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**NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS**

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The Use and Development of Probabilistic Safety Assessment in NEA Member Countries

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- to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

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The Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA) is an international committee made up of senior scientists and engineers. It was set up in 1973 to develop, and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety among the OECD Member countries.

The CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations, which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of the programme of work. It also reviews the state of knowledge on selected topics on nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus on technical issues of common interest. It promotes the co-ordination of work in different Member countries including the establishment of co-operative research projects and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences and specialist meetings.

The greater part of the CSNI's current programme is concerned with the technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment, and severe accidents. The Committee also studies the safety of the nuclear fuel cycle, conducts periodic surveys of the reactor safety research programmes and operates an international mechanism for exchanging reports on safety related nuclear power plant accidents.

In implementing its programme, the CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

* * * * *

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ABSTRACT

The mission of the CSNI is to assist Member countries in maintaining and further developing the scientific and technical knowledge base required to assess the safety of nuclear reactors and fuel cycle facilities.

The mission of the Working Group on Risk Assessment (WGRisk) is to advance the understanding and utilisation of Probabilistic Safety Assessment (PSA) in ensuring continued safety of nuclear installations in Member countries. In pursuing this goal, the Working Group shall recognise the different methodologies for identifying contributors to risk and assessing their importance. While the Working Group shall continue to focus on the more mature PSA methodologies for Level 1, Level 2, internal, external, shutdown, etc. It shall also consider the applicability and maturity of PSA methods for considering evolving issues such as human reliability, software reliability, ageing issues, etc., as appropriate.

This report provides descriptions of the current status of PSA programmes in Member countries including basic background information, guidelines, various PSA applications, major results in recent studies, PSA based plant modifications and research and development topics. While the compilation is a not complete compilation it provides a “snapshot” of the current situation in the Member countries and hence it provides reference information and various insights to both the PSA praticien and others involved in the nuclear industry.

The terms PSA (Probabilistic Safety Assessment) and PRA (Probabilistic Risk Assessment) are utilised to denote this subject.

TABLE OF CONTENTS

ABSTRACT.....	4
FOREWORD	8
CHAPTER 1 - INTRODUCTION	9
1.1 Background.....	9
1.2 Structure of Report	9
CHAPTER 2 – PSA ENVIRONMENT	11
2.1 Belgium	11
2.2 Canada	11
2.3 Czech Republic.....	12
2.4 Finland.....	12
2.5 France	15
2.6 Germany	16
2.7 Hungary	17
2.8 Italy.....	19
2.9 Japan.....	20
2.10 Korea	20
2.11 Mexico.....	20
2.12 Netherlands.....	22
2.13 Spain.....	22
2.14 Sweden	23
2.15 Switzerland.....	23
2.16 United Kingdom	24
2.17 United States.....	26
CHAPTER 3 – QUANTITATIVE SAFETY GUIDELINES	29
3.1 Belgium	29
3.2 Canada	29
3.3 Czech Republic.....	29
3.4 Finland.....	30
3.5 France	30
3.6 Germany	31
3.7 Hungary	31
3.8 Italy.....	35
3.9 Japan.....	36
3.10 Korea	36
3.11 Mexico.....	37
3.12 Netherlands.....	38
3.13 Spain.....	42

3.14	Sweden	42
3.15	Switzerland	42
3.16	United Kingdom	43
3.17	United States	46
CHAPTER 4 – STATUS OF PSA PROGRAMMES		49
4.1	Belgium	49
4.2	Canada	49
4.3	Czech Republic	50
4.4	Finland	51
4.5	France	52
4.6	Germany	57
4.7	Hungary	57
4.8	Italy	58
4.9	Japan	59
4.10	Korea	61
4.11	Mexico	66
4.12	Netherlands	68
4.13	Spain	69
4.14	Sweden	70
4.15	Switzerland	75
4.16	United Kingdom	76
4.17	United States	78
CHAPTER 5 – PSA APPLICATIONS		81
5.1	Belgium	81
5.2	Canada	82
5.3	Czech Republic	82
5.4	Finland	83
5.5	France	87
5.6	Germany	90
5.7	Hungary	90
5.8	Italy	91
5.9	Japan	92
5.10	Korea	93
5.11	Mexico	93
5.12	Netherlands	94
5.13	Spain	97
5.14	Sweden	99
5.15	Switzerland	101
5.16	United Kingdom	102
5.17	United States	103
CHAPTER 6 – PSA RELATED RESEARCH AND DEVELOPMENT		111
6.1	Belgium	111
6.2	Canada	111
6.3	Czech Republic	111
6.4	Finland	112
6.5	France	112
6.6	Germany	114
6.7	Hungary	115

6.8	Italy.....	115
6.9	Japan.....	116
6.10	Korea	117
6.11	Mexico.....	121
6.12	Netherlands.....	122
6.13	Spain.....	129
6.14	Sweden	132
6.15	Switzerland.....	134
6.16	United Kingdom	135
6.17	United States.....	142
6.18	European Commission Joint Research Centre.....	145
CHAPTER 7 – PSA PLANT BASED MODIFICATIONS.....		147
7.1	Belgium	147
7.2	Canada.....	147
7.3	Czech Republic.....	147
7.4	Finland.....	147
7.5	France	152
7.6	Germany	154
7.7	Hungary	156
7.8	Italy.....	156
7.9	Japan.....	156
7.10	Korea	158
7.11	Mexico.....	159
7.12	Netherlands.....	160
7.13	Spain.....	164
7.14	Sweden	171
7.15	Switzerland.....	171
7.16	United Kingdom	176
7.17	United States.....	180
CHAPTER 8 - REFERENCES		183
8.1	Contact Information.....	184
8.2	References	190
APPENDIX A – STATUS OF PSA PROGRAMMES IN NEA MEMBER COUNTRIES		199

FOREWORD

Set-up in 1973, the CSNI is an international committee made up of senior scientists and engineers, with broad responsibilities for safety technology and research programmes. The technical fields of nuclear reactor safety interest into which the CSNI has designated working groups (WGs) are Operating Experience, Analysis and Management of Accidents, Integrity of Components and Structures and Risk Assessment. It also has set up special expert groups (SEGs) on Human and Organisational Factors and Fuel Safety Margins. Along with experts groups formed from time to time, the CSNI also maintains a small working group on nuclear fuel cycle safety. While all of groups have detailed programmes involving important aspects, this paper will focus specifically on the work of PWG No. 5 - Risk Assessment.

The CSNI set up Principal Working Group No. 5 on Risk Assessment (PWG5), composing a group of experts in PSA in 1981. In general terms, the mandate given to PWG5 was to deal with practices and methods of PSA, exchange information on national efforts to develop safety goals and to assess the role they play in licensing and to exchange information on national programmes and current research. Under the CSNI re-structuring carried out in 2000, the work of PWG5 was transferred to the newly created Working Group on Risk Assessment (WGRisk). The mandate and programme of work remained unchanged.

Over the past 20 years, PWG5 and now WGRisk have looked at the technology and methods used for identifying contributors to risk and assessing their importance. Work during much of this period was concentrated on Level 1 PSA methodology, but in recent years, the focus has shifted into specific PSA methodologies modelling issues and risk informed applications.

It is important to note that the information contained in this report represents current practices in these countries as of *1 April 2002*. Since this information is subject to change, due to, advances in methodologies, changes in research programmes, etc., the reader should take these types of occurrences into account.

Belgium	Canada	Czech Republic
Finland	France	Germany
Hungary	Italy	Japan
Korea	Mexico	The Netherlands
Spain	Sweden	Switzerland
United Kingdom	United States	European Commission ¹

The authors would like to extend their appreciation to all those who contributed information and helped in producing this report.

1 . Input have been included from the European Commission in Chapter 6 and 8.

CHAPTER 1 - INTRODUCTION

1.1 Background

This report updates previous reports on the status of PSA programmes and consideration of quantitative safety guidelines and related topics which were produced by the CSNI Principal Working Group No. 5 (PWG5) since 1986. Additionally, the CNRA (Committee on Nuclear Regulatory Activities) produced 2 reports based on Special Issues meetings in 1995 and 1997

The reports referenced above are:

CSNI Report No. 124	A Survey of the Applications made of the Results of PSA of NPPs, issued 1986
CSNI Report No. 172	Status of PSA Programmes in Member Countries, issues 1989
CSNI Report No. 177	Consideration of Quantitative Safety Guidelines in Member Countries, issued 1990
NEA/CSNI/R(91)5	Status of PSA Programmes in Member Countries, issues 1991
NEA/CSNI/R(93)1	National Status Report - PSA Activities in Member and Non-member Countries, issued 1992
NEA/CSNI/R((94)15	The Use of Quantitative Safety Guidelines in Member Countries, issued 1994
NEA/CSNI/R(97)6	PSA Based Plant Modifications and Backfits, issued 1997
NEA/CNRA/R(96)7	Regulatory Use of PSA, issued 1996
NEA/CNRA/R(97)5	Review Procedures and Criteria for Different Regulatory Applications of PSA, issued 1998

Under the new CSNI structure, PWG5 has now been superseded by the CSNI Working Group on Risk Assessment (WGRisk). While the last report was issued in 1997, the status of PSA programmes and quantitative safety guidelines continues to be an essential aspect of the programme of work. At each annual meeting, a round table discussion is held to review the current status and progress over the past year. In recent years the topic of PSA Based Plant Modifications has been added to the discussion.

As one the proposals at its first meeting, WGRisk members supported the idea to update the previous reports. CSNI, at its annual meeting in December 2001 approved this action.

1.2 Structure of Report

In each of the chapters the objective is to present a 'snapshot' of the current status. The main issues considered in the different chapters are Background Information, Quantitative Safety Guidelines, Status of PSA Programmes, PSA Applications, PSA Related Research and Development and PSA Based Plant Modifications.

It is important to note that the information contained in this report represents current practices in these countries as of *1 April 2002*. Since this information is subject to change, due to, advances in methodologies, changes in research programmes, etc., the reader should take these types of occurrences into account.

Chapter 2 – PSA Environment

This chapter defines the background information on the use of PSA in Member countries. The various political and historical development of each nation contributing to this report has led to differences in how the use PSA has matured. The evolution of PSAs, whether or not they are legally required, who performs the PSA and who reviews them are included in this section.

Chapter 3 – Quantitative Safety Guidelines

This chapter builds on the information provided in Chapter 1 and presents an overview of each country's practices in regard to the use of quantitative and probabilistic safety guidelines.

Chapter 4 – Status of PSA Programmes

This chapter provides a summary of the current status of PSA Programmes in Member countries. An appendix is provided to this chapter (Appendix A) which provides a tabular form of the status.

Chapter 5 – PSA Applications

PSA experience in Member countries has grown considerably during the past 10 years. This chapter presents information on how PSAs are being applied and identifies specific applications being used for decision-making.

Chapter 6 – PSA Related Research and Development

While much progress has been made, limitations exist in the methodologies. Weaknesses, such as large uncertainties need to be further studied. Obtaining more and improved data for use in quantification is another area of increased focus. This chapter provides input from the Member countries on current and proposed area of PSA research activities.

Chapter 7 – PSA Plant Based Modifications

Following up on the report produced in 1997, this chapter presents information on insights that have been gained and the role PSA has had in safety decision-making.

References

References are provided to establish a contact point for obtaining further information or details about the PSA Programmes within the contributing countries and for providing information on specific documents

Appendix A

Appendix A provides a tabular form of status of PSA programmes in Member countries.

CHAPTER 2 – PSA ENVIRONMENT

This chapter defines the background information on the use of PSA in Member countries. The various political and historical development of each nation contributing to this report has led to differences in how the use PSA has matured. The evolution of PSAs, whether or not they are legally required, who performs the PSA and who reviews them are included in this section.

2.1 Belgium

For the existing nuclear power plants (NPPs), a periodic safety reevaluation has to be performed every ten years. In this context, a plant specific PSA has been or is being performed for each plant. Also the PSA update process will be linked to this periodic safety review, although intermediate updates are also foreseen.

For future NPPs, a PSA will be required from the licensing phase on.

There are no specific guides, which have been indicated as being strict guides to be followed for the PSA-analyses of the nuclear power plants. For the main tasks of the PSAs (accident sequence delineation, human reliability analysis, CCF modelling, accident sequence quantification, etc.) methodologies have been defined within the projects. Several reference documents (NUREGs, IAEA guidelines, other PSAs, etc.) have been considered for this purpose.

The PSAs for the Doel and Tihange plants are performed by Tractebel Energy Engineering (TEE), the architect-engineer of these plants. AVN, acting as Nuclear Regulatory Body, is reviewing the construction of the PSA models on-line. This means that technical documents (e.g. proposed methodologies, documents describing event tree construction, system reliability studies, etc.) are transmitted continuously to AVN for review. They are discussed with Tractebel and the utilities on an interactive basis. At the end of the project (after publication by TEE of the final report) AVN establishes a PSA evaluation report.

2.2 Canada

In Canada, a PSA is not formally required from a regulation point of view².

However, a consensus is forming in the CNSC on the need for a PSA for every nuclear power plant, and on its use in decision making-process. In this respect, the Cost-Recovery Policy (P-242) [1] under development states the CNSC will accept cost-benefit arguments, implicitly calling for a PSA. Also, CNSC has already announced its intent to require a PSA for every nuclear reactor, and engaged consultations with the Canadian nuclear industry on this subject.

At present, CNSC considers PSA to be an important tool to support the current deterministic approach. The CNSC regulations are based on deterministic criteria such as Defense-In Depth (DID) and Design-Basis

2. References information is contained in Chapter 8.

Accidents (DBA) using conservative assumptions. In the decision-making process, CNSC mostly uses the deterministic criteria in conjunction with the engineering judgement [2].

Nevertheless, the CNSC regulations contain probabilistic criteria for the so-called Special Safety Systems (SSS) (i.e., Shutdown, Emergency Core Cooling, and Containment Systems), and for Serious Process Failures [3]. The licensees shall strive to meet the regulatory targets. In doing so, the licensees are using unavailability models derived from the station's PSA, where one exists, or using other means[3].

To date in Canada, Ontario Power Generation (OPG, formerly Ontario-Hydro) and AECL carried out, on a voluntary basis, a number of PSA studies. These studies have been used for design verification and improvement, for support of license applications, and for assessing potential public risks from operation of nuclear facilities.

All PSAs, that are issued and used to demonstrate the licensees requests compliance with the regulatory requirements, are reviewed by CNSC. The regulator reviews only the formally issued PSAs, and it is not involved in the process of any PSA development.

2.3 Czech Republic

In the Czech Republic there are two WWER type nuclear power plants operated by utility. NPP Dukovany units (4 x WWER 440/213) have been in operation since 1985/1987. At Temelin site unit 1 (WWER 1000/320) was already completed and is under commissioning phase. The second unit is planned to be completed by the end of the year 2001.

A basic PSA study as a first step of typical PSA programme, was for NPP Dukovany unit 1 completed in 1995. The study was performed for internal initiators and full power operation. Since that milestone a "Living" PSA programme has come into force at this site. Temelin NPP basic PSA study was completed in 1996 for all operating modes and internal and external events. A Level 2 analysis was a part of study.

There is not any legal requirement to perform PSA studies by licensee in the Czech Republic. Regulatory body is still building up its own position in this field and therefore PSA activities are mainly initiated by utility based on NPP needs, experience of other countries and taking into account regulatory recommendations.

The PSAs are being performed by teams consist of NPP staff and technical support organisations experts. QA systems implemented within the studies include internal and external reviews. The Regulatory Body does not perform any detailed review of PSAs. This process is partially substituted by regulatory request to IAEA provide IPSART (IPERS) type review. The regulatory effort in review process is supposed to be increased in the future according to use of PSA applications in utility and regulatory practice.

2.4 Finland

2.4.1 Prerequisites for Risk Informed Approach

In Finland, plant-specific level-1 and level-2 PSA studies are a regulatory requirement (STUK, 1984). Plant specific PSAs have been completed for all operating Finnish plants, including internal initiators, fires, flooding, harsh weather conditions, seismic events for operation mode and internal events for low power mode. PSAs are used in support of regulatory decision making and safety management by regulatory agency and by utilities, respectively.

The guidelines for applying Living PSA in Finland are set forth in the Regulatory Guide YVL 2.8 issued by STUK in 1987 and renewed in 1997 . Living PSA is formally integrated in the regulatory process of NPPs already in the early design phase and it is to run through the construction and operation phases all through the plant service time. A condensed picture of the topical content of the Regulatory Guide YVL 2.8 is given in Figure1.

STUK and utilities (Fortum and TVO) made an agreement (1991) on how to introduce the Living PSA and to implement the regulatory and plant safety management applications under common procedure. STUK and the licensees make use of identical plant specific PSA models. Hence both parties have identical PSA information available for issues resolution and for other mutual interaction. In compliance with the requirements posed in the Regulatory Guide YVL 2.8, the licensee has to use the insights of PSA in support of decisions on safety issues at operating plants such as:

- plant changes and backfits
- training of plant personnel
- working up of emergency operation procedures
- applications of Tech Specs
- case by case assessment of risks resulted from component failures
- risk follow-up of Licensee Events
- In- Service Inspections and Testing Programmes
- maintenance and surveillance program planning
- new plant designs

As concerns a possible new plant unit a concise plant specific PSA is required as a prerequisite for issuing Construction Permit and a complete level 2 PSA is a condition for issuing the Operating License.

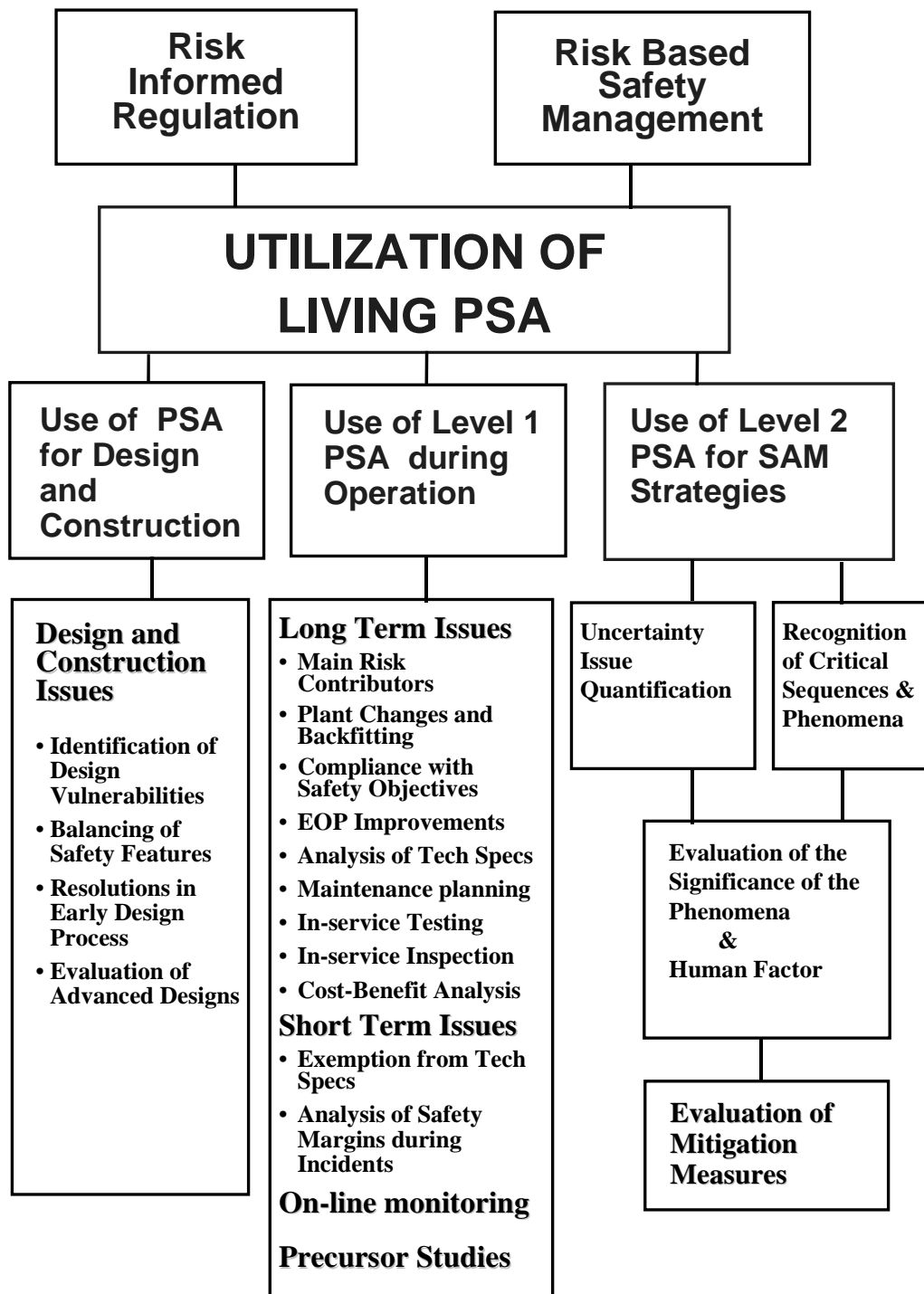


Figure1: Concept of using Living PSA for Risk Informed Regulation and Safety Management

2.5 France

2.5.1 Introduction

Although the safety of French PWRs relies firstly on deterministic principles, the probabilistic approach was considered since the 1970s as an important complement for safety analysis.

PSA is not a regulatory requirement, and probabilistic studies have been carried out without any formal requirement from the Safety Authority. However PSA results led to several specific requirements for improving the plants design or operation. Due to the interest of the PSA applications, the role of PSA in safety analysis becomes more and more important.

2.5.2 Main steps

In 1977, the Safety Authority set probabilistic safety objectives relating to the probability that a plant could be the source of unacceptable consequences (see chapter 2). However these objectives were only considered as orientation values and the demonstration of compliance with these objectives was not required.

Later on, and although it was not a regulatory requirement, partial probabilistic studies were carried out since 1980 by EdF (Electricité de France – the French utility) and IRSN Institut de Radioprotection et de Sûreté Nucléaire - technical support of the Safety Authority), and two global PWR PSAs were completed in 1990.

The first of these studies (PSA 900) concerns a standard reactor of the 900 MWe series, and was carried out by IRSN. The second study (PSA 1300) was carried out by EdF for a unit representative of the 1300 MWe series.

