energy transfer
Underpressure	Inappropriate recovery of isolation failure; Inappropriate operation of filtered vent
Overtemperature	Core-concrete interactions; Direct contact by core debris; Thermal attack of penetrations
Combustion	Detonation; Deflagration to detonation transition; Deflagration
Concrete penetration	Basemat penetration; Pedestal/support failure
Other	Vessel thrust forces; Pipe whip; Random failure of RPV

Table 5.4. Typical containment failure modes and mechanisms.

The APETs/CETs produce a large number of end states, some of them are either identical or similar, in terms of key release attributes. These end states are often grouped together. The APETs/CETs model all the significant, physical, and chemical processes that could occur following a severe accident that challenge the containment or influence the release of radioactive material. Effectively, the PDSs define the initial and boundary conditions for the progression of a severe accident and together with appropriate two steps grouping of large number of CET end states form the interface between the L1 PSA and L2 PSA. The first groups the CET end states on the basis of similar source term phenomena to form source term categories (STCs) and the second one group STCs on the basis of similar environmental consequence to form release categories (RCs). The allocation of STCs to RCs is based on the potential of each source term to cause adverse effects.

To sum up the previous considerations, L2 PSA provides the results of accident progression analysis, containment analysis and estimation of accident source terms based on accident sequence frequencies. The source term analysis itself addresses the phenomena associated with the chemical processes affecting the radionuclide release and formation during the accident progression, and the transport of the radioactive material from the fuel through the containment to the environment. This analysis requires an in-depth understanding of the chemical and physical forms of the radionuclide species.

Thus, the most relevant quantitative results of L2 PSA are the frequencies of the release categories defined in the analysis. The most common results then are large early release (LER)[23] frequencies (LERF), which can be compared with the probabilistic targets for LERF, if they have been defined. However, there is no consensus in the EU Member States on what constitutes a large/early release [32].

It is important that the source term/release categories (sometimes referred to as release groups, release bins or source terms bins) are defined on the basis of appropriate attributes that affect fission product releases and accident consequences. These attributes are specific to the NPP and containment type, and there is no unique way to perform this task. However, Table 5.5 provides a list of important binning attributes for PWRs and BWRs [30].

The source term information that L3 PSA requires for each release category covers [30]:

1. The radionuclides including also the chemical forms of each radionuclide.
2. The frequency of release category.
3. The amount of radionuclides released as a function of time, expressed as fraction of the initial core inventory for each group of radionuclides having similar and chemical characteristics, i. e. having similar volatility.
4. The time of the release, which is related to reactor shutdown.
5. The warning time for implementation of appropriate countermeasures, defined as time from accident initiation to the actual occurrence of a release.
6. The location of the release relative to ground level (ground level release or elevated release).
7. The energy content of the release, which is a function of containment temperature and pressure prior to failure. This is important in determination of the potential for plume rise.
8. The particle size distribution of release aerosols, which will affect the deposition during plume transport.

[23] LER is the rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of off-site emergency response and protective actions [28], i.e. there is the potential for early health effects.

Release attributes	Variations
Timing of release	Very early (containment failure prior to core damage); Early; Intermediate; Late;
Containment bypass/isolation	ISLOCAs; SGTRs; Other initiating events;
Mode/mechanism of release	DBA leakage; BDBA leakage; Rupture; Basemat penetration;
Active fission product removal mechanism Passive fission product removal mechanisms (release pathways)	Sprays; Fan coolers; Suppression pools; Overlying water pools; Ice beds; Filtered vents; Others; Secondary containments; Reactor buildings; Tortuous pathways;
Location of release	Ground level; Elevated;
Energy of release	Low; High and energetic;
Duration of release	Rapid; Protracted;

Table 5.5. Example of binning attributes for APET/CET end states.

Since thermodynamic properties of some relevant fission products indicate the relative volatility of various core materials, they are usually grouped in accordance with their common chemical and physical characteristics. Various group structures have been proposed to date, ranging from very coarse to very fine groupings. Table 5.6 provides a reasonable group structure for such fission products, without going in details [30].

To recapitulate, a source term – that is, the quantity and duration of radionuclide release, is assigned to each STC. With the advances made in the development of so called system, or integrated severe accident computer codes[24], the source terms for specific sequences can be generated directly in recent advanced L2 PSAs. In these codes, the major radionuclide species are grouped just on the basis of similarity in chemical and physical properties and these default groupings are similarly adopted in the PSAs. A summary of the derivation of release categories and the attendant source terms for a number of European L2 PSAs is provided in [33].

[24] E. g. MAAP (developed by EPRI, USA and used especially for most of the US Individual Plant Examination – IPE Programme); MELCOR (developed by Sandia NL, USA and applied in essential reference study NUREG-1150 [13]); ASTEC (jointly developed by IRSN, France and GRS, Germany), etc., see Section 6.1.

Group	Species
Xe	Xe, Kr
I	I, Br
Cs	Cs, Rb
Te	Te, Sb, Se
Ba	Ba, Sr
Ru	Ru, Rh, Pd, Mo, Tc
La	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y
Ce	Ce, Pu, Np
Species	**Chemical forms**
I (gas)	I_2, CH_3I, HI
I (aerosol)	CsI
Cs	CsOH, CsI

Table 5.6. A reasonable group structure for fission products [30].

Another specific and important facet in determining appropriate EP measures is a meteorological assessment. Weather around the NPP site has a large impact on the estimated public doses. The meteorological data used in calculations of the off-site consequences for the severe accidents are based on actual NPP site specific measurements. The data is usually collected at various heights and locations and cover wind direction/speed, temperature, dispersion, and horizontal and vertical stability categories.

The atmospheric stability conditions ranging from super-adiabatic to inversions have been conveniently categorized into discrete classes by Pasquill (1961). This approach was further developed as reported by Gifford (1976) and hence the classes are normally referred to as Pasquill-Gifford (P-G) stability categories. Each class represents a specific atmospheric condition with class A being “unstable” and class F being “stable”. Class G was later added to describe strongly “stable” conditions [34] (Table 5.7).

The atmospheric stability is important in determination of lateral and vertical horizontal dispersion parameters. Neutral atmospheric conditions are given by the ‘D’ or (4) classification.

Stability class	No.	Description
A	1	Very unstable
B	2	Moderately unstable
C	3	Slightly unstable
D	4	Neutral
E	5	Slightly stable
F	6	Moderately stable
G	7	Very stable

Table 5.7. Atmospheric stability classes.

6. KEY SEVERE ACCIDENT ISSUES & PHENOMENA

Probabilistic accident progression and source term analyses, i. e., the L2 PSA addresses the key phenomena and/or processes that can take place during the evolution of severe accidents: the response of containment to the expected mechanical loads and the transport of fission products from damaged core to the environment. Such analyses provide information about the probabilities of accidental radiological releases - source terms - to environment. The analyses also indicate the relative importance of events in terms of radioactivity of the released materials; these provide a basis for development of plant specific accident management strategies [35].

The phenomenology of severe accidents is very complex. The severe accident evaluation methodologies are associated with large uncertainties. Thus quantitative evaluation of uncertainties associated with the results of L2 PSA requires, among other things, knowledge of the uncertainties in the severe accident phenomenology. Such known uncertainties are the major source of uncertainty in the results of L2 PSAs (see Chapter 7).

A Level 2 PSA requires – besides the essential requirements, described in the preceding Chapter 5 - the analysis of complex physical and chemical processes for which only limited experimental data are available. The phenomena to be considered in the course of the accident after the onset of core degradation are usually grouped into two categories [36]:

- Issues and phenomena associated with the thermal-hydraulics of the accident progression and the associated containment response. The associated analysis is generally referred to as*: 'Accident progression and containment performance analysis'.* The main phenomena cover

    1) In-Reactor Pressure Vessel (RPV) phenomena. Apart from the phenomena associated with core melt progression they include in-RPV radionuclide release and transport phenomena, which with the relevant reactor design characteristics impact these phenomena during the in-RPV phase of accident progression, and

    2) Ex-RPV phenomena immediately after the vessel failure and longer term behaviour. They identify the key phenomena during the ex-RPV phase of accident progression and cover the relevant reactor design characteristics, leading to identification of potential containment failure modes and the ex-RPV radionuclide release and transport phenomena.

- Issues and phenomena associated with the chemical processes affecting:

    1) The release of radionuclides and the chemical forms of radioactive materials during the different phases of a severe accident, and

2) The transport of the radioactive material from the fuel through the containment to the environment, if the containment is breached. The associated analysis is generally referred to as: *'Source term analysis'.*

There are two other important aspects of severe accident phenomena: 1) phenomenological issues related to severe accident management, and 2) the perceived level of uncertainty associated with the phenomena and the likely impact on results of analysis. While the former is not covered by this report, a separate Chapter 7 has been devoted to the latter one, i. e. the issue of uncertainties.

6.1 Some Severe Accident Analysis Computer Codes

There are, in general, three types of computer codes [36]:

- Separate phenomena codes: These codes computationally simulate the phenomena by using detailed models, consistent with the state-of-the-art, and the results of available experimental data. These codes are used typically in severe accident research, evaluation of severe accident management alternatives, and as benchmarks for the simpler, more parametric and integrated computer codes.

- System, or integrated PSA codes: These codes address the set of key phenomena that occur during each specific severe accident sequence. They incorporate the thermal-hydraulics, chemical and fission product models into a single code for the core, primary and secondary coolant systems, and the containment building. These codes are designed to run relatively quickly so that they can carry out a large number of calculations necessary for the different severe accident sequences defined for a Level 2 PSA study. To achieve this, they contain much simpler models than the separate phenomena codes.

- Simple parametric codes and computational tools: These are based on simple parametric models which interpolate between fixed points for which calculations with a more detailed code have been performed to determine the values of the parameters. These codes are intended for specific PSA applications in which the assessment of uncertainties on accident progression pathways requires extensive repetitive calculations by varying the input parameters.

Current application in L2 PSAs has tended to make use of the system, integral codes for providing the baseline analysis of accident sequences with limited supplementary analysis provided by other standalone codes for the detailed evaluation of certain phenomena. The further text is restricted to brief overview of three system, or so called *integral codes:* MAAP, MELCOR, and ASTEC.

6.1.1 Modular Accident Analysis Program (MAAP) - MAAP4

The Electrical Power Research Institute (EPRI) Modular Accident Analysis Program (MAAP), as a PSA tool is a fully integrated code that couples thermal-hydraulics with fission product release and transport. It also dynamically couples the Reactor Coolant System (RCS), containment, and reactor/auxiliary building responses. It has been used for many PSAs, especially for most of the U. S. Individual Plant Examinations (IPEs) and Severe Accident Management Guidelines (SAMGs) development and implementation. Different MAAP versions for PWR, BWR, CANDU, and WWER reactor designs are available. Models for advanced LWR plant designs, including their passive features have also been implemented, benchmarked, and accepted for design certification [36]. It simulates the accident progression from a set of initiating events to either a safe, stable, and coolable state, or radioactive releases to the environment. These may result from containment bypass, leakage, or structural failure due to overpressure.

Accidents have been analyzed for a variety of transients, including Loss of Offsite Power (LOOP), Loss of Coolant Accidents (LOCAs), Main Steam Line Breaks (MSLBs), containment bypass, mid-loop operation[25], and shutdown sequences.

The code has been subjected to independent design review and it has also been reviewed by the USNRC. MAAP has been compared with other codes on: (1) pertinent aspects of severe accident phenomena (e.g., core melt progression, source term estimates for plant applications using MELCOR code), (2) containment response, and (3) mass and energy releases for small and intermediate LOCA break sizes (RELAP).

MAAP has been validated against a variety of integral and separate-effects tests. Additionally, many comparisons between the MAAP code and actual plant transients, and accidents have been performed to illustrate the performance of individual models and to provide confidence in the MAAP integral results. The experimental validation status of the MAAP computer code in a form of assessment matrix is given in [36]. The main modelling areas of MAAP are: thermal-hydraulic modelling, core geometry and core melt modelling, radionuclide behaviour, and other physical processes. The latter include: the model for gas flammability, model for combustion completeness, model for burn time, RPV and vessel penetration failure models, the molten debris heat transfer and related models, corium-concrete interaction model, the RPV external cooling model, direct containment heating model and the in-RPV debris cooling model. MAAP4 also calculates the iodine chemistry, useful for long-term iodine behaviour in the containment.

[25] A low water level operation, i.e., mid-loop operation, is carried out for removing the residual heat, particularly in refueling/maintenance/repairing periods.

MAAP models the transport and retention of fission products in the RCS and the generalized containment. The materials released from the core are divided into 12 fission product groups, according to their chemical characteristics. The fission product physical states modelled are: vapour, aerosol, deposited and contained in core or corium. In short, the MAAP aerosol model considers the combined effects of aerosol agglomeration and removal mechanisms, including gravitational settling, wall condensation, inter-compartmental transport by carrier gas, thermophoresis[26], diffusiophoresis[27], and inertial impaction. Re-vaporization is included as a transfer between physical states. The MAAP aerosol model uses correlations based on exact solutions of integro-differential equations written for polydisperse size distribution of aerosols; and these models are extensively validated.

6.1.2 MELCOR Code

The MELCOR code was developed by Sandia National Laboratories (SNL) under the sponsorship of the USNRC. It is a fully integrated code that replaced the formerly used Source Term Code Package developed by Battelle Columbus Division, calculating a spectrum of phenomena leading to radiological releases to the environment. MELCOR calculations have been performed for the essential reference NUREG-1150 study [13]. MELCOR is a generic code applicable for both PWRs and BWRs and it treats a broad spectrum of severe accident phenomena in a unified framework. Originally it was designed to be a fast running PSA severe accident code using simplified parametric models. Today, owing to significant advances in computing power, the latest version of MELCOR serves as the best estimate code for predicting plant response to severe accidents and place special emphasis on core degradation modelling [36].

There is also one more significant aspect of this code. With the consolidation of modelling capabilities from other USNRC codes, *MELCOR today stands as the repository of knowledge concerning severe accident and fission product release phenomena,* benefiting significantly from important international research programs, including PHEBUS[28] for example. The use of parametric models is, in general, limited to areas with great uncertainties where there is no consensus concerning an acceptable mechanistic approach.

MELCOR is intended to be applied by the US NRC for:

- PSA studies for existing and advanced LWRs,

[26] Particle motion in a temperature gradient, from a hotter to a colder region [http://jets.poudres.free.fr/index_fichiers/page0007.html]

[27] A process in a scrubber whereby water vapour moving toward the cold water surface carries particulates with it [http://www.answers.com/topic/diffusiophoresis?cat=technology]

[28] The PHEBUS-FP program is an international cooperative research program to develop experimental data for validating computer codes used for severe reactor accident analysis. The experimental work is done at the Cadarache Centre in France. Partners in this program include the EU, Canada, Japan, South Korea, Switzerland, and the USA.

- Best-estimate accident sequence studies to develop insights into both physical phenomena, as well as hardware performance,

- Audit reviews of PSAs,

- Accident management studies that analyse the progression of accidents and evaluate the detrimental and beneficial effects of various strategies.

Today, MELCOR is being used also to assist the NRC in the design certification process for a number of new plant designs, including Generation III+ AP1000 PWR and the US-EPR, and to assist the USNRC Nuclear Reactor Regulation Office with the evaluation of numerous license amendment requests in the context of risk informing of regulatory processes. Additionally, MELCOR is being used as a code based means of conducting uncertainty analysis in L2 PSA applications. The code is based on specially developed models for thermal hydraulics, core melt, fission product release and transport processes [30, 36].

The main modelling areas of MELCOR are the same as that of MAAP: thermal-hydraulic modelling, core geometry and core melt modelling, other physical processes and radionuclide behaviour. The phenomena modelled are the thermal-hydraulic response in the reactor coolant system starting with the accident initiating event, core uncovery and heatup, hydrogen production, degradation and downward relocation of core materials, lower head failure of the reactor vessel, slumping of molten corium in the reactor cavity, attack on concrete surfaces by corium, heating of containment and confinement buildings, combustion of hydrogen-air mixture, and release of fission products from the core and their transport to the containment or eventually to the environment. The thermal-hydraulic modelling is based on the control volume approach, which - in the reactor core or bundle - is coupled to a special nodalisation scheme consisting of radial and axial cells [37].

6.1.3 ASTEC Code

The ASTEC (Accident Source Term Evaluation Code) is a system code. Compared to MAAP and MELCOR it calculates the fission product (FP) gas phase chemistry along the pathway of the release (primary and secondary loop of RCS).

The code, jointly developed by the French Institut de Radioprotection et de Sûreté Nucléaire (IRSN) and by the German Gesellschaft für Anlagen und Reaktorsicherheit (GRS) is to describe the behaviour of a whole NPP in severe accident (SA) conditions including engineered safety systems and procedures used in SA management [38]. This code is used to simulate FP release experiments; its specific features are described as follows.

The aim of the code is to simulate an entire severe accident sequence from the initiating event through to fission product release and transport to and out the containment. The applications are:

- Source term determination studies;
- Level 2 PSA studies;
- Accident management studies;
- Physical analyses of experiments to improve the understanding of the phenomenology.

The ASTEC V1 series has been developed since 1998. Its main evolutions were:

- Simulation of the front-end phase of the accident, using a new module CESAR for Reactor Coolant System (RCS) thermal-hydraulic behaviour, which allows a complete simulation of the scenarios and to avoid the additional use of other thermal-hydraulics codes;
- Improvement of core degradation models in the new module DIVA that was mostly based on the models of the ICARE2 IRSN mechanistic code.

The version V1.2 rev1 was released in December 2005 to European partners in the frame of the SARNET[29] network of excellence on severe accidents. It started in April 2004 for a 4-year duration in the frame of the 6th FP (Framework Programme) of the European Commission[30]. At present, ASTEC is playing a central role in SARNET in order to progressively become *the reference European integral code*. That is why its description here is more detailed in comparison with the above mentioned codes. The latest ASTEC version V1.3 was released in December 2006.

The ASTEC validation strategy consists [36] in comparison with results of experimental programmes performed at various scales, and by comparison on reactor calculations with reference international codes (so-called ‘benchmarks’), e. g. [37]. The basic validation matrix aims at covering dominant phenomena occurring in severe accidents and to estimate the model uncertainties. Validation is supported by a large set of international experiments:

- On one hand, analytical experiments that address a single phenomenon (separate-effect test) or few phenomena (coupled-effect test);
- On the other hand, integral experiments (for instance PHEBUS-Fission

[29] Severe Accident Research NETwork supported by European Commission [www.sar-net.org].

[30] Contract FI6O-CT-2004-509065, “SARNET. Network of Excellence for a Sustainable Interaction of European research on Severe Accident Phenomenology”, 6th Framework Programme, NUCTECH 2003-3-4.3.1-2, April 2004.

Product Tests at IRSN in France, QUENCH and CORA (see further text) at FZK in Germany); these applications allow checking of the correct reproduction of coupling between the code modules and the completeness of the modelling with respect to significant phenomena. Moreover, this kind of experiment has often been performed at large scale allowing better extrapolation to reactor scale. In a similar way, the application to the TMI-2 severe accident is an essential exercise.

Similarly to the above mentioned experiments, the CORA melt progression experiments performed on electrically heated bundles at FZK (formerly KfK) Karlsruhe were aimed at identifying and quantifying the mechanisms and sequence of events causing severe fuel damage to LWR fuel rods during heat-up and re-flooding. They were supported by separate-effect tests to measure the kinetics of chemical reactions identified as important in the integral experiments[31].

Cooling of an uncovered, overheated PWR core by water is the main accident management measure for terminating a severe accident transient. But, before the water succeeds in cooling the fuel elements, its injection can cause under certain circumstances renewed oxidation of the Zircaloy fuel rod cladding, leading to reheating of the rods, and to a sharp increase in hydrogen production and rod failure followed by the release of additional fission products. The additional hydrogen might threaten the containment, and the increased fission product release increases the source term. Evidence for these effects has been obtained from the analysis of the TMI2 accident, and in CORA experiments. The reasons for this enhanced oxidation are not yet fully understood but it is believed that the cracking of oxide layers due to the thermal shock and subsequent exposure of fresh Zircaloy to steam are significant factors.

Because of the importance of understanding the in-vessel hydrogen source term that results during quenching, the QUENCH program was initiated at FZK Karlsruhe. The large-scale 21 rod bundle experiments in the QUENCH Facility, supported by extensive pre-and post-test calculations, are the highlight of the program. The objectives of the QUENCH experiments are to investigate the physical and chemical behaviour of overheated fuel elements under different flooding conditions, to improve the understanding of the effects of water addition at different stages of a degraded core and to create a data base for model development and code improvement. The main parameters of the test program are: quench medium, i.e. water or steam, fluid injection rate, extent of pre-oxidation at onset of quenching, and the starting temperature at initiation of quenching or cool-down[32]. One of the main objectives of the QUENCH Program is to determine the hydrogen source term during reflooding of an over heated reactor core. For that reason different methods are used to determine the

31 http://www.edata-center.com/proceedings/1bb331655c289a0a,12dba6194fed8cf9,1720a18008afad63.html

32 http://hikwww9.fzk.de/isp45/QUENCH_Program/qprogram.htm, and http://bibliothek.fzk.de/zb/berichte/FZKA6968.pdf]

hydrogen release rates as well as the integral hydrogen production during the complex and - the quench phase – highly transients experiments.

The code validation benefits greatly from the very intensive work performed over more than ten years with the previous codes. Besides, the DIVA module of ASTEC also benefits from the very intensive validation of ICARE2 code in the past years on more than 50 experiments. As far as possible, selected experiments are 'reference experiments' that mostly belong to the list of code benchmark exercises selected by OECD expert groups (International Standard Problem - ISP). For instance ISP 46 (PHEBUS Fission Product T1) is one example of integral calculations coupling all modules. Continuous efforts on interpretation of all the integral PHEBUS Fission Products (FP) tests are performed with modules focusing on some subset of phenomena, e.g. SOPHAEROS for fission products behaviour in the circuit [36].

The PHEBUS-FP experiments simulate the major aspects of a severe accident, beginning with the degradation of irradiated reactor fuel, release of fission products, transport of fission products through a simulated reactor coolant system, and injection of these fission products into a model of a reactor containment. Fission product behaviour within the containment is examined over a period of about five days. This examination includes study of both aerosol behaviour and the chemistry of radioactive iodine[33].

The experiments in the PHEBUS-FP program are providing data that are valuable for validating and refining computer codes used for reactor accident analysis. Data from the tests have been used to refine models of core degradation and fuel relocation, hydrogen production, and fission product speciation. The data indicate needs for refining models of aerosol deposition within the reactor coolant system and models of the aqueous and gaseous chemistry of iodine within the reactor containment.

ASTEC has progressively reached a larger European dimension [36], notably within the 5th Framework Programme with the EVITA project[34] devoted to code validation by independent users. This dimension is still increasing in the frame of the above mentioned SARNET. The ASTEC V1.3 rev.1 latest version was released in December 2006 to 27 European SARNET partners. Its content and capabilities are described in [38]. Some examples of applications of this version include the first phases of the TMI2 accident, containment thermal-hydraulics and a severe accident sequence in a French PWR 900 MWe.

Many applications have been performed on various accident sequences (Station Black-Out (SBO), Small or Medium LOCAs, and Loss of Steam Generator Feedwater (LFW), extending the scope of applicability to Large Break LOCA and to SGTR sequences. The latest version allows complete calculations up to iodine

[33] http://www.nrc.gov/reading-rm/doc-collections/acrs/letters/2002/4921995.html
[34] Allelein, H. J., et al: European validation of the integral code ASTEC (EVITA), Nuclear Engineering and Design 221 (2003).

behaviour in containment of different sequences on PWR 900 and 1300, Konvoi 1300, WWER-440 and WWER-1000. The numerical robustness has been largely improved, in particular on Molten-Core Concrete Interaction (MCCI) situations with presence of water in the cavity [38].

ASTEC has a modular structure, each code module simulates a reactor component or a physical phenomenon in fully coupled or stand-alone mode, see Figure 6.1, taken from [36].

Data is exchanged between the ASTEC modules through a dynamic memory at macro-time steps, i.e. evolving throughout the calculation [36].