The PSAs have been developed independently by IRSN and EdF. However, the important problems related to methods and data were discussed together, and extensive mutual external reviews by EdF and IRSN were very helpful in order to assess the exhaustiveness of the PSAs as well as the validity of the assumptions made. Since PSA was not a regulatory requirement, the relations between EdF and IRSN were more a co-operation and a technical dialogue than a classical safety analysis process.

The results of these studies led to several important plant modifications and backfits. The most important fields of application are:

- Reduction of the risk related to dominant contributions (modifications related to the loss of redundant systems - modifications related to shutdown operating modes)
- Periodic safety assessments
- Analysis of operational events (precursor analysis)
- In the cases where a significant risk contribution is identified, the Safety Authority requires EdF to propose safety improvements in order to reduce the corresponding contribution. The proposals of EdF, and especially the corresponding PSAs, are analysed by IRSN, especially on the basis of his own studies. The acceptability of the proposals is not based on formal criteria, but some orientation values (relative or absolute) can be given case by case.

For the French-German future plant EPR (European Pressurised water Reactor), the French and German Safety Authorities agreed on a wide use of PSA as a complement to the Basic Deterministic Design during all stages of the design.

A PSA was conducted since the early beginning of the design, starting with simplified assumptions. A more complete assessment will allow the verification at the end of the design.

A preliminary EPR PSA (level 1, internal events) has been performed by the designers and included in the Basic Design Report. This preliminary PSA has been analysed by IRSN and GRS for the French and German Safety Authorities, leading to design improvements.

In parallel to PSA applications, EdF and IRSN are still working on PSA developments.

These developments concern updating of existing studies (based on plant evolutions, new data and new knowledge), extension of the scope of the studies (level 2, external events), PSAs for other designs (1450 MWe, EPR), and improvement of methodology.

Another PSA activity is the comparison of French studies with PSAs conducted by other countries operating PWRs of the same Framatome design (Belgium, South Africa, and Korea). Several interesting insights can be drawn from these exercises concerning PSA methods and data, as well as plant safety.

Besides, a PSA « regulatory » activity is in progress. In the past years, PSA studies have been performed out of the regulatory framework. PSA was not required by the Safety Authority and was carried out as an aid for safety analysis, the safety demonstration relying on deterministic principles. For that reason the relations between IRSN and the utility EdF were more a co-operation and a technical dialogue than a classical safety analysis process.

PSA is now recognised as an important tool for safety analysis, and it appears necessary for EdF and for the Safety Authority to define a more precise framework for PSA developments and applications. The preparation of a Basic Safety Rule is in progress, and a first part has been written. It has to be noted that the Basic Safety Rule will not contain any Probabilistic Criteria.

2.6 Germany

Although formalised safety requirements in Germany are still based nearly exclusively on deterministic principles, the probabilistic approach of safety assessment plays an important role since a long time.

The first “PSA” has been published in 1979 as “German Risk Study, Phase A”. This was a level 3 PRA for Biblis B NPP, a 1300 MWe, which followed largely the approach of the US “Reactor Safety Study”. In 1989 Phase B of the “German Risk Study” was published. This was a “level 2-“ PSA for the same reference plant as Phase A but without a quantification of radionuclide release frequencies and without investigation of accident consequences in the environment.

In the meantime level 1 PSAs have been performed for all NPPs in Germany, in most cases as part of the “Periodic Safety Review” (PSR) to be performed every ten years.

The current role of PSA is influenced by the general situation of nuclear power in Germany. In June 2000 the Federal Government and the four main German utilities (EnBW, RWE, VEBA, VIAG) have signed an agreement concerning the further operation of NPPs in Germany. In the agreement it is fixed that the 19 operating NPPs can produce an amount of 2516 TWh of electrical power (counting from 1 Jan 2000). This amount of energy has been calculated on the basis of a 32 years overall lifetime for each plant.

Additionally 107 TWh can be produced as a compensation for Muelheim-Kaerlich NPP which never got a final operating license. Transfer of “remaining amounts” – normally from older to younger plants – is possible.

The utilities have declared, that in the current political situation in Germany the agreement was the only rational way to allow a politically undisturbed operation of their NPPs for a foreseeable future. They do not expect that this will be the final stop for nuclear power in Germany.

The agreement has influenced the regulatory application of PSA, since a second round of “Periodical Safety Review” for all NPPs has been fixed to be performed until the end of 2000 for Stade NPP and until the end of 2009 for Neckarwestheim 2. Mandatory part of these reviews will be plant-specific PSAs. In order to become formally valid, the agreement has been converted into an amendment of the Atomic Law in November 2001.

The PSAs for the PSR are performed by, resp. on behalf of, the respective utility and reviewed – on behalf of the regulator – by technical safety organisations (TÜV, GRS).

Research in the field of PSA is performed mainly by GRS, which has also performed – and is performing - PSAs for a number of NPPs as R&D projects (PWR: Biblis B, Obrigheim, Neckarwestheim2; BWR: Gundremmingen, Philippsburg 1).

2.7 Hungary

In February 1992 the Advanced General and New Evaluation of Safety Project (so-called AGNES Project) was launched in Hungary by the Hungarian Atomic Energy Authority (HAEA). The project was aimed at a comprehensive reassessment of the safety level of the Paks NPP by the use of internationally accepted and state-of-the-art safety principles, requirements, analysis methods and tools. The first level 1 probabilistic safety assessment (PSA) study was performed for unit 3 of the Paks NPP as an integral part of the AGNES Project. That initial PSA study covered analysis of internal initiating events during full power operation of the reactor.

The AGNES project, including the full power PSA, was completed in 1994. Its major conclusion was that the general safety level of the plant corresponded well to that of other PWR’s of the same vintage. In addition, recommendations were made to extend the safety-upgrading programme of the plant and prioritise the necessary safety measures based on the results gained. In particular, quantitative results and qualitative findings from the PSA study were used for prioritising safety enhancement efforts as well as for identifying areas of safety concern that needed further investigation following the AGNES Project. An important recommendation was to make extensions to the PSA study in a number of areas. This recommendation has been taken into consideration since the AGNES Project ended. Also, emerging requirements from the regulatory authority have added momentum to continue and extend the PSA programme for the Paks plant. Accordingly, in the recent years substantial improvements and extensions have been made to the original PSA to ensure a credible, up-to-date safety assessment and to support safety enhancement at the plant by PSA applications.

To date the most important extensions of the initial level 1 PSA are as follows:

- The level 1 PSA of anticipated internal initiators at full power operation has been extended to all of the four units operating at Paks.
- The PSA for plant operation at full power has been extended with a probabilistic analysis of an annual refuelling outage (so-called shutdown PSA) including all phases of cooling down,

refuelling and start-up. Unit 2 is the reference unit for the shutdown PSA, and the results have been found applicable to the other units at Paks too.

- The most important internal hazards have been taken into account in the Paks PSA study by an analysis of internal fires and flooding for Unit 1 of Paks.
- A level 1 PSA of seismic events has been started that is to be completed in 2001. It is an important extension that ensures the inclusion of the most important external hazard in the PSA for Paks.

The PSA studies performed for the Paks NPP had the following **general objectives**:

- quantification of core damage risk
- identification and evaluation of dominant contributors to risk in low power and shutdown operating modes
- development of recommendations for safety upgrading measures to reduce the risk contribution of the most dominant initiating events and event sequences.

The general objectives listed above implicitly included some specific goals of the Paks PSA studies as follows:

- identification of plant systems, equipment and human actions important to safety during plant operation in low power and shutdown modes
- quantification of expected risk reduction by assuming the implementation backfitting proposals
- comparative evaluation of the safety level of the Paks NPP with internationally accepted target values (safety goals).

It is noted that, initially, the PSA studies had no specific objectives to directly support plant operation and outage management (e.g. by applying risk based maintenance planning during an outage, or reviewing limiting conditions of operation prescribed in the Technical Specifications). These aspects were considered as potential application areas of the developed PSA models.

Several factors affected **the scope of the PSA studies** for NPP Paks which were taken as initial conditions at the beginning of the analysis as follows. The most important attributes that characterise the scope of the Paks PSA studies are as follows:

- The existing PSA studies are all of level 1. Accordingly, the most important undesirable end-state in these analyses is core damage. In addition, boiling of primary coolant in the reactor core was treated as a separate end-state in those operational states within the shutdown PSA where it can lead to direct increase in radiation exposure of plant personnel.
- The reactor core was considered as the source of potential radioactivity release. In general, fuel handling, fuel storage and other ex-core situations have not been addressed.
- Full power operation and plant operational states of a regular, planned refuelling outage were considered in the analyses.

- In the full power PSA internal initiating events due to equipment failures, malfunction, or/and inappropriate human actions as well as internal hazards are taken into account. Currently the shutdown PSA covers internal initiating events.
- Naturally, the resources which were available for the purpose of the analysis determined the level of detail in modelling and quantification. A detailed human reliability assessment, dependence analysis was performed and accident sequence quantification included uncertainty analysis for input data and sensitivity analysis for important model parameters and assumptions. Also, availability or lack of some information/data on plant behaviour during accidental conditions in certain plant states were important determinants of level of detail and depths of the analysis too.
- The full power PSA models are unit specific with stand-alone models for each unit. The shutdown PSA uses Unit 2 as a reference unit and a representative for the other units as well.
- The studies reflect plant conditions that always correspond to the state of the Paks NPP during the latest PSA update. The reference plant state relates to plant system design and operation, operating conditions, characteristics of emergency responses, and all plant modifications effective at the date of the latest PSA update

As a summary it can be noted, that the basic PSA studies are legally required by the Nuclear Safety Directorate (NSD) of the HAEA within the framework of the Periodic Safety Reviews. The studies have been generally performed by Hungarian institutions, namely through the co-operation of the VEIKI Institute for Electric Power Research Co., the KFKI Atomic Energy Research Institute, and the Paks NPP itself. In some cases foreign engineering companies have supported the safety assessment (e.g. by probabilistic failure analysis of the containment structure).

The PSA studies are generally reviewed by the staff of the regulatory body itself (HSD), in essential cases the IAEA IPERS review services have supported the regulatory review (e.g. in case of the first full-power and off-power internal event PSAs).

2.8 Italy

In Italy there are no Nuclear Power Plants in operation - after the Chernobyl accident the Government took the decision to shutdown the operating nuclear power plants and stop the construction of new ones.

As consequence the activity and the research related to PSA aspects underwent a significant reduction, the remaining activities were aimed principally at maintaining competencies and skills, as well as R&D capabilities.

However in the nineties ENEL (Italian National Electric Utility), ANPA (the National Agency for Environment Protection acting as Nuclear Regulatory Body) and ENEA (Italian National Agency for New Technologies, Energy and the Environment) have been involved in the safety assessment on the probabilistic standpoint of the so called Innovative Reactors, like SBWR, AP 600 and PIUS through a series of international collaborations with industries, utilities and research organisations.

In more recent years PSA activities have mostly been directed towards nuclear installations other than nuclear reactors and particularly have concerned the application of the probabilistic approach with regard to the safety assessment of international fusion projects, as e.g. fusion reactor ITER (International Thermonuclear Experimental Reactor) (ref.1, 2) and experimental facility IFMIF (International Fusion Materials Irradiation Facility) (ref.3, 4).

Other topics that fall within the PSA research activities pertain to the reliability of accelerators in the frame of ADS (Accelerator Driven Systems) development and the reliability of Passive Systems, which is the object of an ongoing project (2001-2004) supported by the European Community in the context of the Fifth Framework Programme.

Moreover there is a significant participation to meetings, supported by international organisations, like IAEA, concerning the diffusion and application of the probabilistic safety assessment in nuclear field and the establishment of a consistent framework for conducting a PSA in non-reactor nuclear facilities (NRNFs), in this context a specific interest is in regard of installations for intermediate and final storage of radwaste.

2.9 Japan

The safety of Nuclear Power Plants (NPPs) in Japan is secured by stringent safety regulations based on the deterministic method, minimising the possibility of a severe accident to a technologically negligible level. Though PSA is recognised as the convincing tool of supplementing the deterministic method to discuss balanced design and procedures and examine accident management of NPPs, PSA itself is not required in the current regulatory procedures. With the progress of PSA technology and study of severe accident phenomenology, application area of PSA has been expanded in Japan. Accident management (AM) strategy has been made based on PSA; namely AM measures have been extracted based on PSA results and the effectiveness of the AM implemented has been confirmed by PSA. In periodic safety review (PSR) PSA has been revised to assess the current plant situation of safety every ten years. The technical specification has been revised in detail, especially after JCO accident, laying stress on examination, where PSA has been applied on trial to determine allowed outage time (AOT) and the safety criteria to be used have been also investigated. Under these circumstances the tendency toward the establishment of safety goal has been shown.

2.10 Korea

Even PSA is not formally required, the utility has been strongly recommended to utilise the probabilistic evaluation. The initiative to perform PSA was taken by the regulatory body in Korea to ensure operational safety of the nuclear power plants (NPPs) especially after the TMI-2 accident.

Korean PSA activities are carried out in many organisations: KEPRI (Korea Electric Power Research Institute) in charge of utility (Korea Hydro Nuclear Company; KHNC), KAERI (Korea Atomic Energy Research Institute), and KOPEC (Korea Power Electric Company). These activities are focused on the development of PSA models and methods, use of PSA in design, as well as PSA applications for operational safety improvement. Regulatory reviews are in charge of the Korean regulatory body, KINS (Korea Institute of Nuclear Safety).

2.11 Mexico

The PSA program in Mexico formally started in the early 80's with the conformation of different PSA groups within the different institutions of the nuclear sector: the utility (Comisión Federal de Electricidad), the regulatory agency (Comisión Nacional de Seguridad Nuclear y Salvaguardias) and the national research institutes (Instituto de Investigaciones Eléctricas and Instituto Nacional de Investigaciones Nucleares).

In 1985 a multi-institutional PSA group was formed in order to apply the PSA techniques to the evaluation of the core damage frequency for Laguna Verde Nuclear Power Plant unit 1. The group was integrated with staff members from the above mentioned organisations, under the technical project management of

the Instituto de Investigaciones Eléctricas (the Mexican Electric Research Institute). This project was developed on a voluntary basis, since there was no regulatory requirement at that time to perform a PSA.

Once this project was completed, the PSA groups within the different institutions continued their probabilistic safety assessment related activities at various levels of effort. The CFE began a project to update the PSA developed and the regulatory agency initiated their first PSA application to the safety evaluation of Laguna Verde NPP, the analysis of the station blackout scenario. The station blackout event tree was developed along with the development of front line and support systems fault trees. During the development of this analysis, and as a result of the lacking of a regulatory probabilistic model of LVNPP that would establish the initial and boundary conditions for a posterior containment response analysis, as well as the need to have an adequate tool for decision making and for benchmarking the results of the licensee plant specific evaluation, the objectives and scope of the station blackout analysis were reoriented to yield a full Internal Event Analysis for Laguna Verde NPP unit 1.

The scope of the Internal Event Analysis is the so called PSA level 1, excluding the external events and considering the full power operation of LVNPP unit 1 as initial condition. The initiating events considered involved 3 types of LOCA's inside the primary containment, one interfacing LOCA and seven transient categories. Systemic event trees were developed for each initiating event depicting the possible plant response to the initiating event and solving the core vulnerable sequences. Over 30 fault trees for front line and support systems were developed. Generic data, compiled from different sources, were used to quantify the accident sequences as well as the total core damage frequency. Uncertainty and importance analyses were performed for the total core damage frequency.

The Mexican regulatory authority, following the USNRC generic letter 88-20, requested the utility to perform an Individual Plant Examination (IPE) of Laguna Verde NPP. The IPE involved a thorough examination of the plant design and operation to identify dominant severe accident sequences and their contributors. Then the CFE proceeded to assess areas of potential improvements and to implement them when a cost-benefit analysis justifies it. The scope of the IPE is equivalent to a Level 1 and Level 2 analysis for events initiated during full power operation by internal initiating events and internal flooding events. The utility performed the front-end analysis of the IPE by updating the PSA level 1 that had been developed by the above multi-institutional project. The Instituto de Investigaciones Eléctricas was commissioned by the utility to perform the back-end analysis of the IPE using the NSAC 159 methodology. The IPE was submitted to the Mexican regulatory authority in the early 1996, and subjected to a detailed review process.

The primary objective of the IPE review process was addressed to determine whether the CFE met the intent of the Generic Letter 88-20, i.e., that the CFE (1) develop an overall appreciation of severe accident behaviour through their involvement in the IPE process; (2) understand the most likely severe accident sequences that could occur at the Laguna Verde NPP; (3) gain a quantitative understanding of the overall probability of core damage and radioactive material release; and (4) reduced the overall probability of core damage and radioactive release by modifying procedures and hardware to prevent or mitigate severe accidents.

The IPE review process checked the study for completeness and consistency and also checked that the utility took into account the full range of phenomena and/or processes that can take place during the evolution of a severe accident and that can have an important effect on the containment behaviour. Also, it was checked that the IPE explicitly addressed Unresolved Safety Issues and Generic Safety Issues. The review process involved an in-depth examination of many aspects of the Laguna Verde IPE technical tasks, e.g., examination of analytic models, pertinent input data, assessment of human factors, quantification process, and so forth. Interaction between the CNSNS staff and the utility was an important part of the review process. The review staff formulated and transmitted both general and specific questions to the

licensee in areas identified as being important. Once the licensee's response was available, it was reviewed and a meeting was held to discuss any differences that might exist between the review team findings and the licensee. An attempt was made to solve, at that moment, any contentious issues in order to yield a modification and improvement of the IPE or a clarification of the review team findings.

2.12 Netherlands

2.12.1 Safety Objectives

Safety policy in the nuclear field is based on the following safety objectives.

The General Nuclear Safety Objective (see IAEA NUSSAG report entitled 'The Safety of Nuclear Installations - Safety Fundamentals', Safety Series No 110):

To protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defences against radiological hazards.

This general nuclear safety objective is supported by two complementary safety objectives:

- The Technical Safety Objective:

To take all reasonably practicable measures to prevent accidents in nuclear installations and to mitigate their consequences should they occur; to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the installation, including those of very low probability, any radiological consequences would be minor and below prescribed limits; and to ensure that the likelihood of accidents with serious radiological consequences is extremely low.

- The Radiological Safety Objective:

To ensure that in all operational states radiation exposure within the installation or due to any planned release of radioactive material from the installation is kept below prescribed limits and as low as reasonably achievable, and to ensure mitigation of the radiological consequences of any accidents.

2.13 Spain

As it was described in previous yearly reports on the Spanish PSA Programme Status for former CSNI PGW5, since 1986 the "Integrated Programme on Performance and Use of PSA in Spain" (IP) has been the conductive document of the activities that the CSN, and Spain in general, have been carrying out in relation to the PSA. The IP was revised in 1998, when its current second edition was issued by the CSN.

In the years of experience with the IP first edition, the activities in the country went more along the line of PSA performance, first of the objectives indicated in the title of the IP. The activities in relation to the second great objective, related to the use of PSA, had been more sporadic and, in general, carried out in an exploratory way.

The second edition of the Integrated Program states the same objectives, although now the emphasis is directed towards the needed activities to apply the PSA to different fields. This current edition remarks that objective, although it also includes the remaining CSN activities on PSA review and acceptance and the utility activities to revise and update the previous PSA projects. The second edition also discusses the

remaining activities to reach to a final and common scope of all the Spanish PSA. These types of activities are the basis for the development of PSA applications.

The PSA applications, and also the improvement or validation of some of their methodological aspects, are sometimes going to need development activities, as well as research. One of the best ways to carry out that type of activities will be always that the utilities and the CSN collaborate in those developments and, as a product of that collaboration, a consensus can be reached that, within the objectives of each institution, allows that PSA become one of the bases for the rationalisation of the regulatory processes. The IP promotes that collaboration and tries to be a framework for the activities in that direction, which, in some way, might in fact imply in the future a reform of the regulatory system.

2.14 Sweden

The Swedish Government has stated in the proposition 1980/81:90, that all licensees have to publish an ASAR report every 8-10th year for the regulatory body SKI. The ASAR report have to explain the operating history from the last 8-10 years of operation and draw lessons learnt from this time period for the future safe operation of NPPs.

During the 80:ies and 90:ies the Swedish PSA work was very much linked to the program of the domestic ASAR-programs (ASAR80 and ASAR90 programs) (ASAR = As Operated Safety Analysis Report). In the ASAR80 program, the licensees had to perform their LOCA and transient PSA level-1 studies. In the ASAR90 program, e.g., the PSA level-2 studies, CCIs, shutdown studies, was planned to be performed.

A PSA had to be published and reported to the regulatory body SKI, every 8-10th year and as an appendix to the respective ASAR report. The regulatory body SKI reviewed both the ASAR reports and the PSA studies and a recommendation was thereafter given to the Swedish government.

Since 1998 and after that **The Swedish Nuclear Power Inspectorate's Regulations Concerning Safety in Certain Nuclear Facilities and General Recommendations Concerning the Application of the Swedish Nuclear Power Inspectorate's Regulations above.** The Regulation SKIFS 1998:1 went into force in 1998 and the PSA activities are after that time more and more split from the earlier Swedish ASAR-program. Domestic PSAs are nowadays planned to be updated on an annual bases. The SKI inspection and follow-up of the PSA-activities at the licensees are therefore also nowadays based on what is said and demanded in the SKIFS 1998:1. In the SKIFS 1998.1, the regulation is very much "activity oriented" and put demands on licensees to have e.g., an active safety work and organisation, to be responsible for necessary researching. The review of the Swedish PSA work is also more activity oriented today than earlier. Random inspections or checks are performed on the structure of fault-trees and fault-tree logic.

2.15 Switzerland

The first contract for the development of a probabilistic safety assessment (PSA) for a Swiss nuclear power plant was signed in 1983. The contract aimed at the development of a level 1 PSA for the Beznau nuclear power plant. In 1987, level 1 and level 2 full power PSAs were required by the Swiss Federal Nuclear Safety Inspectorate (HSK) for all Swiss nuclear power plants. Four years later, HSK additionally required the licensees to develop plant specific low power and shutdown PSAs including external events.

In the meantime, PSAs for all Swiss nuclear power plants have been both completed by the plant operators and reviewed by HSK. The plant specific PSAs include internal and external events such as fires, flooding, earthquakes, air craft impacts and high winds. Level 1 PSAs have been developed for full power as well as for shutdown mode. Several intermediate updates of the PSAs have been performed. For every periodic

safety review an entirely updated PSA has to be submitted to HSK. Entirely updated PSAs for the Beznau and Mühleberg nuclear power plants are currently under review at HSK. Furthermore, HSK required an update of the probabilistic seismic hazard analysis. In particular, the systematic assessment of uncertainties will receive high priority in the approach to be used (NUREG/CR 6372).

PSA studies are being used as the basis for the development of accident management guidance, aimed at reducing the vulnerabilities to severe accidents. A move to update HSK's licensing requirements towards risk-informed regulation has also been initiated at HSK. HSK has completed the development of its own plant specific living PSA models for all Swiss nuclear power plants. Also, by the end of 2002, HSK will have completed the development of the essential parts of the Accident Diagnostics, Analysis and Management (ADAM) system. ADAM is used for PSA level-1/-2 applications (e.g., review of success criteria, containment loads and accident source terms) and to assist the HSK emergency response team.

2.16 United Kingdom

2.16.1 PSA in the UK - Background

The legal requirement in the UK is that the operators of nuclear plants should conform to the Health and Safety at Work etc. Act 1974 (HSW Act) which requires them, so far as is reasonably practicable, to ensure that their employees and members of the public are not exposed to risks to their health and safety. This means that measures to avert risk must be taken unless the cost of these measures, whether in money, time or trouble, is grossly disproportionate to the risk which would be averted. Hence, the risk should be reduced to a level which is as low as reasonably practicable – the ALARP principle. The term “reasonably practicable” is not defined in the legislation but has been established in the courts as a result of cases brought under the HSW Act.

The application of the ALARP principle requires that risk assessment is carried out which, for nuclear plants, involves assessments against both qualitative/ deterministic criteria and Probabilistic Safety Analysis (PSA).

Probabilistic techniques and targets have been used in the UK since the early 1970s in the design of the Advanced Gas-cooled Reactors (AGRs). In particular, for Hartlepool and Heysham 1, a probabilistic analysis which looked at individual fault sequences was used to complement the deterministic approach that had been used until then. This was followed by Heysham 2 and Torness where Level 1 PSAs were carried out during the design process for internal initiating events.

For the PWR at Sizewell B, PSA was carried out throughout the design process. The initial Level 1 PSA at the PSR stage was followed by two PSAs at the PCSR stage – a Level 1 PSA by the architect-engineer (NNC) and a Level 3 PSA by the vendors (Westinghouse). For the POSR, a full scope Level 3 PSA was produced which addressed all internal initiating events and all internal and external hazards, and covered all the modes of operation of the plant including full power operation, and low power and shutdown modes.

Since then, PSAs have been progressively carried out for the earlier reactors. These have been done as part of the Long Term Safety Reviews (LTSRs) carried out for the Magnox reactors and continued with the Periodic Safety Review (PSRs) which are now carried out every 10 years for all nuclear facilities. One of NII's requirements for the PSR safety case is that it includes a plant specific PSA.

The application of PSA to nuclear chemical plants first started in the UK in the 1980s. It is worth noting that these facilities differ from reactors in several important respects - greater diversity of technologies,

fissile material and waste is handled throughout the nuclear installation and thus the hazard is more distributed. The systematic use of PSAs within the safety case began in the mid 80s as part of the programme of Fully Developed Safety Cases (FDSCs), though some PSA had been in use earlier. The UK nuclear chemical plants undergo PSRs in the same way as nuclear reactors and the existing FDSCs are in the process of being revisited. PSA is incorporated into the Continued Operation Safety Reviews (COSRs) which are the same as PSRs. As a result of dialogue and agreement between the licensees and the regulator, the COSRs benefit from improved hazard and fault identification processes together with much more closely defined and explicit links between the PSA and the plant engineering and operation. As with reactor PSRs, a key element of COSRs is to consider whether or not the risks are ALARP and suggest improvements where necessary. Recently IAEA has published the TECDOC on "Procedures for Conducting Probabilistic Safety Assessment for Non-Reactor Nuclear Facilities" (IAEA-TECDOC-1267; Jan 2002). Both the UK nuclear industry and the regulator contributed to this report in separate phases of the work

2.16.2 Requirement for a PSA

PSAs are required for all nuclear installations in the UK to evaluate the design of the plant and to demonstrate that the risk to workers and members of the public is both tolerable and as low as reasonably practicable (ALARP). The PSAs need to address the probabilistic criteria given in the Safety Assessment Principles for Nuclear Plants (SAPs) which relate to the maximum effective dose to a member of the public, the individual risk of death of a worker, a large release of radioactivity, plant damage (which is defined as a degraded core in the case of a reactor) and an inadvertent criticality (for a plant other than a nuclear reactor).

The PSAs which currently exist have been produced or updated either as part of the design process for a new nuclear facility or as part of a Periodic Safety Review for an existing plant. The production of the PSA is the responsibility of the licensee. However, it is usually the case that the detailed work is subcontracted to consultants. NII does not require licensees to use any specific analysis methods, models or data in their PSAs so that the licensees are free to carry out the analysis in any way they choose as long as it can be justified that they are suitable/ fit for purpose. Although there was a high degree of variability in the scope, level of detail and quality of the early analyses, there is now a relatively high level of uniformity in the PSAs currently being produced.

All the PSAs produced are required to undergo an independent peer review by the licensees before they are submitted to NII. NII then carries out its own regulatory review of the PSA. There is no agreed standard procedure for carrying out the assessment of a PSA and this generally takes account of the level of risk from the plant, the complexity of the PSA and political considerations. For reactors and other high risk plants, the review usually involves a relatively detailed assessment carried out in-house often with the support of external consultants.

Guidance to NII assessors in carrying out an assessment of a PSA is given in the SAPs and the Technical Assessment Guides (TAGs) which give guidance on the interpretation of the SAPs and the specific topics that an assessor may need to address. They do not give formal acceptance criteria for safety case issues or the PSA and this relies heavily on the judgement of the assessor who is carrying out the assessment.

In addition, the licensees are required to produce their own safety principles which provide the framework for their staff to produce safety cases and PSAs. For example, British Energy and BNFL Magnox Generation have both produced their own Nuclear Safety Principles for their reactors which have been incorporated into the formal company standards. This provides a framework for assessing the level of safety of existing plants by applying both deterministic and probabilistic criteria together with specific advice to analysts on the quantitative aspects of performing ALARP arguments. The comparisons which

have been made between the SAPs and the licensee's guidance have identified some differences and attempts are being made to resolve them.

2.17 United States

The NRC has for many years developed and adapted methods for doing probabilistic safety assessments (PSAs) and performance assessments (PAs) to better understand risks from licensed activities. The NRC has supported development of the science, the calculation tools, the experimental results, and the guidance necessary and sufficient to provide a basis for risk-informed regulation. By the mid-1990s, the NRC had a sufficient basis to support a broad range of regulatory activities. The Commission's 1995 PSA policy statement provides guidance on risk-informing regulatory activities. The 1995 policy statement on the use of probabilistic risk assessment (PSA) methods in nuclear regulatory activities says the following:

The use of PSA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PSA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defence-in-depth philosophy.

PSA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PSA should be used to support the proposal of additional regulatory requirements in accordance with the Backfit Rule (Code of Federal Regulations, *Title 10, Energy*, Part 50.109, "Backfitting"). Appropriate procedures for including PSA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.

PSA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgements on the need for proposing and backfitting new generic requirements on nuclear power plants licensees.

The Commission also said -

Given the dissimilarities in the nature and consequences of the use of nuclear materials in reactors, industrial situations, waste disposal facilities, and medical applications, the Commission recognises that a single approach for incorporating risk analyses into the regulatory process is not appropriate. However, PSA methods and insights will be broadly applied to ensure that the best use is made of available techniques to foster consistency in NRC risk-based decision-making.

In issuing the policy statement, the Commission said it expected that implementation of the policy statement would improve the regulatory process in three ways: by incorporating PSA insights in regulatory decisions, by conserving agency resources, and by reducing unnecessary burden on licensees. The movement toward risk-informed regulation has indeed sharpened the agency's (and, therefore, the licensees') focus on safety, reduced unnecessary regulatory burden, and an effective, efficient regulatory process. A collateral benefit is the opportunity to update the technical bases of the regulations to reflect advances in knowledge and methods and decades of operating experience. In line with the NRC's goal of

increasing public confidence, the agency is considering risk-informed regulation openly, giving the public and the nuclear industry clear and accurate information and a meaningful role in the process.

In 1998 the agency formally defined risk-informed regulation as an approach to regulatory decision-making that uses risk insights as well as traditional considerations to focus regulatory and licensee attention on design and operational issues commensurate with their importance to health and safety. A risk-informed approach enhances the traditional approach by: (a) explicitly considering a broader range of safety challenges; (b) prioritising these challenges on the basis of risk significance, operating experience, and/or engineering judgement; (c) considering a broader range of counter measures against these challenges; (d) explicitly identifying and quantifying uncertainties in analyses; and (e) testing the sensitivity of the results to key assumptions. A risk-informed regulatory approach can also be used to identify insufficient conservatism and provide a basis for additional requirements or regulatory actions.

CHAPTER 3 – QUANTITATIVE SAFETY GUIDELINES

This chapter builds on the information provided in Chapter 1 and presents an overview of each country's practices in regard to the use of quantitative and probabilistic safety guidelines.

3.1 Belgium

Except for the evaluation of the required protection against external events (where the probabilistic criteria of SRP section 2.2.3 are used), no probabilistic safety criteria have been defined in Belgium to evaluate the safety of the operating nuclear power plants. As a direct consequence, the results of the PSAs are not used to show compliance with any criteria.

Until now, the PSAs were not used for quantified cost/benefit analysis, nor is it foreseen in the near future.

3.2 Canada

The CNSC regulations include the quantitative criteria for the unavailability of Special Safety Systems. There is no other quantitative criteria in CNSC regulatory documents with regard to PSAs.

In spite of the fact that Canada has presently no formal policy on PSA and its application, two PSA related regulatory documents, that refer to quantitative probabilistic criteria, are in project but they have not reached the level where a draft could be accepted for publication [4]:

- Regulatory Policy on the Use of both Deterministic and Probabilistic Criteria for Regulatory Decision Making (P-151) (presently on hold), policy on balanced regulatory decision-making process, and
- Guidance on the Balanced Use of Deterministic and Probabilistic Criteria in the Decision-Making (G-152) (presently on hold), guidance on balanced regulatory decision-making process.

Given CNSC's intent to use PSA for risk-informed decision making process, it is expected the two documents on hold to be revisited to make them consistent with the PRA policy.

3.3 Czech Republic

Safety demonstration of Czech Nuclear Power Plants currently relies on deterministic principles. Probabilistic safety assessment is considered as a support decision tool, always used in complement of the deterministic approach. There are not any quantitative safety goals/limits, although some probabilistic guidelines, based on IAEA and US NRC documents have been set up as indicative values.

3.4 Finland

3.4.1 Numerical Safety Objectives

The following numerical design objectives cover the whole nuclear power plant:

- The mean value of the probability of core damage is less than $1E-5/a$.
- The mean value of the probability of a release exceeding the target value defined in section 12 of the Council of State Decision (359/91) must be smaller than $5E-7/a$. However the containment has to be designed in such a way that its integrity is maintained with a high likelihood in case of both low and high pressure core damage.

The risks associated with various accident sequences of the PSA are to be compared with each other to ensure that no dominant risk factors deviating from the common risk level remain at a plant.

3.5 France

In 1977, during the examination of the major technical options for the 1300 MWe plants, the Safety Authority set an overall probabilistic objective expressed as follows:

- In general terms, the design of a plant that includes a pressurised water nuclear reactor should be such that the overall probability that the plant could be the source of unacceptable consequences should not exceed 10^{-6} per year.

This implies that, whenever a probabilistic approach is used to assess whether a family of events must be taken into account in the reactor design, the family must effectively be taken into account if its probability to lead to unacceptable consequences exceeds 10^{-7} per year.. »

It has to be noted that :

- a) The overall objective is stipulated in terms of « unacceptable consequences », but these « unacceptable consequences » are not specified by legislation or regulation.
- b) The 10^{-6} value is an « objective » for a PWR plant, but EdF is not required to demonstrate that this objective has been achieved.
- c) The 10^{-7} per year value is more practical for operational uses and is used in the approach to determine the risks generated by external events ; for example, the value is applied to several families of air crash events.

During the PSA applications, the acceptability of the utility proposals are not based on formal criteria, but some orientation values (relative or absolute) can be given case by case. Some examples are the following:

- A probabilistic target of 10^{-6} per reactor/year for the CMF related to shutdown conditions was set by the Safety Authority (considering in particular that during shutdown containment integrity is not guaranteed).

- In the framework of the 900 MWe series Periodic Safety Review, each sequence of the PSA 900 with a CMF $> 10^{-7}$ per year was analysed, in order to investigate the interest and the feasibility of plant improvements. Particular attention was paid to sequences potentially resulting in early containment failure.
- A probabilistic analysis of operating events is carried out in France since 1994. The aim of the quantitative analysis is to assess the risk increase (in term of core damage probability) due to the incident. An incident is considered as a precursor if the risk increase is higher than 10^{-6} per event.

The Safety Authority required to take particular measures if the risk increase is higher than 10^{-4} , and to assess the benefit of these measures.

For the French-German project EPR (European Pressurised Reactor), the French and German Safety Authorities gave the following very general probabilistic objectives:

- a reduced CMF compared to existing plants
- « practical elimination » of sequences with potential for large early releases.

In order to fulfil these objectives, the designers have proposed probabilistic safety objectives as orientation values that give useful guidance but are not strict limits and do not correspond to a requirement of the Safety Authorities. Examples of these probabilistic objectives are a value of 10^{-6} per year for the CMF due to internal events, respectively for power states and for shutdown states.

Generally speaking, the French Safety Authority considers PSA as a fruitful tool, notably for improving the safety of French PWRs by identifying where design and operating modifications are worthwhile, and for ranking problems in order of importance. However, they are not in favour of setting probabilistic criteria. As the French Safety Authority's aim is to improve safety, they believe that the use of PSAs for relative consideration is more efficient than the use of absolute criteria.

3.6 Germany

There are no quantitative safety guidelines in Germany. Decision making based on PSA results is made in an informal way, applying the results of former (domestic and foreign) PSAs for orientation about an “de-facto accepted” level of risk.

For the French-German EPR project quantitative criteria have been formulated (see French contribution, section 3.5).

3.7 Hungary

The implementation of the Act CXVI on Nuclear Energy is performed through decrees (governmental and ministerial), and the Governmental decree No. 87/1997. (V.28.) Korm. regulates the statute of HAEC and HAEA. Governmental decree No. 108/1997. (VI.25.) Korm. and its five enclosures, the Nuclear Safety Code define the mandatory safety requirements for nuclear facilities. The Nuclear Safety Code is made up of the following five volumes:

1. Regulatory Procedures of Nuclear Power Plants
2. Quality Assurance of Nuclear Power Plants

3. Design Requirements of Nuclear Power Plants
4. Operation Requirements of Nuclear Power Plants
5. Regulation of Research Reactors

The code covers the regulation of both nuclear safety and radiation protection to the extent HAEA has responsibility in these two areas. The structure of these regulations is in good conformity with that of the relevant IAEA codes. Volume No. 1 contains the conditions for the procedures of a regulatory body as required by legal documents.

The scope of authorisation, inspection and other procedures (types of licences, authorisation of personnel, reporting requirements for the licensees, etc.) are in line with the IAEA safety guides to code No. 50-C-G. The details of the procedures comply with the Hungarian legislation and practice. Volume No. 2 includes the quality assurance requirements adapted from the IAEA Code No. 50-C-QA (Rev. 2). The requirements for design and for operation (Volumes No. 3 and No. 4) also take an account of the relevant and applicable IAEA codes and guides. Volume No. 5 is a stand-alone regulatory document for small-scale nuclear facilities (practically the research reactor and the training reactor in Budapest) prepared as a synthesised and simplified version of the other four volumes of the Nuclear Safety Code.

Volume No. 3 contains the regulatory requirements for the design of NPP. Among other issues, this volume summarises basic principles for the justification of safety, as well as general requirements for the safety analysis and PSAs, too. These items are given in the Sub-section 3.7 of the Code, Volume 3 as follows:

3.7. JUSTIFICATION OF SAFETY

3.7.1. Basic principles

3.050. The fulfilment of the general safety requirements relating to the design process shall be assessed and validated during the design, construction, commissioning and operation of the nuclear power plant.

3.051. The analysis shall be performed by using well-documented and validated methods of analysis and on the basis of a specific and realistic (representative) database. The methods of analysis shall be validated by a comparison with actual processes, appropriate experiments or tests. If this is not possible, a comparison with other, different calculation methods is also acceptable.

3.052. The safety analysis of the nuclear power plant shall cover all-important sources of ionising radiation and all planned modes of operation of the facility.

3.053. The analysis of the anticipated operational occurrences and design basis accidents shall be performed in a sufficiently conservative manner.

3.054. It shall be demonstrated that in the case of the failure of any of the safety protection functions the frequency of the anticipated operational occurrences and design basis accidents is lower than 10^{-5} /year (for each of the reactor blocks in the case of a multi-unit nuclear power plant), proving in this way the fulfilment of the acceptance criteria relating to the anticipated operational occurrences and design basis accidents.

3.055. *It shall be demonstrated that*

- a) *the strength characteristics of structural materials correspond to the calculated maximum stresses with adequate margin to allow for the effects of ageing and to provide safety in all operational states and anticipated accident conditions;*
- b) *in any part of the structure the intensity of stresses may not exceed the fracture toughness calculated for the actual temperature - i.e. discontinuities in the material of the structure may not propagate;*
- c) *in the case of design basis accidents the loads on the reactor's primary circuit, main steam system and the loads on the containment will remain below the acceptable design value, including embrittlement and yield faults and the load on the fuel rods;*
- d) *in spite of dimensional and geometric changes of the core, that proper cooling of the reactor core can be ensured.*

3.056. *The probability of the occurrence of a heat transfer crisis in the case of anticipated operational occurrences shall be sufficiently low at any position of the core.*

3.057. *In the case of design basis accidents, the primary circuit of the reactor must remain in such a condition that the short- and long-term cooling of the fuel rods can be maintained.*

3.7.2. Safety analysis

3.058. *The analysis should be started by making the list of possible initiating events.*

3.059. *The fault sequences ensuing from the initiating events should be identified.*

3.060. *The effect of the fault sequences on the technological processes of the power plant should be determined.*

3.061. *For the fault sequences which would lead to radioactive releases and radiation doses originating from a direct and/or an indirect radiation exposure, the maximum effective radiation dose for persons within and outside the plant site should be estimated.*

3.062. *The initiating events may be grouped. For each group it is possible to specify an initiating event which, compared to the other initiating events in the same group, will have the most unfavourable consequences. It is then sufficient to perform the analysis relating just to this initiating event.*

3.063. *Without the assumption of a further independent fault, none of the anticipated operational occurrences and design basis accidents may cause consequences which are more severe than those specified for the given group of initiating events.*

3.064. *The fault sequences can be grouped for the purpose of fault analyses, and for each group a „bounding” case can be specified. Bounding cases should be selected having regard to the relevant physical and chemical processes involved and the activation of the safety-related systems and components. The consequences of the bounding case should be at least as severe as every member of the groups of fault sequences which they are claimed to bound.*

3.065. *The radiation protection calculations should cover the exposure to direct radiation, the inhalation and ingress of radioactive materials and take the physical and chemical characteristics of the released radioactive material into consideration.*

3.066. *The analysis should also prove that the adverse effects which may arise as a consequence of the fault sequence would not jeopardise the claimed operational capability and performance of the required safety-related systems and components.*

3.067. *During the analysis of operator interventions the circumstances of monitoring the plant, diagnosing plant state, decision-making and implementing actions should be taken into consideration. The analysis should demonstrate that the required activities are feasible and can be performed in the time available.*

3.068. *The validity of data used for the analyses should be proven by comparing them with well-established actual data, using experimental results or other means and any extrapolation of data should be shown to be valid.*

3.069. *Sensitivity analyses should be performed in order to evaluate the uncertainty of fault analyses to the assumptions made, the data used and the methods of calculation. The results of the sensitivity analyses and the conclusions that can be drawn from those results should be summarised. Where these results prove to be sensitive to the assumptions of the model, if possible, additional analyses should be performed using methods and procedures which are independent of those used previously.*

3.070. *In order to ensure the appropriate reliability of the analyses an independent check should be performed, where possible using different methods.*

3.071. *The fault analysis performed during the design phase should be reviewed upon closing the design phase and, where necessary revised to take account of:*

- a) *changes to the plant or the system of operation at the design, construction stage or during its operation*
- b) *any new relevant technical and scientific knowledge concerning the behaviour of the plant and fault potential*
- c) *any material property changes and deterioration due to ageing not previously taken into account.*

3.7.3. Probabilistic safety analysis

3.072. *In order to evaluate the level of risk arising from the plant to be made, a probabilistic safety analysis of the operational system should be performed and the reliability of this analysis should be judged. The probabilistic safety analysis should demonstrate that the power plant has a balanced design, which means that it does not have any particular series of anticipated operational occurrences or characteristics which would make a disproportionate contribution to the overall risk level.*

3.073. *In the probabilistic safety analysis, the initiating events leading to a considerable radiation dose rate should be evaluated. The analysis should be performed for the entire spectrum of the fault sequences which could occur, even including the severe accidents, taking account of the possibilities of individual system and component failures, the non-availability of systems and components due to maintenance or testing, common cause failures, human errors and also those failures which occur as a consequence of events preceding the given initiating event.*

3.074. *It is necessary to identify the external and internal hazards affecting the safety of the facility. These hazards should be treated as potential initiating events of fault sequences and, where necessary, they should be taken into account together with other fault combinations of the facility systems and components.*

3.075. *The estimated frequency of occurrence and consequences of each of the identified fault sequences should be determined. Where a group of sequences of events is represented by a so-called bounding case, this bounding case should be given a frequency value equal to the frequency sum of the group represented by it.*

3.076. *For the thermal-hydraulic analyses, radiation protection analyses and for the determination of the frequency and probability values used in the probabilistic safety analysis, best estimate methods and data are preferred. Where this is not practicable, reasonably conservative assumptions should be made.*

3.077. *Only reliable and authentic statistical data may be used. The source of the data, the size of the sample and the uncertainty in the data should be specified. In the case of a change in the source data the differences between the design data and the plant operating conditions should be taken into account and evaluated.*

3.078. *Where no relevant statistical data are available, estimations should be made and their basis should be stated. In case of such estimated data special attention should be paid to determining the sensitivity of the results obtained from the probabilistic safety analysis.*

3.079. *The probability data of human errors should cover the complexity of the individual tasks, the psychological effects (such as stress, physical condition, level of inspection, working methods), the physical environment and potential dependencies between separate activities (between staff members performing either similar or different activities). All circumstances relating to the systems, components or procedural requirements which promote reliable human activities, should be taken into consideration.*

3.080. *Where due to the inadequacy of the data or to the non-availability of an appropriate model it is not possible to calculate the frequency of occurrence, an engineering estimation should be used to judge the contribution to the estimated frequency value of the faults.*

3.081. *The probabilistic safety analysis should also provide information about reliability and about the maintenance and testing of the safety-related systems and components.*

As a summary it can be noted that very few real quantitative probabilistic requirements are explicitly defined within the safety code, the acceptance decisions generally consider international – mainly IAEA – recommendations and practices.