- *CESAR module* deals with two-phase thermal-hydraulics in the Reactor Coolant System (RCS). The RCS discretization is an assembly of 0D elements (volume with homogeneous fluid or with a swollen water level for two-phase flow situations) and 1D elements (pipe with axial mesh)

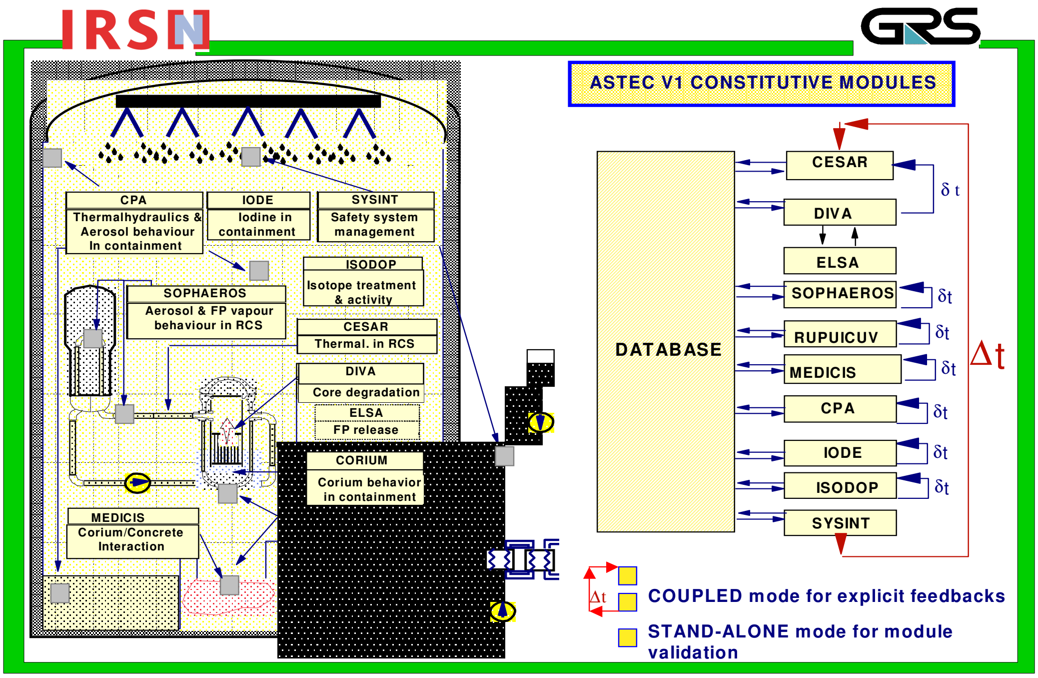


Figure 6.1. ASTEC Modules and Structure [36].

connected with junction elements. Two phases are considered: water and gas mixture.

- *DIVA module* simulates the in-vessel core degradation (most models are derived from the ICARE2 IRSN mechanistic code for core degradation).

The main phenomena are modelled, namely: fuel-rod ballooning and failure, exothermic oxidation of Zircaloy cladding by steam and concurrent hydrogen production, core heat-up and formation of mixtures of molten materials (corium) at high temperatures (as high as 3000K) with corium pool formation and growth, corium slumping through internal core structures until it reaches the vessel lower head, corium accumulation heating the lower head until melt-through or mechanical failure of the vessel. Degradation of control rods (Ag-In-Cd or B_4C materials) is also modelled.

- *ELSA module* calculates the release of fission products, actinides and structural materials (Ag, In, Cd, Sn, etc.) from the core. The semi-empirical approaches deal with three fission product classes (volatile, semi-volatile and non-volatile) for solid fuel (intact rods, debris) by only modelling the dominant phenomena for each class. For example in the case of volatiles: species intra-granular diffusion through UO_2 fuel grains, taking into account the fuel oxidation and the fuel grain-size distribution. Actinides and low volatile fission products are released after a severe degradation of the fuel rods. The release from a corium molten pool is modelled by a non-ideal solution for phase distribution and an ideal solution for fission products release.

- *SOPHAEROS module* computes the aerosol and vapour transport through the RCS via gas flow to the containment. Using five physical states (suspended aerosols, suspended vapours, vapour condensed on walls, deposited aerosols, and adsorbed vapours), the mechanistic or semi-empirical approaches model the main vapour-phase phenomena (such as gas phase equilibrium chemistry, homogeneous and heterogeneous nucleation) and aerosol phenomena (agglomeration, turbulent diffusion, thermophoresis, diffusiophoresis, impaction in bends and constrictions and remobilization of deposits). The module calculates aerosol mechanical resuspension and abrupt changes of pipe sections as well. The main feature of SOPHAEROS module is the materials database including the thermo-chemical properties of about one thousand chemical species. This allows to calculate the chemical forms composed from fission product elements and to identify the most significant ones from the multitude of possible chemical forms.

- *RUPUICUV module* aims at evaluating Direct Containment Heating (DCH). After vessel lower head rupture, corium is discharged at high temperature driven by primary pressure into the cavity (vessel blowdown, cavity pressurization), where some part of the ejected corium may be entrained into the containment and contribute to its heat-up. Two kinds of cavities are accounted for: one with an annular space around the vessel like in European PWRs, and one with several intermediate compartments between cavity and containment like in USA PWRs.

- *MEDICIS module* for MCCI with a lumped-parameter approach for corium layers. Corium remaining in the cavity interacts with substrate concrete leading to concrete ablation and release of incondensable gases (H_2, CO, CO_2) into the containment. Water injection upon the corium surface is accounted for (including water ingression and melt eruption). Corium pool configurations may be either mixed or stratified (metal/oxide).

- *CPA module* for containment thermal hydraulics and aerosol behaviour, based on a "lumped-parameter" approach. Most models are derived from former GRS codes (RALOC and FIPLOC). The containment can be nodalised as several 0-D zones, each representing one real compartment surrounded by walls (such as dome or casemates). The containment atmosphere heats up under the effect of sources of steam, fission product gases and aerosols, and pressure increases. CPA describes phenomena such as gas distribution, pressure build up, hydrogen combustion and the behaviour of engineered safety systems such as Passive Autocatalytic Recombiners (PAR) and sprays. The code describes aerosol and fission products transport and depletion where, for example, much of the aerosol settles on horizontal surfaces.

- *IODE module* for iodine behaviour in the containment. This module describes in a kinetic way (i.e. in non-equilibrium) chemical transformations of iodine in aqueous and gaseous phases, iodine mass transfer between sump and atmosphere and its adsorption/desorption on painted and metal walls.

The Material Data Bank (MDB) library groups together all material characteristics: properties of simple materials and corium mixtures (such as enthalpy or density), ideal chemistry (equilibrium reactions), iodine chemistry (kinetics), fission product isotopes (decay heat and transmutation rates). A significant effort was done on validation of the two-phase thermal-hydraulics in RCS and of MCCI. With respect to the previous version, the newest released ASTEC V1.3 version presents the following main model improvements [38]:

- General model of RCS pumps, applicable to any type of reactor,

- Improvements of models of corium behaviour in the vessel lower plenum,

- Quenching of fission products at their arrival in the containment,

- Release of structure materials from the core and the in-vessel corium molten pools.

The ASTEC code was used for IRSN L2 PSA studies of the 900 and 1300 MWe French PWRs. A significant number of scenarios that differ in initiating event and in the actuation of safety systems are being analysed. Furthermore, in the EVITA project, several benchmarks on plant applications have been performed including comparison with other integral codes such as the MELCOR and MAAP4 codes. These were performed on the following reactor types: French PWR 900, German Konvoi 1300, Westinghouse AP1000, VVER-440/V-213 and V-230, and VVER-1000. These benchmarks are currently completed or extended in the SARNET [36].

7. TREATING UNCERTAINTIES

Since L2 PSA is a major contributor of information to EZ, this chapter, though generic in its nature (the concepts shown are applicable to many systems where uncertainty is an issue), will focus on the treatment of uncertainties in this area of nuclear safety. L2 PSA is a systematic way to study, from the point of view of safety and with the restrictions of a specific methodology, the behaviour of a system (NPP under accident or quasi-accident conditions) when uncertainty is present and widespread. The starting point of level 2 is the result of a L1 PSA. The outcome of such study is a huge quantity of accident sequences that are grouped, according to different criteria regarding accident characteristics and potential containment responses, into a manageable number of plant damage states (PDS). After an appropriate screening of very low probability sequences, the probabilistic progression of accidents is studied using event trees, commonly known as accident progression event trees (APET) or containment event trees (CET), under two possibilities: large event trees (virtually all questions regarding severe accident are included as top events) and small event trees (only main questions regarding severe accident phenomena are included as top events). The use of these event trees leads to getting a huge quantity of end states, which have to be grouped (see Chapter 5), as in the case of PDSs, to get a more manageable set of release categories (RCs), later used to estimate all the variety of different possible source terms.

Uncertainty is really pervasive in a L2 PSA. The first matter of concern is the starting point. Is really complete the picture obtained as an output from the L1 PSA? Could there be any 'hole' in the picture? Is completely meaningful the set of PDS obtained? In order to get the end states, computer codes that simulate the behaviour of the plant in a deterministic way have to be run under the containment event tree structure. Do they reproduce the behaviour of the system in an accurate way? Are we able to estimate accurately branching probabilities? Are we able to model the accident event sequence in the right order and with the right intervals between events? Do we have the necessary data to feed the computer codes we are using? These and other questions are key issues when dealing with uncertainties in L2 PSA.

7.1 Uncertainty sources

The last paragraph could be summarised saying that uncertainty arises in three areas of the L2 PSA: 1) definition of plant damage states and release categories, 2) simulation of the problem, including event tree construction and phenomenological models (computer codes) used to simulate the physical-chemical processes involved, and 3) data used to feed models. This is what typically has been classified as scenario, model and parameter uncertainty. Scenario uncertainty is sometimes called completeness uncertainty.

Nevertheless, attending to their real origin, these types of uncertainty may be re-classified as aleatory and epistemic or lack of knowledge uncertainty.

Aleatory uncertainties are usually associated to parameters with some inherent variability. Aleatory uncertainties arise when an experiment is repeated several times under equivalent conditions and the results obtained differ from each other. An example of a parameter affected by this kind of uncertainty is the time taken by a safety system to start after its actuation is demanded. In this case, variability comes from the set of physical and chemical processes involved. Increasing the number of observations (experiments) does not make aleatory uncertainty to decrease, the standard deviation of the variable itself will not decrease, but will allow knowing with more accuracy the probability density function (PDF) followed by that parameter, i.e. the type of PDF and the parameters that characterise it. So, if that time follows a Weibull distribution, increasing the number of observations will allow knowing more accurately the standard deviation and the shape parameter.

Epistemic uncertainties are related to the existence of lack of knowledge about the problem. This type of uncertainty affects not only parameters, but also models, PDSs and source terms (STs). A parameter will be affected by epistemic uncertainty when it is not random, but we cannot measure it, either because it is impossible or because it is extremely expensive to do it. This type of uncertainty is completely different of the aleatory uncertainty. Parameters affected by aleatory uncertainty are fully described by their associated PDFs. In the case of parameters affected by epistemic uncertainty, what we do is to characterise our uncertainty about the parameter, and we do it through PDFs. Those PDFs summarise our state of knowledge about what values the parameter could be close to more likely or less likely. Many parameters (coefficients) of models used in the area of severe accidents are affected by lack of knowledge uncertainty; they are not random, but their values are unknown, so PDFs have to be used to characterise them. This type of uncertainty affects most of the parameters used in severe accident codes.

Epistemic uncertainty does also affect models. Sometimes, there are several models to describe the behaviour of the system; some of them describe the behaviour of the system under some circumstances and others under other circumstances, and it is not clear at all how to take that fact into account in the analysis. Think just of two different nodalisations for the same computer code. Some authors consider [55] appropriate to assign probabilities to the different alternative models and to run one of them or another one according to those probabilities. Another alternative is to build up a metamodel that includes, as sub-models, the different models and runs either one sub-model, or another one depending on the values sampled and which models fit better experimental results under those circumstances. Under any circumstances, only validated codes, or at least non-invalidated, should be used.

In a PSA uncertainties propagate through the whole study. Those that arise in L1 PSA propagate into L2 PSA and those, together with the ones that arise in the L2 PSA itself propagate into L3 PSA, where they are combined with the intrinsic uncertainties of L3 PSA.

Current L1 PSA is a very mature area of knowledge. The main uncertainty arises from the possibility of omitting some not very unlikely sequences that could lead to a significant PDS (completeness or scenario uncertainty). This could produce an underestimation of that PDS. The binning of L1 PSA sequences into PDSs can also introduce uncertainty. The binning process consists in classifying sequences in different PDSs according to a set of variables that provides relevant information about the physical status (the pressure of the coolant in the primary system being the most important one in PWRs) of the plant and the availability of safety systems. Variables used in the classification are continuous and any classification based on setting limits on continuous variables is somewhat arbitrary. An inadequate classification could lead to a wrong propagation of uncertainties through the L2 PSA model.

A key step in the L2 PSA is the design of the APETs. The correct definition of the APETs involves the inclusion of the right events, no omission of any relevant event or process, the right order and the right number of events, including also the right timing between successive events, which usually depends on the correct simulation of stochastic events. Additionally, phenomenological models (i.e. MELCOR, ASTEC, etc.) are used to simulate the behaviour of the plant under such accident sequence. An example of this is the construction of an APET to study sequences with important generation of hydrogen (H_2). Uncertainty appears everywhere in such a sequence: possibility of more than one combustion in the same sequence (model uncertainty), occurrence of ignition sources (random uncertainty, correct simulation of stochastic events), distribution of H_2 /air/steam in the containment (model uncertainty), percentage of H_2 burnt in a given combustion/detonation (model uncertainty), conditions for reaching different flammability regimes (parameter uncertainty), etc. Many other physical-chemical phenomena that can happen in an accident sequence are subject to important model and parameter uncertainties such as direct containment heating (DCH), in-vessel and ex-vessel steam explosions, vessel rocketing, fission products resuspension and molten core – concrete interaction (MCCI), among others.

Uncertainties may also arise in the last step of L2 PSA, the binning of accident sequences in RCs. As in the case of PDSs, a set of variables must be identified. Later on, the binning of sequences is done according to the values of these variables in the sequences. The omission of important variables and the selection of non-appropriate limits in each variable may produce an inadequate final set of RCs.

L3 PSA is also deeply affected by uncertainty. Main uncertainties affect atmospheric dispersion models used to simulate the radioactive plume transport (model uncertainty and parameter uncertainty). The meteorological database used is a good example of random parametric uncertainty, while many of the parameters used in the atmospheric models, such as centreline concentrations (χ_c/Q) and standard deviations of crosswind locations at release height (σ_y) are good examples of epistemic parametric uncertainty.

7.2. Input uncertainty assessment

Input uncertainty assessment is the process of characterising through probability density functions (PDFs) or probability mass functions the uncertainty about continuous and discrete input random variables. There are essentially two ways how to do it: Using classical inferential methods and using Bayesian methods. Expert judgment is a third way to do it, which constitutes, in this context, an extension of Bayesian methods.

7.2.1 Classical inferential methods

Classical inference methods are based on the assumptions of having a random sample and knowing the probability model from which the data come from. There are several methods, some of them recently developed, like Jackknife and Bootstrap[35] [56], but the best known and most widely used methods are Maximum Likelihood method and the Method of Moments. The main shortcoming of all these methods is the need of large sample sizes not easily obtainable under the restrictions of a complicated engineering facility in order to get good quality estimates, which limits their applicability in this field.

Method of Moments

This is probably the oldest inferential method to estimate the parameters of a PDF. K. Pearson developed the method of moments by the end of 19th century. The idea is quite simple. It consists in taking as an estimator of a parameter its equivalent sample quantity. So, the sample mean is the estimator for the mean, the sample variance is the estimator for the variance and so on.

Maximum Likelihood Method

Maximum Likelihood Method is the most widely known and most powerful estimation method in the classical context. Let us assume that we wish to study a random variable X (parameter affected by uncertainty) whose type of distribution function $f(X/\theta)$ is known, but whose parameter θ is unknown. In order to estimate θ we take a random sample - $\vec{X} = (X_1, X_2, ..., X_n)$, which is supposed to be a random vector, whose components are independent and identically distributed, so that its joint probability density function is

$$f(\vec{X}/\theta) = f(X_1, ..., X_n/\theta) = \prod_{i=1}^{n} f(X_i/\theta). \quad (1)$$

[35] Jackknife and Bootstrap are resampling methods, which are based on treating the sample obtained as the population itself and resampling from it to get estimates of statistics of interest (mean, variance, etc.)

It is important to notice that in this expression, under the classical statistics view, before sampling, θ is unknown, but has a given value. The objective is to determine what value, among the infinite values that θ could take, makes more likely obtaining the sample actually obtained. So, the problem is to find the value of θ for which function (1) gets its maximum value. Specific applications and guidance may be found in standard Statistics books such as reference [57].

7.2.2 Bayesian inference methods

Bayesian interpretation of probability makes Bayes' formula a powerful tool to update degrees of belief when new information is available about an event or a proposition. Let H be the knowledge of a person, and let $\{z_i\}_{i\in I}$ be a partition of the sample space of events. The Bayesian probability provided by the person for an event z_k, is $P(z_k / H)$. The acquisition of a set of new evidence *H'* induces a change in the probability given by Bayes' formula

$$P(z_k / H, H') = \frac{P(H' / H, z_k) \cdot P(z_k / H)}{P(H' / H)}, \tag{2}$$

where $P(z_k / H, H')$ is the 'a posteriori' probability of z_k, $P(z_k / H)$ is the 'a priori' probability of z_k and $P(H' / H, z_k)$ is the likelihood of evidence conditional on knowledge H and the occurrence of event z_k. $P(H' / H)$ is the probability of new evidence conditional on previous knowledge. That means that the a posteriori probability is proportional to the a priori probability and to the likelihood of evidence.

Two remarkable results are obtained from (2). If the a priori probability of an event is zero, the a posteriori probability will remain zero, even though the evidence against it could be very strong. So, much care should be taken when providing a priori probabilities. Null a priori probabilities should be avoided, unless total evidence of the impossibility of the events or propositions under study is available. The second result is related to the existence of strong evidence. In that case, the likelihood will be completely dominant and the a priori probability will be almost irrelevant (the a posteriori probability and the likelihood will be almost equal). This is the case of large sample sizes, for which relative frequencies and Bayesian probabilities will be almost equal. Specific applications and guidance may be found in standard Bayesian Statistics books such as reference [58].

Bayesian estimation is mostly adequate when there are not many specific available data but generic information from other providers or other similar NPPs can be obtained and used to create a first estimate (a priori distribution). Bayesian methods are widely applied in L1 PSA and, to a less extent in L2/L3

PSA.

7.2.3 Expert judgment

The use of Expert Judgment (EJ) techniques is unavoidable in a L2 PSA due to the lack of data about many of the involved phenomena. In some cases, it is almost impossible, from a physical point of view, to get the data needed to feed computer codes. In other cases the cost of getting them is so high that only a few of them may be obtained.

Report NUREG-1150 [13] shows the results of the whole PSA methodology applied to estimate the risk associated to five NPPs in USA. This is the first large-scale project where expert judgment was extensively and systematically used to collect relevant information. Regarding L1 PSA, two expert panels addressed six issues; while five expert panels addressed twenty-two issues in L2 PSA. This study highlighted the importance of EJ as an unavoidable source of information to deal with very sophisticated safety problems.

Between 1990 and 1999, the European Commission (EC) and the US Nuclear Regulatory Commission (USNRC) sponsored a large joint uncertainty study of accident consequence codes (MACCS and COSYMA) for NPP using EJ. Several expert panels participated in the elicitation of many code input parameters in six fields: 1) atmospheric dispersion and deposition, 2) deposited material and external doses, 3) food chain, 4) internal dosimetry and 5) late health effects. A summary of main results of this study and a short description of the methodology used is available in reference [59].

The bullet list shown in the next lines provides the steps of a generic EJ protocol based on the experience accumulated during the last decades in this area of knowledge:

- Selection of project team;
- Definition of the questions to be studied;
- Selection of experts;
- Training;
- Tasks definition;
- Individual experts' work;
- Elicitation of experts' opinions;
- Analysis and aggregation of results;
- Documentation.

Selection of project team

The project team consists of analysts and generalists. Analysts are in charge of organising the steps of the protocol, so, they should have a sound background in

Probability and Statistics Theory, in Knowledge Psychology and in Elicitation Techniques. Additionally, they should be skilful in working with people, since they have to extensively interact with experts. The number of analysts needed depends on the extent and scope of the EJ application, though usually a couple of analysts will be enough, even for large applications. Generalists provide help to the analysts in all subjects related to the specific area of knowledge of the problem to be solved. They should be able to help experts when decomposing a problem and they should be skilful at getting information sources as needed. So, they should have a good general knowledge about the problem at hand, though they do not need to be leading experts in that field. The organisation interested in the EZ study usually provides the generalists.

Definition of the questions to be studied

Once the project team has been made, analysts and generalists must define the questions to be evaluated by the experts. The starting point for any question to be solved is usually vague. It is completely necessary to arrive at a *complete definition* of the parameters whose uncertainty we want to characterise. *Complete definition* of a parameter means the full definition of the parameter, the initial conditions to evaluate it and any other implicit hypothesis under the initial conditions. The final definition should be extremely clear and accurate, with no ambiguity.

After the full definition of the question, a list with all relevant sources of information should be done. Potential decompositions of the parameters could be done. The list of references to be considered in the list must show the actual state of knowledge in that area, but independence and reliability of the sources should always be kept in mind.

When experts are expected to use computer codes for their assessment, the project team should foresee the potential training of experts in uncertainty propagation techniques (sampling, response surfaces, estimation, order statistics, etc.).

Selection of experts

The only objective of this phase is to select the most qualified experts to perform the assessment. Qualified experts are those that: 1) have the necessary knowledge and experience to perform the assessment, 2) are willing to participate in the assessment, and 3) do not have important motivational biases.

The first step to get the final list of experts is to start with a large list of potential experts. That first list could be based on the opinion of the generalist plus a thorough search in the scientific literature about that area. A screening should be done checking the three points in the list above. If necessary, interviews should be done to check those conditions, mainly the third one. After performing the screening, a shorter list should be obtained, from which the final selection of experts will be done. In order to arrive at the final list, two criteria should be taken into account: The number of experts to assess each question should preferably be between three and five (based on Bayesian combination of opinions criteria) and the experts should have as much diversity as possible (different background, different types of institutions, etc.).

Training

The objective of this phase is to let experts know normative aspects of EJ elicitation processes. This main objective may be decomposed as the following sub-objectives: a) motivate experts to let them provide rigorous assessments, b) let them remember basic concepts of Probability and Statistics, c) provide them training in the assessment of Bayesian probabilities, and d) let them be aware of basic issues related to knowledge biases.

During the motivation phase the experts must get information to point out the importance of the work they are going to do. Firstly, the project team explains to the experts the study frame where their opinions will be used, stressing the part of the study where their opinions are relevant. Secondly, the necessity of EJ will be explained, letting them be aware of the concept of *Lack of Knowledge Uncertainty*, and how it links to them. Thirdly, the project team should let them know that the key issue is not to predict a single value of each parameter under study, but characterising their uncertainty, allowing others to know the actual state of knowledge in that area.

After remembering basic Probability and Statistics concepts, the experts get some training about assessing Bayesian probabilities, which includes: accurate definition of questions to be assessed (making explicit implicit hypotheses, showing well/non-well posed questions), decomposition as a way to simplify assessments (use of influence diagrams, event trees and uncertainty propagation techniques) and adequate evaluation of different evidences in order to assess probabilities (use of Bayes' theorem and concepts of independence and reliability of information sources).

The last part of the training session is dedicated to explain Knowledge biases to the experts in order to teach them to provide more reliable opinions, i.e. representativeness, availability, and anchor and adjustment. Experts should be informed of the hazard of being overconfident. A calibration exercise could be appropriate. The whole training session should not take more than one morning.

Tasks definition

This step is done through an interactive session of the project team and the experts. The issue at hand is to explain to the experts, in a detailed way, the questions to be assessed and to make a schedule of the activities to be developed by each expert. All the work developed by the project team during the *Definition of the Problem* phase should be used now. The session should start with a presentation by the generalist of the parameters to be assessed, including all relevant sources of information previously identified. Experts should provide their own view of the problem and the definition of the parameters, pointing out, if needed, further information sources, computations to be made, etc. The result of this session, eventually, would be a refined definition of the parameters under study. Common definitions to all the experts should be agreed.

The second step in this meeting is to study the possible ways to decompose each parameter. The project team should provide a seminal decomposition that should be discussed with the experts. The objective is to help the experts to develop their own decompositions. Decompositions could be quite different from one expert to another one. Experts will have to assess uncertainties of variables in the lowest levels. The analysts will usually do its aggregation. This is the point to introduce propagation of uncertainties concepts to the experts and to let them know all the potential variety of tools that the analysts could provide them to pre-process and post-process probabilistic runs of computer codes, or of the simple decomposition model developed by experts.

Individual experts' work

Experts develop their analysis during this phase, according to the schedule agreed in the previous step. Each expert will write by the end of this period a report summarising the main hypothesis and procedures used during his/her work, the conclusions achieved and, if he/she wishes, a preliminary assessment of uncertainties. Whenever needed during this period, the project team should be available to each expert in order to provide statistical support, or to solve any doubt about the parameters to be assessed.

At the end of this phase, the project team organises a meeting with all the experts. Each expert presents his/her work and the conclusions achieved. This meeting allows each expert to get some hints about alternative ways to tackle the problem.

Elicitation of experts' opinions

The elicitation of each expert's opinions is individual and should be done in a quiet environment, if possible without interruptions. It is convenient to have the presence of an analyst and a generalist, in addition to the expert. In a systematic way, the analyst gets the opinion of the expert for each parameter, asking for supporting reasons whenever necessary. The role of the generalist in this session is to provide additional information when needed, to provide general support and to audit the session in order to avoid irregularities (bias induction, etc.). Whenever needed, the analyst could ask questions in a different way to check potential inconsistencies. The session must be recorded as much as possible (tape recorders, video or extensive hand annotations).

The techniques used to help the expert when assessing uncertainties are quite standard: quantile assessment for continuous variables and probability estimations for discrete variables (direct or indirect methods); in the case of experts with some skills in probability other techniques like direct parameter assessment or drawings are acceptable.

Analysis and aggregation of results

Assessments provided by experts are studied in this phase. The objective is to check that there are not important biases and the logic correctness of their rationale. If biases and logic faults are not present in experts' assessments, next step is to check if individual opinions may be aggregated to get a unique distribution for each parameter.

Before aggregating individual distributions, one condition should be checked. It is related to the overlap between distributions of different experts. If the distributions do not overlap, it means that essentially the experts disagree. In that case aggregation should be avoided. Under these circumstances a reconciliation session could be of help. An analyst should lead the session and should organise it according to the following steps: 1) exposition of different opinions, 2) identification of differences, 3) discussion about the reasons for each original assessment, 4) discussion about the different sources of information used, and 5) re-elaboration of individual opinions in posterior elicitation sessions or joint assessment (through consensus) of a common distribution, if agreed by experts.

In the case that a consensus distribution is obtained, that is the final step (before documentation). If further elicitation sessions are needed, the consistency of the opinions is checked again and aggregation is done if acceptable overlap is achieved. Otherwise, the project team should choose what opinions could be aggregated as main opinion of the group (after aggregation), and what opinions should be left as an alternative to perform sensitivity analysis. The main strategies for aggregation are the following ones: 1) linear combination, 2) log-linear combination and 3) Bayesian combination.

Documentation

Documentation of the application must be as complete as possible, including results and description of the ways to obtain them. The contents of the documentation should follow the order of application of the procedure, recording, in each step, *what* has been done, *why* it has been done, *how* it has been done and *who* has done it. In order to achieve this degree of documentation, a schedule of standardised documentation activities should be made for each phase. It should always be completely clear to the reader what is a result assessed by an expert and what results is the outcome of an aggregation, sensitivity analysis or any other analysis not provided explicitly by an expert.

Given the general validity and the wide experience acquired in this type of protocols designed to provide best quality information and PDFs, its inclusion as one more task in the EZ decision making process would be desirable and could improve it.

Reference [60] provides an up to date review of EJ techniques and protocols.

Reference [61] provides a set of applications of EJ in different technology areas. Reference [62] is a recent and good review article about methods to elicit PDFs.

7.3 Uncertainty propagation

In a L2 PSA it is necessary to estimate as accurately as possible all relevant output variables. The full characterisation of a random variable is given by its probability density function or, equivalently, by its integral, the cumulative distribution function (CDF), or its complementary curve to 1 (CCDF). A part of the information contained in those curves is summarised by some numeric statistics like the mean, the standard deviation and order statistics, among others. Additionally, there are several graphics that provide visual information about the shapes of the aforementioned functions.

The Monte Carlo method is based on sampling the vector of input parameters, running the system model computer code for each sample of that vector in order to get a sample of the vector of output variables, and estimating the characteristics of the output variables using the output samples obtained. One of the benefits of using the Monte Carlo method is that many statistical standard estimation methods and tests may be used to estimate the output variables distributions and to test any hypothesis. This makes it the most straightforward and powerful method available in the scientific literature to deal with uncertainty propagation in complex models.

7.3.1 Most commonly used statistics

The most widely used statistics to summarise the information contained in a sample are the sample mean ($\hat{x}$) and the sample standard deviation ($\hat{S}$). They are respectively estimators of the mean and the standard deviation of the variable. Additionally, both of them are used to provide approximate confidence interval for the mean of the studied variable.

Quite important statistics are the order statistics. Given a sample of size n $X=\{x_1,x_2,...,n_n\}$, the order statistics obtained from this sample are $x_{(1)},x_{(2)},...,x_{(n)}$, where the number between brackets stands for the number of order ($x_{(1)}$ stands for the smallest observation in the sample, $x_{(n)}$ stands for the largest observation). There are two important facts related to these statistics: 1) any order statistic $x_{(i)}$ is an asymptotically unbiased estimator of the i/n quantile of the studied variable, and 2) based on their distributional properties, exact distribution-free confidence intervals may be provided for any quantile. Distribution-free means that the confidence intervals generated do not rely on assuming any specific type of distribution for the variable. This is an extremely important result since it allows providing estimates for any percentile (the median, 95%, 99%, etc.) of a random variable and exact confidence intervals (for large enough sample sizes).

7.3.2 Uncertainty range and confidence interval estimation

Usually, Safety Goals are defined in terms of some specific statistical quantities, normally means or some pre-specified quantiles (50%, 75%, 90%, 95%, 99%, 99.9%, 99.99%, etc.). Nevertheless, providing just the estimates of those quantities does not say much about the quantities themselves. One observation is enough to provide an estimate of the mean and two observations are enough to provide an estimate of the median (quantile 50%) and the standard deviation, but, certainly, nobody would trust the estimates provided by samples as small as those mentioned in the last sentence. Thus, the question is: how large should our sample be in order to provide meaningful and believable estimates? The answer to this question is provided by the theory of interval estimation (confidence intervals) and by tolerance intervals results (based on order statistics theory), and depends on what we want to estimate and to guarantee.

Interval estimation is a part of statistical inference whose objective is to generate confidence intervals for different quantities. A confidence interval of a given quantity, like for example of a mean, a standard deviation, a probability or a median, is, roughly speaking, an interval that will be based on the sampled data. A priori, before computing it, we know that it will contain, with a pre-specified probability, the quantity to be estimated. As previously mentioned, provided large enough samples, exact confidence intervals may be provided for any quantile. The case of the mean is completely different, exact confidence intervals are available only for a few distributions, such as normal or exponential distributions. For any other distribution, only approximate intervals are available, which are based on the asymptotic normality of the mean. Such property is not easy to demonstrate.

A tolerance interval (L,U), usually determined by the maximum and the minimum observations in a sample, is defined as an interval for which we can guarantee that, with a pre-specified probability (β), a given proportion (γ) of the random variable is contained within them. In mathematical notation this is

$$\Pr\left(\int_L^U f_x(x)dx \geq \gamma\right) = \beta \qquad (3)$$

In this sense we speak about, for instance, a 95%/99% tolerance interval ($\gamma=0.95$, $\beta=0.99$), as an interval that will contain 95% of the values of the random variable under study with probability 99%. The problem to be solved is how large should the sample be to guarantee with probability (at least) β that a proportion (at least) γ of the studied variable is contained between the limits L and U to be estimated. 'At least' comes from the fact that is impossible to fit exactly those to conditions imposed, so that we try to be on the safe side. Tolerance intervals may be two-sided or one sided. A two sided interval is obtained when no restriction is imposed on U and L; a one-sided interval is obtained when either U or L are fixed (either to a physical threshold or to $\pm\infty$). The following table shows minimum sample sizes to get such tolerance intervals.

	Two sided tolerance intervals			One-sided tolerance intervals		
B \ γ	0.90	0.95	0.99	0.90	0.95	0.99
0.90	38	77	388	22	45	230
0.95	46	93	473	29	59	299
0.99	64	130	662	44	90	459

Table 7.1. Minimum sample size for one and two sided tolerance intervals for different values of β and γ.

7.3.3 Variance reduction techniques

In many actual applications, applying simple Monte Carlo may be prohibitive, conditional on the code used. In those cases, variance reduction techniques could be of use to reduce the computational costs. Variance reduction techniques allow the user to get the same accuracy with a lower computational cost. Main techniques are Latin Hypercube Sampling (LHS), stratified sampling, control variates, importance sampling and antithetic variates, among others.

7.3.4 The use of surrogate models

Some of the computer codes used to simulate severe accidents or some L2 PSA models are very expensive in computational terms, so that, in many cases, it is unthinkable to run them as many times as needed to get accurate probabilistic results. In those cases it could be of interest to seek for good predictors that could be used as surrogate models (or synonym "response surface" or "metamodel") to be run instead of the real models.

7.4 Sensitivity analysis

Sensitivity Analysis (SA) is a formal task included in PSA, though it is not actually implemented in PSA codes. PSA practitioners with a good background in statistics and sensitivity analysis apply these techniques to PSA results with the help of SA specific software. SA techniques may be used pursuing different objectives, all of them related to getting knowledge about the behaviour of the model/system studied, in other words, related to getting information about the input-output relation. For examples, they could provide guidance as to where to improve the state of knowledge in order to reduce the output uncertainties most effectively, to steer research and development efforts, or better understand the modelling, or to obtain a good confidence in the results (potentially large uncertainties). Reference [63] is a good introduction to the different methods of SA and their use in industry.

They may be divided into numerical and graphical techniques, and may in many cases be used simultaneously, using the same data set. There are many different SA techniques that may be divided according to different criteria. Let us divide them into the following types:

- Regression based techniques;
- Non parametric statistics used to identify relations between regions of input parameters and output variables;
- Variance based techniques;
- Distribution sensitivity techniques.

All these methods are "complementary" and could be used according to the characteristics of the physical models and the statistics of interest. All these methods assume that the random input variables are statistically independent.

7.4.1 Regression based techniques

These techniques are based on using a sample (inputs plus outputs), obtained either by random sampling, or LHS or stratified sampling, though they could also be based on data obtained when applying design of experiments concepts, and try to identify if there is any linear, monotonic or polynomial, possibly considering also interactions, relation between inputs and outputs. Main tools are:

- Pearson correlation coefficient, which identifies linear relations between one input parameter and one output variable;
- Spearman rank correlation coefficient, which identifies monotonic relations between one input parameter and one output variable;
- Partial correlation coefficients (PCC) and standardised regression coefficients (SRC), which identify linear relations between inputs and outputs taking into account the effect of correlation between inputs;
- Partial rank correlation coefficients (PRCC) and standardised rank regression coefficients (SRRC), which identify monotonic relations between inputs and outputs taking into account the effect of rank correlation between inputs.

In all these cases, the validity of the indices obtained relies on the quality of the linear (or monotonic) approach adopted. If such hypothesis is not right, other more suitable techniques should be used. In the case of the PCCs and related statistic, a measure of the goodness of the linear (or monotonic) hypothesis is the coefficient of determination (R^2). Coefficients of determination close to one indicate good global validity of the hypothesis, which means that the sensitivity indices are reliable in that case. Coefficients of determination close to zero indicate lack of validity of the hypotheses.

7.4.2 Non-parametric statistics based techniques

There are several statistics commonly used in data analysis that may be oriented to study specific relations between different regions of input parameters and output variable. In some cases, there is no clear trend in the behaviour of an output variable with respect to a given input parameter, but there could be some close relation between, for example, the 10% lowest observations of one of them and the 15% of largest observations of the other one. This kind of relation, under some circumstances could be not easy to be detected by regression-based techniques, while they are easily detected by non-parametric statistics. In what follows there is a list of those amongst the most useful:

- Mann-Whitney test [63];
- Smirnov test [63];
- Kruskal-Wallis test [64].

7.4.3 Variance based techniques

Variance decomposition for computer results is based on Sobol's theorem [65] of decomposition or expansion of any integrable function $f(\mathbf{x})$ in the n-dimensional unit hypercube (K^n). This expansion was named '*expansion of a function into summands of different dimension*' or *'High Dimensional Model Representation*' (HDMR) by its author. As a result of such decomposition, ways to estimate the contribution of main effects and interactions of different order may be studied. There are three well-known techniques to estimate those contributions to the global variability:

- Correlation ratios [63]. This is the cheapest of these techniques in computational term, at least for models with many input parameters, but it allows to estimate only the contribution of main effects;
- Sobol's sensitivity indices [63]. This is the most expensive technique, but

allows to compute all the contributions;

- Fourier Amplitude Sensitivity Test (FAST) [63].

7.5 Uncertainty combination

Uncertainty evaluation has specific aspects if performed for a series of codes used in PSA, as it is usually the case. L2 PSA and L3 PSA use both results from the L1 PSA and, respectively L2 PSA combined with specific inputs for those steps in PSA evaluation, and also different codes combined between them. This can result in a more realistic situation defined in Figure 7.1.

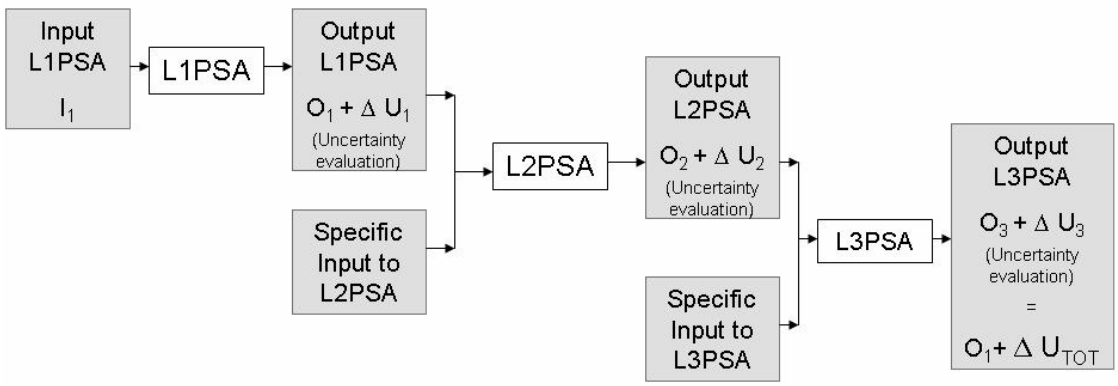


Figure 7.1. Uncertainty calculations for the case of using different codes in PSA phases.

It is assumed in this case that specific approaches for the evaluation of the total uncertainty are defined, as for instance in formula (4) below, in which a calculation of the output of L3 PSA is connected with the output from calculations for L1 PSA and uncertainties at each phase L2 PSA and L3 PSA:

$$O_3 + \Delta U_3 = O_1 + \Delta U_{TOT} = O_1 + f(\Delta U_1, \Delta U_2, \Delta U_3) \quad (4)$$

where f could be a combination (as for instance square root sum) of all the uncertainties from all levels of PSA.

8. PSA BASED & DETERMINISTIC NPP EZ

8.1 Some General Aspects of PSA Application for EZ

The use of PSA approach as a model and its methods, as included in the existing standard procedures and tasks already highlighted in previous chapters, is driven for any PSA application by a series of generic and specific aspects. In this section the main generic aspects having a high impact on the use of PSA for EZ are summarized.

1. **PSA goals and objectives – status and limitations**

The PSA objectives and context are of high impact for its use on any application, including for EZ. In [47; 49] a set of results for various risk metrics in PSA studies is presented for all the period since early 1980's. These surveys and the information on PSA referred in previous chapters present the PSA studies status. PSA studies are performed for various objectives and goals and with various limitations. Their intended use for various applications is also very diverse.

Therefore, for all those situations there are some limitations well known for PSA, which have a direct high impact if they are to be used for EZ application. Some of those limitations are as follows:

- The environmental and demographic aspects have limitations for an individual country, but also could be evaluated diverse approaches in different countries, which for cross-border accidents pose a serious problem of initial common databases.
- The calculations performed for a given plant have limitations and not yet commonly agreed approaches even if the question is about the same type of plant, as follows:
    - Environmental source term and emergency plan;
    - Approach to the consideration of core melt in shutdown states with open containment or containment bypass;
    - Modelling of high pressure core melt is considered;
    - Modelling of large early releases resulting from containment failure;
    - Modelling of the mitigation of low pressure core melt and vessel melt-through;

    - Interface between non PSA approaches and other methods used for the identification of severe accidents from methodology and acceptance criteria point of view.

- PSA technique issues in the use for EZ application require also evaluation of issues common to any approach in modelling severe accidents, as follows:

    - In-containment source term and radiological releases;
    - Reliability of passive systems;
    - Containment by-pass;
    - Hydrogen risk,
    - Assumptions on corium cooling;
    - In addition, some specific PSA type issues are to be solved, too, as for instance:

        - Definition of the PSA level and hence the risk metrics to be used;
        - Definition of scenarios and their evaluation in separate calculations;
        - Qualification of systems for severe accidents;
        - Requirements to probabilistic and risk metrics in the given regulatory environment.

In this context it is important to mention that the objectives of PSA, for instance of level 2, as described in [49] and which have previously been mentioned in section 5.4 of this report, will not change if they are to be used for EZ application.

2. PSA metrics

As it was shown in the previous paragraph, the present status of PSA and hence the level of PSA performed are expected to have an important impact on the EZ application. The existing situation of PSA studies is summarized in [49] for the whole period since PSA started to be developed. As it is shown in Figure 8.1, there was a continuous change of requirements to risk analysis and thus a certain evolution of risk metrics can be noticed. By risk metrics it is understood further mainly CDF (as the main result from L1 PSA), LERF (as the main result from L2 PSA) and risk (as the main result from L3 PSA).

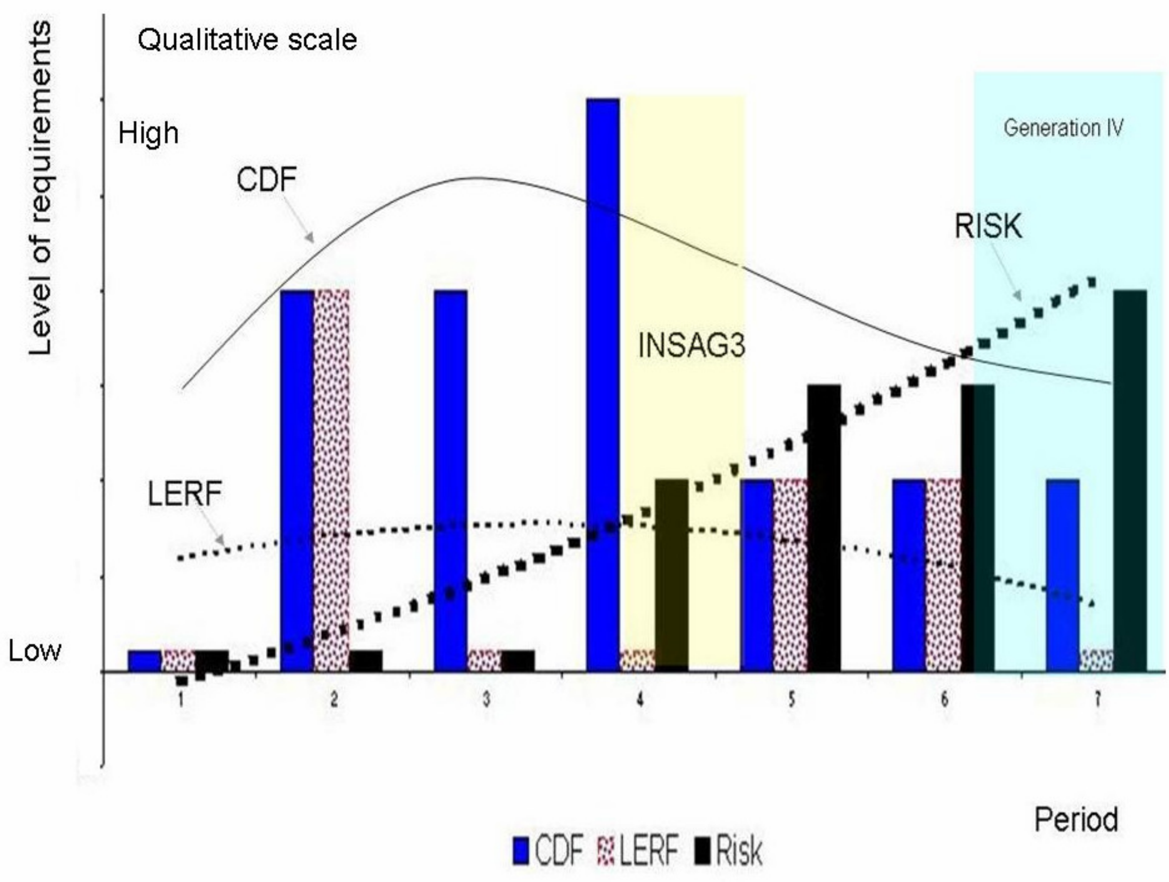


Figure 8.1. Sample representation based on PSA published results [49] of the evolution of risk metrics.

Figure 8.1 is a sample representation of the evolution of risk metrics during the years (split into 7 periods since 1980 by now) as results from the PSA published results [49]. The evaluation of the level of requirements is done in a qualitative manner as represented on the vertical axis. The results show the possibility during all the periods to use various risk metrics in order to define requirements and targets for PSA analyses and their use.

However, if it is possible to have a choice on which risk metrics to use, it is also important to consider the fact (as shown later in this chapter) that the risk metric should be preferred.

The probabilistic criteria in the sense of probabilistic targets, which are defined in INSAG 3 (shown in comparison with the PSA studies published results in Figure 8.1) requirements are set for the risk metrics so that :

- The cumulated frequency CDF leading to a core melt has to be kept below 10^{-5} per reactor year for all plant states, considering internal as well as external events;

- The cumulated frequency of sequences leading to unacceptable releases (i.e. in excess of maximum allowable releases for design extension conditions) has to be kept below 10^{-6} per reactor year;

- The cumulated frequency of sequences leading to early containment failure or to very large releases – LERF has to be kept below 10^{-7} per reactor year.

In some specific cases [53] a set of risk based criteria are set by the regulatory environment and this is illustrated in Figure 8.2.