3.8 Italy

The general design criteria for PWR NPP issued in eighties in Italy defined the following objectives to be verified by Probabilistic Safety Study:

- for each single sequence the annual probability of exceeding the core coolability limits shall not be higher than $10^{-6} - 10^{-7}$

- the annual overall probability of exceeding the above mentioned coolability limits shall not be higher than $10^{-5} - 10^{-6}$

3.9 Japan

3.9.1 Safety Goal

The Nuclear Safety Commission (NSC) stated in the White Paper on Nuclear Safety published in March, 1999, that NSC promotes the discussion on the establishment of safety goals from a comprehensive point of view, taking into consideration the international trends and outcomes from various PSA studies. It is expected that the establishment of the safety goals will contribute to the further clarification of the roles of various activities for assuring safety and more systematic approaches for enhancement of safety.

This policy was reconfirmed by the NSC's official statement announced on January 17, 2000, on its basic policy for near term activities. Here the NSC stated that it will propose a direction for establishment of its safety goals and will soon set up a committee to discuss on this matter. It also emphasised the importance of risk management based on quantitative risk analysis and that the discussion in the new committee will be directed to promotion of risk based safety management.

3.9.2 Safety Criteria to determine AOT

Advisory committee on operations management, under the Ministry of International, Trade and Industry (MITI) has temporarily decided the safety criteria for the determination of AOT in technical specification using PSA as follows;

- Safety criteria for risk increment for one outage
 - Allowed incremental conditional core damage probability (ICCDP) 5E-7
 - Allowed incremental conditional large early release probability (ICLERP) 5E-8
- Safety criteria for annual risk increment
 - Allowed annual increment of core damage frequency (CDF) 1E-6/ry
 - Allowed annual increment of large early release frequency (LERF) 1E-7/ry

These criteria will be authorised by the upstream WGs.

3.10 Korea

It has been recommended by KINS to perform plant-specific level 2 PSA, including external events (mainly fires, floods, and seismic) analyses, identifying the mitigating plant features against severe accidents. As a result, in the case of new plants, level 2 PSAs are done or being performed, depending on their construction schedule. On the other hand APR 1400, i.e. the next generation plant in Korea, decided to do full scope (Level 3) PSA to use PSA insights in a standardised design and operation.

Recently, Korean regulatory policy statement for the safety assurance against severe accident is announced. The objective of this statement is to establish the following essential elements in Korea:

- 1) Setting-up of probabilistic safety criteria;
- 2) Implementing PSA in the design and operation;
- 3) Providing capability for the mitigating and preventing features against severe accidents;
- 4) Establishing of severe accident management program (SAMP).

These will be used to provide the means of safety improvement for nuclear facilities, to assure the consistency in the execution of PSA review, and to establish the basic guidelines to the technical requirements for each specific PSA areas. For example, the policy states as:

“PSAs should be performed in order to determine countermeasures such that the risk from the NPPs is reduced to as low as reasonably achievable. As for relatively high probability accident scenarios producing plant damage, available means for accident prevention and mitigation should be identified and implemented in the design and operating procedures of NPPs, through cost-benefit analysis.”

The regulatory guidelines and criteria have been mainly developed by KINS. It was also urgently needed that a requirement with respect to the new design concepts, such as APR 1400, i.e. Korean Next Generation Reactor (KNGR), be developed. On the basis of a thorough investigation of past PSA experiences, the fundamentals of basic and specific PSA requirements were provided. Furthermore, issues arising from the research results of severe accident and PSA, as well as foreign country's experiences, were addressed in order to establish a better regulatory position. The regulatory requirements are hierarchically structured, consisting of:

- General safety criteria,
- Specific technical requirements, and
- Safety regulatory guides.

The specific technical requirements consist of common requirements and specific requirements. The specific requirements on PSA are directly linked with general safety criteria, giving final details of the mandatory requirements, and also linked with safety regulatory guides. The major contents of specific safety requirements consist of the work scope and methodology, the evaluation of PSA results, as well as the acceptance criteria. During the current licensing period for the preliminary design, we are willing to reflect the actual review insights to these requirements.

3.11 Mexico

It is now widely recognised that PSAs furnish figures that can be complementary used to assist safety decisions. As a result of this, the Mexican regulatory authority has initiated activities and devoted efforts to the development of Probabilistic Safety Criteria.

Once the Individual Plant Examination for Laguna Verde NPP was reviewed and approved by the CNSNS, and further based on the recommendations of the review team, it is been subjected to an updating and improvement process. This will lead to an updated and living PSA model that could be used to support

different applications related with changes to the licensing basis, technical specifications and operation and maintenance activities.

The CNSNS initiated a project aimed to develop an adequate framework to evaluate the above mentioned applications. Based on the USNRC regulatory guides, the CNSNS has developed and issued for comments the Regulatory Guide 1.00, which formally settles an approved methodology for using probabilistic safety assessment in risk informed decisions on temporary and permanent plant-specific changes to the licensing basis for Laguna Verde NPP. This regulatory guide establishes safety criteria based on the loss of integrity of the reactor core, i.e., core damage frequency, and on the magnitude of a large radioactive release, i.e., large early release frequency, which have to be met in order for the regulatory authority to approve the PSA application. It is important to remark that the Regulatory Guide compels the licensee to comply with both criteria.

For permanent changes, the risk acceptance guidelines established tend to reject applications that result in increase in CDF above 10^{-5} per reactor year, and to consider applications with CDF calculated increase in the range of 10^{-6} to 10^{-5} per reactor year if it can be reasonably shown that the total CDF is less than 10^{-4} per reactor year. When the calculated increase in CDF is very small, less than 10^{-6} per reactor year, the change is considered regardless of whether there is a calculation or not of the total CDF, except in those cases when there is indication that the CDF may be considerable higher than 10^{-4} per reactor year.

Regarding the large early release frequency, the applications are not acceptable if they result in an increase in LERF above 10^{-6} per reactor year. When the LERF calculated increase is in the range of 10^{-7} to 10^{-6} per reactor year the applications are considered if it can be reasonably shown that the total LERF is less than 10^{-5} per reactor year. When the calculated increase in LERF is very small, less than 10^{-7} per reactor year, the change is considered regardless of whether there is a calculation or not of the total LERF, except in those cases when there is indication that the LERF may be considerable higher than 10^{-5} per reactor year.

These guidelines are intended for comparison with a full-scope, including internal events, external events, full power, low power, and shutdown, assessment of the change in CDF and LERF, and when necessary, as discussed above, the baseline value of this risk metrics.

3.12 Netherlands

3.12.1 *The Technical Safety Objective*

As was discussed in the sections on the various articles of the Convention, extensive rules and regulations, derived from the IAEA NUSS Safety Codes and Guides, have been defined and formally established. No licence is issued unless the applicant satisfies the regulations. Inspections are carried out to monitor compliance with the rules. Priority is given to safety, and the licensee is aware of its responsibility for safety. Periodical safety re-evaluations are performed, to ensure that account is taken of new safety insights.

The Dutch government therefore believes that all echelons of the defence-in-depth principle have been preserved, so that there is a low probability of accidents and, should accidents occur, the probability of radiological releases is very low. Even for accidents beyond the design basis - those that might lead to serious radiological releases - measures have been taken to further reduce their probability and to mitigate the consequences, should they occur.

In the light of these measures, the Technical Safety Objective has been fulfilled.

3.12.2 *The Radiological Safety Objective*

Under the Radiological Safety Objective, the formal legal limit for the radiation levels to which members of the public are exposed is based on the Euratom 1996 Basic Safety Standards. The government has also formulated an environmental risk policy, which is taken into account.

3.12.2.1 Environmental risk policy

The concept of risk management and risk assessment was first introduced in environmental policy in the 1986-1990 Long-term Programme for Environmental Management. This concept was reassessed following debates in parliament. As part of the Dutch National Environmental Policy Plan [Lower House of the States General, 1988-1989 session, 21137, Nos. 1-2, The Hague 1989], the Minister of Housing, Spatial Planning and the Environment, the Minister of Economic Affairs, the Minister of Agriculture, Nature Management and Fisheries, and the Minister of Transport, Public Works and Water Management set out a renewed risk management policy in a document called 'Premises for Risk Management; Risk Limits in the Context of Environmental Policy' [Lower House of the States General, 1988-1989 session, 21137, No. 5, The Hague 1989]. In the following year, a separate document was issued dealing with the risk associated with radiation: 'Radiation Protection and Risk Management; Dutch Policy on the Protection of the Public and Workers against Ionising Radiation' [Lower House of the States General, 1989-1990 session, 21483, No. 1, The Hague 1990]. These two documents still form the basis for government policy on risk management.

The Nuclear Installations, Fissionable Materials and Ores Decree has recently been amended to incorporate this risk policy in the licensing process. Risk criteria are explicitly included as assessment principles for licences to be granted to nuclear power plants. The outcomes of a level-3 PSA must be compared with these risk criteria and objectives.

This concept of environmental risk management has the following objectives and steps:

- Verifying that pre-set criteria and objectives for individual and societal risk have been met. This includes identifying, quantifying and assessing the risk.
- Reducing the risk, where feasible, until an optimum level is reached (i.e. based on the ALARA principle).
- Maintaining the risk at this optimum level.

This means assuming a maximum total individual dose of 1 mSv in any year for the consequences of normal operation of all man-made sources of ionising radiation (i.e. NPPs, isotope laboratories, sealed sources, X-ray machines, etc.). For a single source, the maximum individual dose has been set at 0.1 mSv per year. In addition, as a first step in the ALARA process, a general dose constraint for any single source has been prescribed at 0.04 mSv per year. The latter value corresponds with an individual human mortality risk of 10^{-6} per year (based on a mortality factor of $2.5 \cdot 10^{-2}$ per Sv).

For the prevention of major accidents, the maximum permissible level for the individual mortality risk (i.e. acute and/or late death) has been set at 10^{-5} per year for all sources together and 10^{-6} per year for a single source.

As far as major accidents are concerned, both the individual mortality risk and the group risk (= societal risk) must be taken into account. In order to avoid large-scale disruptions to society, the probability of an accident in which at least 10 people suffer acute death is restricted to a level of 10^{-5} per year. If the number

of fatalities increases by a factor of n , the probability should decrease by a factor of n^2 . Acute death means death within a few weeks; long-term effects are not included in the group risk.

In demonstrating compliance with the risk criteria, one has to assume that only the usual forms of preventive action (i.e. fire brigades, hospitals, etc.) have been taken. Evacuation, iodine prophylaxis and sheltering may therefore not be included in these measures.

This risk management concept is used in licensing procedures for nuclear installations and all other applications of radiation sources. Guidelines for the calculation of the various risk levels have been drafted for all sources and situations. In principle, the calculations must be as realistic as possible (i.e. they should be 'best estimates').

For NPPs, this means that the level-3 PSA plays a leading role in the verification process. Specific procedure guides have therefore been drafted in The Netherlands for performing full-scope PSAs. The level-1 PSA guide is an amended version of the IAEA Safety Practice: 'Procedures for conducting level-1 PSAs' (Safety Series No. 50-P-4) and the level-2 guide is based on the IAEA Safety Practice: 'Procedures for conducting level-2 PSAs' (Safety Series No. 50-P-8).

The procedure guide for level-3 PSAs is a specifically Dutch initiative, in which the COSYMA code for atmospheric dispersion and deposition is used. It gives instructions on the pathways which should be considered, the individuals (i.e. critical groups) for whom the risks should be assessed and the type of calculations which should be performed. It also describes how the results should be presented.

Since it has been recognised that PSAs produce figures that can be used as a yardstick in safety decisions, a number of countries have developed probabilistic safety criteria. The regulatory body in The Netherlands has taken note of the INSAG-3 safety objective, i.e. the maximum acceptable frequency for core damage is 10^{-5} per year for new NPPs and 10^{-4} per year for existing NPPs.

In addition, the objective of accident management strategies should be that the majority of potential accident releases will not require any immediate off-site action such as sheltering, iodine prophylaxis or evacuation. This means that the dose to which members of the public are exposed in the first 24 hours after the start of the release should not exceed 5 mSv. The PSA can help in fixing these figures. For example, the limit of 5 mSv was used as an acceptance criterion in the design of the containment emergency venting filter for the Borssele NPP.

3.12.2.2 Minimisation of residual risk

The Rasmussen study (WASH-1400) showed that risk was not dominated by the design basis accidents, as was made very clear by the TMI-2 incident and the Chernobyl accident. For this reason, the government felt it would be useful to enhance the reactor safety concept, which had to date been based mainly on deterministically defined events such as a large break LOCA, by incorporating certain risk elements. In addition to the radiological hazard criteria already mentioned, it was decided to make various changes to the Code of Practice on Design that would define the required safety level more clearly and require the licensee to make a reasonable effort to minimise the risk. The following text was added under the heading 'Postulated Initiating Events (PIEs)':

The nuclear power plant shall be designed to cope with PIEs in such a way that it can be demonstrated in a probabilistic safety assessment that the probability of a large release is no greater than 10^{-6} per reactor-year. These PIEs may be of internal or external origin, or a combination of the two.

Large releases are releases that could lead to doses outside the plant exceeding the acceptable limits for accident conditions (see paragraphs 315 and 1003 of the Code of Practice on Design). They might necessitate the consideration of external measures (i.e. off-site countermeasures). Evidence must be provided that there is no sharp increase in risk just below the probability of 10^{-6} per reactor-year.

In the section on ‘Severe Accidents’, a more stringent form of wording was chosen in paragraph 317 (i.e. ‘shall’ instead of ‘should’):

Although the probability of severe accidents occurring is very low, these accidents shall be considered in the design so as to further reduce risks wherever these risks can be reduced by reasonable means.

3.12.2.3 Design basis accidents

The public health risks due to incidents or accidents in the design basis area are also bound to the criteria of the individual risk concept. However, a conservative deterministic analysis of the respective design basis accidents is more effective than a PSA, which is based on a probabilistic approach, for the purpose of ensuring that the engineered safety features of a particular NPP are adequate. There are a number of reasons why a conservative, deterministic approach has certain advantages over a probabilistic approach:

Design basis accidents are postulated to encompass a whole range of related possible initiating events that can challenge the plant in a similar way. These other related initiating events do not therefore need to be analysed separately.

It is much easier to introduce the required conservatism. With a probabilistic approach, uncertainty analyses need to be performed to calculate confidence levels.

By definition, design basis accidents are events that are controlled successfully by the engineered safety features. Hence, they do not result in core melt scenarios, and are considered in a PSA as being ‘success sequences’. The related radioactive releases are negligible compared with the uncontrolled large releases associated with some of the beyond-design basis accidents. In other words, a general ‘state-of-the-art’ PSA, which focuses primarily on core melt scenarios and associated large off-site releases, does not take account of the consequences of design basis accidents.

Clearly, the above dose and risk criteria are not suitable for use as rigid criteria in the conservative and deterministic approach used in traditional accident analyses. A separate set of safety criteria was therefore formulated, as is prescribed by NVR 1.1, § 1201. This set, which is part of the amended Nuclear Installations, Fissionable Materials and Ores Decree, reads as follows:

Frequency of event F per year	Effective dose (H_{eff} , 50 years)	
	Adult	Child (1 year old)
$F \geq 10^{-1}$	0.1 mSv	0.04 mSv
$10^{-1} > F \geq 10^{-2}$	1 mSv	0.4 mSv
$10^{-2} > F \geq 10^{-4}$	10 mSv	4 mSv
$F < 10^{-4}$	100 mSv	40 mSv

An additional limit of 500 mSv thyroid dose (H_{th}) must be observed in all cases.

Correspondingly the provisions concerning the dose related to normal operation as a first step in the ALARA process, a general dose constraint has been prescribed at values of 40% of the above mentioned.

3.13 Spain

No quantitative safety guidelines have been officially used in Spain. PSA results within the usual range of published results all over the world were intended and, in many cases, this intention was the basis for plant modifications. It was believed by the CSN, since the IP first edition, that formal or legal quantitative goals would not be positive for a risk analysis methodology where many aspects were, and still are, needed of research and development activities. A legal goal might also have made the objective of the analysis to derive from identifying and making safety improvements to just trying to demonstrate the goal achievement.

Nevertheless, the situation may change in the future, since some PSA applications may be needed of some kind of quantitative acceptance criteria or guidelines. In Chapter 4, this possibility is also mentioned.

3.14 Sweden

Safety demonstration of Swedish Nuclear Power Plants relies on deterministic principles documented in the Safety Analysis Reports of the individual NPPs. PSA is a supporting decision tool, used in complement with the deterministic approach.

At present there are quantitative safety goals/limits established at the Swedish NPPs, and they are as follows:

The overall objective are stipulated in terms of “unacceptable consequences”, but these “unacceptable consequences” are not specified by legislation or in regulations of SKI in terms of exact probabilities. However, SKI appraises this as a good way of working with safety and safety matters. The licensees have expressed safety goals to be followed for their own quality assured safety work, and they are;

- The 10-5/y value is an objective for the BWR and PWR plants, for the overall and annual CDF frequency. Safe shutdown has to be demonstrated for this level. Such measures are given high priority at the licensees, that have to be taken, to be able reach this safety goal.
- The 10-7/y value is for the BWR and PWR plants, is the CDF frequency for unplanned release of core inventory larger than 0,1% of the total core inventory exclusive noble gas. Such measures are given high priority at the licensees, that have to be taken, to be able reach this safety goal.

The dominating sequences to the CDF frequency shall not diverge more than a factor 10, from each other.

3.15 Switzerland

With regard to risk-informed regulation, probabilistic safety criteria had been proposed by HSK a few years ago. Currently, these criteria are being reviewed. Revised safety criteria are expected to be available for review and comment in the near future. It is intended to develop cost/benefit criteria concerning backfitting. Pilot projects are in progress concerning both the in-service inspection and the ageing surveillance program. Corresponding guidelines will be developed in the next years.

3.16 United Kingdom

3.16.1 *Tolerability of Risk From Nuclear Power Stations*

One of the recommendations that came out of the Sizewell B Public Inquiry was that HSE should “*formulate and publish guidelines on the tolerability of levels of individual and societal risk to workers and the public from nuclear power stations*”. This was done in the document “**The Tolerability of Risk from Nuclear Power Stations**” (TOR) which was issued for comment in February 1988 and in its final version in October 1992.

This specified a framework for defining the risk criteria which identified three regions of risk as follows:

- an **unacceptable region** where the risk is regarded as intolerable and cannot be justified in any ordinary circumstances,
- a **tolerable region** where the nuclear plant would be allowed to operate provided that the associated risks have been reduced to a level that is as low as reasonably practicable (ALARP) such that the costs of further improvement would be grossly disproportionate to the reduction in the risk, and
- a **broadly acceptable region** where the risk is so small that the regulator need not seek further improvements provided that they are satisfied that these low levels are attained in practice.

Regarding the levels of risk which define these three regions, TOR proposed the following:

- 10^{-4} per year as the limit of tolerability for the risk of death for a worker on a nuclear plant,
- 10^{-5} per year as the benchmark for the risk of death for a member of the public from a new nuclear power plant, and
- 10^{-6} per year for the broadly acceptable level of risk of death for a member of the public,

and these numerical values were used as the basis of the accident frequency criteria given in the SAPs.

This has recently been updated in the publication “**Reducing Risks, Protecting People; HSE’s Decision-Making Process**” which sets out an overall framework for decision taking by HSE which would ensure consistency and coherence across the full range of risks falling within the scope of the HSW Act. This framework is based on the approach described in TOR for nuclear power plants.

This report emphasises the role of Risk Assessment, both quantitative and qualitative, in the decision-making process and expands on the role of good practice in determining the control measures that must be put in place for addressing hazards.

3.16.2 *Accident frequency criteria*

The framework defined in TOR has been used as the basis for defining the numerical accident frequency criteria given in the NII Safety Assessment Principles (SAPs) which were published in 1992. The concept of a limit of tolerability has been translated into a **Basic Safety Limit** (BSL) so that the risk from the plant must be below this limit before it can be considered for licensing. In addition, a **Basic Safety Objective**

(BSO) is defined which is the point beyond which the risk is so small that assessors need not seek further safety improvements. However, the licensee of the plant is still legally required to make further improvements where reasonably practicable.

Following the publications of the SAPs, the nuclear power plant operators updated their own Nuclear Safety Principles for Gas Cooled Reactors which are now formal company standards. They include numerical accident frequency criteria which are broadly equivalent to those defined in the SAPs.

Numerical criteria are defined in the SAPs as follows:

P42	doses to the public
P43	risk to workers
P44	large release
P45	plant damage
P46	criticality incidents

These were chosen to address the risks discussed in TOR supplemented by the consideration of societal effects of lesser accidents and to emphasise defence in depth. A guiding aim was to focus assessment on the design and operation of the plant and to minimise the extent to which judgements on the safety of the plant depend on the numbers of people who live and work in the vicinity of the site. Hence, the risks of offsite consequences of accidents are not addressed directly, but rather via surrogate measures related to the plant so that a Level 3 PSA is not required

P42 Doses to the public:

The total predicted frequencies of accidents on the plant, which would give doses to a person outside the site, should be less than the values given in the following table:

Maximum Effective dose (nSv)	Total predicted frequency (per year)	
	BSL	BSO
0.1 – 1	1	10 ⁻²
1 – 10	10 ⁻¹	10 ⁻³
10 – 100	10 ⁻²	10 ⁻⁴
100 – 1000	10 ⁻³	10 ⁻⁵
> 10000	10 ⁻⁴	10 ⁻⁶

Note. A subsidiary aim should be for no single class of accident to contribute more than about one tenth of the total frequency in any dose band, to avoid placing excessive reliance on particular features of the plant or on particular assumptions in the analysis.

The measure chosen to represent the severity of the accident is the maximum effective dose which would be received by a person at the nearest habitation which is typically 1 km downwind from the plant. This is considered to be a generally adequate surrogate for the whole range of offsite effects which an accident can lead to including the individual risk of death (prompt and delayed) and of other health effects to local people, contamination of land, disruption of people's lives from the application of countermeasures such as evacuation, fear and alarm in the general public, economic loss etc.

When account is taken of the variability of wind and weather conditions, a plant which just met the BSLs/BSOs would give a maximum individual risk of death to a person outside the site of about 10⁻⁵ per year/10⁻⁷ per year respectively which meet the criteria given in TOR.

P43 Risk to workers:

The total predicted individual risk of death (early or delayed) to any worker on the plant attributable to doses of radiation from accidents should be less than:

BSL	BSO
10^{-4} per year	10^{-6} per year

Note 1. It is recognised that the calculation of individual risk to workers may be difficult and hence only a broad estimate will normally be required, sufficient to show that the BSL is very unlikely to be exceeded and that ALARP has been appropriately applied.

Note 2. This principle is not intended to apply to personnel returning to perform recovery actions after an accident.

The risk of death to workers on the plant from accidents does not involve consideration of offsite effects, and so the individual risk is used directly. To maintain consistency with ICRP, the major part of the tolerable level of risk is allocated to normal operation, and hence the BSL for accidents in P43 is set out at 10^{-4} per year. The BSO value is chosen as 10^{-6} per year as being reasonably consistent with the broadly acceptable level of 10^{-6} per year in TOR, bearing in mind that, while the latter includes normal operation, it is directed principally at members of the public.

P44 Large release:

The total predicted frequency of accidents on the plant with the potential to give a release to the environment of more than:

- 10 000 TBq of Iodine 131**
- or** **200 TBq of Caesium 137**
- or** **quantities of any other isotope or mixture of isotopes which would lead to similar consequences to either of these**

should be less than:

BSL	BSO
10^{-5} per year	10^{-7} per year

For a major accident, the dose to a person close to the plant may be into the range which would cause prompt death, so the particular level of dose is no longer an appropriate measure of its severity. In this situation the number of people affected and the land contamination become dominant concerns. A more appropriate surrogate for these effects, but one which is still related to the design and operation of the plant, is the quantity of radioactive material released in the accident.

The quantities of radioactive material chosen are consistent with the definition of a major accident given in TOR and in line with international thinking on large releases. The BSL frequency is consistent with the value proposed in the Barnes report for Hinkley Point C.

P45 - Plant damage:

The total predicted frequency with which the plant suffers damage and a significant quantity of radioactive material is permitted to escape from its designed point of residence or confinement, in circumstances which pose a threat to the integrity of the next physical barrier to its release, should be less than:

BSL	BSO
10^{-4} per year	10^{-5} per year

Note. Such plant damage is interpreted as a degraded core in the case of a reactor. For other plant, it would include a major breach of vessel pipework etc, together with the potential for events such as fire, explosion, or aggressive chemical attack which might lead to degradation of the containing cell or its ventilation/ filtration system even though there may be a safety system provided to prevent such degradation.

This principle is included to reinforce the objective of defence-in-depth which looks for a series of physical barriers to a release of radioactive material. The safety of the plant should not rely predominantly on the integrity of the final barrier to the release: there should be sufficient reliability in each of the barriers to make a challenge to the final barrier very unlikely.

The BSL frequency is set at 10^{-4} per year on the basis of a judgement that a higher frequency would be intolerable in terms of the alarm, concern and loss of confidence that would be caused by such an accident, even without a release, and because it would indicate an intolerable weakness in the design of the plant or laxity in the control of its operation. The BSO frequency of 10^{-5} per year is that given in the same report as the goal for future plants.

P46 - Criticality incidents:

The total predicted frequency of an accidental criticality excursion on a plant other than a nuclear reactor should be less than:

BSL	BSO
10^{-3} per year	10^{-4} per year

This principle also applies to plants handling or storing fissile material outside the reactor core on a nuclear power station.

This principle is included by analogy with P45 to address defence-in-depth for the protection of workers against radiation from accidental criticality incidents, which are an important concern on some non-reactor plants. Such an incident would represent a loss of control and might impose a challenge to the shielding and to the emergency arrangements for personnel protection. The potential consequences, however, are more limited than those of plant damage and so the BSL and BSO frequencies are set a factor of ten higher.

3.17 United States

As a result of the recommendations from the President's Commission on the Accident at Three Mile Island, the NRC issued the *Safety Goals for the Operations of Nuclear Power Plants; Policy Statement* in 1986. This policy statement expressed safety policy using both qualitative and quantitative methods. The

policy statement was not a regulation, but influenced various regulatory actions, primarily the development of the Regulatory Analysis Guidelines used to backfit analyses and the guidance developed for risk-informing reactor regulatory activities. The reactor Safety Goals broadly define an acceptable level of radiological risk and apply to reactor accidents and do not address environmental considerations, worker protection, routine operation, sabotage, non-reactor activities, or safeguards matters.

There is a level of safety that is referred to as “adequate protection.” This is the level that must be assured without regard to cost and, thus, without invoking the procedures required by the NRC’s Backfit Rule. Beyond adequate protection, if the NRC decides to consider enhancements to safety, costs must be considered, and the cost-benefit analysis required by the Backfit Rule must be performed. The Safety Goals, on the other hand, are silent on the issue of cost but do provide a definition of “how safe is safe enough” that should be seen as guidance on how far to go when proposing safety enhancements, including those to be considered under the Backfit Rule.

The Commission has established two qualitative safety goals, which are supported by two quantitative objectives. These two supporting objectives are based on the principle that nuclear risks should not be a significant addition to other societal risks.

The qualitative safety goals are as follows:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.
- Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The following quantitative objectives are to be used in determining achievement of the above safety goals:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

The Commission believes that this ratio of 0.1 percent appropriately reflects both of the qualitative goals—to provide that individuals and society bear no significant additional risk. However, this does not necessarily mean that an additional risk that exceeds 0.1 percent would by itself constitute a significant additional risk. The 0.1 percent ratio to other risks is low enough to support an expectation that people living and working near nuclear power plants would have no special concern due to the plant’s proximity.

CHAPTER 4 – STATUS OF PSA PROGRAMMES

This chapter provides a summary of the current status of PSA Programmes in Member countries. An appendix is provided to this chapter (Appendix A) which provides a tabular form of the status.

4.1 Belgium

In the level 1 part, power and non-power states are analysed. Internal hazards (fire and flooding) and external hazards are not covered.

All level 2 analyses performed for the Belgian NPPs are limited to the analysis of the containment response, with the aim to investigate dominant containment failure modes. No source term analyses have been performed.

For the Doel 3 and Tihange 2 PSAs, the analysis was limited to a binning into Plant Damage States (PDS) and a deterministic analysis of the containment behaviour for some dominant core damage sequences with the STCP code. In this way, the phenomena (hydrogen burning, basemat melt-through, etc.) threatening the containment integrity could be identified for each of these typical accident scenarios, however without obtaining information on the probabilities of the failure modes.

For the Doel 1 and 2 and Tihange 1 PSAs, a probabilistic analysis using Containment Event Trees (CETs) is performed, using MELCOR for the analysis of the severe accident progression. The review of these level 2 analyses is still in progress.

Appendix A gives an overview of the time schedule of the different PSA projects.

4.2 Canada

The status of PSA for Canadian plants is as follows:

- Pt Lepreau NPP in New Brunswick: Risk Baseline PSA was produced to identify design improvements for plant life extension. These design improvements will be later analysed for benefit cost assessment.
- Gentilly-2 NPP in Quebec: PSA work to identify design improvements for plant life extension has just started.
- AECL produced a Generic PSA to which elucidates the PSA methodology and tools for external events and level 2 PSA for a reference CANDU 6 and CANDU 9 plants. Applications of these methods and tools have been performed for selected scenarios and systems.

- Bruce B Risk Assessment is complete. Presently Bruce Power use the EPRI EOOS system for maintenance planning in managing risk. The BBRA is also used for other applications such as environmental qualification component lists, cost benefit assessments, and system unavailability reporting.
- Bruce A Risk Assessment is half complete. Bruce Power intends to complete the Bruce A Risk Assessment.
- Pickering A Risk Assessment is complete. It is being used as a tool to identify design improvements in the Pickering A Return to Service effort.
- Darlington Risk Assessment is complete and is issued for internal Ontario Power Generation (OPG) review.
- Pickering B Risk Assessment is scheduled for completion in 2002. Both Darlington and Pickering B PSAs are Level 3.

4.3 Czech Republic

Three main project tasks have been established within ongoing Living PSA Programme in Dukovany NPP:

- Risk management (executive task)
- Data Collection and Information Exchange (data support task)
- Maintenance and continuous improvement PSA models (maintenance task)

Specific Living PSA QA guidelines have been developed to assure consistency of all current activities and to establish a control system for information transfer, which is necessary for Dukovany PSA models use, maintenance and improvement. For all regular activities specific schedules have been defined.

The current Dukovany Living PSA model, which includes analysis of internal initiators as far as internal fires and floods, reflects all power plant modifications up to December 2000. The PSA model is valid for NPP unit within a full power operation.

Analysis of Shutdown and Low Power unit states was completed in 1999. Study reflects NPP design and operational practices of year 1998 and includes analysis of internal initiators, fires & floods, analysis of other sources of radioactive releases (spent fuel storage tank) as far as heavy load drops initiators. Results showed for WWER 440/213 type unit comparable risk level of shutdown and low power states with full power operation. The analysis is being updated within year 2001 to take into account all unit modifications since 1998.

In Temelin NPP a limited, PSA Level 1 for internal and external (fire, floods and seismic) initiating events and for all unit states was completed (there was not available all information - NPP was under construction). A Level 2 analysis was a part of study.

As the plant was under construction and a lot of safety improvements were being implemented into design, the information used in PSA were based on a number of conservative assumptions concerning the ultimate design and future operation in some areas. Some information which was used was based on the previous plant design status and some information related to I&C, cable installation, and control layout

were not available at the time of the analysis. All the unresolved issues were carefully documented and integrated into PSA information database so that they could be confirmed, corrected or amplified when the final plant information becomes available.

To overcome above mentioned deficiencies and conservatism contained in the PSA study an update of all NPP Temelin models and documentation is being performed. Living Level 2 PSA study is supposed to be completed in 2002 to ensure that information and assumptions in PSA reflect the design and construction when the plant is commissioning and to provide risk models that are suitable for PSA applications and use in licensee-related work.

4.4 Finland

4.4.1 Summary of Olkiluoto PSA programme

Since 1994 TVO has submitted to STUK the updated risk analyses on harsh weather conditions, internal flooding, fire, shut down mode and internal initiators where the aforementioned plant changes are embedded in. Since 1995 some major plant changes have been made in order to compensate the power upgrade (TVO is applying for Operating Licence for upgraded power of 115%):

- two diverse safety relief valves have been installed to upgrade the reactor overpressure capacity and to enhance the system reliability as well.
- the hydraulic protection and control system of turbine has been replaced by a system based on digital and analogue electronics.
- Inertia of main coolant pumps has been increased and the control system has been replaced by digital and analogue electronics.

In the newest version TVO has refined the PSA model and reduced some overly conservative assumptions used in PSA. The latest analysis results (2000) are as follows:

- internal initiators, $5.6 \times 10^{-6}/a$
- fires, $5 \times 10^{-7}/a$
- internal flooding, $1.4 \times 10^{-6}/a$
- harsh weather conditions, $1.8 \times 10^{-6}/a$
- seismic events, $4 \times 10^{-6}/a$
- shut down mode (internal events only), $3.9 \times 10^{-7}/a$

TVO submitted to STUK also the level 2PSA which showed that the average probability of large release (atmospheric release of Cesium-137 is more than 100 Tbq) is about $4 \times 10^{-6}/a$. The majority of the risk comes from early high pressure transients and the remainder mainly from low pressure transients and the shut down mode initiators.

4.4.2 Summary of Loviisa PSA programme

Since 1994 Fortum (former IVO) has submitted to STUK the updated risk analyses on harsh weather conditions, internal flooding, fire, shut down mode and internal initiators where the aforementioned plant changes are embedded in. The latest analysis results (2000) are as follows:

- internal initiators, $1.6 \times 10^{-5}/a$
- fires, $4 \times 10^{-5}/a$
- internal flooding, $1 \times 10^{-5}/a$
- harsh weather conditions, $4.3 \times 10^{-5}/a$
- seismic events, $3.6 \times 10^{-6}/a$
- shut down mode (internal events only), $2.8 \times 10^{-5}/a$

STUK has reviewed the risk analyses above for the renewal of the Operating Licence at upgraded power of IVO plant units.

Fortum submitted also the level 2 PSA to STUK which showed that the probability of large release (atmospheric release of Cesium-137 is more than 100 Tbq) is about $5 \times 10^{-6}/a$). The majority of the risk comes from leaks between primary and secondary circuits and other by-pass sequences of the containment.

4.5 France

4.5.1 Introduction

French PSA activities are carried out in mainly two organisations: IRSN and EdF. These activities are the development of PSA models and methods, as well as PSA applications for various safety analysis problems. Moreover, for the French-German project of a future plant (the European Pressurised water Reactor-EPR), a PSA was performed by the designers since the beginning of the design, and analysed by the Safety Authorities.

In the past years, PSA studies have been performed out of the regulatory framework. PSA was not required by the Safety Authority and was carried out as an aid for safety analysis, the safety demonstration relying on deterministic principles. PSA is now recognised as an important tool for safety analysis, and it appears necessary for EdF and for the Safety Authority to define a more precise framework for PSA developments and applications. The preparation of a Basic Safety Rule is in progress.

4.5.2 PSA activities in IRSN

4.5.2.1 Developments

Three projects are on-going concerning PSA development: updating of the level 1 PSA, and extension of the scope by a fire PSA and a level 2 PSA. These three projects concern the 900 MWe PWR standardised series.

4.5.2.1.1 *PSA level 1 updating*

An updating of the level 1 PSA carried out for the 900 MWe standardised PWRs is nearly finished. The updating takes into account plant modifications, new data and knowledge (for example a revised I & C model).

The recent studies are related to the updated analysis of :

- The accident sequences related to the loss of the electrical supplies
- The accident sequences related to containment bypass (SGTR, Interfacing System LOCA)

Moreover a general harmonisation of the assumptions for the whole study is in progress.

4.5.2.1.2 *Fire PSA*

The important recent steps of the fire PSA are the following:

- Methodology development for fire scenarios determination (fire event tree construction): the methodology has been applied to all the critical zones.
- Functional analysis of fire scenarios consequences related to fixed combustible during plant power states.
- Evaluation of damage time for safety equipment by use of FLAMME-S computer code, for the state reactor in power.
- Methodology development for estimation of core damage frequency linked to each fire scenario including the probabilistic quantification of human error. This methodology has been applied to critical compartments for the state reactor in power.
- Methodology development for the study of fire scenarios during Low Power Shutdown state.
- Evaluation of damage time for safety equipment by use of FLAMME-S for confined compartments and with simplified formula inside containment, for the Low Power Shutdown state is in progress.
- Control Room fires analysis.
- Development of computerised data banks for fire initiators and for equipment location.

Moreover, in order to reduce uncertainty associated with fire modelling :

- Fire tests have been carried out about electrical damage. These tests consisted of studying the behaviour of CCI and power cables under thermal load
- Fire tests concerning combustion of electrical cabinets have begun and preparation of tests of fire propagation from a compartment to the adjacent ones is continuing.

- A model related to the propagation of thermal effects from one compartment to the adjacent one has been added in FLAMME-S computer code. This model has been qualified by fire test results already performed.

4.5.2.1.3 *PSA Level 2*

A Level 2 Probabilistic Safety Assessment is underway for the French 900 MWe PWR.

The expected impact of the results should be:

- to better assess the safety level of existing 900 MWe PWRs and to identify the potentially most vulnerable points in the design or operation;
- to estimate the contribution of emergency procedures and guides to the safety level and the radioactive release mitigation;
- to evaluate the interest of improvements in the design or the operation mode;
- to prioritise necessary R&D actions to reduce the uncertainties on phenomena with a potentially high impact on safety and better evaluate the effect of proposed upgradings.

Three principal tasks have been identified for PSA implementation:

- a) The regrouping, into categories of Plant Degradation States (PDS), of the sequences resulting in a core-melt, as they were identified in the PSA level 1: this task delineates the interface between levels 1 and 2 of PSA;
- b) The design of the event-tree of the further severe accident progressions, starting from each of the initial states previously defined from level 1-PSA results; this event tree, called Accident Progression Event Tree (APET) represents all the events having an influence on the course of the accidents : system operation, human actions and physical phenomena ; when quantifying the APET, uncertainties on physical phenomena are taken into account ;
- c) The regrouping of severe accident sequences into categories, called Accident Progression Families (APF) in the containment, with relevant probability assessment; such a regrouping of sequences will be arranged as a function of the mode of containment failure and the fission product behaviour; the assessment of the radioactive releases is performed for each APF.

In order to quantify the APET, a large number of supporting studies has been performed within the framework of the Level 2 PSA. They are dealing with:

- The operation of safeguards systems : an exhaustive study of ultimate means for recovering water injection into primary and secondary circuits has been performed ; a study of the survivability of equipment under severe accident conditions, including small scale experiments, is under way ; an exhaustive assessment of containment leakage paths has been also conducted ;

- The assessment of human reliability : a dedicated method has been developed in order to quantify the human actions during severe accidents, taking into account the whole emergency organisation and the emergency guides ;
- The physical phenomena: these studies are related to the core uncovering and degradation, the hydrogen production and combustion, the temperature induced breaks of the primary circuit under high pressure situations, the corium flowing down into the vessel bottom head, the steam explosion, the vessel rupture and the containment direct heating, the corium-concrete interaction, the thermalhydraulics in the containment and the fission products behaviour in the primary circuit and in the containment.

In order to quantify the physical phenomena, French codes such as ESCADRE-ASTEC, CATHARE, CASTEM and MC3D have been used. When some physical phenomena have been judged inadequately modelled in available codes, some specific simplified parametric models have been developed within the framework of the project, especially in the field of advanced core degradation. Corresponding uncertainties have been assessed. Recent experimental results, in particular as regards fission product behaviour (PHEBUS PF program), have been also taken into account.

In the framework of the development of the level-2 PSA project, an intermediate step has been completed, based on the 1990 version of IRSN level-1 PSA for French 900 MW PWR units: only the major accident families at full power have been treated in that step, taking into account the major uncertainties of physical phenomena related to early containment rupture modes. The results of this preliminary step are presently reviewed internally.

The final version of the level-2 PSA will consider all initial plant states and be based on an updated version of the level-1 PSA, currently under development, on the results of the PSA related to fire events, and will take into account recovery actions using available site resources, as indicated by the emergency teams, the results of the program under way on equipment survivability and improved assessment on physical phenomena.

4.5.2.2 PSA applications for safety analysis

Developed in the chapter 5.

4.5.2.3 PSA comparisons

A comparison has been conducted between the French 900 MWe series PWR PSA and the Belgian Tihange1 PWR PSA which have a comparable design.

The main technical objective of the comparison was to understand the important differences between the PSA results, and especially to identify if the differences are due to design differences or to differences in PSA methods and data.

Several important insights were drawn from this exercise concerning plant safety, as well as PSA methodology and model improvements.

Similar exercises are undertaken with Korea and South Africa, which are also operating PWRs with a Framatome design.

4.5.3 *PSA activities at EdF*

4.5.3.1 Model developments and main applications for design

4.5.3.1.1 900 MWe PSA

The level 2 PSA has been completed and presented for internal EdF review. Some improvements have been requested, in order to take into account hydrogen recombiners which have been recently decided, and ultimate actions and strategies which would be suggested by the technical crisis organisation, in addition to accident procedures and Severe Accident Management Guidelines, both before and after the beginning of core uncovering.

4.5.3.1.2 1300 MWe PSA

After the 1300 MWe level 1 PSA transfer on RISK-SPECTRUM in 1999, the updating work is done, as well as the identification of Plant Damage States for sequences in power and hot shutdown initial conditions.

The resulting PSA model is now used for the purpose of the Periodic Safety Review of this plant series, according to a new methodology. In that process, the risk is no longer distributed into accident families, but into functional sequences, which are characterised by the ultimate measure (equipment or operator action) preventing the core degradation. Severe Accidents are now taken into account, by extrapolation of the results of 900 MWe level 2 PSA.

A methodology for the use of PSA in the analysis of multiple failures situations has also been developed and presented in April 2000 to the French Safety Authority. An application of this methodology, based on the new 1300 MWe PSA, has led to a new list of Beyond Design Basis Accidents which are under study in preparation of the Safety Review. The results of these transient studies will be integrated in the reference PSA model by the end of year 2001.

4.5.3.1.3 1450 MWe PSA

The level 1+ (including PDS definition) 1450 MWe PSA Project is in its ending phase, using the last PSA methodologies available at EdF for sequence analysis, human factor and I&C dependability evaluation. This Project should be completed by the end of 2000, and issued in 2001.

4.5.3.1.4 EPR project

A level 1+ PSA (including definition of Plant Damage States) has been developed in parallel to the Basic Design studies of the European Pressurised Reactor. The PSA versions already issued have contributed to design choices of some safety related systems during and after the Optimisation Phase of the Basic Design. The last one allows the assessment of the effect of maintenance during power operation. Further development of the model will take into account I&C unavailability and a wider range of initiating events. Events occurring in shutdown states were integrated from the beginning of the work. A new PSA improvement program has been formed for the detailed design phase, including a level 2 development, in order to demonstrate the ability of EPR design to face severe accidents.

4.5.3.2 PSA applications

See chapter 5.

4.5.3.3 Methodology evolutions

See chapter 6.

4.5.4 PSA in the regulatory process

In the past years, PSA studies have been performed out of the regulatory framework. PSA was not required by the Safety Authority and was carried out as an aid for safety analysis, the safety demonstration relying on deterministic principles; For that reason the relations between IRSN and the utility Electricité De France (EdF) were a co-operation and a technical dialogue rather than a classical safety analysis process.

After the completion of the two PSA studies in 1990 (one for a standardised 900MWe PWR and one for a 1300MWe PWR, respectively by IRSN and by EdF), several applications have been performed, leading to plant modifications and backfits. PSA is now recognised as an important tool for safety analysis, and it appears necessary for EdF and for the Safety Authority to define a more precise framework for PSA developments and applications.