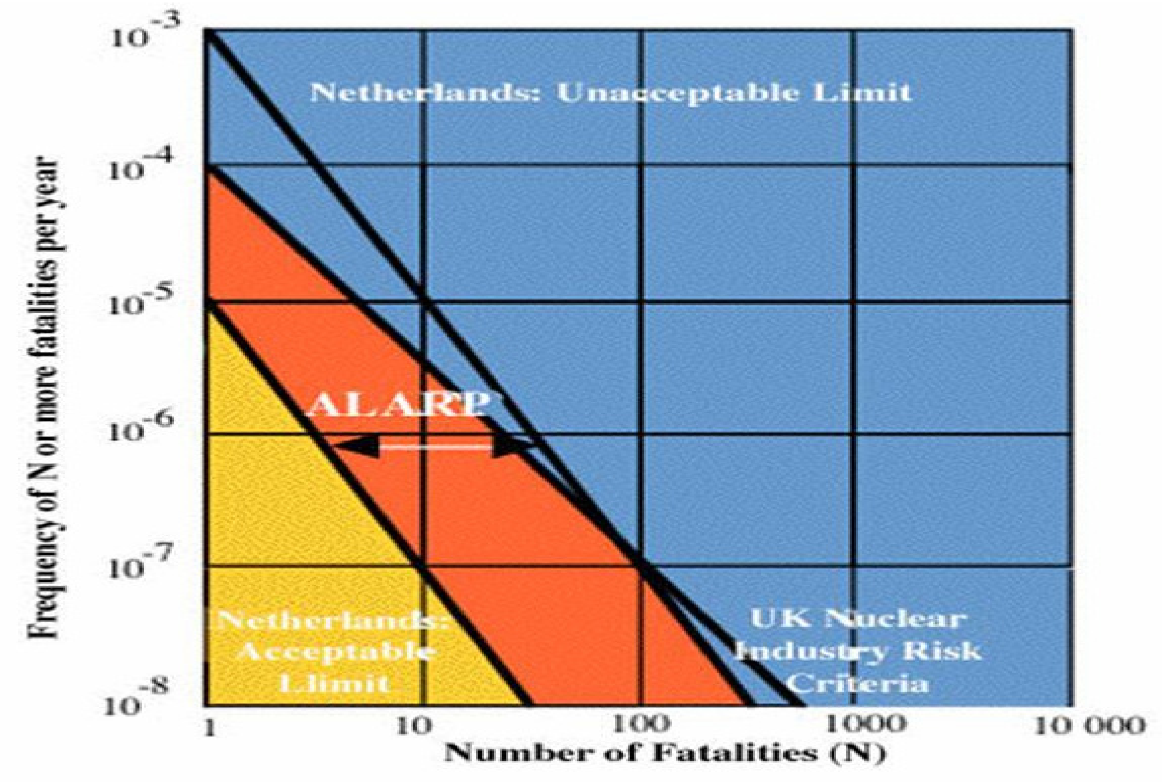


Figure 8.2. Risk levels for individuals in a Critically Exposed Group [53].

For all the world regions preparing for next generations of NPP it is expected to see a general move towards more intensive use of the risk as a risk metric for all NPP aspects and applications. In this context, for instance in USA expected future developments are (as shown in [48]) to move to the use of safety goals including Quantitative Health Objectives, which require a more extensive use of risk for all NPP aspects including specific applications of PSA. It is expected to have this evolution as part of the existing risk informed regulatory environment.

In some cases foreseen for future requirements, in Europe (like in EUR requirements case illustrated in more details in section 5.3 of this report) the use of risk targets and much stronger other risk metrics is expected to lead to no needs for expected EZ. However, this still definitely will be a matter of future analysis and clarification.

The clarifications and refinements for decisions on EZ radii are needed for various situations and aspects to be considered, as follows:

- The requirements for new reactors built at new sites;
- The requirements for new reactors built at old sites, close to other reactors of old generations, for which the EZ radii have already been defined;
- The requirements for new NPP with multiple units at the same site;
- The agreed policies for EZ of NPP (new and/or old) in areas affecting more than one country.

8.2 Some Specific Aspects of PSA Application for EZ

Assuming that the generic aspects of PSA procedures are considered as discussed above and illustrated in previous chapters, then the next step in the use of PSA for EZ application is to define how specific tasks of PSA level 1, 2 and/or 3 are applicable and which are the differences (if any).

In this section it is considered that, in principle, the tasks of PSA are applied as defined by standards for each level of PSA without modifications in order to use them in EZ application.

However, some of the tasks need either special attention or some modifications for such a case. The next part presents those specific aspects for the tasks, which are considered to be of higher impact for EZ applications and also some details on how some of the tasks in PSA have to be performed.

The tasks will be coded as ***Task PSA_EX_X***. The coding is used in order to underline the tasks which are important and to which more attention should be devoted.

There are also some references not only to NPP of generation II+ and III, but also to generation IV.

PSA starts by considering diverse and all sources of radiation and all scenarios challenging them, and therefore, it is highly suitable for EZ application. The results and insights from L2 PSA, in the format of LERF calculations based on various scenarios combined between L1 and L2 PSA in a process called “binning”, which is presented at PSA_EZ_1 below, lead to a conservative envelope of the EZ parameters.

This process is possible by application of the PSA procedure, which combines inputs from source term evaluation with containment impact - in event trees for containment, CETs, and by including results on phenomenological evolution of various scenarios calculated in the severe accidents codes, as it was described in previous chapters.

In fulfilling all below tasks for EZ application, no major change from standard procedures is expected. On the contrary, it is expected that the PSA approach of addressing all scenarios and challenges might be highly beneficial, providing more conservatism in comparison with the deterministic evaluations.

The logic of combining initiating scenarios and end states of containment and the final proposal of source terms might be the most important specific set of tasks from L2 PSA, making the difference between the deterministic and probabilistic approaches in EZ application.

Task PSA_EZ_1: Source Terms Evaluation

The identification of radioactive sources, of the timing of the release, of the quantity and chemical form of radioactivity released and the modelling of dispersion inside containment is a very important part requiring special calculations. In case of this EZ task a special attention is allocated to the choice of the source of radiation and the scenarios postulated. The PSA approach could bring, as a new part in this task, the possibility to evaluate more comprehensively all the range of initiating events (as postulated in PSA) and also to perform a series of severe accident calculations to define and refine the source term parts.

Task PSA_EZ_2: Sensitivity and uncertainty (S&U) analyses in L2 PSA methodology

The S&U analyses might be the next significant specific set of tasks from L2 PSA of high importance for the EZ application. This is due to how the following items are performed:

- Definition of PDSs;
- Number of nodes and endpoints defined in the containment event trees;
- Number of source terms and release categories defined;
- The assumptions resulted from the phenomenological codes runs;

- The independent alternative approaches are used in severe accident analyses;
- The independent alternatives perform a correlation between the probabilistic and deterministic descriptions;
- The S&U are actually performed.

Task PSA_EZ_3: Definition of the plant damage states

Definition of fault sequences that lead to core damage, which are identified in L1 PSA are taken forward into the L2 PSA. The groups obtained, called plant damage states (PDS), are defined in terms of the attributes that would influence the way that the accident progresses to challenge the containment integrity and to release of radioactive material to the environment. The PDS attributes are specific to the type of reactors (PWR, BWR, heavy water channel type, etc.) as well as also for gas reactors. For generation IV gas reactors, for which there is no sense to consider core damage, but only release categories (RC), binning process is of much higher importance than for LWR. Things are also more sensitive to systematic errors for channel reactors.

The binning rules and results of the binning for PDS are of high importance and need to be subject to careful and independent reviews in order to assure accurate L2 PSA results.

Task PSA_EZ_4: Accident progression analysis

This L2 PSA task model the progression of the accident from core damage to the challenges to the containment and the subsequent release of radioactive material for each of the PDSs by using an event tree approach in the format of CETs or APETs. These event trees need to model all the significant physical and chemical processes, which might be actually the source of potential important systematic modelling errors. Those event trees require also inputs from specialized codes calculations. For the generation IV gas reactors with confinements the release categories defined for the CET are of special importance.

The latest developments in PSA technique also take the advantage of integrated PSA models (including internal and external events, all modes of operation PSA models in one unitary model). This is of special help for the performance of intensive sensitivity calculations, which are considered in order to evaluate the impact of the modelling aspects on the results.

Task PSA_EZ_5: Severe accident modelling

The tasks of L2 PSA related to severe accident modelling are considered also to be subject of intensive review and check. This is mainly due to the fact that the physical and chemical processes that are expected to occur during severe accidents typically involve many simultaneous phenomenological interactions for which detailed experimental information may be sparse or not available and therefore they use mathematical and computer simulation. For the generation IV reactors this is of one of the highest priorities.

Task PSA_EZ_6: Containment performance analysis

L2 PSA quality and accuracy of results potentially to be used in EZ applications depends on the containment performance analysis. For the water reactors of generation II+ and III, a series of containment integrity issues were identified during the experience accumulated so far and they could be found in [47, 49].

Mechanisms challenging the containment function and the containment failure modes were extensively illustrated in previous section 5.4 and in Table 5.4.

Typical gas reactor confinement has, however, other problems and the whole mechanism is different. An illustration of such a confinement is shown in [52]. The difference is given by the energies of the released gas, the radioactivity carried away, and the timing, which have very high impact on severe accident concepts and the definition of EZ. Nevertheless, the process required by this task is the same as the similar L2 PSA task, performed not for EZ application.

Task PSA_EZ_7: Quantification of L2 PSA model

The tasks of quantification in all PSA levels, including L2 PSA are important and related to the accuracy of the models, which are built using various software codes. The PSA models include also assumptions and interface with results from deterministic analyses. The quantification of the frequency of the various sequences from the containment event trees uses the data on frequencies of the PDSs, derived from the L1 PSA, and the conditional probabilities of the event trees. These probabilities include failure of safety systems such as the containment spray system (quantified also using fault trees) structural failures of the containment (quantified using a model of the performance of the structure), and the occurrence of physical phenomena where the split fractions relate to the analyst's evaluation. For the split fractions the numerical values are derived from judgment supported by available sources of information.

After obtaining frequencies for PDS, fatalities are calculated for each release category (in case of generation IV gas reactors), or for PDS (for the water reactors) as shown in Figure 8.3. The results of L2 PSA are then post processed and used for PSA applications as licensing or EZ in the form of fatalities.

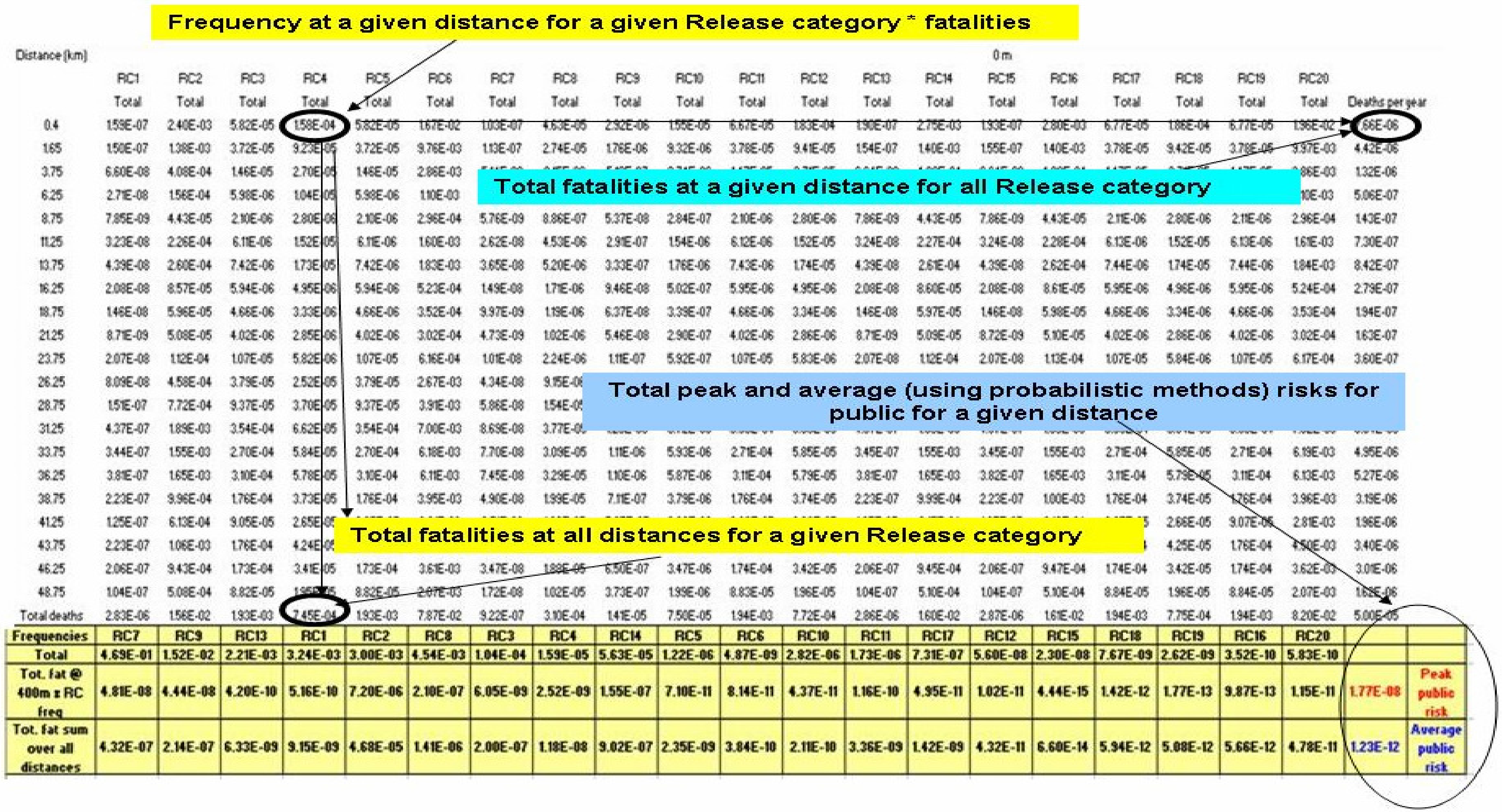


Figure 8.3. Sample representation of the calculation of fatalities and risks for each distance level.

It is important to mention that in Figure 8.3 the summary table for all the release categories and the total fatalities for all distances are already summed and normalized for the risk metric of L3 PSA, because the example is actually illustrating such a case.

Task PSA_EZ_8: Use of computer codes and various models

A significant set of problems has to be solved for new applications in PSA for the computer codes used. The situation is increasingly complicated from L1 to L3 PSA due to the fact that more advanced and higher level codes are used and coupled, that results in dependency of their interface on connecting assumptions.

A special category represents the separate phenomena codes for L2 PSA, which are basically of two groups as it was mentioned in section 6.1 of this report.

For each of those codes extensive verification and validation (V&V) was performed for water reactors. Some examples of the V&V actions for MAAP codes are presented in [47, 49] Though for those codes their V&V process is very important, the most important aspect for PSA calculations is to be able to define and perform V&V for all the PSA flow path of the calculations using diverse codes.

As it was shown in [52], in case of performing such calculations for a generation IV gas NPP, there are some very important aspects to mention:

- The error evaluation and uncertainty calculations should consider the fact that a set of codes are used for the full L3 PSA calculation;
- It was established that some diffusion codes have an error variation with the distance from the source (i. e NPP in EZ application);
- Many phenomenological codes are providing results with their own uncertainties and limitations, which have to be considered while being prepared as inputs to other codes;
- There is a need to define a procedure for uncertainty calculation of the whole calculation flow path for the risk metrics adopted in the EZ application.

Task PSA_EZ_9: L3 PSA process

In the L3 PSA, a large number of CET end-points are grouped to provide the interface between the L2 PSA and L3 PSA consequence analyses. This grouping and classification for L2 PSA and L3 PSA interfaces is called also "binning", like the similar action between L1 PSA and L2 PSA. This subtask is of utmost importance for the PSA results and subject to extensive sensitivity analyses.

The flow path of L3 PSA as shown in [52] in a format of a series of code calculations and other assumptions, and this aspect is not usually mentioned. However, the definition of the calculation sequences and the codes to be used is one of the most important in order to obtain the risk metrics. The results are presented usually in risk metrics (risk for instance) and its uncertainty band.

Task PSA_EZ_10: Use of results and various risk metrics

PSA results are mainly in a form of risk metrics. As it was shown previously in Figure 8.1, there was a certain development of risk metrics requirements during the years. One reason for that is that not all the PSA like risk metrics are suitable for decision making process of many PSA applications.

This statement is fully applicable for EZ, for which the use of CDF is the less desirable and adequate and the use of risk is the best option. This is also illustrated by the latest developments as shown for a case of using L3 PSA in applications similar to EZ [50].

In this case the risk metrics are represented in early fatalities/year, early injuries/ year, latent fatalities/year, thyroid cancer/year, whole-body person-rem/year, based on a series of sensitivity calculations to derive the envelope of the EZ parameters.

PSA calculations are done so that they lead to a reasonable envelope of the risk metrics of various scenarios and this is the main difference from deterministic calculations valid also for EZ applications of PSA. The risk metrics are then represented with the range of their variation for all scenarios [52] for any type of NPP, including generation IV ones, as illustrated in Figure 8.4.

If the dependence of the risk metrics of a large set of parameters is considered, then one can actually obtain a set of acceptable risk surfaces as shown in Figure 8.6.

To conclude on the use of various risk metrics, Figure 8.4 shows that the applicability of L3 PSA risk metrics to NPP EZ is much better than L2 PSA, while L1 PSA risk metrics is not expected to be of some help for the definition of EZ.

8.3 Evaluation of PSA Results for EZ

PSA results for risk metrics as decided by the analysts (but considering the limitations mentioned above) can be used in order to evaluate parameters important to EZ like for instance PAZ and UPZ.

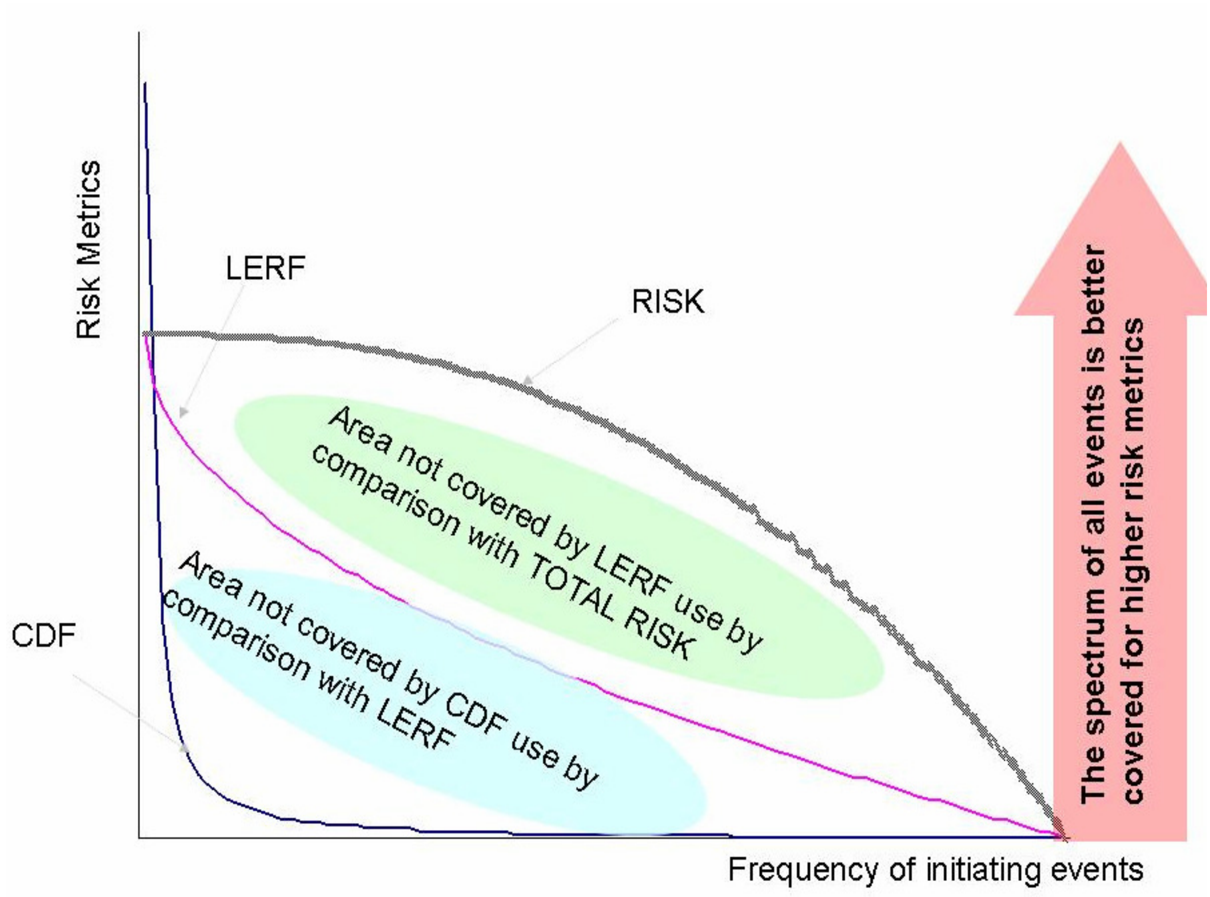


Figure 8.4. Sample representations of the PSA risk metrics limitations for the EZ application use.

Since PAZ and UPZ should be roughly circular areas around the facility, the results should be represented in a corresponding format. The PSA calculations are practically able to evaluate suggested PAZ and UPZ radii.

Task PSA_EZ_11: Use of PSA results for defining NPP EZ

PSA application for EZ includes the modelled barriers and scenarios aspects, common in nuclear safety for any kind of analyses (deterministic or probabilistic) as for instance DBA, BDBA, SA, fission product characteristics, meteorological considerations, exposure pathways, adverse health effects, and avoiding adverse health effects.

PSA performs evaluation of risk metrics considering all those aspects but using the strengths of the PSA method able to derive an envelope of all the challenges to the installation (initiating events) in one single unitary and systematic approach. However, there are limitations due to PSA performance and methodology, specific to each country and group of users, which could produce supplementary difficulties in the interpretation of PSA results for applications like EZ. For example, grouping of NPP events including accidents by frequency of their occurrence differs in different countries.

Nevertheless, as shown in Figures 8.5, 8.8 and 8.9, the expected PAZ and UPZ are distributed within a range of values. In order to decide on the final values, more information is needed to be available for the decision makers.

It can be also mentioned, as shown in [52] that practically there is no expected fundamental difference for the calculations of EZ parameters of radii in case of a gas NPP of generation IV in comparison with a water reactor NPP. This is true even if decision on whether to have or not PAZ/UPZ and which are to be their magnitudes is still a debated issue.

For the sake of underlying the computational aspects of the radii in a deterministic like approach versus a probabilistic like approach, a set of simplified (but not superficial) formulas can be derived as per (1) to (3):

$$Rad_d = S_d * R_d * C_d * Diff_d * D_d \pm \Delta U_d \qquad (1)$$

$$Rad_p = S_p * R_p * C_p * Diff_p * D_p = \qquad (2)$$

$$\cong S_d * R_d * C_d * Diff_d * D_d * \int f_1(S_p) * f_2(R_p) * f_3(C_p) * f_4(Diff_p) * f_5(D_p)\,dx \pm \Delta Up$$

$$Rad_p \cong Rad_d * F_1(S_p) * F_2(R_p) * F_3(C_p) * F_4(Diff_p) * F_5(D_p) \pm \Delta U \qquad (3)$$

Where

S_d Source term in deterministic approach
R_d Reactor failure criterion in deterministic approach
C_d Containment failure criterion in deterministic approach
$Diff_d$ Diffusion criterion in deterministic approach
D_d Fatalities criterion in deterministic approach
S_p Source term in probabilistic approach
R_p Reactor failure criterion in probabilistic approach
C_p Containment failure criterion in probabilistic approach
$Diff_p$ Diffusion criterion in probabilistic approach
D_p Fatalities criterion in probabilistic approach
$\Delta U_{d,p}$ Uncertainties in deterministic, respectively probabilistic calculations
ΔU Final total uncertainties
$f_1(S_p)$, $f_2(R_p)$, $f_3(C_p)$, $f_4(Diff_p)$, $f_5(D_p)$ Distribution functions for the probabilistic criteria
F_{TOTAL} Convolution of f1 to f5.

Combination cases (where D and F are weather categories)	50mSv D75% 1day	50mSV F100% 1 day	50mSv D75% 7day	50mSv F100% 7day	500mSv D75% 1day	500mSv F100% 1day	500mSv D75% 7day	500mSv F100% 7day
Abbreviations for the cases represented in Fig. 8.5	UPZL_50D1	UPZU_50F1	UPZBE_50D7	UPZEXU_50F7	PAZEXL_500D1	PAZBE_500F1	PAZL_500D7	PAZU_500F7

Table 8.1. Codes for the EZ calculations in Figure 8.5.

For the cases represented in Table 8.1, a representation of PAZ and UPZ is shown in Figure 8.5.

The calculations from probabilistic point of view require combination of all the probabilistic criteria distributions, which is done by calculating convolution integral as shown in Figure 8.6.

If the calculations have been performed for generation IV reactors, then there are not expected any changes in the type of results.

Final results of L1, L2, or L3 PSA are actually represented by a set of surfaces within a certain error band, as a function of the probabilities of events and parameters governing the model, as shown in Figure 8.7.