From her part the DSIN (Direction de Sûreté des Installations Nucléaires - Safety Authority) has decided to issue a Basic Safety Rule related to the development and use of PSA in France. Presently IRSN is working on the writing of this Safety Rule and the preliminary chapters are discussed with EdF.

4.6 Germany

For all German NPPs level 1 PSAs are available. For nearly all plants these PSAs do not consider accidents during low-power and shutdown operation, in some PSAs plant internal accident management measures have not been taken into account.

GRS has performed for Gundremmingen, units B and C ("BWR-72"), and for Neckarwestheim, unit 2 ("Konvoi"-PWR), level 2 PSAs for accidents during normal power operation, including level 1 PSAs for low-power & shutdown states. GRS has started a PSA of the same scope for Philippsburg, unit 1 ("BWR-69").

On behalf of the Federal Regulatory Authority GRS routinely performs accident sequence precursor (ASP) analyses for selected reportable events in German NPPs.

4.7 Hungary

Concerning plant operation at *full power*, four stand-alone, unit specific PSA models and the corresponding results are now available for NPP Paks for internal initiating events, e.g. loss of coolant accidents, transients initiated in the secondary circuit, in the electric power supply system, in the instrumentation and control systems, etc. PSA for internal hazards, including, fire and flooding, has been performed for Unit 1. A similar analysis is currently underway for Unit 2 that is due to be finished in 2001. The need for unit specific PSAs of internal hazards has emerged mostly due to the differences in cable routing. Also, the fire and flood PSA is planned to be extended to Units 3 and 4 in 2002. A PSA project for seismic events has already been launched and it will be completed in 2001.

Concerning *low power and shutdown states* of NPP Paks, PSA is available for a regular outage for refuelling covering internal initiating events. The current intention is to extend this analysis to include the contribution of internal fires and flooding and, potentially, seismic events. In addition, there are outages other than a refuelling outage (e.g. shutdown due to plant disturbances or due to requirements of Technical

Specifications). A PSA for low power and shutdown states arising from such unplanned outages is also to be performed.

The table below gives a summary overview of the status of level 1 PSA for NPP Paks, including available analyses, ongoing actions and short-term plans.

Plant Operating Mode	Available Level 1 PSA	Ongoing Analysis	Near-Term Objectives
Full power operation	Detailed unit specific PSAs for internal events Fire and flood (internal hazard) PSA for Unit 1	Internal hazard PSA for Unit 2 Seismic PSA using Unit 3 as reference unit	Internal hazard PSA for Units 3 and 4
Low power and shutdown states	PSA for internal events using Unit 2 as reference unit	None	Extension to internal hazards and to seismic events Analysis of non-refuelling shutdown

All the available logic models, databases, results and documentation for the Paks PSA are regularly updated using a living PSA procedure. Safety related plant modifications and changes in the reliability characteristics of plant equipment and/or plant personnel are modelled and quantified. The updating is performed in co-operation between plant personnel and PSA analysts of VEIKI. Operation of this living PSA helps to follow changes in the safety level of the plant, and it also ensures that risk based decisions can be supported by up-to-date risk models and data. Living PSA enables a range of PSA applications and it also ensures usefulness and credibility of results gained from the applications. Both the utility and the regulatory body possess the very same living PSA models. The latest PSA update was made in 2000 following the regular refuelling outages for the four units of NPP Paks.

In addition to level 1 PSA, a level 2 analysis was started for NPP Paks in late 2000 using Unit 1 as reference unit. This ongoing level 2 PSA covers all the plant operational modes and initiating events that have been treated by the level 1 PSA studies. The objectives of the level 2 PSA are to (1) quantify the risk of radioactivity large release from potential severe accidents, (2) identify plant vulnerabilities to large releases, and (3) provide a technical base for developing accident management strategies, tools and procedures.

4.8 Italy

As mentioned in chapter 1 PSAs aren't currently being carried out on a plant: at present the ongoing PSA related Programmes in Italy are concerning, above all, the development of the probabilistic approach to be adopted in the frame of the safety analysis regarding international Fusion projects, specifically fusion reactor ITER (International Thermonuclear Experimental Reactor) and experimental facility IFMIF (International Fusion Materials Irradiation Facility).

With reference to the international activities, Italian experts took part to several consultant meetings convened by the IAEA in order to prepare the current guidance on conducting a PSA for a non-reactor nuclear facility (NRNF) (ref.5).

Another PSA related activity is the participation to COOPRA (International Co-operative Research Program in Probabilistic Risk Analysis – promoted by US NRC) (ref.6).

4.9 Japan

Utilities and NUPEC have made several PSAs in accordance with PSA standards for full power operation and shutdown operation.

4.9.1 *Development of PSA Standards*

1. PSA Standards for level 1 and level 2 PSAs at Full Power Operation

It is recommended that PSA on individual NPPs should be performed in accordance with the guidebooks issued by Nuclear Safety Research Association (NSRA) in 1992 for level 1 PSA and in 1993 for level 2 PSA, which have been prepared by the voluntary committee (chairman: Professor Kondo of University of Tokyo) consisting of representative PSA-specialists from governmental organisations and industry groups, and the fundamental concept and methodologies of the standards are the same as NUREG/CR-2300.

2. PSA Standard for level 1 PSA during low power and shutdown operation

A subcommittee on PSA in the Power Reactor Technical Committee initiated to prepare the draft of Procedures Guide for the low power and shutdown PSA. This will be utilised in the low power and shutdown PSA in PSR. The Power Reactor Technical Committee is one of three technical committees set up in the Standard Committee of the Atomic Energy Society of Japan (AESJ), organised in September 1999 in order to issue the standards for nuclear technology from the viewpoint of effectively securing the safety and reliability of nuclear facilities based on the state of the art technology. The Standard Committee makes and revises the standards, guides and guidance on design, construction, operation and decommissioning of nuclear facilities. The members of committees are elected widely among industrial and academic circles to secure the neutrality, impartiality, accountability and transparency. The proceedings of the committees are made public.

4.9.2 *PSAs on typical NPPs*

Both governmental organisations and industry groups are carrying out various kinds of PSAs in objective, scope and method on several typical NPPs selected in the light of plant features, such as reactor type, containment type, power, configuration of emergency core cooling system (ECCS) and so forth, in order to investigate relative vulnerability, grasp risk profile and confirm safety margin of them. They have been making efforts through the execution of such PSAs to develop and upgrade their PSA methodologies to date.

4.9.3 *Accident Management Strategies based on PSA*

- NSC issued the severe accident management policy statement in May 1992 as follows; Though the frequencies of core damage and containment failure due to severe accidents at Japanese typical NPPs are evaluated to be sufficiently small from an engineering perspective, NSC decided to introduce accident management based on PSA in order to further reduce plant risks, which does not directly lead to the licensing conditions for constructing or operating NPPs.
- Based on NSC's decision, the competent regulatory authority MITI, prepared own policy on implementing accident management to cope with severe accidents, and in July 1992 strongly

recommended and encouraged the owners of NPPs to take the appropriate measures to perform PSA and establish PSA-based accident management.

- The utilities together with vendors conducted 43 level 1 and level 2- PSAs on each of all Japanese operating NPPs (Individual Plant Examination: IPE) (51 NPPs including several NPPs under construction). Since Japanese NPPs have been progressed in improvement and standardisation and can be classified into several groups from the viewpoint of plant design and operation, their own accident management strategies have been fundamentally established for respective groups. Results of 43 PSAs were submitted to MITI at the end of March 1994.
- The virtual review on the results of IPEs has been executed by MITI and the Technical Advisory Committee in support of Nuclear Power Engineering Corporation (NUPEC) after the formal submission by the utilities at the end of March 1994. MITI and the Advisory Committee have approved the fundamental adequacy of the methodologies, database and results of IPEs from viewpoints of state-of-the-art of PSA methodology and the recent objective of comprehensive and quantitative understanding for safety characteristics of individual NPPs in order to develop accident management program. The review report written by MITI was presented to NSC in October 1994. NSC reviewed and admitted it to be approved in November 1995.
- MITI has studied the basic requirements in implementing AMs, taking expert opinions of Technical Advisors for Nuclear Power Generation into consideration, and has extracted the “basic requirements for implementing AMs” related to the following from a standpoint of securing the effectiveness of the AM as counter-measures to SA.
 - a) Implementation system for AM
 - b) Facilities and equipment, etc. related to implementation of AM
 - c) Knowledge base related to implementation of AM (procedures of actions which are deemed to be effective and appropriate to be studied beforehand)
 - d) Notice and communication related to implementation of AM
 - e) Education and training of personnel engaging in implementation of AM
- Utilities are now implementing AM (preparing the equipment for AM and preparing procedures related etc.) for operating and constructing NPPs. In near future an Implementation Report for each NPP-site will be submitted to the Ministry of Economy, Trade and Industry (METI)³. The effectiveness of the AM on CDF and containment failure frequency will be evaluated through level 1 and level 2- PSAs mainly for eight typical NPPs, namely BWR3 with Mark-I containment vessel (CV), BWR4 with Mark-I CV, BWR5 with Mark-II CV, ABWR with ABWR CV, 2 loop-PWR with dry-type CV, 3 loop PWR with dry-type CV, 4 loop-PWR with dry-type CV and 4 loop-PWR with ice-condenser type CV.
- METI will review these AM Implementation Reports including the effectiveness of the AM measures on CDF and containment failure frequency (CFF) under the support of NUPEC. NUPEC also implement level 1 and level 2- PSAs for the above eight NPPs. The review

3. Former MITI, which was reorganized to METI in January 2001.

report written by METI will be presented to NSC. Plant specific AM measures will be reviewed in the domain of PSR.

4.9.4 PSA in PSR

In Japan PSR is introduced as so-called voluntary measures for safety activities done by utilities under close deliberation with MITI, which requested utilities PSR in June 1992, in order to assess periodically (about every 10 years) and comprehensively the current situation of safety and reliability of each existing NPPs in the light of up-to-date technical knowledge.

Until now PSRs have been conducted seven times for 29 plants. In the first two PSRs, PSA was not included. In the third PSR PSAs conducted in 1994 to examine candidates for accident management were quoted without update. From the fourth PSR, PSAs were updated to take into account accident management measures prepared for its realisation. Especially plant-specific AMs different from the standard AMs are taken into account in PSA. From the seventh PSR, PSAs for shutdown operation states are included to secure safety during low power and shutdown operation.

4.9.5 Evaluation of Allowed Outage Time (AOT) based on PSA

In Japan the technical specifications in NPPs have been required to be made detailed with accountability and transparency especially since JCO accident. Japanese utilities now have revised the technical specifications as detailed as those in Standard Technical Specification of USA. In the process of the revision the applicability of level 1 and 2 PSAs has been pursued in both utilities and NUPEC in order to have the accountability and the transparency of setting up AOTs for the safety systems with redundancy. NUPEC, under the sponsor of METI, has estimated incremental conditional core damage probabilities (ICCDP) and incremental conditional large early release probabilities (ICLERP) during AOT for Japanese BWR and PWR, using level 1 and 2- PSAs. The effects on ICCDP of surveillance tests, conducted for the remaining system during AOT, are taken into account. Allowed ICCDP should be essential to setting up AOT using PSA. The allowed ICCDP was provisionally set up taking into account ICCDP under the current technical specification, ICCDP for outage experiences, ICCDP during manual trip and the conceivable safety goal.

4.10 Korea

4.10.1 PSA of operating nuclear power plants

4.10.1.1 Kori Units 3&4 Level 1 PSA

The PSA plays a unique function in linking the design, operation, maintenance, testing, and safety of the plant. Kori 3&4 PSA, which is the first one performed on an operating NPP, was started in September 1989 and ended in October 1992. The main objective of this PSA is to evaluate the overall safety of Kori Units 3 &4, and to identify practically applicable safety improvement items. The other diverse objectives are summarised as follows:

- Identify plant-specific vulnerabilities to core damage sequences and gain a perspective on severe accidents;
- Determine potential weak points in design, operation, test and maintenance;

- Provide a prioritised modification planning of hardware and/or procedures to reduce the overall frequency of core damage;
- Provide information and guidance to the operators on how to cope with severe accidents;
- Establish Level 1 PSA methodology.

The scope of the Kori Units 3&4 PSA is a detailed level 1 PSA, including selected external events such as seismic events, internal fires, internal flooding, and extreme wind (typhoons). Because the PSA methods were not well defined at the beginning of the study, it was determined for this study to follow the approach and method described in NUREG/CR-2300. However, this study also incorporated new methods and updated reference materials collected during project period.

The most significant contributor to total core damage frequency (CDF) in terms of initiating events is station blackout (26.1% of total CDF), followed by fire (23.0%), earthquake (17.5%), extreme wind (9.7%), flooding (6.0%) and small LOCA (6.0%). The plant design features contributing to the CDF from internally initiated events are similar to those of other Westinghouse designs. As the plant does not have an alternate electric power supply other than the normal emergency diesel generators (EDGs), the performance of the turbine driven auxiliary feedwater (AFW) pump and the battery capacity are governing the CDF.

In case of fire events, main control room fire and switchgear room fire dominate the core damage frequency, as similarly reported in other PSA results. However, the plant design features associated with externally initiated events are a little different from those of other plants, for example:

- In case of seismic events, it is shown that the seismic hazard for Kori site largely depend on expert opinions due to lack of seismographic data, and thus the resulting hazard curve has a relatively high frequency in the high acceleration range.
- In case of extreme wind induced event, the CDF is relatively high due to the low fragility of transmission towers.
- In case of flooding events, the CDF is relatively high since the flooding from the essential service water system is not automatically isolated; instead the essential service water standby pump is automatically started and increases the flow rate of flooding.

The recommendation was provided to reduce the plant risk from deep review of above results and extensive importance/sensitivity analyses. The risk characteristics of Kori Units 3&4 was assessed not only by performing plant specific accident analysis, system interaction review, and assessment of operator recovery factor, but also by comprehensively collecting plant-specific experience information and site-specific characteristics (e.g., typhoon). As a result of this study, various plant modification alternatives were identified.

4.10.2 PSA applications on the design of nuclear power plants

4.10.2.1 Yonggwang Units (YGN) 3&4 Level 2 PSA

The objective of YGN 3&4 PSA, which is the first study for Korean Standard Nuclear Plant (KSNP), is to evaluate quantitatively the containment integrity against severe accidents using the probabilistic methodology. Via utility's joining in this evaluation, not only utility staff's site experience but also their understanding of the severe accidents can improve the plant safety. Another objectives of the study are to

identify the plant vulnerabilities and to suggest ideas, which supplement the vulnerabilities by decreasing the CDF or probability/quantity of radioactive material release. The work scope includes finding some enhancing options of the facility or operator-responding procedures, which will prevent or mitigate the severe accidents, and the output will be used to set up the SAMP.

The containment failure probability from the internal/external events out of total CDF is about 0.25, about 17% from the internal events and about 83% from the external events. The seismic event contributes about 64% to the total value caused by the external events. It is identified, from the analysis of containment failure mode distribution, that the early containment failure in which mode fission product release is most risky, contributed least (1.1%) because the containment is designed robustly relative to the other foreign containment. Regarding the containment failure from the internal events, the containment bypass due to the steam generator tube rupture (SGTR) accident has the probability of 10%, which is the second contributing failure mode. In the case of external events, the frequency due to the fire event is 1.3 times higher than that of seismic event. However, the seismic event has a great influence on the containment failure and source term release because this event contributes more than 55% to the late containment failure.

The overall methods and analytical tools applied to the various areas in the analysis were found to be technically sound and consistent with other similar PSA studies. From the analysis results and sensitivity studies, following essential risk management items to improve the containment integrity are recommended:

- In the case of station blackout accidents in which safety injection system (SIS) and containment spray system (CSS) is not available, the ex-vessel cooling should be considered using the cavity water injected from the non-safety grade system (e.g., emergency fire protection system) to prevent the reactor vessel failure. This will lower the failure probability of the reactor vessel.
- The recirculation mode of CSS is proved to be important to prevent or mitigate the fission product release outside the containment. So, the reliability of this function should be enhanced via setting up an inspection/maintenance plan and the procedures to exclude common cause failure.
- The improved instrumentation should be established to trace up the accident progression. Temperature, pressure, hydrogen concentration, and radiation level should be indicated by this instrumentation under harsh environment.

4.10.2.2. Wolsong Units 2,3&4 Level 2 PSA

Wolsong Units 2,3,&4 PSA project was finished in 1997, and submitted to KINS for the regulatory review. External event analysis has revealed that estimated core damage frequency is high compared to that of internal events. It has also revealed that Wolsong Units 2,3,&4 responds to severe accident initiators uniquely by using abundant water inventories in the various process systems, as well as using the containment dousing.

Several potential vulnerabilities were found during the assessment, which could disable one or more of the essential systems during a severe accident. Many plant design and procedure modifications were done with the support of a feasibility study. The significant finding was that the high confidence of low probability of failure (HCLPF) ground motion acceleration of the emergency water supply building is very low, relative to other buildings. Sensitivity analysis confirmed that a reinforcement of this building would greatly reduce the total CDF. Consequently, the reinforcement of the EWS building was done, including a backfit for Wolsong Unit 1.

4.10.2.3 Ulchin Units (UCN) 3&4 Level 2 PSA

UCN 3&4 PSA, with similar work scope of YGN 3&4 PSA, has generally been conducted by taking into account the past experiences of reference plants. Based on the final results of the internal events analysis for UCN 3&4, the following conclusions were drawn:

- Human errors show up as the most important contributors to the total CDF. Important operator actions are related to the aggressive cool down for low pressure safety injection, and feed-and-bleed operation for short-term core cooling,
- The safety-related components and systems appear very robust, and behave extremely well for a design basis earthquake.
- The dominant seismic accident sequences are due to failures of condensate storage tanks, EDGs, and batteries with rack, which are assumed to lead directly to core damage.

Using insights gained from the results, KHNC has planned to prepare and implement the SAMP, which enables utility's staff to deal with such rare events. By doing that, KHNC has provided a SAMP guidance and corresponding technical basis report for responding to potential severe accidents in UCN 3&4.

4.10.2.4 Yonggwang Units (YGN) 5&6 Level 2 PSA

YGN 5&6 Level 2 PSA was finished in 2001. The special characteristic of YGN 5&6 PSA is that it is the first low power/shutdown (LP&S) PSA in Korea. YGN 5&6 PSA has a purpose to identify the risk during low power and shutdown operation and improve plant safety. YGN 5&6 is under the construction, which are also KSNP. The scope of the study is the internal level 1 PSA and the level 2 PSA for full power operation including external events analysis that are doing simultaneously. The preliminary evaluation was established in 1996 and the quantification is finished in early this year.

It is noted that related application programs of LP&S PSA have covered a limited scope because their results have much uncertainty, compared with full power PSA. The LP&S PSA methodology development program in Korea has been performed since 1994 by KAERI (Korea Atomic Energy Research Institute).

Based on the result of the research program, some information on outage experience for reference plant (YGN 3&4) was reviewed and 17 plant operational states (POSSs) were defined. The accident sequence analysis, plant information gathering, and HRA were extensively done. General PSA approach and procedure were applied for LP&S PSA, like fault tree/event tree method. It is well known that the significant problem of this operational mode is the quantification of human intervention. The THERP (NUREG/CR-1278) and ASEP HRA procedure were used as main methodology for HRA. However, it was recognised that more detailed HRA methodology should be developed to evaluate various shutdown-specific situations.

Some decision trees for quantifying dynamic human actions are developed to allow a human reliability analyst to perform a systematic and consistent HRA. The performance shaping factors considered in HRA

are available time, stress level, level of operator training/experiences, existence of hesitancy, type of task, failure of makeup at mid-loop POS, and so on. Three kinds of dependencies are considered as follows:

- Dependencies between the activities that make up an action,
- Dependencies between operators,
- Dependencies between various parallel human actions in accident sequences; e.g. fails to shutdown cooling operation using standby shutdown cooling pump or containment cooling water system.

The dependencies between operators are explicitly considered in the quantification of recovery probability of execution error. The dependency level between multiple human actions having same goal in accident sequences is determined mainly on the cues of operator actions, time difference between two human actions, structures of procedures, and interviews with operators.

External events are not included in current work scope. The thermal hydraulic analyses are required in detail to evaluate operator response time as an input to the HRA. The use of various computer codes, such as MAAP4 and RELAP5 as a supporting tool depending on the phenomena of concerns, was recommended.

The initiating events (IEs) were identified by similar methodology of full power PSA. The actual plant experiences were intensively reviewed using plant document such as, operation logbooks. The followings IEs are selected, with the consideration of the grouping:

- Transients,
- LOCAs,
- Loss of shutdown cooling function,
- Loss of off-site power, and
- Some plant-specific IE (e.g., loss of supporting systems).

The typical POS duration time is calculated using reference plant, YGN 3&4, refuelling outage experience.

Four thermal hydraulic calculations were performed to determine need of containment spray recirculation cooling as ultimate heat sink using RELAP5/MOD3. There are two type's calculations. The first one, as a base case, is to simulate the response after loss of SCS (Shutdown Cooling System). The other is to calculate the primary system behaviour with forced feed-and-bleed operation. The results indicated that containment spray cooling is not required for mitigating the loss of SCS event in POS 4A (Drain w/o large vent).

The developed LPS PSA model is used to update and verify the safety improvement options in the following UCN 5&6 PSA. The design stage LPS PSAs for APR 1400 (Korea Next Generation Reactor) and New Kori 1&2 (Korean Standard Nuclear Plant Plus) are being performed to verify the design, and to improve the safety of these NPPs.

4.10.2.5 APR 1400 (KNGR) Level 3 PSA

Next generation nuclear power plants to be constructed after 2004 in Korea, known as APR 1400 (previously KNGR) with the capacity of 3983 MWt, utilise many evolutionary advanced design concepts. The second stage preliminary PSA was finished in 2000 and was submitted to KINS for review. The final

third stage PSA is being performed. The APR 1400 PSA has been performed to satisfy the objectives required for the design certification, for example, it was desired to show that the design satisfies the safety goals imposed by utilities.

Based on the preliminary results of this PSA, the following insights were given:

- If the strategy to avoid the common cause failure would be developed and adopted at the detailed design stage, especially for safety injection systems, it would be enhance the safety of the plant,
- In-containment refuelling water storage tank is quite efficient to reduce CDF,
- Internal fires during power operation do not represent a significant contribution to the overall CDF,
- Operator training and management control of plant configuration are important to reducing shutdown events and risk,
- During plant shutdown operation, the integrity of fire and flood barriers between areas in the same division, where systems comprising the alternate shutdown are located should be maintained,
- New advanced design features can significantly enhance the containment performance.