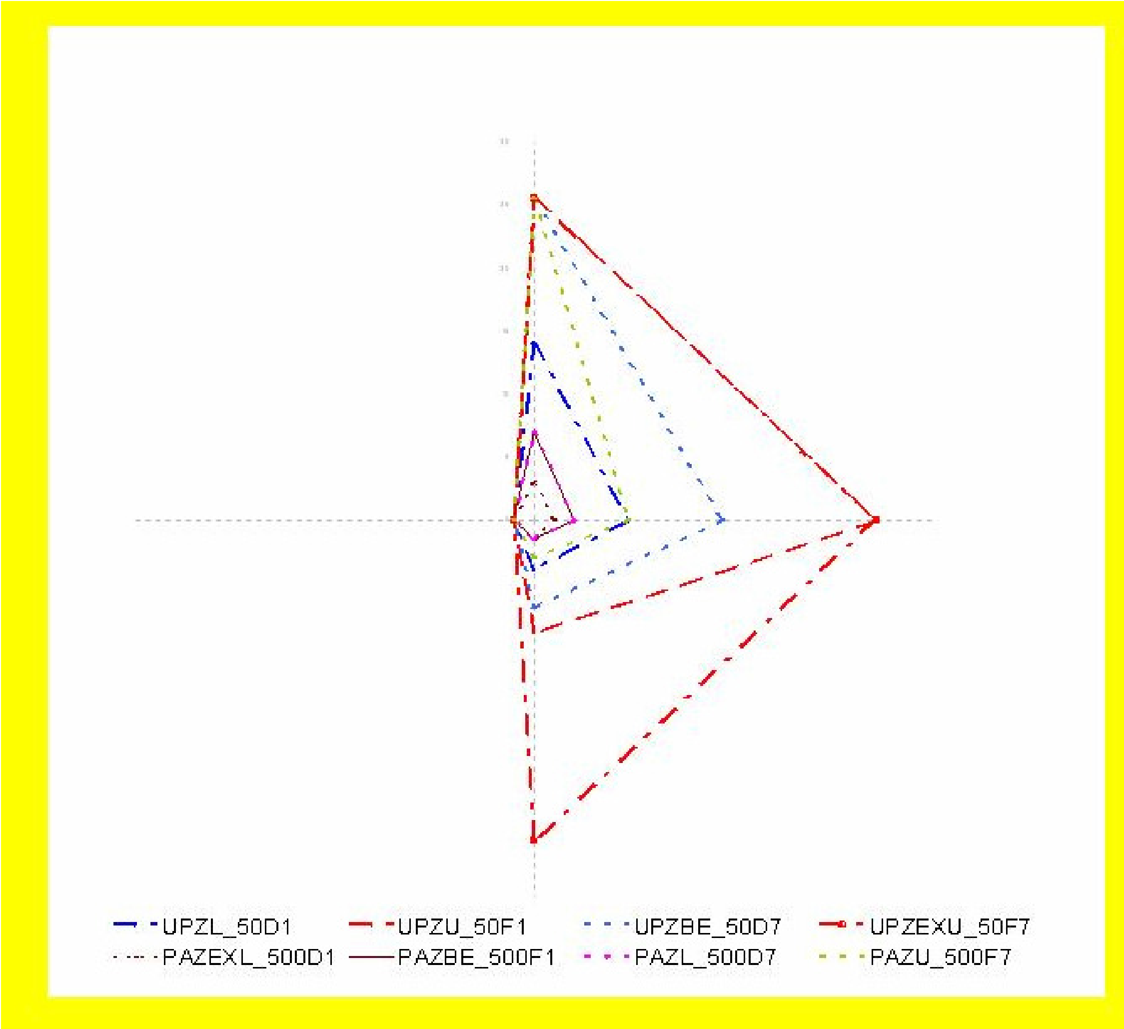


Figure 8.5. Sample representation of the PSA risk metrics limitations for the EZ application

After performing those calculations, the results are obtained in the form from (4) with some uncertainty band and a certain connection with the expected deterministic like result:

$$Rad_p \cong Rad_d * F_{TOTAL} +/- \Delta U \quad (4)$$

The calculation of the convolution integral is embedded in the PSA codes calculation and the flow of calculation was already shown in [52]. The formulas shown above are illustrating the fact that there is a traceable connection between the deterministic type of results and the probabilistic/risk metrics ones.

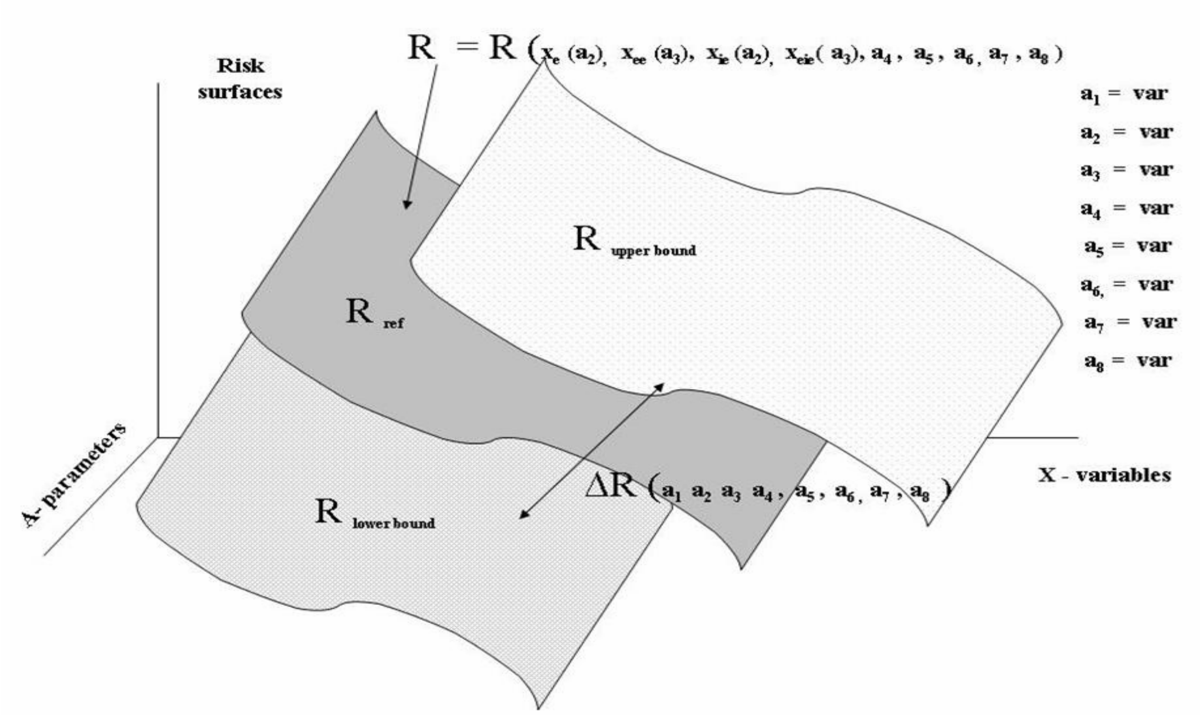


Figure 8.6. Sample representation of the calculation of convolution integral and the surface of risk results generated in L3 PSA.

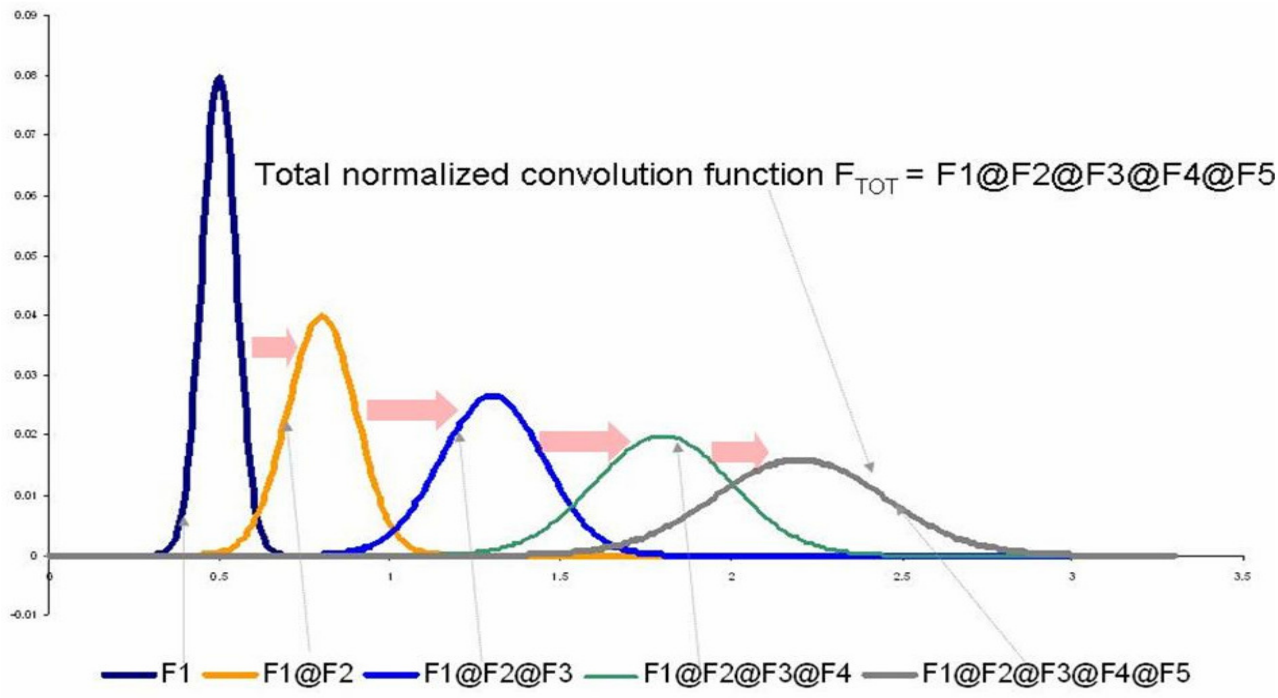


Figure 8.7. Sample representations of the calculation of convolution integral and the surface of risk results generated in a L3 PSA.

8.4 Conclusions & Comments to the Discussed Specific Aspects

In the previous paragraphs there were illustrated some specific aspects and details of implementing PSA for EZ application, including some samples of PSA practical results. However, it is of the highest importance to mention that obtaining risk metrics based EZ parameters does not constitute the end of the EZ application in PSA approach.

On the contrary, if the PSA based results are not using a specific approach in reasoning, which is called “risk informed decision making” (RIDM), then the conclusions could be fundamentally wrong. In order to apply RIDM one has to use logical connectors between deterministic (D) probabilistic (P) interface (F) and correlation statements (Rg) to be included in a set of decision tables

The decisions should consider the fact, that deterministic and probabilistic areas for their inherent best applications are different.

The important aspects to be noted in relation to the use of PSA like results in the decision making process based on the use of decision tables is (as shown in [52]) that it is highly recommended to use a risk informed type of approach in formulating the final decision. This is due to the fact that risk results require probabilistic type of inferences in the judgements to build decision tables. This involves also a very clear description of the limits and strengths of deterministic and probabilistic results for EZ parameters.

Based on the results of combination of various approaches (optimistic, pessimistic, etc.) using insights from all methods, i.e. deterministic and probabilistic, a decision on the EZ parameters can be taken.

To summarize, it is highly recommended or even ***necessary to consider deterministic and probabilistic approaches being complementary.***

An example of formulation of results interpretation of the EZ parameters by using different approaches, i.e. deterministic and probabilistic, and for various events and for various risk zones could be as follows [54]:

- If the decision is aimed at evaluating high foreseen risk situations above the acceptable limits, then the deterministic pessimistic statements may lead to the most conservative decision, even if that happens under less credibility than for the probabilistic ones. On the other hand, due to other reasons than technical ones, the deterministic based decisions could be expected.

- If the decision is aimed at evaluating high or moderate foreseen risk situations below the acceptable limits, then there is no difference between the very pessimistic way of thinking and optimistic one, or a probabilistic one. However, there is an exception based on the fact, that the probabilistic evaluation has more credibility, which could make it the best option to choose for the decision.

- If the decision is aimed at evaluating low and very low foreseen risk situations below the acceptable limits, then it may be based on the probabilistic approach, giving the fact that it generates the most conservative results with highest credibility. Evaluation of risk impact using extensive sensitivity cases is one of the key issues to support the probabilistic type of thinking and its more extensive use in decision making process. This is integrated in the verification and validation process, of which independent review and benchmarking play a very important role in confirming the truth value of probabilistic statements.

In a geometric representation that means, that the EZ radii could be illustrated as a set of spectrum available values from low bound to upper bound with a certain best estimate set of values, as shown in Figures 8.8 and 8.9.

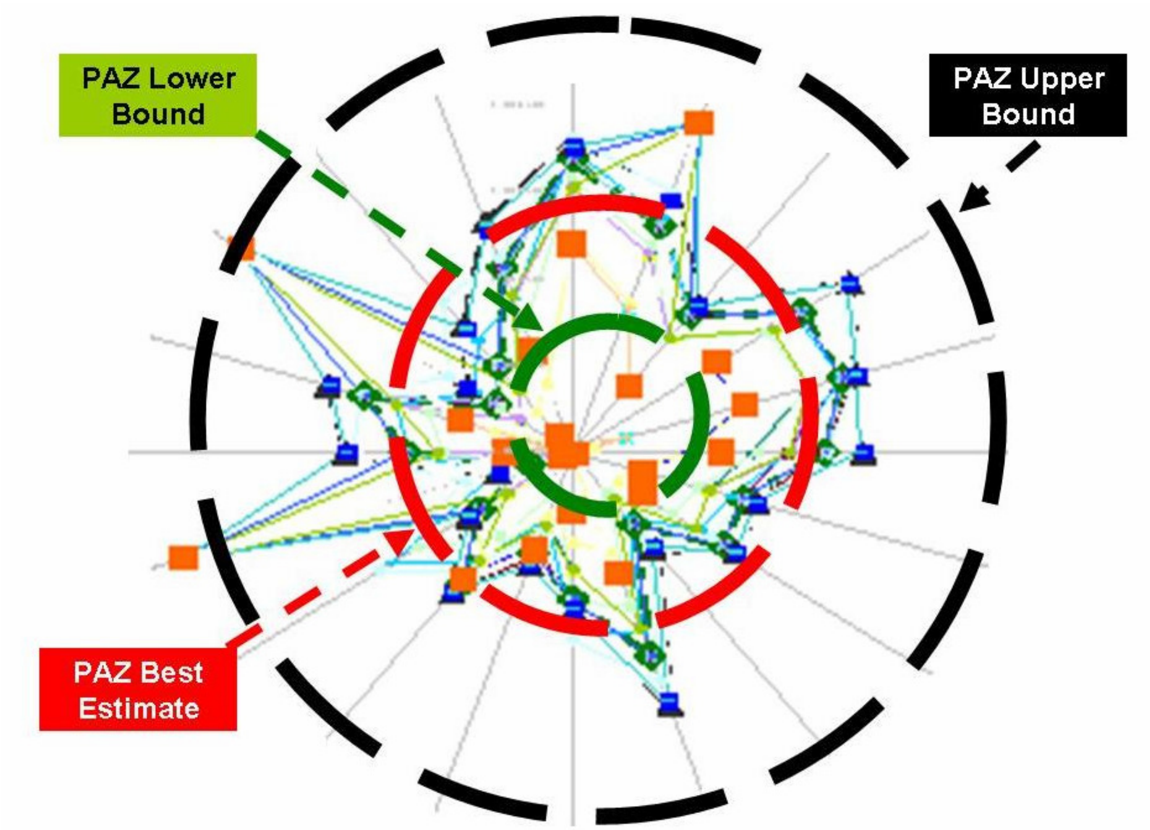


Figure 8.8. Sample representations of the EZ radiuses calculated based on PSA sensitivity analyses.

The last very important point of the performed evaluations is related to EZ parameters of multiple NPP units from various generations on the same site, as shown in Figure 8.10.

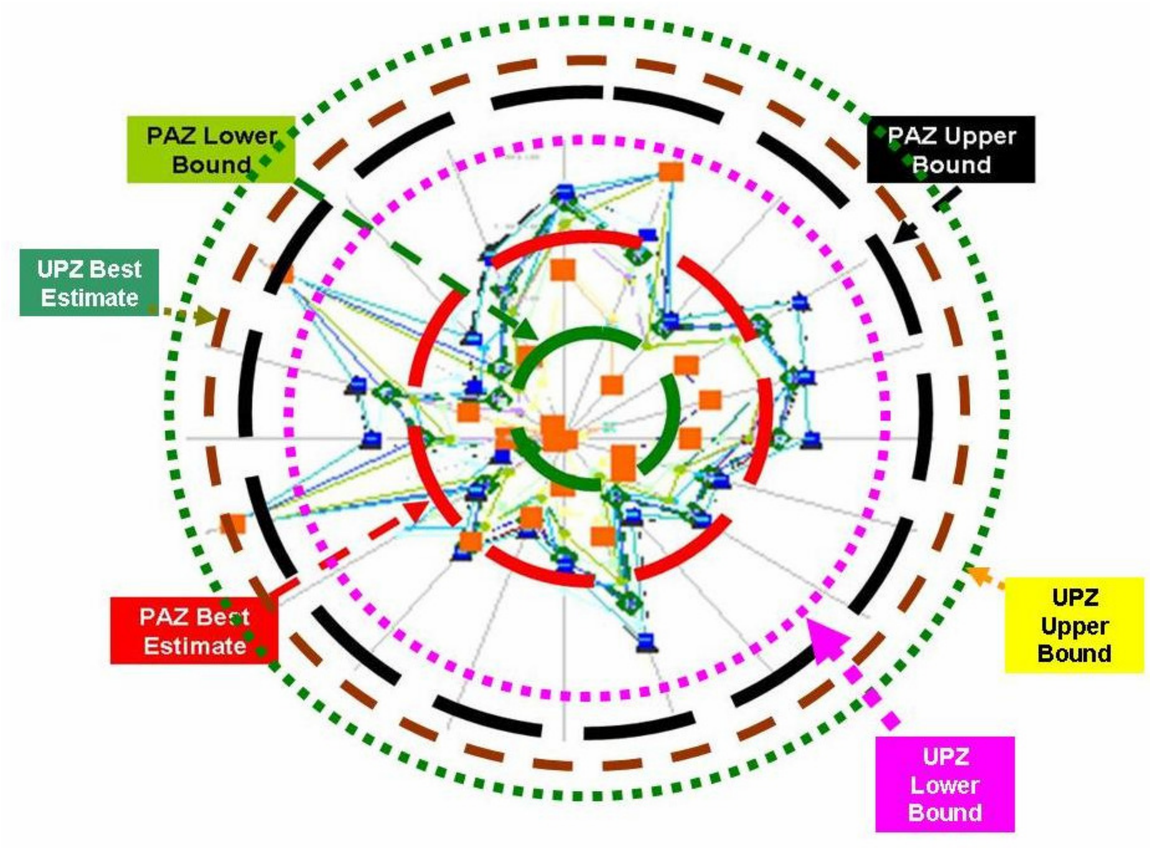


Figure 8.9. Sample representations of the EZ radiuses calculated based on PSA sensitivity analyses.

As a conclusion on the practical aspects of using PSA results for EZ application, it is considered a highly recommended approach that the evaluations performed using *deterministic and probabilistic analyses have to be considered complementary,* given the strengths and limits of each of them.

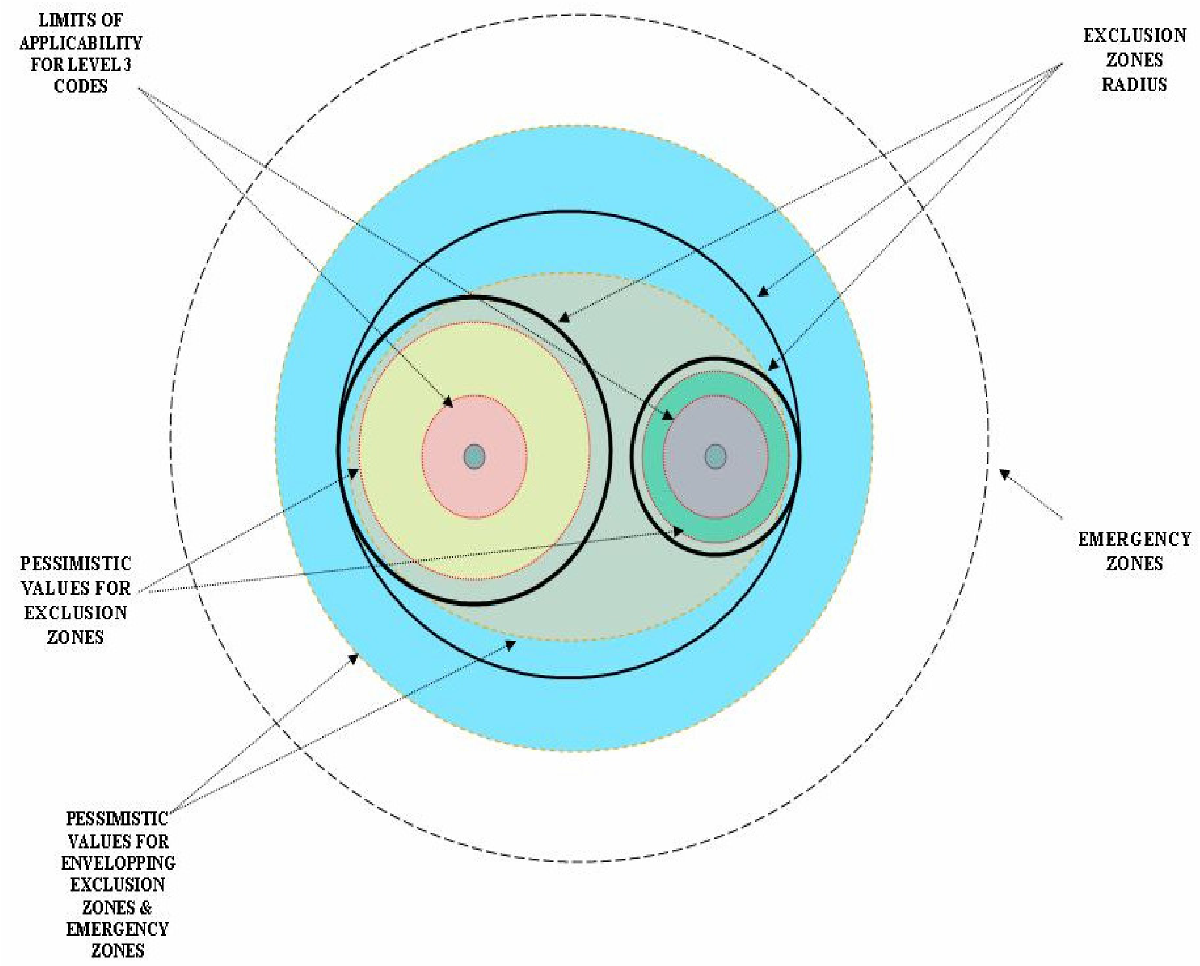


Figure 8.10. Sample representations of the EZ radiuses calculated based on PSA for a plant site with multiple NPP units [52].

9. CONCLUDING REMARKS

One of the key challenges in dependable RIDM is the reconciliation of PSA results and insights with traditional deterministic safety analysis. This is particularly true when it comes to defense in depth and safety margins. PSA results may and often conflict with deterministic insights. If a method of reconciling these conflicts is not defined, then RIDM can become deterministic assessment, along with PSA. This results in PSA being an additional layer of requirements rather than a tool for optimised decision making [39].

Within this report, an attempt to reach a balanced approach, using PSA technology as a complementary tool has been done and illustrated on some specific examples, resulting in the realistic, feasible outcome from NPP emergency zoning practice. There is a general agreement that RIDM has the potential to contribute towards maintaining and improving nuclear safety. It can complement the deterministic approach to nuclear safety and maintain the concepts of defence in depth and adequate safety margins. However, RIDM is broader concept than just the use of PSA in NPP applications. RIDM uses the results of PSA as one input to the decision making process, but allows for consideration of other factors, in particular aspects of safety management and safety culture. At present these aspects are included in PSA only to the extent that they are reflected in the plant-specific data used, but they are not explicitly modelled in PSA [40]. RIDM in NPP emergency zoning is a process, which can be used by the utility and the regulator, and provides the framework for risk informed regulation in this area. The objective should be to enhance regulatory effectiveness, using risk information to optimise nuclear safety regulation.