4.11 Mexico

The current PSA model for Laguna Verde developed by the utility and approved by the Mexican regulatory authority, is a detailed one that involved the utilisation of safety and non-safety systems. That assess the impact of the containment status on the continued core cooling (i.e. the model evaluates the harsh environment as result of primary containment venting or failure and evaluates the failure probability of the core cooling systems components to survive such conditions) but it has some deficiencies. The PSA model uses a generic database to evaluate the failure rates and the initiating event frequencies, and it does not represent in an accurate way the status of the plant. In order to overcome such deficiencies and to have a suitable PSA model for applications and for decision making, the Mexican regulatory authority has asked the utility to take the necessary steps to make the actual PSA model a Living PSA. The utility is currently updating the PSA model by incorporating plant specific data for component failure rates and initiating event frequencies. The information is being compiled from the Licensee Event Reports (LER), actual data base for example for the maintenance rule, the reliability program for the emergency diesel generators, by means of the Bayesian analysis. This updating will also incorporate the modifications to the plant design and operational features, as well as the level of understanding of the thermal-hydraulic performance and improvements made to the modelling techniques. The frequency of subsequent updating will be at the end of each refuelling outage, or whenever there were significant changes that impact the figures of merit, i.e., CDF or LERF.

The utility offered to develop a Risk Monitor (RM) in order to comply with the paragraph 4 of the Maintenance Rule, which states that before performing maintenance activities the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. The Risk Monitor was developed using the Equipment-Out-Of-Service (EOOS) computer package from EPRI/SAIC and is being used on a trial basis. The Mexican Regulatory Agency has been involved in the review of the implementation of the PSA models into the EOOS software. The RM will be updated accordant with the updating of the PSA used to support such tool, once the updated model is approved by the CNSNS.

As it was mentioned above, in order to have an adequate framework to evaluate the applications, the Mexican regulatory authority has issued the draft for comments of the Regulatory Guide 1.00 which is based on the USNRC Regulatory Guides and has been adapted to the Mexican case. This regulatory guide establish an approved methodology for using probabilistic risk assessment in risk-informed decisions on temporary and permanent plant-specific changes to the licensing basis for Laguna Verde NPP.

Another activity that has been initiated recently is the establishment of the Risk Based Performance Indicators (RBPI) for Laguna Verde NPP. The objective is to monitor performance in three broad areas: reactor safety (avoiding accidents and mitigating the consequences of accidents if they occur); radiation safety for both plant workers and the public; and protection of the plant against security threats (safeguards). In the context of the RBPI, performance refers to those activities in design, procurement, construction, operation and maintenance that support the achievement of the objectives of the fundamental aspects of safety. To monitor plant performance some specific fundamental aspects of safety have been defined which support the safety of plant operations in the three above mentioned broad areas: Initiating Event, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Occupational Radiation Safety, Public Radiation Safety, and Physical Protection. The implementation of the Risk Based Performance Indicators has been divided in three phases. In the first one, a set of performance indicators for the Initiating Event, Mitigating Systems, Barrier Integrity and Emergency Preparedness aspects of safety has been defined. These performance indicators are based on the results and insights of the Individual Plant Examination developed by the licensee and reviewed by the Mexican Regulatory Agency. These RBPI are currently being subjected to a review and open for comments. Once the comments are received and incorporated a one-year trial period has been set. The second phase will involve the validation and verification of these RBPI and new ones will be defined to cover the rest of the fundamental aspects of safety. The third phase will verify and validate the implementation of the RBPI in order to assure that they identify declining trends before performance becomes unacceptable, without incorrectly identifying normal variations as degradations. One important aspect of the implementation of the RBPI is the definition of an indicator that will provide the capability to consistently assess the integrated risk significance of the performance indicators and the inspections findings on the overall plant performance.

Also a PSA project has been initiated within the Mexican regulatory authority to support the activities of the Radiation Protection Division. The project entitled "Probabilistic Risk Assessment Associated with the Operation of an Industrial Irradiator" has been presented to the IAEA as a Co-ordinated Research Project. In facilities using radiation sources the safety improvements have been based on the perceived weaknesses in the systems, procedures and processes or on lessons learned from accidents that have occurred. This project is aimed to apply the PSA techniques and tools to the safety evaluation of an industrial irradiator installation, and to find out if such tools are also valuable in the identification of vulnerabilities or weaknesses in the safety systems or if some changes to the methodology are needed. A deep familiarisation with the facility is under way through the review of the design, review of the operation and maintenance procedures as well as the training of the operator. A technical visit and plant walkdowns have been performed in order to gain a deep knowledge of the operation and safety systems of the irradiator. The determination and selection of the plant operating states of concern have been already performed. The risk measures associated with the irradiator facility have also been defined. A thorough review of the incidents that have taken place in such facilities around the world has been performed to identify potential initiating events. A failure mode effect analysis (FMEA) has carried out to define the possible events that give rise to disturbances in the plant and have the potential to lead to consequences of concern. As a result, a complete list of initiating events has been set up. The analysis of the response of the plant due to the occurrence of disturbances that required a plant shutdown by means of event trees and fault trees is currently underway.

Another PSA activity is currently underway at the Instituto Nacional de Investigaciones Nucleares (ININ) to estimate the Laguna Verde NPP core damage frequency associated with fire. Based on the internal flooding analysis performed by the ININ for the utility, as part of the Individual Plant Examination, the

ININ was commissioned to extend the scope of the internal flooding analysis and to include the fire initiating events, propagation and quantification of the CDF associated. This study is being accomplished on a voluntary basis, since there is no regulatory requirement to perform an Individual Plant Examination for External Events (IPEEE), yet. The study is actually being subjected to an internal peer review by the utility and will be submitted to the regulatory authority for review.

4.12 Netherlands

Both Dutch nuclear power plants launched their PSA programmes in 1989. The main objective of these PSAs was to identify and assess the relative weak points in the design and operation of the power plants, and thus to facilitate the design of accident management measures, and also to support backfitting. An assessment of source terms, public health risks, etc., was regarded as unnecessary at that time.

The licensees translated the regulatory requirements as well as their own wishes regarding the objectives of the PSAs into their original bid specifications:

- To identify and analyse accident sequences, initiated by internal and area events, that may contribute to core damage and quantify the frequency of core damage.
- To identify those components or plant systems whose absence most significantly contributes to core damage and to isolate the underlying causes of their significance.
- To identify weak spots in the operating, test, maintenance and emergency procedures that contribute significantly to the core damage frequency.
- To identify any functional, spatial and human-induced dependencies within the plant configuration that contribute significantly to the core damage frequency.
- To rank the weak spots according their relative importance and to easily determine the effectiveness of potential plant modifications (both backfitting and accident management). See Annex 1 for a more detailed description of the PSA-based backfitting and modifications at the Borssele NPP.
- To provide a computerised level-1 PSA to support other living PSA activities such as the optimisation of Technical Specifications, maintenance planning, etc.
- To transfer technology and expertise to the licensee to allow it to evaluate future changes in system design, operating procedures and to incorporate these changes in a ‘Living’ PSA.

4.12.1 Living PSA applications

After the PSA relating to the modification project was completed, the focus shifted towards ‘Living PSA’ (LPSA) applications. Even, the new licence for the modified Borssele plant required the licensee to have an operational ‘Living’ PSA, but gave no further details of the concept and applicability of such a LPSA. Both the licensee and the regulatory body are in the process of defining the boundary conditions for possible applications. The use of PSAs for configuration control, the optimisation of Technical Specifications, or event analysis are potential applications. The current ongoing LPSA applications, such as support for backfitting measures, support for periodic safety reviews, support for licensing activities, retrospective use of the risk monitor, optimisation of test and maintenance strategies, incipience of reliability-centred maintenance, etc., will be continued or intensified. However, the number of applications

might need to be expanded in order to make maximum use of the LPSA. For this reason, in 1999 the IAEA was asked to produce a Peer Advisory Report on LPSA applications tailored to the specific conditions in The Netherlands. Because the regulatory authorities expressed its wish to make greater use of LPSA insights and to move to a more risk-informed form of regulation, the IAEA has also been asked to include these aspects in its report. The main conclusion and recommendation of this Peer Advice was:

In order to make use of risk information in regulatory decisions in a formal and predictable way, the authority should develop an appropriate framework.

In appendix 5 an outline is given of the conclusions and recommendations of this IAEA peer advice, and on the follow-up actions with respect to Risk-informed regulation.

4.12.2 Guidance and review of the PSAs

No national PSA guidelines existed at the onset of the Dutch PSA programmes in 1988/1989. To make matters worse, both the licensee and the regulatory body had very little experience with the development of a complete PSA for a nuclear power plant. Both licensees therefore requested foreign contractors to develop the two PSAs. At the first round of talks (in 1988) between one of the licensees (i.e. the Borssele NPP) and the regulatory body, only general requirements, and the scope and objectives of the PSA were discussed. One of the key elements in these talks was the need for technology transfer from the contractor to the plant staff. Much of the available knowledge came from studying literature, such as NUREG reports, rather than from any hands-on experience. It is fair to say that the ongoing regulatory guidance and assessment benefited greatly from this technology transfer, as well as from the peer reviews that were held. This was equally true for the licensees. The regulatory requirements set and instructions given concerned the scope, the level of detail, whether or not best-estimate techniques could be used for modelling purposes, etc. As far as more detailed technical matters were concerned, the USNRC PRA Procedures Guide (NUREG/CR-2300) and the PSA Procedures Guide (NUREG/CR-2815) were considered to be acceptable at that time.

Because the Dutch authorities and their traditional technical support organisations had only limited experience with nuclear PSA programmes, and also because of regulatory body had limited staff resources, the IAEA was asked for support. This support was provided in the form of peer reviews of the PSAs (IAEA-IPSART-missions, formerly known as IPERS missions), and training courses in PSA techniques and PSA review techniques. The PSAs of both plants were scrutinised by IPERS reviews at various stages of their conduct. For example, the first stage of a peer review of the Borssele PSA by the IAEA took place at the onset of the PSA programme. This review involved looking at the agreed scope of the PSA and assessing how this had been translated into a project proposal by the contractor. Another example was a limited IPERS mission which took place with the aim of checking whether all the issues raised in previous IPERS missions had been adequately resolved in the final report. This review showed that all the issues raised in previous IPERS missions had indeed been adequately resolved, and that the PSA was of high quality.

4.13 Spain

The common scope established by the PSA Integrated Programme second edition for the Spanish NPP PSA is that of Level 1 and 2 analyses, including all reactor operating modes and all external events. As it was described in previous yearly reports, the original scopes of the plant specific PSA was progressively increased and none of the seven original PSA were done up to that level. Therefore, all the PSA are being revised and updated to get to the mentioned common scope. Up to now, only Vandellos, Asco and Garona NPP have got to that scope, although these three PSA have still to be revised to incorporate the comments obtained from the review done by the CSN to the Level 2 and to the Low Power and Shutdown PSA (SPSA).

Current status can be summarised as follows:

- Garona. (BWR, GE Mark I). CSN review completed for Level 1, Level 2 and internal (fires, floods) and external hazards. PSA to be revised by the utility in 2001. SPSA submitted to the CSN by the end of 2000. SPSA CSN review ongoing until 2002. Ongoing CSN review of seismic risk (margins).
- Almaraz. (PWR, W 3 loops). CSN review completed for Level 1, Level 2 and internal and external hazards. PSA is being revised and updated in a continuous maintenance process. SPSA to be started by the utility in 2001. Ongoing CSN review of seismic risk.
- Asco. (PWR, W 3 loops). CSN review completed for Level 1, Level 2, SPSA and fires and internal floods. SPSA to be revised by the utility in 2001. Whole PSA also revised and updated in 2001. Ongoing CSN review of the rest of external events.
- Cofrentes. (BWR, GE Mark III). CSN review completed for Level 1, Level 2 and internal and external hazards. PSA is revised and updated in a continuous maintenance process. Decision on SPSA or other methodology for shutdown risk analysis after CSN review of the Garona (the other Spanish BWR) SPSA.
- Jose Cabrera. (PWR, W 1 loop). CSN review completed for Level 1, Level 2 and fires and internal floods. Ongoing CSN review of the seismic risk. Whole PSA to be revised and updated in 2001. SPSA and the rest of external events to be started in 2002. PSA used extensively in the Periodic Safety Review process, linked to the operating licence after 2002.
- Vandellos. (PWR, W 3 loops). CSN review completed for Level 1, Level 2, SPSA and internal and external hazards. Level 1 PSA revised and updated in 2000. The rest of the PSA still to be revised to incorporate the CSN review comments.
- Trillo. (PWR, Siemens 3 loops). CSN review completed in 2001 for Level 1, internal floods and external hazards. Ongoing CSN review of Level 2. Fire and seismic risk analyses still to be completed by the utility. SPSA to be started after the Almaraz SPSA.

In Appendix A, this summary is also given by means of a table.

According to the plant specific PSA status above summarised, it can be seen that the PSA performance activities are only in the fields of SPSA and external events and that the CSN review is completed for most of the analyses already submitted by the utilities. Therefore, from 2001, the PSA activities in Spain are mainly directed towards PSA applications and related tasks, like the PSA maintenance and updating process and the still needed research in some areas.

4.14 Sweden

The PSA goals and - program in Sweden can be summarised as follows (Status October 2001):

The Swedish Government has specified the following;

Within the reactor safety area SKI's supervision is to ensure that Swedish nuclear facilities implement and maintain adequate protection based on the concept of multiple physical barriers to prevent the occurrence of severe incidents and accidents originating from technology, organisation or human competence as well

as to prevent or mitigate radioactive releases to the environment in the event of an accident. Thus, safety must be based on the internationally accepted defence-in-depth principle, which has been adopted in the international convention on nuclear safety, in order to protect man and the environment from the harmful effects of nuclear activities.

Important components of the defence-in-depth principle include ensuring that:

- The plant design is robust so that:
 - plant operation is undisturbed and that sources of abnormalities are few
 - the plant has multiple physical barriers to mitigate radioactive releases
 - the plant has several safety systems to protect the physical barriers from damage in the event of abnormal events and accidents.
- The technical plant reliability is verified through a suitable programme for control and testing, of the condition of the physical barriers as well as of the reliable performance of the safety systems.
- The plant is operated and maintained by a robust organisation, e.g. characterised by a clear division of responsibility and adequate financial and human resources.
- The quality of all processes relating to the interaction between man-technology-organisation (human factors) is verified through a clear safety policy, clear management and follow-up, unambiguous procedures, recurrent training and exercises as well as efficient internal safety audits and quality assurance.
- The plant has an efficient system for analysing and learning from its own and others' operating experience and research.
- Periodic Safety Reviews (ASAR reports) are performed periodically in order to renew the safety assessment of the plant

SKI's regulatory criteria for each component of the defence-in-depth system are promulgated in SKI's regulations and recommendations as well as in the conditions specified by the Government and SKI in licences to conduct nuclear activities. SSI has, correspondingly, promulgated its radiation protection criteria through its regulations.

PSA activities of today, in Sweden have as earlier indicated to fulfil the regulations in SKIFS 1998:1.

The applicability of the SKIFS 1998:1 is:

Chapter 1. Applicability and Definitions

1 § These regulations apply to the following types of nuclear facilities for which permission to conduct nuclear activities is issued by the Government on the basis of 5 § of the Act (1984:3) on Nuclear Activities or corresponding provisions of the Atomic Energy Act (1956:306).

- a nuclear power reactor,
- a research or materials testing reactor,
- a facility for the fabrication of uranium pellets and fuel bundles,
- a facility for the storage or other handling of spent nuclear fuel,
- a facility for the storage, processing or final disposal of nuclear waste.

2 § The following terms and definitions are used in these regulations:

Barrier: Physical containment of radioactive substances.

Physical protection: Engineered, administrative and organisational measures which aim at protecting facilities against unauthorised intrusion, sabotage or other such impacts which can result in a radiological accident as well as at preventing the unlawful possession of nuclear material or nuclear waste.

Normal operation: Operation within the conditions and limitations stipulated in the Technical Specifications for a facility.

Radiological accident: Any deficiency arising in a barrier or any other condition leading to the dispersion of radioactive substances or which leads to radiation doses exceeding permissible limits during normal operation.

Safe state: Cold shutdown or another operating state which minimises the risk of a radiological accident.

The PSA program and activities nowadays in Sweden have the take into account and analyse all the states belonging to the normal operation (full power, shut-down, refuelling, start-up).

Status of the Swedish PSA program, is as follows (as of October 2001);

Table 1. Status year 2001, last time reported PSA studies to SKI and PSA work in progress.

<i>Plant</i>	Level 1		Level 2		Fire, Flooding		Refuelling	Up- and shutdown
	Op	Low	Op	Low	Op	Low	Low	Low
Barsebäck (1) and 2	1998		1995		1998 2002		1995 2001, method development 2003 4)	1998 2001, method development 2003 4)
Forsmark 1 and 2	1995 2001		- 2001		1997 (only fire) 2001		-	-
Forsmark 3	1995		1995		1995*		1995*)	-
Oskarshamn 1	1997		1998		1997		1998*)	-
Oskarshamn 2	1999 2001 1, 2)		1999 2001 1)		1999		- 2002 2)	- 2002 2)
Oskarshamn 3	1998		1998		1998 2002 3)		- 2002 3)	- 2002 3)
Ringhals 1	1992		1996		1997		- 2001, method development	- 2001, method development
Ringhals 2	1992		1994		1994		1995 2002 3)	- 2002 3)
Ringhals 3 and 4	1992		-		1997 (only fire)		- 2002 3)	- 2002 3)

Legends: Op = at operating mode. Low = at low power operation

*) *Simplified study, no complete PSA.*

1) *The fall of 2001*

2) *Level-1 and Level-2 studies for all normal operating modes, planned for year 2002*

3) *Planned for year 2002*

4) *Planned for 2003*

Periodic Safety Review history in Sweden.. ASAR history. ASAR reports for the bold marked plants are now being reviewed

Plant	Starting Year	ASAR 1, # 1			ASAR 2, #2			ASAR 3, #3		
		Period	Report to SKI	SKI report	Period	Report to SKI	SKI-rapp	Period	Report to SKI	SKI report
Barsebäck 1/2	1977	-77—83	1984	1985	1984 - 1994	1995	1996	1995 - 2004		
Forsmark 1	1980	-80—89	1990	1991	1990 - 2000	2001	2002	2002-2012		
Forsmark 2	1981	-81—90	1991	1991	1990 - 2000	2001	2002	2001-2011		
Forsmark 3	1985	-85—95	1996	1997						
Oskarshamn 1	1972	-72—81	1982	1983	1982 -1991	1992	1995	1992-2002		
Oskarshamn 2	1975	-75—85	1986	1990	1986 -1998	1999	2001	1999-2009		
Oskarshamn 3	1985	-85—95	1996	1997	1996-2006					
Ringhals 1	1976	-76—83	1985	1986 ⁴	1984 -1994	1996 + 1999	2000	1995 - 2005		
Ringhals 2	1975	-74—83	1983	1985 ⁵	1983 -1992	1994	1995	1993 - 2003		
Ringhals 3	1981	-81—91		1994 ^{6, 7}	1992 - 2001	2002	2002	2003-2013		
Ringhals 4	1983	-83—91		1994 ^{3, 4}	1992 - 2001	2002	2002	2003-2013		
Studsvik										
Bränslefabriken										
CLAB	1985	-85—96	1997	2000	1997-2007					
SFR (Updating of final safety report)	1988	-88—00	2001*)	2002	2001-2011				*) SAFE = Safety Assessment of Final repository for operational radioactive waste.	

4. SKI Rapport – SKI-ASAR-R1, 1986
5. SKI Rapport 95:35 – SKI-ASAR-R2, augusti 1995
6. SKI Rapport - 94:24, SKI-ASAR-R3/4, December 1994
7. SKI Rapport – 95:38, Granskning av Ringhals 3 och 4 Säkerhetsstudie.

The licensees in Sweden are nowadays planning and preparing themselves for using more risk informed decision making and use of results from PSAs in optimisation of e.g., maintenance activities and frequencies of tests.

In our Westinghouse PWR plants, the licensee is planning for adopting new standard Technical Specifications according to the NUREG-1431. New standard Technical Specifications are planned to be taken into use in January 2003. PSA (generic USA) is used by the licensee to judge, weather some components shall remain in the Technical Specifications or not.

4.14.1 PSA data

The component data handbook T-Book - Reliability data of components in Nordic Nuclear Power Plants, is updated on a regular basis. The most recent edition is - T-Book 5th version. ISBN 91-631-0426-1

The T-Book can be ordered from:

The TUD office
SwedPower AB, Energy Technology
PO Box 527
SE-16216 Stockholm, Sweden

Phone: +46 8 7397320

4.14.2 PSA researching

In Sweden and Finland the Nordic licensees have established a special working group, called Nordic PSA group (NPSAG), in which PSA related researching is discussed, initiated, followed-up.

The regulatory body SKI in Sweden and STUK in Finland are also members of that NPSAG group, adjourning members.

Matters that concern general reactor safety issues and touch PSA questions are items on the agenda for the NPSAG.

4.15 Switzerland

All Swiss licensee PSAs are full scope level-1 and level-2 studies. They include analyses of:

- Internal and external events
- Low power and shutdown states.

No level 2 PSAs have been conducted for low power and shutdown modes. In addition, for each nuclear power plant HSK maintains an independent regulatory PSA model that has been developed within the process of reviewing the licensee PSA. The regulatory PSAs are focused on full power state, and are, in general, level 2 (including the analysis of radiological release activities to the environment, to help assess the potential risk impact of SAM actions and/or plant modifications).

HSK is committed to move into the domain of risk-informed/guided regulatory process. Therefore, the current programs are specifically designed to help achieve this outcome.

4.16 United Kingdom

4.16.1 PSA for the Sizewell B PWR

The PSA that has been produced for Sizewell B is a full scope Level 3 PSA. It addresses all modes of operation of the plant (full power, low power and shutdown modes), and all internal initiating events and internal and external hazards.

The PSA that was produced as part of the safety case leading up to fuel load in September 1994 has been revised so that it can be used as a Living PSA during station operation. The Level 1 part of the analysis has been changed from a large fault tree approach to one that is based on linked event trees and fault trees using the Risk Spectrum software. The Level 2 and 3 parts of the analysis have been factored in using invariant transformation matrices.