Whether risk informed regulation is of benefit to utilities depends to a large extent on the common understanding developed with the regulatory authorities. Since the preparation of a PSA imposes a considerable burden in terms of the human and financial resources that need to be expended, it is of utmost importance to define clearly what is expected from the utility and how the results will be used. This common understanding can be developed in a dialogue that includes all stakeholders. RIDM would strengthen the perception that the operator is assuming the primary responsibility for safe operation.

RIDM in areas that affect licensee requirements necessitates review (and, ultimately, approval) of PSA and supporting information by the regulatory body. A suitable regulatory framework and regulatory staff with considerable technical capabilities in the areas of PSA and risk informed decision making are prerequisites for such review and approval. This constitutes a considerable burden for countries with small nuclear programmes and limited numbers of regulatory staff [40].

It is necessary to ensure the availability of high quality PSA to support RIDM. The meaning of “high quality” in this context can vary and is defined as being commensurate with the intended use. Several IAEA as well as EU Member States have developed national PSA guidelines, and the IAEA has prepared guidance on PSA quality for applications in NPP at the international level [41].

The American Society of Mechanical Engineers (ASME) has developed a standard on PSA [28, 42]. Additional efforts to promote the production of high quality PSA include peer reviews, establishment of user groups for similar type of plants, pooling of data and preparation of reference PSA [40]. **RIDM in NPP emergency zoning can be successful -** like in other areas **- only if all stakeholders understand the process and the results obtained.**

In addition to the main nuclear regulatory body, a licensee has to deal with several other regulatory organizations, e.g. those responsible for environmental protection. If the concept of RIDM in NPP emergency zoning is not shared by these other authorities, this might complicate the decision making process. Thus, consistency between the approaches followed by different authorities would be beneficial.

Owing to the state-of-the-art understanding and increased characterisation of NPP severe accidents as well as advanced understanding of PSA technology, which can be currently considered mature enough, overall management of NPP severe accidents could be – and also should be - analysed as an integrated complex process. The interrelationship of NPP emergency operating procedures, safety and risk assessments, severe accident management guidelines, and emergency off-site actions should be planned and organized to minimize the consequences of such accidents. This approach might be a contribution to ensure the continued safety of NPPs and to improve effectiveness of regulatory practices in EU Member States.

As the **transition to risk informed regulation** is taking place gradually more or less worldwide, activities conducted within this project represent comprehensive application of PSA technology to contribute to NPP emergency zoning issues.

This report indicates clearly that the current, state-of-the-art **PSA technology** is significantly able to contribute – **as a complementary tool** - to the traditional engineering, deterministic approach to addressing various issues of NPP emergency planning practices, especially emergency zoning and might be highly topical at present in terms of regulatory effectiveness in EU Member States.

And finally, there is one more facet of the subject matter: some safety consequences resulting from economic pressure on NPP operators as a result of deregulation of electricity markets. Although deregulation is not the only reason why nuclear operators have intensified their efforts to reduce costs and become more efficient, it is clear that the industry is changing and that regulators must prepare for this new situation. This report would not like to give outright advice regarding any prioritising. This must follow from the assessment of the **national situation in each EU Member State**. The authors of this report hope that it will be of some help in this assessment and in thorough consideration to the subject.

10. REFERENCES

1. Kubanyi, J., Kirchsteiger, C.: Risk-Informed Support of Nuclear Power Plant Emergency Zoning, Report EUR 21580 EN, JRC Petten, 2005.

2. EUR 20163 EN, TSO Study Project on Development of a Common Safety Approach in the EU for Large Evolutionary Pressurised Water Reactors, European Commission-Nuclear Safety and the Environment; Summary Report, Oct. 2001.

3. IAEA, Method for the Developing Arrangements for Response to a Nuclear or Radiological Emergency, EPR-METHOD (2003), ISBN 92-0-111503-2, IAEA, Vienna,2003,at http://www-pub.iaea.org/MTCD/publications/PDF/Method2003_web.pdf

4. IAEA, Jointly Sponsored by FAO, IAEA, ILO, OECD/NEA, PAHO, OCHA, WHO, Preparedness and Response for a Nuclear or Radiological Emergency. REQUIREMENTS. IAEA Safety Standard Series No. GS-R-2, IAEA, Vienna, 2002, available at http://www-pub.iaea.org/MTCD/publications/PDF/Pub1133_scr.pdf

5. IAEA, Arrangements for Preparedness for a Nuclear or Radiological Emergency. IAEA Safety Standards, Draft Safety Guide DS105 (version 7.0), IAEA, Vienna, 2005, at: http://www.iaea.org/googleResult.html?q=Draft+Safety+Guide+DS105+Version+7.0%2C+Feb.+2005&domains=http%3A%2F%2Fwww.iaea.org%2F&sitesearch=IAEA.org&cx=004828748078731094376%3Am_jpm98tdns&cof=FORID%3A11&x=12&y=6#171

6. Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, NUREG-0654, US NRC, Nov.1980, at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0654/

7. Report RELKO/1R1204 Benchmarking and Harmonising Strategic Planning Practices for Emergency Zoning and Information to the Public (Report prepared for JRC), December 2004, Bratislava.

8. Qu, J., Cao, J., Li, H.: Discussions on Emergency Planning For a Pilot Commercial HTR Plant in China. Paper at EC DG JRC-IE/OECD NEA International Seminar on Emergency & Risk Zoning around Nuclear Power Plants, 26-27 April 2005, Petten, Netherlands.

9. Charpin, F., Raimond, E.: Chaumont, B.: Technical basis for off-site emergency planning in France. Proceedings of the EC DG JRC-IE/OECD NEA International Seminar on Emergency & Risk Zoning around Nuclear Power Plants, 26-27 April 2005, Petten, Netherlands.

10. Perryman, L. J.: Risk informed emergency planning requirements for Koeberg Nuclear Power Station. Paper sent to be presented in EC DG JRC-IE/OECD NEA International Seminar on Emergency & Risk Zoning around Nuclear Power Plants, 26-27 April 2005, Petten, Netherlands.

11. Navert, S.: Determination and Justification of the Reference Scenarios and Associated Source Terms for Emergency Planning in Switzerland. EC DG JRC-IE/OECD NEA International Seminar on Emergency & Risk Zoning around Nuclear Power Plants, 26-27 April 2005, Petten, Netherlands.

12. RASCAL 3.0, Description of Model and Methods, NUREG-1741, US NRC, Washington DC, 2001.

13. Severe Accident Risk: An Assessment for five US Nuclear Power Plants, NUREG -1150, US NRC, Washington DC, 1990, at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1150/

14. Generic Environmental Impact Statement for License Renewal of Nuclear Plants. NUREG-1437, Vol. 1, US NRC, Washington DC, May 2001, at http://www.nrc.gov/reading-rm/doccollections/nuregs/staff/sr1437/v1/part05.html#_1_129

15. INTERNATIONAL ELECTROTECHNICAL COMMISSION, Risk Analysis of Technological Systems, International Standard IEC 300-3-9, IEC, Geneva, 1995.

16. IAEA, Safety Report Series No. 23, Accident Analysis for Nuclear Power Plants, IAEA Vienna, 2002, at http://www-pub.iaea.org/MTCD/publications/PDF/Pub1131_scr.pdf

17. International Commission on Radiological Protection, Principles for Intervention for Protection of the Public in a Radiological Emergency, ICRP Publication No. 63, Pergamon Press, Oxford, 1992.

18. IAEA International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No. 115, IAEA, Vienna, 1996, at http://www-pub.iaea.org/MTCD/publications/PDF/SS-115-Web/Start.pdf

19. Libmann, J.: Elements of nuclear safety. IPSN, Les Editions de Physique, 1996 (ISBN: 2-86883-286-5).

20. Apostolakis, G. E.: How Useful Is Quantitative Risk Assessment? Risk Analysis, Vol. 24, No. 3, 2004, p. 515-520.

21. Hayns, M., R.: The Evolution of Probabilistic Risk Assessment in the Nuclear Industry. Trans IChemE, Vo. 77, Part B, May 1999.

22. Vidard, M., Bassanelli A.: Emergency Planning: EUR Original Approach. Comments and Practical Implementation. EC DG JRC-IE/OECD NEA

International Seminar on Emergency & Risk Zoning around Nuclear Power Plants, 26-27 April 2005, Petten, Netherlands.

23. Shepherd, C., Kouzmina, I.: Rapporteurs' Notes of the Meeting. EC DG JRC-IE/OECD NEA International Seminar on Emergency & Risk Zoning around Nuclear Power Plants, 26-27 April 2005, Petten, Netherlands.

24. Kubanyi, J., Kirchsteiger, C., Vetere, A. L.: Summary and Recommendations Report on the EC DG JRC-IE/OECD NEA International Seminar on Emergency & Risk Zoning around Nuclear Power Plants, 26-27 April 2005, Petten, Netherlands.

25. IAEA INSAG, Basic Safety Principles for Nuclear Power Plants, Safety Series No. 75-INSAG 3, IAEA Vienna, 1998.

26. Kirchsteiger, C., Bolado-Lavin, R.: Current use of probabilistic safety evaluations for advanced reactor concepts. KERNTECHNIK, Vol. 71 (2006) 3.

27. Sandberg, J., Vilkamo, O.: Emergency Zoning and PSA Applications in Finland. EC DG JRC-IE/OECD NEA International Seminar on Emergency & Risk Zoning around Nuclear Power Plants, 26-27 April 2005, Petten, Netherlands.

28. Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME RA-S-2002. An American National Standard, The American Society of Mechanical Engineers, New York, NY 10016, USA, 2002.

29. Kumamoto, H., Henley, E. J.: Probabilistic Risk Assessment and Management for Engineers and Scientists. IEEE Press, New York, 1996. ISBN 0-7803-6017-6.

30. IAEA, Procedures for Conducting Probabilistic Safety Assessment of Nuclear Power Plants (Level 2). Safety Series No. 50-P-8, IAEA, Vienna, 1995.

31. IAEA, Safety Reports Series No. 25, Review of Probabilistic Safety Assessments by Regulatory Bodies. Jointly sponsored by IAEA, OECD/NEA, IAEA Vienna, 2002.

32. NEA OECD, Level-2 PSA for Nuclear Power Plants. CSNI Technical Opinion Papers No. 9, OECD 2007, NEA No. 5322, ISBN 978-92-64-99008-1.

33. Chaumont, B.: Overview of SARNET Progress on PSA 2 Topic; The First European Review Meeting on Severe Accident Research (ERMSAR). Aix-en-Provence, 14-16 November 2005.

34. Cooper, J. R., Randle, K., Ranjeet, S. S.: Radioactive Releases in the Environment. Impact and Assessment. John Wiley and Sons, 2003, ISBN 047 1899240.

35. Nourbakhsh, H. P., Kress, T. S.: Assessment of Phenomenological Uncertainties in Level 2 PRAs. The First European Review Meeting on Severe Accident Research (ERMSAR). Aix-en-Provence, 14-16 November 2005.

36. OECD NEA, Recent Developments in Level 2 PSA and Severe Accident Management. A Supplement to Report NEA/CSNI/R91997)11, OECD NEA, (Draft).

37. Toth, B., Mueller, K., Birchley, J., Wada, H., Jamond, C., Trambauer, K.: Benchmark study on fuel bundle degradation in the PFEBUS FPT3 test using the severe accident codes ATHLET-CD, ICARE2 and MELCOR. Nuclear Technology, Vol. 157, Feb. 2007.

38. Van Dorsselaere, J. P., Schwinges, B.: Overview of the integral code ASTEC V1.3. Note technique/Technische Notiz DPAM/ASTEC-2006-362/GRS ASTEC 06/02. IRSN – GRS, December 2006.

39. Niehaus, F., Szikszai, T., Risk-informed Decision Making, Topical Issues Paper No. 1, Proceedings of International Conference on Topical Issues in Nuclear Safety, IAEA, 3-6 September 2001, Vienna, Austria.

40. Risk-informed Decision Making, Topical Issues Summary. Proceedings of International Conference on Topical Issues in Nuclear Safety, IAEA, 3-6 September 2001, Vienna, Austria.

41. IAEA, Determining the quality of probabilistic safety assessment (PSA) for applications in nuclear power plants. IAEA-TECDOC-1511, IAEA Vienna, 2006.

42. An American National Standard ASME RA-Sa-2003, Addenda to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications. The American Society of Mechanical Engineers, New York, NY 10016, USA, 2002.

43. Jay Wu; Yung-Muh Yang; Ing-Jane Chen; Huan-Tong Chen; Keh-Shih Chuang: Reevaluation of the emergency planning zone for nuclear power plants in Taiwan using MAACCS2 code. Applied Radiation and Isotopes 64 (2006), pp. 448-454, ELSEVIER 2006, at www.elsevier.com/locate/apradiso

44. Perryman, L. J.: EP Evacuation Requirements. Report Eskom Generation RRM04/0156, Revision R1, Eskom Generation, Dec. 2004.

45. SEVERE ACCIDENT RISKS POSED BY THE TEMELIN NUCLEAR POWER PLANT AND SEVERE ACCIDENT MITIGATION ALTERNATIVES. Bericht an die Österreichische Bundesregierung zur Gesamt-UVP Temelin, Juni 2001, at http://www.umweltbundesamt.at/fileadmin/site/umweltthemen/kernenergie/temelin/Melk/GesamtUVP/UVPBericht/Teil5.pdf

46. Additional information to severe accidents and emergency preparedness on nuclear power plant Temelin. *Compiled by State Office for Nuclear Safety, Praha, as response to questions raised by Austrian party during and after the public hearing under item V of the "Melk protocol"in Vienna on 26 June 2001.* Praha, 8 August 2001, at http://www.mzv.cz/EIA/eia/severe2.pdf.

47. OECD, Level 2 PSA Methodology and Severe Accident Management. Prepared by the CNRA Working Group on Inspection Practices (WGIP), CDE/GD(97)198 , Paris, 1997.

48. Durham, F.A., Camp, A. L., Apostolakis, G., and Golay, M.: A Framework for Regulatory Requirements and Industry & Standards for New Nuclear Power. SAND2000-1598C, International Conference PSAM 5, 27 November-01 December 2000, Osaka, Japan.

49. NEA OECD, Level-2 PSA for Nuclear Power Plants. CSNI Technical Opinion Papers No. 9, OECD 2007, NEA No. 5322, ISBN 978-92-64-99008-1.

50. Levinson, S., H., Doug, P.: Beyond level 1 & 2 PSA's: B&WOG's development of level 3 models. International Conference PSAM 6, 23-28 June 2002, Puerto Rico, USA. 0-0804-4120-3, Elsevier Science, 2002.

51. Jae-Joo, Ha: Nuclear Power Reactor Technology. Module 2. Probabilistic Safety Assessment (PSA), KAERI, Integrated Safety Assessment Team.

52. Serbanescu, D.: Sensitivity and Uncertainty Issues in the Integrated PSA Studies, EC DG JRC-IE/OECD NEA International Seminar on Emergency & Risk Zoning around Nuclear Power Plants, 26-27 April 2005, Petten, Netherlands.

53. INTERNATIONAL ELECTROTECHNICAL COMMISSION, Functional safety of electrical/electronic/programmable electronic safety-related systems. International Standard IEC 61508, Geneva, 1998-2000.

54. Serbanescu, D.: Some specifics of the use of probabilistic risk analyses as a support to the evaluation of safety margins and the interface with the deterministic based decisions. IAEA Technical Meeting on Effective Integration of Deterministic and Probabilistic Safety Analysis in Plant Safety Management, Barcelona, Spain, 4 - 8 September 2006.

55. Glaeser, H., Hofer, E., Kloos, M., Skorek, T.: Uncertainty and sensitivity analysis of a post-experiment calculation in Thermal hydraulics. Reliability Engineering and System Safety, 45 (1), pp. 19-33. January, 1994.

56. Efron, B., Tibshirani, R.J.: An introduction to the bootstrap. Chapman & Hall. 1993. ISBN 0-412-04231-2.

57. Johnson, L. J., Kots, S., Balakrishnan, N.: Continuous univariate distributions. Vols.1 and 2. Wiley. 1994. ISBNs 0-471-58495-9 and 0-471-58494-0.

58. Martz, H. F., Waller, R.A.: Bayesian reliability analysis. Wiley, 1982. ISBN 0-471-86425-0.

59. Cooke, R. M., Goossens, L. J. H.: Procedures guide for structured expert judgment. EUR 18820 EN. 2000.

60. Simola, K., Mengolini, A., Bolado, R.: Formal Expert Judgement. An overview. EUR 21772 EN, 2005.

61. Bolado, R., Devictor, N.: Proceedings of the workshop on The use of Expert Judgement in Decision-Making. EUR 21873 EN. 2005.

62. Garthwaite, P. H., Kadane, J. B., O'Hagan, A.: Statistical methods for eliciting probability distributions. JASA, Vol. 100, No. 470, pp. 680-700. 2005.

63. Saltelli, A., Chan, K., Scott, E. M.: Sensitivity Analysis. Wiley. 2000. ISBN 0-471-99892-3.

64. Conover, W. J.: Practical nonparametric statistics. Second edition. Wiley. 1980. ISBN 0-471-08457-3.

65. Sobol, I. M.: Sensitivity analysis for non-linear mathematical models. Math. Modelling Comput. Exp. Vol. 1, pp 407-414. 1993.

11. APPENDICES

List of Appendices

APPENDIX A

Re-evaluation of EPZ for Chinshan NPP in Taiwan

This section provides a summary of the NPP EPZ defining practice as applied in Taiwan. The information used in this Appendix A has directly been taken and adapted from ELSEVIER publication Applied Radiation and Isotopes 64 (2006) [43] [36], because it provides a very good and informative example of practical use of risk informed approach in support of NPP emergency zoning.

According to the government regulations, the EPZ of a NPP in Taiwan must be defined before operation and re-evaluated every 5 years. Corresponding emergency response planning (ERP) has to be made in advance to guarantee that all necessary resources are available under accidental releases of radioisotopes. In that study [43], the EPZ for each of the three operating NPPs, Chinshan, Kuosheng, and Maanshan in Taiwan was re-evaluated using the MELCOR Accident Consequence Code System 2 (MACCS2) developed by Sandia National Laboratory. Meteorological data around the nuclear power plant were collected during 2003. The source term data including inventory, sensible heat content, and timing duration, were based on previous PSA information of each plant. The effective dose equivalent and thyroid dose together with the related individual risk and societal risk were calculated.

An EPZ is considered the area where actions should be taken first to protect the general public when a nuclear accident occurs. The corresponding ERP in the EPZ, therefore, has to be made in advance to ensure that all necessary resources are available to protect the population from radiation exposure. According to the government regulations revised in March 2005 in Taiwan, the EPZ of a NPP must be specified again and re-evaluated every 5 years according to the latest environmental data. Therefore, the EPZs of the three existing NPPs had to be re-evaluated.

The previous, original EPZ results[37] obtained for these three operating NPPs using the CRAC2[38] code were less than 5.0 km radius (3.6, 4.6, and 4.4 km, respectively). Therefore, the government set an EPZ of 5.0 km radius for all three plants. Afterwards, in due time, it was necessary to re-evaluate the EPZ for each plant using the MACCS2 code with the updated population distribution and meteorological data to fulfil the revised regulations. The effective dose equivalent and thyroid dose together with the individual risk and societal risk for each

36 www.elsevier.com/locate/apradiso

[37] Yin, H. L.: CRAC2 evaluation of emergency planning zone for Kuosheng nuclear power plant; Institute of Nuclear Energy Research, INER-T1784, 1993; for Chinshan and Maanshan NPPs in analogous report of Chen et all., 1992 in 1992.

[38] Calculations of Reactor Accident Consequences Code (CRAC) was developed in support of Reactor Safety Study WASH-1400, US NRC, 1975. The updated version, which was released in 1982 incorporated major improvements over the CRAC in terms of weather sequence sampling and emergency response modelling (Ritche, L. T., Jonson, J. D., Blond, R. M.: Calculation of reactor accident consequences version 2 CRAC computer code user's guide. Sandia National Laboratories, NUREG/CR-2326).

category of accidents were evaluated and then weighted to achieve the final outcome. By comparing the results with the Protective Action Guide (PAG)[39] and the related criteria, a reasonable conservative EPZ was proposed for each plant.

The MACCS2 code was used to estimate radiological doses, health effects, and economic consequences that could result from accidental releases of radioactive materials to the atmosphere. The treated phenomena consist of building wake effects, buoyant plume rise, plume dispersion during transport (Pasquill-Gifford dispersion parameters), wet and dry deposition, and radioactive decay and in-growth. Two kinds of doses were calculated: acute dose used for the estimates of early fatalities and injuries, and lifetime dose commitment used for the estimates of associated excess cancer risks resulting from early exposure. The dose calculation for each exposure pathway is spatially variant and is the product of the following quantities: radionuclide concentration, dose conversion factors, duration of exposure, and shielding factors. Evacuation, sheltering, and relocation could have been chosen as the protective actions in this calculation module, but none of them was specified because the most conservative results were required.

The calculations pertaining to the intermediate and long-term phases were also performed. The associated exposure pathways during the intermediate phase are groundshine and resuspension inhalation, and the pathways during the long-term phase are groundshine as well as food and water ingestion. A polar-coordinate grid divided into 16 compass directions with an angle of 22.5° each is centred at the location of the release. The results outputted from the MACCS2 are stored subsequently on the basis of this spatial grid system. Fig. A.1 shows the polar coordinate system built in the MACCS2 code and the numbering system associated with 16 compass directions.

To evaluate the EPZ, some specific data, such as source terms, meteorological data, and population distribution were required. The source terms used in the re-evaluation were identical to those used in the former evaluation using the CRAC2 code in 1992 – 93, based on the preliminary design of the facility. For example, Table A.I shows a sample of some important parameters associated with 15 release categories for the Chinshan NPP (2 x BWR 600 MWe), Table A.II shows a sample of the inventory of 60 radionuclides contained in that facility, and Table A.III shows a sample of the release fractions of nine radionuclide groups of the 60 radionuclides for each release category. The hourly meteorological data, including wind direction, velocity, and stability, were collected at the weather tower inside each plant during 2003.

Dose conversion factors (DCFs) of the 60 radionuclides considered important for NPP releases were required as input data for the MACCS2 code. For the exposure pathways of cloudshine and groundshine, the DCFs were extracted from the database of ICRP[40], and for the pathways of inhalation and ingestion

39 There was not a reference found; however, obviously it is a regulation or a legislative document in Taiwan.

40 Radionuclide transformations; energy and intensity of emissions. ICRP Publication 38, Annals of the ICRP vol. 11-13

the DCFs were adopted from the Federal Guidance Reports 11[41] and 12[42].

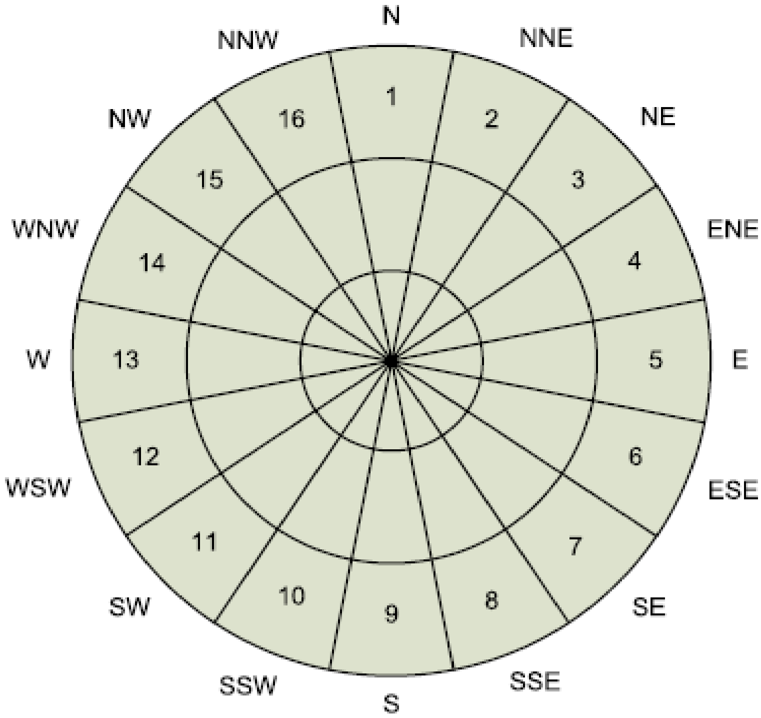


Fig. A.1. The polar coordinate and the numbering system associated with 16 compass directions built in the MACCS2 code.

The MACCS2 code itself is only a consequence modelling code. For the purpose of EPZ calculation, some safety criteria should be provided as a reference to achieve conservative and reasonable results. According to the relevant regulations, the following four guidelines were proposed as the basis:

- The risk of prompt fatality to an individual or to the population in the vicinity of a NPP that might result from reactor accidents should not exceed 0.1% of the sum of prompt fatality risks resulting from all other causes.

- The risk of cancer fatality to an individual or to the population in the vicinity of a NPP that might result from reactor accidents should not exceed 0.1% of the sum of cancer fatality risks resulting from all other causes.

- The anticipated whole body dose and thyroid dose beyond the EPZ should not exceed the national Protective Action Guide levels in the design basis accidents and most of the core-melt accidents.

- There is no prompt fatality beyond the EPZ even if the most severe

[41] Eckerman, K.F., Wolbarst, A.B., Richardson, A.C.B., 1988. Limiting values of radionuclide intake and air concentration and dose conversion factors for inhalation, submersion, and ingestion. Federal Guidance Report 11, DE89-011065.

[42] Eckerman, K.F., Ryman, J.C., 1993. External exposure to radionuclides in air, water, and soil. Federal Guidance Report 12, PB94-114451.

accident occurs.

Release Category	Frequency [$year^{-1}$]	TTRAR [h]	Release duration [h]	TTPN [h]	Heat rate [$cal \times s^{-1}$]	Release height [m]
1	2.00×10^{-6}	6.97	48.00	9.54	1.80×10^{7}	240. 00
2	7.60×10^{-9}	0.07	0.83	0.70	7.20×10^{6}	29. 80
.	.	.	.	.	.	.
.	.	.	.	.	.	.
.	.	.	.	.	.	.
14	1.60×10^{-6}	1. 06	1. 50	1. 32	8.60×10^{5}	29. 80
15	1.90×10^{-8}	0. 26	1. 07	0. 67	4.10×10^{6}	29. 80

TTRAR: Time between reactor shutdown and radioactive material release [h].
TTPN: Time between notification of the public and release [h].

Table A.I. Sample of important parameters associated with 15 release categories for Chinshan NPP.

According to the above guidelines and the prompt and cancer fatality data collected from other accidents in Taiwan, the safety criteria for calculating the boundary of an EPZ can be derived as follows:

- The individual risk < 6.41×10^{-7} per year;
- The societal risk[43] < 2.18×10^{-6} per year;
- The frequency of the whole body dose exceeding 0.1 Sv < 3.0×10^{-5} per year;
- The frequency of the thyroid dose exceeding 1.0 Sv < 3.0×10^{-5} per year;
- The frequency of the whole body dose exceeding 2.0 Sv (prompt fatality dose) < 3.0×10^{-6} per year.

[43] The societal risk is the ratio of latent cancer fatality (expected fatal cancers) to total population within the radius.

Number	Isotope	Group	Inventory [Bq]
1	Co-58	6	1.005×10^{16}
9	Sr-90	5	1.289×10^{17}
16	Zr-95	7	2.926×10^{18}
29	Te-129	4	4.976×10^{17}
33	I-131	2	1.695×10^{18}
38	Xe-133	1	3.563×10^{18}
40	Cs-134	3	2.777×10^{17}
44	Ba-140	9	3.236×10^{18}
48	Ce-141	8	2.939×10^{18}
60	Cm-244	7	2.052×10^{15}

Table A.II. Sample of the inventory of 60 radionuclides contained in Chinshan NPP.

By comparing the consequences of individual risk, societal risk, whole body dose, and thyroid dose versus distance to the corresponding safety criteria listed above, a reasonably conservative suggestion for the EPZ of each of the three NPPs could be proposed.

Using the MACCS2 code, the radiological doses and the associated risks that could result from each postulated accidental release category were calculated The consequences were then summed up by the probability weighting factor[44] of each category. The complementary cumulative distribution function (CCDF)[45] was used to estimate the probability of exceeding the safety criteria. Fig. A.2 plots the CCDFs of whole body dose of 0.1 Sv, whole body dose of 2.0 Sv, and thyroid dose of 1.0 Sv versus distance from the release site.

---

[44] A multiplier that is used for converting the equivalent dose to a specific organ or tissue into what is called the "effective dose." The goal is to express the dose to a portion of the body in terms of an equivalent dose to the whole body that would carry with it an equivalent risk in terms of the associated fatal cancer probability. It applies only to the stochastic effects of radiation; see e.g. http://hps.org/publicinformation/radterms/radfact153.html

[45] In risk assessment it addresses "the exceedance question." It is the probability that the release will exceed a certain value. This question can be answered by a summing, or integration operation, on the probability density function (Figure B.1[b]). The result of such a summation is called the cumulative distribution function. The complement-that is, one minus the parameter (here, the cumulative probability)-and the log-log scale are the additional steps taken to achieve the desired form (Figures B.1[c] and [d]). These steps result in a compact form for representing parameters that cover an extremely wide range of values; see e.g. http://books.nap.edu/openbook.php?record_id=5269&page=111

Group »» » Category	1 Xe - Kr	2 I - Br	3, 4, 5, 6, 7 3: Cs-Rb 4: Te-Sb… 5: Sr 6: Co-Mo 7: La-Y	8 Ce - Pu	9 Ba
1	9.96×10^{-1}	3.50×10^{-5}		0. 00	4.30×10^{-6}
2	8.50×10^{-1}	8.60×10^{-2}		0. 00	9.70×10^{-4}
3	9.94×10^{-1}	1.80×10^{-2}		4.20×10^{-3}	9.90×10^{-2}
. . .					
14	9.78×10^{-1}	8.40×10^{-2}		1.70×10^{-2}	2.00×10^{-2}
15	9.99×10^{-1}	9.46×10^{-1}		1.30×10^{-2}	2.25×10^{-1}

Table A.III. Sample of the release fractions of 9 radionuclide groups of the 60 radionuclides for 15 release categories for Chinshan NPP.

The results showed that the whole body dose of 2.0 Sv was the most critical dose criterion and hence should be selected for the conservative purpose. The resulted EPZ for the Chinshan NPP with respect to dose criteria was less than1.5 km. Fig. A.3 shows the individual risk[46] and societal risk for the plant. From the aspect of risk, the estimated EPZ for Chinshan NPP met the dose criteria requirements.

Following these results, a decision was made that a radius of 5.0 km is still a conservative value for this operating NPP.

[46] By definition, the individual risk at a given radius from the plant is the ratio of acute fatality to total population within the diameter.

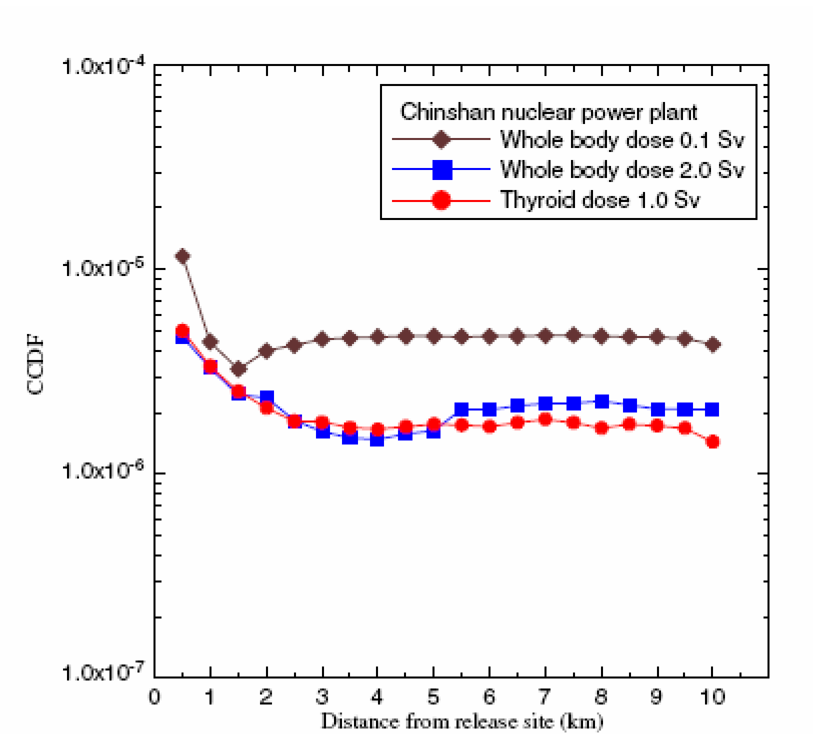


Fig. A.2. The probability of exceeding various doses versus distance from Chinshan NPP.

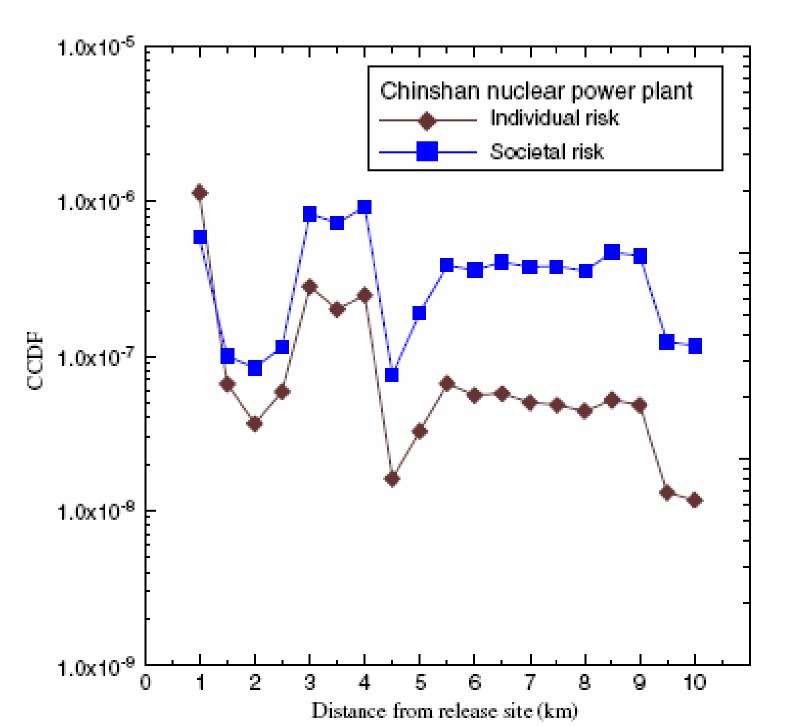


Fig. A.3. The probability of exceeding individual/societal risks for Chinshan

APPENDIX B

Risk Informed Support of EP for Koeberg NPP in South Africa

This section provides an overview of the risk insights to aid the derivation of the requirements for the Koeberg NPP emergency plan as applied in South Africa. The information has been taken and adapted from [10, 44], because it provides a very good and informative example of practical use of risk informed approach in support of NPP emergency zoning.

The methodology used is based on using the Koeberg probabilistic risk assessment (PSA) results to the worst credible severe accident scenario, i. e. the reference accident. The off site consequences of this reference accident are assessed and compared to the criteria for protective actions as sheltering and evacuation. The approach used is a blend of deterministic and probabilistic approaches which conforms to international standards and can be used to optimise emergency planning. It is also a holistic approach that allows the NPP modifications that improve plant safety to be balanced against a reduction in the magnitude of the off-site emergency planning requirements.

Koeberg NPP has two 922 MWe units. Each unit is a 3 loop Framatome (French version of Westinghouse PWR) unit and is located approx 35 km from Cape Town, a major city. With the growth in the local population, housing development is taking place nearer to the Koeberg NPP site. This development impacts on the ability to respond effectively in the event of a severe nuclear accident. In an effort to develop effective emergency plans, the utility implemented a program to derive the risks to the public and to use these risk insights to aid the optimisation of the emergency planning actions, zones and response times. In accordance with the current international practice, the Koeberg NPP has defined three zones surrounding the site: PAZ, UPZ, and FRPZ. The Koeberg emergency planning intervention levels meet national[47] as well as international guidance[48, 49] and are listed in Table B.I and B.II.

The severe accidents that were derived for the EP reference accidents were obtained by using the iodine release as a measure of consequence, since iodine dominates the early public risks (Fig. B.1). It plots typical element contributions to whole body effective 7 day dose. Figure B.1 was derived for a specific accident scenario assuming a large early release of radioactivity and specified at a distance of 16 km from the site. However, the Koeberg risk assessment indicates that iodine dominates all risk significant accidents at all distances [10].

[47] National Nuclear Regulator, The NNR Report on The Technical Basis For Emergency Planning At Koeberg Nuclear Power Station, South Africa, June 2000.

[48] IAEA, Method for the Development of Emergency Response Preparedness for Nuclear or Radiological Accidents, IAEA-TECDOC-953, Vienna, 1997. See also the updating in [3].

[49] http://www-pub.iaea.org/MTCD/publications/PDF/Pub1133_scr.pdf

Protective Action	Avertable dose	Dose Integration Period
Sheltering	5 to 50 mSv	2 days
Evacuation	50 to 500 mSv	7 days
Iodine prophylaxis	100 mGy	Lifetime
Temporary relocation	Initiate at 30 mSv Terminate at 10 mSv	30 days
Permanent resettlement	1Sv	Lifetime

Table B.I. Koeberg NPP intervention levels for EP actions.

Radionuclides in foods destined for consumption	**kBq/kg**
CS-134, CS-137, I-131, Ru-103, Ru-106, Sr-89	1
Sr-90	0.1
Am-241, Pu-238, Pu-239, Pu-242	0.01
Radionuclides in milk, infant foods, and drinking water	**kBq/kg**
Cs-134, Cs-137, Ru-103, Ru-106, Sr-89	1
I-131, Sr-90	0.1
Am-241, Pu-238, Pu-239, Pu-242	0.001

Table B.II. Koeberg NPP intervention levels for foodstuffs.

A similar chart could be developed for a 50 year exposure which would then indicate that caesium dominates the overall consequences. It is therefore caesium that often determines the requirements for late phase emergency planning actions such as relocation and permanent resettlement [10].

All the severe accidents from the plant risk assessment (L2 PSA) were then listed in order of their iodine release fraction with their associated accident frequencies. These severe accidents included those in a traditional PSA and those initiated during plant shutdown, initiated by external events (floods and even terrorism), and also those associated with the spent fuel pool.

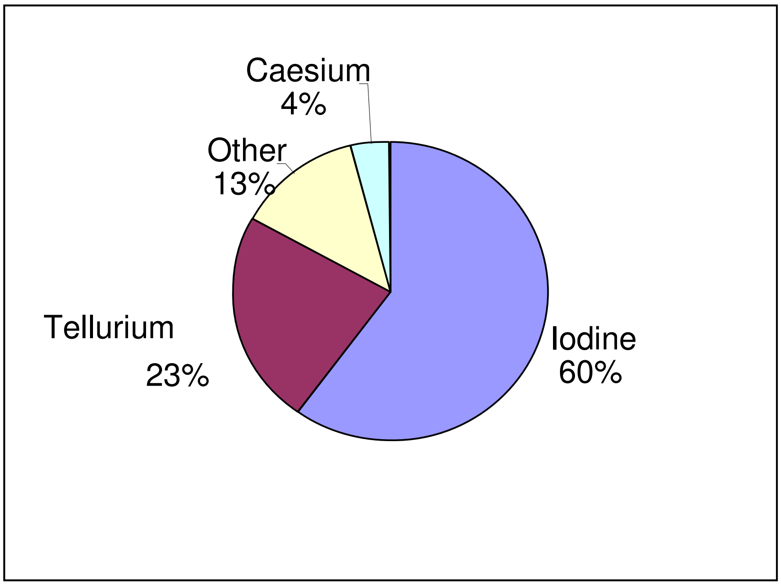


Fig. B.1. Typical element contributions to whole body effective 7 day dose.

In discussion with the regulatory body, it was decided that a cut-off of 10^{-6} per year can be used such that the cumulative frequency of the worst severe accident scenarios with a value of less than 10^{-6} per year can be deemed “incredible”. Thus, the worst credible severe accident scenario which must be planned for is the scenario, which has a cumulative frequency of 10^{-6} per year. This worst credible severe accident scenario became the reference accident for EP. Figure B.2 summarises the output of this assessment which is the plot of the cumulative frequency of severe accident sequences in order of their iodine release. Where the line crosses the cumulative frequency of 10^{-6} per year gives the maximum credible iodine release. Thus, the graph shows that the worst credible severe accident scenario leads to an iodine release of approximately 0.85% of the core inventory. A 1% iodine release was then conservatively taken as the maximum credible release. This conservatism is appropriate given the shallow gradient of the curve over a wide range of iodine releases. The top left of the curve is linked to the total fuel melt frequency for the plant and the cumulative frequency of a 10% iodine release is associated with the Koeberg NPP estimated large early release frequency.

Reference severe accident scenarios can be selected from the Fig. B.2. These scenarios have iodine release fractions (IRF) of approx 1E-1, 1E-2, and 1E-4 and are labelled “Likely” with given core damage (CD), “Credible” (maximum), and “Incredible”; details are shown in Table B.III. The severe accident scenarios from the plant PSA associated with these releases provide additional details such as time to release, duration of the release, height and energy of release, and magnitude of the other associated fission products. The dominant isotope releases are given on the next page in Table B.IV [10].

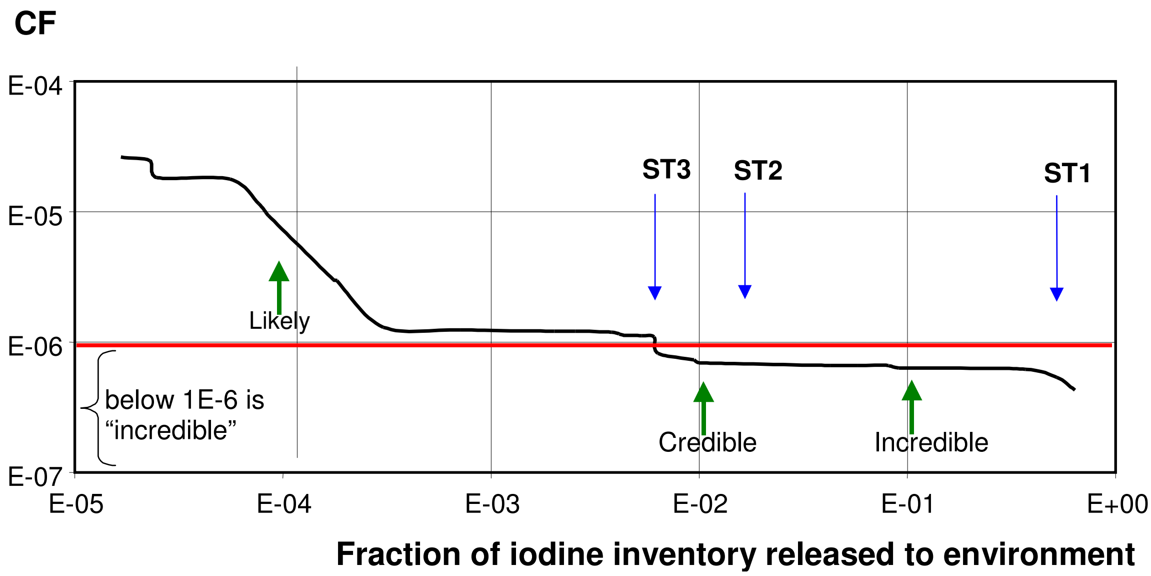


Fig. B.2. Iodine release vs. cumulative frequency **CF** of severe accident sequences.

PSA Label	Description	Representative approx IRF [%]	Approx release duration [h]	Frequency [1/year]
RC1T3	Likely (given CD)	0.01	8	$5x10^{-5}$ to 10^{-5}
RC2T1	(Maximum) Credible	1.00	2	10^{-5} to 10^{-6}
RC3T1	Incredible	10	1	$<10^{-6}$

Table B.III. Koeberg NPP reference accidents.

The off site consequences of the emergency planning reference accidents are then assessed in terms of public doses and ground contamination. This is then compared to the criteria given in Table B.I and B.II to determine the distance from the site that each EP protective action may be required in the maximum credible severe accident scenario, which then leads to the deviation of the EP zone radii. The emphasis is placed on the worst credible release which equates to 1% of the total iodine inventory with the other radionuclides as determined by the severe accident analyses. The use of the other reference accidents, i. e. "Likely" (given core damage) and "Incredible" serve only to aid sensitivity analyses.

The timing of the release in the credible reference accident and its duration is also used to aid the deviation of the time available for the implementation of some emergency planning protective actions as sheltering and evacuation.

For comparison, some considerations are possible concerning basic classes of source term in France, which are given in Table 2.2 in section 5.2 of this report. That is why the ST1, ST2, ST3 source terms are also plotted in Figure B.2. Therefore, the comparison should be between the Koeberg NPP credible source term and ST3 source term as practised in France.

PSA Label » » » »	**RC1T3**	**RC2T1**	**RC3T1**
Containment failure mode »»»»	**No containment failure**	**Small release**	**Large early release**
Description»»»»	**Likely (given CD)**	**(Maximum) Credible**	**Incredible**
I-131	2.32×10^{14}	2.71×10^{16}	2.47×10^{17}
I-132	3.40×10^{14}	3.99×10^{16}	3.64×10^{17}
I-133	4.89×10^{14}	5.72×10^{16}	5.22×10^{17}
Te-132	3.53×10^{14}	4.20×10^{16}	3.08×10^{17}
Nb-95	8.14×10^{13}	2.01×10^{16}	2.99×10^{16}
Ru-103	6.47×10^{13}	1.60×10^{16}	2.38×10^{16}
Ru-106	1.94×10^{13}	4.79×10^{15}	7.14×10^{15}
Cs-134	3.78×10^{13}	4.33×10^{15}	4.02×10^{16}
Cs-137	2.81×10^{13}	3.22×10^{15}	2.98×10^{16}
Ba-140	4.25×10^{13}	5.82×10^{15}	3.77×10^{16}

Table B.IV. Reference accident dominant isotope releases [Bq].

As results from the Figure B.2, only ST3 source term is used in off-site consequence assessments, such as the requirements of emergency planning.

Two computer codes were used to aid the assessment of the off-site consequences of the reference accidents. These were PC Cosyma[50] and Hotspot[51]. The further text focuses only on some results, obtained by Cosyma (Code SYstem from MAria), which is a software package for assessing the offsite consequences of accidental releases of radioactive material to the atmosphere. It was developed as a part of the European Communities program Methods for

[50] PC COSYMA National Radiological Protection Board, Kernforschungszentrum Karlsruhe, EUR 16240 EN (NRPB - SR280).
[51] Lawrence Livermore National Laboratory, Hotspot Version 2.05, 2003. [http://www.llnl.gov/nhi/hotspot/].

Assessing the Radiological Impact of Accidents (MARIA).

The doses were calculated with the assumption that the core contains the specified fuel assemblies with an enrichment of 5,0% U-235. As is customary, it has been assumed that the core is a three-cycle core and the accident occurs at end-of-life conditions. Thus, the reactor is assumed to have been operated for an extended period at steady full power conditions. During this time it is assumed that the fuel has been changed in accordance with normal refuelling procedures in which case the oldest third of the fuel is replaced at each of the previous refuelling outages.

Only the active zone has been considered. The other zones (gas plenums, bottom and top nozzles of the fuel assemblies) do not significantly contribute to the overall source terms and so have been ignored.

PC Cosyma allows a minimum of 4 and a maximum of 72 sectors. The number of distance bands allowed is 2 to 25. For these calculations, the radial grid comprises 21 distance bands and 72 sectors at 5 degree intervals (Figure B.3). The distances are up to 80 km around Koeberg NPP site [44]. The code calculates the dose at the centre of each distance band in the direction of interest.

The site boundary at Koeberg NPP extends from 1.3 km to 2.5 km from the site. It was agreed that the site boundary be taken as 1.5 km. The PAZ extends from 2 to 5 km from site whereas the UPZ extends from 5 to 16 km from site. The distance bands are chosen in such a way that the mid-point of the distance bands coincides with these boundaries.

Table B.V gives the 7-day effective whole body dose for the maximum credible radioactive release. The red cells mark where the doses exceed the upper level of evacuation (500 mSv) and the yellow cells mark where the doses exceed the lower level of evacuation (50 mSv), but they are lower than the upper intervention level for evacuation. The 7-day effective dose is an over-estimate of the avertable dose. This is because evacuation is based on environmental monitoring once the plume has passed. This means that the cloudshine and inhaled dose (committed dose) cannot be averted and only the groundshine (and any associated resuspension) can be averted. The 7-day effective dose is also the guideline used in the national nuclear regulator (NNR) EP technical bases.[52]

In comparison, EDF uses a dose integration time of 1 day instead of 7 days and so their off-site consequence assessments of their ST3 source term are generally at least a factor of 7 less than the results given in Table B.V.

[52] The NNR Report on The Technical Basis For Emergency Planning At Koeberg Nuclear Power Station, National Nuclear Regulator, June 2000.

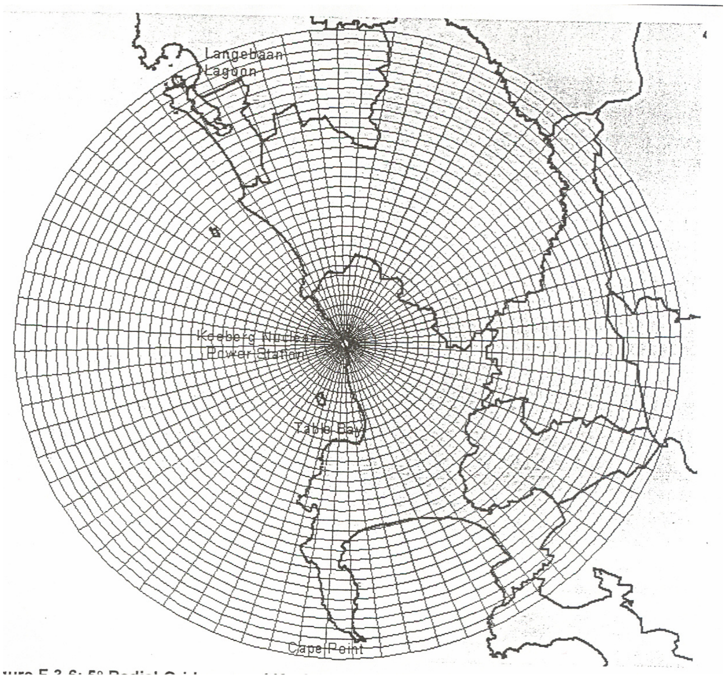


Figure B.3. Radial representation.

As far as weather data is concerned, weather category D was used with a site-specific wind speed of 3.8 m/s, which complies with Koeberg specific meteorological data. Within a sensitivity study, weather category F was considered. Weather Category F is more stable than D and so the area contaminated is longer but also narrower, leading to a similar overall area for evacuation. Changes in wind direction would have the effect of spreading a lower dose over a wider area such that the area exceeding 50 mSv would be reduced. Thus, changes in wind direction would reduce the area requiring evacuation. Rain was not considered in the calculations.

The PC Cosyma calculations indicate that using a 5-degree angle for estimating the area to be evacuated in the UPZ would be sufficient.

Distance (km)	-20°	-15°	-10°	-5°	0°	5°	10°	15°	20°
0,5	**84**	**561**	**2146**	**4765**	**6215**	**4765**	**2146**	**561**	**84**
1,2	5	**61**	**400**	**1275**	**1881**	**1275**	**400**	**61**	5
1,5	2	33	**249**	**890**	**1362**	**890**	**249**	33	2
1,7	1	23	**190**	**725**	**1134**	**725**	**190**	23	1
2,0	1	14	**132**	**554**	**895**	**554**	**132**	14	1
3,1	0	4	47	**263**	**466**	**263**	47	4	0
5,0	0	1	14	**113**	**227**	**113**	14	1	0
6,5	0	0	7	**70**	**152**	**70**	7	0	0
7,5	0	0	4	**54**	**122**	**54**	4	0	0
8,5	0	0	3	43	**101**	43	3	0	0
9,5	0	0	2	35	**85**	35	2	0	0
10,5	0	0	2	29	**73**	29	2	0	0
11,5	0	0	1	24	**64**	24	1	0	0
12,5	0	0	1	21	**56**	21	1	0	0
13,5	0	0	1	18	**50**	18	1	0	0
14,5	0	0	1	16	47	16	1	0	0
16,0	0	0	0	14	43	14	0	0	0

Table B.V. Credible accident 7-day whole body effective dose [mSv].

To determine the impact on plume width and area to be evacuated, sensitivity analysis the off-site consequences of the likely release (given core damage) and the incredible release were also conducted. The sensitivity analysis resulted in no significant dose received off-site in the event of the most probable core damage scenario [44].

Table B.VI provides the criteria for determining the inner EPZ, i.e. PAZ and outer EPZ, i. e UPZ radii. This is based on the criteria specified in the above Table B.I and the national nuclear regulatory body technical assessment[53].

CONSEQUENCES: 50 mSv and 500 mSv doses at a distance							
50 mSv				**500 mSv**			
1 Day Exposure		**7 Days Exposure**		**1 Day Exposure**		**7 Day Exposure**	
Weather D Covers 75%	**Weather F Covers 100%**	**Weather D Covers 75%**	**Weather F Covers 100%**	**Weather D Covers 75%**	**Weather F Covers 75%**	**Weather D Covers 75%**	**Weather F Covers 100%**
UPZ Lower Bound	**UPZ** Upper Bound	**UPZ** Best Estimate*	**UPZ** Extreme Upper Bound	**PAZ** Extreme Lower Bound	**PAZ** Best Estimate*	**PAZ** Lower Bound	**PAZ** Upper Bound

Table B.VI. PAZ and UPZ Dose Criteria.

Table B.VI essentially states that the PAZ radius is obtained by determining where the EP reference accident with worst case weather gives a dose of 500 mSv in 1 day. The PAZ radius upper bound is obtained using the same accident but assuming a 7-day exposure of 500 mSv. The lower bound would be obtained assuming average weather conditions and a 7 day exposure of 500 mSv.

The same table essentially states that the UPZ radius is obtained by determining where the EP reference accident with average weather gives a dose of 50 mSv in 7 days. The UPZ radius upper bound is obtained using the same accident but assuming worst-case weather and a 1-day exposure of 50 mSv. The lower bound would be obtained assuming average weather conditions and a 1 day exposure of 50 mSv.

Table B.VII presents the off-site consequence results for a range of severe accident releases for 3.9% enriched fuel with the plant specified average assembly burn-up. Thus, for a severe accident which releases 1.0% of the iodine (an iodine release fraction of 1.00E-2, see the above Fig. B.2), a PAZ of 3 km and a UPZ of 14 km would be appropriate.

[53] See footnotes 47 - 49

* Best estimate is the point estimate of a parameter that is not biased by conservatism or optimism. Generally, the best estimate of a parameter is represented as a mean value [28].

Iodine Release Fraction	CONSEQUENCES: 50 mSv and 500 mSv Doses at Distance X (km)							
	50 mSv				500 mSv			
	Exposure = 1 Day		Exposure = 7 Days		Exposure = 1 Day		Exposure = 7 Days	
	Weather D Covers 75%	Weather F Covers 100%	Weather D Covers 75%	Weather F Covers 100%	Weather D Covers 75%	Weather F Covers 100%	Weather D Covers 75%	Weather F Covers 100%
	UPZ Lower Bound	UPZ Upper Bound	UPZ Best Estimate	UPZ Extreme Upper Bound	PAZ Extreme Lower Bound	PAZ Best Estimate	PAZ Lower Bound	PAZ Upper Bound
1,00E-01	14 km	>25 km	>25 km	>25 km	3 km	7 km	7 km	>25 km
1,00E-02	7 km	>25 km	***14 km***	>25 km	<2 km	***3 km***	3 km	7 km
1,00E-03	4 km	9 km	7 km	>25 km	<2 km	<2 km	<2 km	3 km
1,00E-04	<2 km	<2 km	<2 km	<2 km	<2 km	<2 km	<2 km	<2 km

Table B.VII. Off-site consequence results for a range of severe accident releases.

To conclude, as it results from the presented example, emergency planning requirements can be enhanced by transition towards a more risk informed approach where the requirements take into account information supplied from the NPP PSA via the derivation of risk informed reference accidents. Although three reference accidents were developed, the focus was on the worst credible accident. The off-site consequences of the worst credible accident indicate that the off-site emergency plans presented in the following Table B.VIII are appropriate.

As stated in [10], the analysis also indicated that no planning for permanent resettlement is warranted. The planning for temporary relocation is only required out to a distance of 6 km for a maximum of 2 months even though the environmental monitoring will need to extend to 80 km since it is possible that there could be small localised areas of high contamination. Food restrictions only need to be planned for the long term protective zone FRPZ (80 km) but with the explicit understanding that actual measures may need to be expanded beyond this planning zone.

Finally, the choice of weather stability category has little impact on the results. Weather stability category F is more stable than D and so the area contaminated is longer, but also narrower leading to a similar overall area for EP actions.

In addition, in [10] there is also some information on evacuation traffic modelling, as it is important in determining the credibility of implementing effective evacuation. The model used the predicted housing and road developments to estimate the time required for evacuation. The modelling took into account the income category of the developments, the amount of public transport that would be required, as well as the traffic flow rate. The estimated population growth near Koeberg is expected to grow from ~ 95000 in 2005 by a factor of 4 over the next 25 years, to reach ~ 380000 in 2030.

ZONE	SIZE [km]	ACTION	IMPLEMENTATION TIME [h]
PAZ	0 - 5	Evacuation (all sectors) based on in-plant conditions	4[54]
UPZ	5 - 16	Shelter (downwind sectors)	4[54]
		Evacuation based on in-plant condition leading to 12-16 hour advance warning	16
		Thyroid blocking (downwind sectors)	10[54]
FRPZ	0 - 80	Relocation (based on environmental monitoring)	Long-term action
		Food ban (based on environmental monitoring)	Long-term action

Table B.VIII. Koeberg NPP emergency plan requirements.

The estimated evacuation time in 2010 is ~ 9 hours, and in 2030 ~ 11 hours, respectively. The results of modelling indicate, that the time to evacuate remains below 12 hours for the life of the plant. The overall evacuation modelling process is still being refined.

[54] The implementation time from declaration of a general emergency i.e. the time required for public (e.g. siren) notification system

APPENDIX C

PSA Support of EPZ for Temelin NPP in Czech Republic

This section provides a summary of severe accident analyses and the use of their relevant results for emergency zoning practices as applied in Czech Republic for Temelin NPP. The information has been taken and adapted from[55],[56] [45, 46], and it provides another good and very informative example of practical use of risk informed approach in support of NPP emergency zoning.

The above referred materials were developed in "Temelin case", aiming at achieving more concerted approach between Czech and Austrian party towards the emergency planning and response to a radiological accident, which could have – with very low probability – trans-boundary impact. It provides in a comprehensive way the description of background and approach applied in Czech Republic for emergency preparedness and planning in general.

The Temelin NPP (two PWR units, each of 1000MWe of WWER reactor type) is located about 25 kilometres north of Ceske Budejovice, the regional capital of South Bohemia and 5 kilometres south of the small town of Tyn nad Vltavou. The site is approximately 55 km from the Czech-Austrian border. During the licensing process of the Temelin NPP, Czech nuclear regulatory authority kept in mind and carried out - together with other aspects of nuclear safety and radiological protection - also assessments of radiological consequences of both DBAs as well as BDBAs and severe accidents, which normally are not covered by the NPP Safety Analysis Report and other licensing documentation. For the EPZ size determination, as the worst case approximation of first two hypothetical sequences with maximal consequences were analysed, both of occurrence frequency of the order of 10^{-10}/year.

Further step took into consideration more realistic scenarios and their radiological consequences. According to the Czech national regulation[57], the licensee shall provide for the regulatory authority's decision a list of possible radiological accidents for the particular nuclear facility of the occurrence frequency higher or equal of 10^{-7}/year with evaluation of their consequences. These scenarios were selected on the basis of L1 and L2 PSA results. The requirements of the above mentioned national regulation resulted in analyses of the radiological consequences for scenarios of occurrence frequency of one order lower than accepted in international practice based on for example [15, 16], Table 5.1, Chapter 5.

According to the Czech legislation there is no advance planning of response and countermeasures for consequences of events of the probability of occurrence lower than 10^{-7} per reactor year. Nevertheless, like in various countries within EU and beyond, there is the general emergency response system established, which

[55] http://www.mzv.cz/EIA/eia/severe2.pdf

[56] http://www.umweltbundesamt.at/fileadmin/site/umweltthemen/kernenergie/temelin/Melk/GesamtUVP/UVPBericht/Teil5.pdf

[57] Government Ordinance No. 11/1999

has to be able to cope with even less likely disasters with more severe consequences ad hoc depending on the extent, development and actual and predicted consequences. The basic technical and organisational tools for such hypothetical cases are the nation-wide radiation monitoring network and off-site national emergency response plan on the level of the region and the whole country.

The selection of severe accident scenarios meeting the frequency of occurrence higher or equal to the value of 10^{-7}/year given in the national regulation for the Temelin NPP was based on the following criteria:

- Sequences with the highest frequency, i.e. with the highest probability of occurrence;
- Sequences with the highest significance, i.e. with the highest source term related to the frequency.

Application of this approach resulted in two main events/associated sequences, which would contribute to the considerable radiological consequences, both of the highest frequency and the highest significance criteria: 1) major leak from primary to secondary circuit, and 2) large LOCA.

The first sequence is defined as a major leak from primary to secondary circuit when the operator fails to cool down and depressurize the primary circuit. Damage of the core and significant release of radionuclides will occur after the loss of inventory to cool the core.

The second sequence is defined as large LOCA with the failure of low pressure injection (LPI) system of emergency core cooling system (ECCS). Other emergency systems remain available. Due to insufficient capacity of these systems, there is a severe damage to the core with subsequent damage to reactor pressure vessel (RPV). Since the spray system is in operation, the containment is not challenged due to overpressure.

Many variations and sensitivity cases of both above specified scenarios were analyzed by MELCOR code, aiming at studying various phenomena influencing the containment behaviour during the severe accident progression. To mention the first of them, major leak from primary to secondary circuit was combined with simultaneous complete loss of alternating current (AC) electric power (in further text mentioned as scenario "V" and source term "STV") with frequency of 10^{-10}/year. The second one is the simultaneous occurrence of large LOCA with complete loss of AC electric power (frequency of 10^{-10}/year), which is further mentioned as various "AB" scenarios. The most relevant scenarios were as follows:

1. Large leaks from primary to secondary circuit with containment bypass without any operator action[58] with and without thermal creep of the hot leg

[58] Due to operator failure to intervene, leading to core degradation containment bypass occurs. In terms of the NPP personnel training, plant human factor philosophy, instrumentation and control devices available to operators, operating/contingency planning procedures, etc. is more than evident this assumption is entirely unrealistic.

piping. Thermal creep of the hot leg as one case, and high pressure scenario of primary circuit hermetically tight and direct containment heating after RPV bottom failure as another case were analysed. For both cases time course of the radionuclides releasing from the containment, i. e. the source terms were estimated. In further text they are mentioned as ST1.1 and ST1.2.

2. Large LOCA on the pressurizer surge line with simultaneous unavailability of ECCS. Calculated source term (ST2) corresponds with the scenario leading to severe core damage. Within this, spray system in operation was modelled, enabling to analyse also the effects related to hydrogen issue. Hydrogen deflagration in containment during the severe accident progression and molten core concrete interaction after the RPV bottom failure with corium pool surface of approx 100m^2 were analysed[59].

3. Containment integrity challenge after hydrogen detonation during LOCA with simultaneous unavailability of ECCS. The same initiating event and assumptions as in 2, but the function of catalytic recombiners and hydrogen deflagration was not considered. Two source terms ST3.1 and ST3.2 correspond with analysed cases, both considering breach of the containment integrity after hydrogen detonation. However, the hydrogen detonation can easily be excluded by rigorous adherence to severe accident management measures (SAMG). Only complete failure of operating staff could lead to conditions enabling the hydrogen detonation in the containment[60].

4. Station blackout with permanent loss of all active safety systems. Following phenomena were studied in this case: containment slow overpressure due to loss of heat sink, direct containment heating after the failure of the RPV bottom, molten core concrete interaction in the cavity and in vertical neutron measurement channels with the corium pool surface of approx 25 m^2. Source term ST4 corresponds with this analysed case.

5. Large LOCA on pressurizer surge line with equivalent diameter of 200 mm with ECCS reinitiated after RPV failure. The progress of molten core concrete interaction when the water layer covers the corium pool was studied in this case[61]. The analysis results in source term ST5.

The analysed beyond design basis and severe accidents scenarios, i.e. the above “AB” and “V” sequences as well as other sequences meeting probabilistic criterion specified in the national regulation (the frequency of occurrence for the particular nuclear facility higher or equal 10^{-7}/year) would lead to exceeding of selected dose levels - intervention levels - at distances summarised in Tab. C.I.

[59] See footnote 58
[60] See footnote 58
[61] The event was analysed in principle only for confirmation of possibility to stop the molten core penetration through concrete of the containment basement plate by flooding the melt in the late phase of severe accident progression.

Sequence/ Source term	Intervention level for ≤2 days SHELTERING 10 mSv[62] [km]	Intervention level for ≤2 days SHELTERING 50 mSv[63] [km]	Intervention level for ≤7 days EVACUATION 50 mSv[64] [km]	Intervention level for ≤7 days EVACUATION 500 mSv[65] [km]
AB_01	5	<1	1	<1
AB_02	8	2	2	<1
AB_03	11	3	4	<1
AB_04	9	1	2	<1
STV	40	<1	<1	<1
ST1.1	23	2	3	2
ST1.2	17	5	5	2
ST2	<1	<1	<1	<1
ST3.1	19	2	2	<1
ST3.2	14	2	3	<1
ST4	<1	<1	<1	<1
ST5	2	<1	<1	<1

Table C.I. Most relevant results of radiological consequence calculations for selected accidents.

As the worst case, conservative assumption of weather category F (virtually without any dispersion) was used for the dispersion model.

[62] The generic optimized intervention level for sheltering is 10 mSv of avertable dose in a period of no more than 2 days [http://www-pub.iaea.org/MTCD/publications/PDF/SS-115-Web/Pub996_web-5.pdf]

[63] International Commission on Radiological Protection, Principles for Intervention for Protection of the Public in a Radiological Emergency, ICRP Publication No. 63:, Pergamon Press, Oxford, 1992

[64] The generic optimized intervention value for temporary evacuation is 50 mSv of avertable dose in a period of no more than 1 week; in: as in footnote 62.

[65] See footnote 63

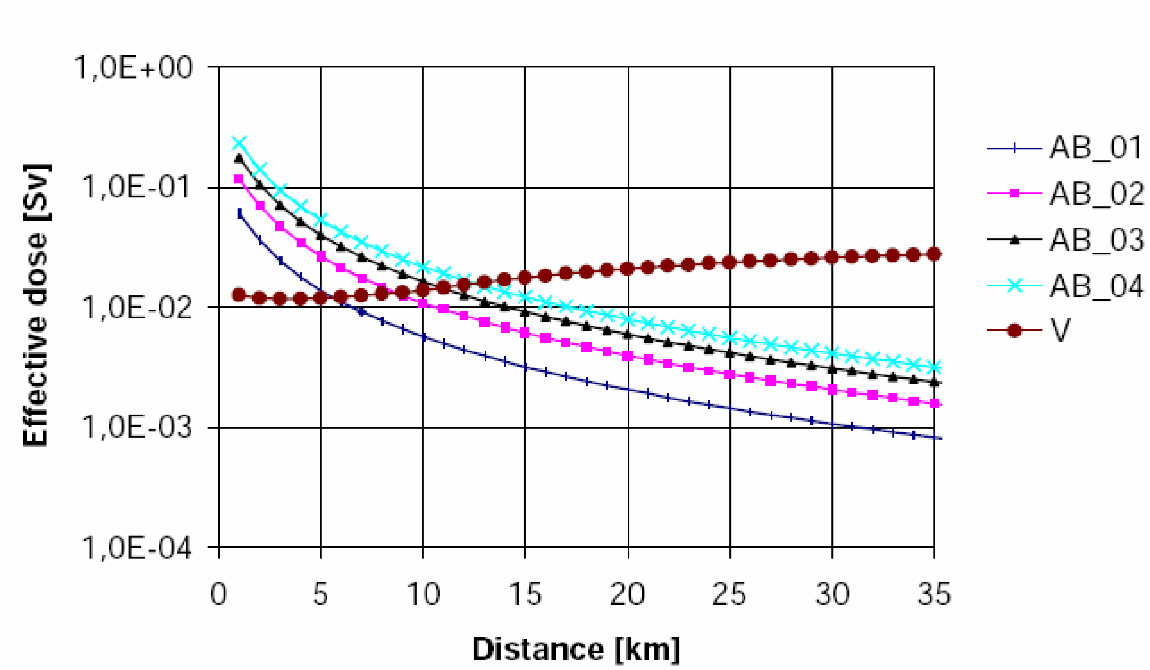


Fig. C.1. Effective dose-distance courses for the worst case of conservative assumption of weather category F.

Results in Tab. C.I are relevant for urgent (short term) countermeasures, i. e. sheltering (2 days), iodine prophylaxis, and evacuation (7 days). The countermeasures do not have to be introduced at distances at which the doses do not exceed upper bounds of intervention levels interval, i.e. at distances where the dose is lower than 50 mSv for sheltering and iodine prophylaxis (ICRP recommendations [17]) and lower than 500 mSv for evacuation, according to the same recommendations. For all urgent countermeasures the upper bound of intervention levels interval is not exceeded at distances greater than 5 km from the plant.

The course of effective dose in 7 days along the distance from the plant for the most important sequences is as illustrated in Fig. C.1. As the worst case, conservative assumption of weather category F (virtually without any dispersion) was used again for the dispersion model. What results from the figure is, that for the distances from the plant longer than 35 km the dose is lower than 5 mSv except for V sequence, having, however, the frequency 10^{-10} per year.

To conclude the above considerations it should be mentioned, that the shortest distance from Temelin NPP to the border between Czech Republic and Austria is approx 50 km.

European Commission

EUR 23280 EN– Joint Research Centre – Institute for Energy
Title: Risk Informed Support of Decision Making in Nuclear Power Plant Emergency Zoning – Generic Framework towards Harmonising NPP Emergency Planning Practices
Author(s): Jozef Kubanyi, Ricardo Bolado Lavin, Dan Serbanescu, Bela Toth, Heinz Wilkening
Luxembourg: Office for Official Publications of the European Communities
2008 – 129 pp. – 21 x 29.7 cm
EUR – Scientific and Technical Research series – ISSN 1018-5593

Abstract
The report provides a systematic overview of the essential aspects of risk informed support of decision making (RIDM) in nuclear power plant (NPP) emergency zoning (EZ) as a contribution to harmonizing strategic practices in this area. Owing to the state-of-the-art understanding and increased characterisation of NPP severe accidents, overall management of them should be analysed as an integrated complex process. The interrelationship of NPP emergency operating procedures, safety and risk assessments, severe accident management guidelines, and emergency off-site actions should be planned and organised to minimize the consequences of such accidents. A deterministic approach, coupled with both probabilistic safety assessment (PSA) technology and PSA results can play significant role in the development of relevant nuclear utility, regulatory and all stakeholders policies. The report describes the background, objectives and current state of a corresponding activity within JRC-IE's Analysis and Management of Nuclear Accidents (AMA) Action on probabilistic safety / risk assessment methodologies and practices for RIDM approach applied to NPP EZ. The approach is interdisciplinary, based on integration of PSA technology, severe accident phenomenology, and radiological protection.

How to obtain EU publications

Our priced publications are available from EU Bookshop (http://bookshop.europa.eu), where you can place an order with the sales agent of your choice.

The Publications Office has a worldwide network of sales agents. You can obtain their contact details by sending a fax to (352) 29 29-42758.

The mission of the JRC is to provide customer-driven scientific and technical support for the conception, development, implementation and monitoring of EU policies. As a service of the European Commission, the JRC functions as a reference centre of science and technology for the Union. Close to the policy-making process, it serves the common interest of the Member States, while being independent of special interests, whether private or national.

EUROPEAN COMMISSION

Publications Office
Publications.eu.int