This new PSA gives a better estimate of the risk by removing some of the conservatism which were in the licensing PSA and will be used by the licensee to advise on configuration control during plant outages, to assist in monitoring the validity of the Technical Specifications and to produce risk profiles with the aim of maintaining the risk as low as reasonably practicable at all times.

The PSA has been used to provide operational support in a number of areas including the following:

- increasing the enrichment of the fuel used in the reactor,
- increasing the time given in the Technical Specifications for the period between refuelling outages from 18 months to 2 years, and
- considering the best options available for managing the risk during refuelling outages. This addressed the risk which would arise when the reactor coolant system inventory was reduced to mid-loop level.

4.16.2 PSA for the AGRs

PSAs have been produced for all the AGRs as part of the Periodic Safety Reviews which are carried out every 10 years. The aim of the analysis is to address the dose-frequency criteria given (see SAP P42 above) so that Level 2 PSAs have been carried out which determine the radiological consequences of the fault sequences identified.

The most recent PSAs are those carried out for the Heysham 2 and Torness Periodic Safety Review which was completed in 1999. These are Level 2 PSAs for internal initiating events occurring during full power operation only (the licensee has argued that the level of risk during shutdown conditions would be very low). These PSAs address internal initiating events fully but have a limited treatment of internal and external hazards.

- The PSA has been used to provide operational support in a number of areas including the following:
- increasing the time given between refuelling outages from 2 to 3 years, and
- extending the duration of shifts at Hinkley from 8 to 12 hours.

These PSAs have been enhanced to produce a four-quadrant model which explicitly represents initiating events occurring in each of the four quadrants. The four-quadrant model is being maintained as the Living PSAs for the stations and will be used as the basis for the updated Risk Monitors that are being implemented.

4.16.3 *PSA for the Magnox reactors*

A Long Term Safety Review (LTSR) was carried out for each of the Magnox reactors to determine whether it was safety to allow operation to continue beyond about 30 years. The LTSRs reviewed the plant against both engineering/ deterministic and probabilistic principles. As a result of this, a number of changes were identified to the design and operation of the plant which were required to meet modern standards and to reduce the risk (see Chapter 5). In addition, significant deficiencies were identified in the scope and contents of the PSAs and it was agreed that they should be improved.

The PSAs for all the Magnox reactors have now been completed and updated as part of the PSR process. These are Level 1+ PSAs which address internal initiating events occurring during full power operation only. There is only a limited treatment of internal hazards and, for natural external hazards such as seismic events, a deterministic approach has been used supported by some probabilistic analysis

In the analysis, it has been assumed that failure to trip, shutdown or provide post trip cooling would lead to a large release - that is, an off-site dose of greater than 1000 mSv. Sequences which resulted in smaller releases (generally less than 100 mSv) were also assessed in terms of frequency and consequence and the results brought into the consideration of whether the risk from the station was ALARP. The smaller releases arise from sequences following a successful reactor trip and shutdown where there was nevertheless the statistical possibility of limited fuel failures. In addition, the analysis has addressed faults involving water ingress into the reactor with safety relief valve lift, and fuel route faults.

The PSAs have been used for a number of activities as follows:

- to inform decision making with respect to requirements for heating and ventilation dampers used to protect equipment from the effects of a loss of primary coolant accident (hot gas release),
- to assist in setting the design requirements for modifications to boiler headers whose failure would result in a loss of primary coolant accident (hot gas release),
- to inform on the benefits of modifications in terms of reductions in expected accident cost, and
- to check that operating rules related to plant unavailability deliver satisfactory control of risk.

Lifetimes for the Magnox reactors have now been declared so that further updates of the PSAs will not be required with the exception of Wylfa where a revision of the LTSR PSA is currently being undertaken and is expected to be completed in 2003.

4.16.4 *PSA for the naval nuclear reactors*

The facilities which carry out the refit and refuel activities for the naval nuclear reactors are being refurbished to bring them up to a modern standard. Since they are licensed nuclear sites, they follow the same requirements as for any new nuclear facility. In particular, there is a requirement to produce Pre-

Construction and Pre-Operational Safety Reports (PCSR/ POSR) which need to include a PSA to address the relevant accident frequency criteria given in the SAPs.

For the new facility that has recently started operation, a Level 1 PSA has been produced which addresses all the shutdown modes that occur during the submarine refit/ refuel process. The PSA addresses all internal initiating events and internal and external hazards (including a full PSA for seismic events). The licensee is in the process of extending this to a full Level 3 PSA.

The aim is to use the PSA to provide risk information during the operation of the facility to control the equipment that is removed from service during the refit process.

4.16.5 PSA for research reactors

The CONSORT reactor is a small 100kW research reactor operated by Imperial College. A PSA has been produced as part of the Periodic Safety Review to determine whether it was safe to operate the reactor beyond 30 years.

Although the design of the reactor is such that failures would not lead to overheating of the fuel, a PSA that has been produced which has identified the fault sequences which would lead to a failure to shut the reactor down, failure of the decay heat removal systems, over-exposure of reactor operators and people carrying out experiments, etc. This analysis is broadly equivalent to a Level 1 PSA and addresses a wide range of initiating events and hazards. It was used to identify the relatively weak areas in the design and operation of the plant where improvements are being made.

4.16.6 PSA for nuclear chemical facilities

As a result of the audit of safety carried out by NII of the nuclear fuel reprocessing activities on the Sellafield site, the licensee was required to produce Fully Developed Safety Cases (FDSCs) which include a PSA. Many of these safety cases have been produced and assessed by NII. As a result of this, improvements have been made to the design and operation of the plants.

The current status of the PSA programme is now closely linked to the Continued Operation Safety Reviews (COSRs) process which is agreed between the licensees and the regulator.

4.16.7 PSA for AWE sites

Following the decision by the government that the sites where atomic weapons are manufactured would be licensed, safety cases needed to be produced for these facilities which were of the same standard as those that NII would expect to see for nuclear power plants. These have included PSAs which mainly address the risk to workers (since the risk to the public from these facilities is relatively small) which includes the risk from an inadvertent criticality incident.

4.17 United States

The Nuclear Regulatory Commission's (NRC) program in probabilistic risk analysis includes a spectrum of activities, from basic research to regulatory applications. The PSA methods and tools developed and used in NRC's program support the NRC's strategic plan and its four strategic performance goals of: (1) maintaining safety, (2) improving staff regulatory effectiveness, efficiency and realism, (3) reducing unnecessary burden, and (4) increasing public confidence. The NRC strategic plan is on the Internet at the following address: <http://www.nrc.gov/reading-rm/basic-ref.html#statutes>

The Risk-Informed Regulation Implementation Plan (RIRIP) describes the NRC's strategy to risk-inform NRC regulatory activities in all arenas. The key topics discussed in the RIRIP are:

- Risk-Informed Regulation - Guidelines, Factors to Consider, Communication Plans, and Training Plans
- Risk-Informed Regulation Implementation Activities - Reactor Safety Arena, Nuclear Materials Safety, and Nuclear Waste Safety Arenas

The plan is updated semi-annually. For the current version of RIRIP visit the following website: <http://www.nrc.gov/what-we-do/regulatory/rulemaking/risk-informed.html>.

CHAPTER 5 – PSA APPLICATIONS

PSA experience in Member countries has grown considerably during the past 10 years. This chapter presents information on how PSAs are being applied and identifies specific applications being used for decision-making.

5.1 Belgium

5.1.1 *Design evaluation*

Up to now, the main application concerns design evaluation. Indeed, the primary objective is to use the PSA, in the framework of the periodic safety review, as a complementary tool to the deterministic safety analysis. It should mainly provide valuable insights in the balance of the design, identify important contributions to the core melt frequency and constitute a useful tool to evaluate the effectiveness of proposed plant modifications.

5.1.2 *Accident management*

Based on the results of the first level 1+ studies performed for the Doel 3 and Tihange 2 plants, the utility decided to install catalytic hydrogen recombiners in the containment, for all 7 nuclear power plants. This action is now implemented for all units.

Further discussions with the utility on other accident management measures are to be continued in the framework of the periodic safety reviews of the plants. The PSAs are expected to play a role in these discussions.

5.1.3 *Evaluation of Technical Specifications*

Insights gained from the PSAs have been used by AVN for arguing in some Technical Specifications related matters (for instance, requirements on the availability of some systems in shutdown). Until now, this has not yet led to formal modifications of the Technical Specifications.

So far, no requests have been discussed with the utility for modifications to the Technical Specifications based on PSA insights.

In the opinion of AVN PSA can contribute to the optimisation (not only relaxation) of Technical Specifications. This is especially the case for the Technical Specifications in shutdown conditions, where the justification for allowed unavailabilities for instance is even weaker than in the power conditions. It is our expectation that in future this kind of application will be discussed with the utility.

5.1.4 Other

Discussions are on going between the utility, TEE and AVN concerning the future use of the PSAs. In this context a discussion is being organised on the application of R.G. 1.174 in the Belgian framework.

A risk informed training programme of operators and plant management is ongoing. Its objective is to focus attention on the high safety significant components, important accident sequences, and on pre- and post-accidental human actions.

An international survey of on-going activities in the area of Risk-Based In Service Inspections has been started at TEE.

5.2 Canada

The PSAs play an increasingly important role in CNSC activities, as the following examples indicate:

- Based on the Pickering A Risk Assessment (PARA) insights, CNSC put a number of conditions on the restart of the four units at Pickering A. More specific, CNSC required that:
 - the core melt frequency (CMF) be significantly reduced such that the OPG internal CMF target to be met, and
 - the event sequences that do not meet the single failure criterion to be removed as much as possible.
- Using Bruce B Risk Assessment (BBRA) results, OPG requested CNSC to drop an action item.
- OPG used the results of Darlington A Risk Assessment (DARA) to support a request for functional changes.

It is expected that in the future PSAs will be applied more in many areas, such as configuration management, significant event analysis, maintenance, operator training, operating procedures, design changes or backfit, operational safety system test program, etc., to support safe operation of nuclear plants. [3] Consequently, both the CNSC and the Canadian nuclear industry will be dealing with submissions supported by risk arguments from the PSAs.

5.3 Czech Republic

Level of use of PSA applications is dependent on PSAs status and scope, particular NPPs needs and objectives and on Regulatory body PSA policy. In the Czech Republic three types of PSA applications have been currently used.

At Dukovany site there is a systematic process with aim to decrease units risk level and therefore PSA is currently used mostly for identification of weak points, evaluation of design and procedures modifications and improvements, comparison of alternatives and making priorities for implementation of such safety measures.

Modified Living PSA models have been also used for an evaluation of Allowed Outage Times (AOTs) adequacy and STIs optimisation. In parallel way a Risk Monitor system has been developed and

implemented for NPP units operation at full power. The system is being used within NPP risk management process to support a short term risk - based decision making. In the near future all units states are supposed to be implemented into the Risk monitor system at Dukovany site.

Temelin plant PSA models have been also used for application on the field of AOTs evaluation within plant Technical Specifications to be consistent with risk insights. Besides this, the PSA model has been used for other application on the design and I&C field to evaluate and confirm some design criteria and implemented backfitting measures.

It is expected that also at Temelin site, similar to Dukovany NPP practice, the PSA will be used extensively in the future for various risk informed applications to assess the impact on safety of design changes, procedural changes, changes in the technical specifications, and changes in test and maintenance procedures and strategies based on risk minimisation (IST, ISI). Also, it is expected that the PSAs will be used by plants training personnel to gain a better understanding of certain accident sequences and thus enhance the training given to the plant personnel. Evaluation of events, outage risk management or providing risk profiles associated with the plant operation could be the other areas of PSA applications.

Safety Monitor system for Temelin has been developed since 1996, both specific top logic and databases while the software was customised for the Czech language and Temelin needs. The software pilot on-line operation on the plant local computer network was started in the mid of 1999 with aim to get users feedback and experience from on-line software operation. It is expected that the Safety Monitor will be used extensively by the maintenance division in scheduling preventive maintenance activities, both at power and during refuelling as well as by operator personnel for information about a risk associated with an upcoming plant configuration prior such configuration will be entered, including recommended limiting time to be allowed for such configuration or to provide risk reduction advise for such configuration.

5.4 Finland

5.4.1 Use of PSA for analysis of plant modifications and operational events

PSA has been applied much in evaluation of plant modifications. As a matter of fact it is a regulatory requirement that the licensee provides STUK with the assessment of safety significance of the candidate modification in conjunction with the related pre-inspection documentation. The assessment has to be submitted to STUK independent of the safety class that the modified systems belong to. Up to now a number of plant modifications have been performed based on the insights from level 1 PSAs, assigning highest priority to modifications with most risk impact.

In the area of operational events PSA is a standard tool (called Risk Follow up) to assess the safety significance of component failures and incidents. Accordingly systematic risk follow-up studies are being made at STUK. Since 1995 the licensee events from all units have been analysed on annual basis. Based on the insights received from the risk follow- up studies, STUK has set forth an internal risk based objective for operational events at Finnish NPPs. The objective is that the annual share of operational events (component failures, preventive maintenance, exemptions from Tech Specs, events) is equal to or less than 5 % in the predicted annual core damage probability. This objective constitutes the strategy by STUK to lessen the number and contribution of operational events at NPPs.

5.4.2 Use of PSA for analysing Technical Specifications

PSA is used to give arguments for relaxing Technical Specifications. In such a case however it is provided that the short exceedance of the allowed outage time contributes only a tiny increment to the core damage probability compared with normal operation.

Furthermore, the meaningfulness of some Allowed Outage Times given in Technical Specifications has been re-evaluated by PSA, although such failure situations have not been met in practice. Certain inconsistency with the deterministic AOT's and actual risk impact has been identified.

It has also been concluded that in certain faulted states (i.e. at loss of equipment important for decay heat removal) it is more safe to continue operation than to shutdown immediately, as required by the current Tech Specs. Accordingly a licensee has asked for an exemption of Tech Specs in certain plant configuration (with specific safety system trains inoperable) on the basis of risk studies. A respective general statement was inserted into Tech Specs to advise the licensee not to shut the plant down immediately if the heat sink systems are seriously ineffective.

Additional items have been included in the Technical Specifications for Shutdown States, based on results from shutdown mode PSA. In 1994 the STUK set forth a new requirement to keep the lower air lock of the containment closed during the maintenance of main circulation pumps because this task contributed to increase the probability of large bottom LOCA of the reactor vessel. If the large LOCA took place and the lower air lock was open, the water would escape out of the containment preventing any core cooling measures and leading to core damage within short time with open reactor vessel and open containment. The deterministic rules do not require keeping the lower air lock closed during the aforementioned maintenance but the complementary PSA based review prompted the STUK to make such a requirement.

A pilot project on RI-Tech Specs will be completed at the end of year 2001.

5.4.3 Use of PSA for preventive maintenance

Usually a majority of maintenance's are performed during annual overhaul. However STUK allows Preventive Maintenance for selected safety systems also during power operation if deterministic criteria are fulfilled (e.g. single failure criteria...). PSA is used to minimise the risk deriving from on-line PM. At the BWR plant it is allowed to take one redundancy out of service at a time. In 1989, the risk contribution of on-line PM was approx. 5 %. Later on the schedule was upgraded further and currently the risk contribution of on-line PM to the mean CDF is approximately 1 %.

An important reason for on-line PM is to reduce the duration of annual overhaul. This leads obviously to a small risk increase, while parts of the safety systems are unavailable. On the other hand however the shutdown risks are supposed to be reduced and the reliability of respective components maintained.

It is possible to minimise the risk deriving from on-line PM with the help of PSA. Example case of on-line preventive maintenance optimisation is from TVO BWR plant, which has four redundant subsystems A, B, C and D in parallel. The capacity of safety systems is 4x50 %. In TVO case it is allowed to take one redundancy out of service at a time. The figure 1 below shows the risk contribution of all four subsystems PM to mean core damage frequency for three different maintenance schedule alternatives.

When TVO PSA was first completed in 1989 (internal events), the risk contribution of on-line PM was approx. 1,25 % per subsystem, altogether more than 5 %. This was considered to be too much. After the first change to PM schedule the risk contribution dropped significantly as can be seen in the figure. Later

on the schedule was upgraded further and currently the risk contribution of on-line PM to the mean CDF is approximately 1 %.

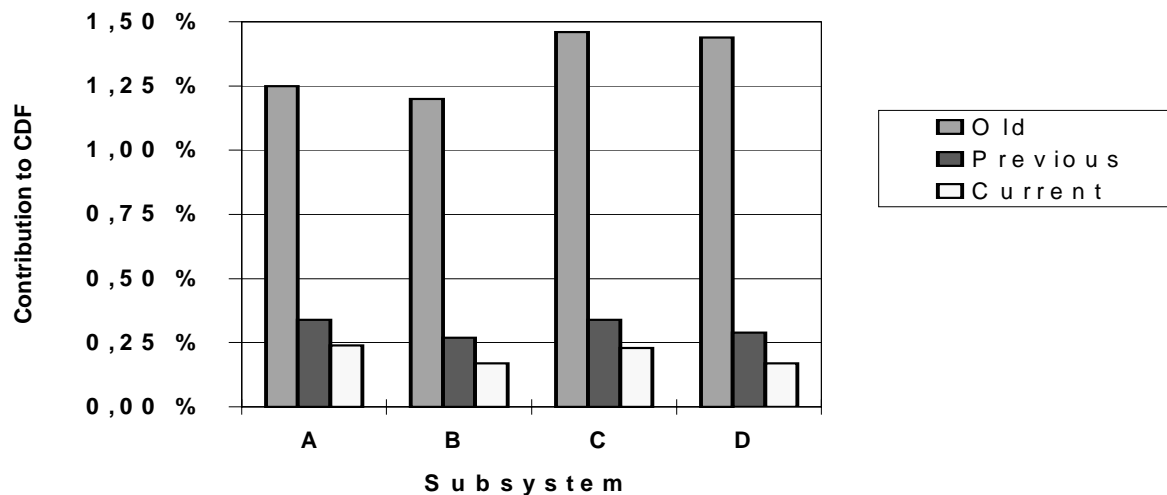


Figure 1. Preventive maintenance' contribution (%) to core damage frequency

5.4.4 Use of PSA for ISI/IST

STUK is in the process of extending the scope of risk informed regulatory activities to ISI, IST and Risk Informed Technical Specifications. A new project dealing with PSA support to regulatory audits has been initiated at STUK in 1998. The aim of the project is to develop a risk-informed method and apply them to consolidate specific regulatory tasks such as ISI, IST and Technical Specifications. Use of PSA by STUK has up to now been rather limited for regulating and controlling in-service testing and inspections (ISI/IST).

5.4.4.1 Pilot studies on IST

A pilot study to optimise the MOVs changing program has been performed for Loviisa NPP. Originally, some MOVs were equipped with over-dimensioned actuators, which may damage the valve and result in an external leakage provided that the limit protection function of the valve was unavailable. Because the need for changes for the major portion of the valves was not evident, the risk assessment study was necessary for ranking the valves, which may cause the highest risk contributions. The leading aspect in the study turned out to be the small LOCA induced by the damages in valves, located at the pressure retaining parts of the reactor coolant system, due to over-dimensioned actuators.

The study indicated that the modification of 64 valves from 500 candidates reduced the LOCA contribution by one order of magnitude, which is an insignificant contribution to the total core melt probability. Further the study showed that the importance of MOVs in terms of risk significance deviates in a large range. Accordingly the majority of components were left untouched while a minority of MOVs were modified.

Some testing procedures for diesel generators have been modified at Olkiluoto NPP unit 1 and 2 in order to reduce negative impact of tests to the equipment's ageing.

5.4.4.2 Pilot study on ISI

The pilot applications on ISI of piping both in PWR (Loviisa) and BWR (Olkiluoto) plants have been completed. The pilot study contains the high-pressure injection system and the emergency feed water system at the PWR and the shutdown cooling system and the service water system at the BWR plant. The Finnish licensees contributed to the pilot study by providing qualified systems information data to STUK.

STUK's risk-informed procedure combines both the plant specific PSA information and the traditional insights in support of the system specific detailed ISI program selection. At the starting point all systems important to safety are exposed to the selection procedure irrespective of the ASME class (1, 2, 3 or even non-code piping).

The procedure includes several steps such as selection of systems and identification of the evaluation boundaries and functions, evaluation of the qualitative degradation mechanisms of piping, evaluation of consequences and division of the segments into different categories.

Division of pipe segments into various degradation categories is to be based mainly on qualitative identification of the mechanism to which the pipe segment is exposed (such as erosion corrosion, vibration fatigue, water hammer, thermal fatigue, stress corrosion cracking and others). Recently few probabilistic fracture mechanics methods to estimate the potential pipe break probabilities have been presented. An alternative method is to use expert opinion and pipe failure experience to determine the degradation category of each pipe segment.

The division of pipe segments into various consequence categories is based on conditional core damage probability estimated by PSA applications. The pipe segments are divided into different categories containing high, medium and low risk segments, respectively. Finally the expert panel combines the traditional and probabilistic information. The experts in the panel represent extensive areas of operational and safety disciplines such as plant design, operation, maintenance, structural and material engineering, probabilistic risk assessment and in-service inspection. The panel itself can be seen both as quality assurance or critical review of the preliminary results and as a support for the decision making for the final categorisation of the pipe segments.

The pilot study on ISI of piping produced essential experience for further RI-ISI applications [18]. Furthermore, the study also gave guidance to further development of the chosen method. The study produced also a new testing strategy for the chosen systems. Based on these results and the overall experiences the general suitability of the method and the PSA application guidelines will be evaluated. It is anticipated that the gain of the pilot application is improvement of safety, more effective use of regulatory resources, and if the optimisation is well accomplished, reduction of unnecessary burden and cost of the licensee. The RI-ISI approach is being included in the respective Regulatory Guide YVL 3.8.

5.5 France

5.5.1 Existing plants

5.5.1.1 Reduction of the risk related to dominant contributions

5.5.1.1.1 Loss of redundant systems

Following the 1977 letter (see above), partial probabilistic studies were carried out by EdF for investigating the probabilities and consequences of the loss of redundant safety systems.

These studies showed the need of complementary measures to achieve a satisfactory safety level. Specific procedures, called "H" procedures (H for "hors dimensionnement", i.e. beyond design basis), if necessary implementing supplementary equipment, have been established .

5.5.1.1.2 Shutdown situations for the 900 MWe and 1300 MWe PWRs

An interesting finding of the French PSAs is the significant contribution of shutdown operating conditions to core melt frequency (CMF) - 32 % of the total CMF for PSA 900 and 56 % for PSA 1300.

The CMF was particularly high for a loss of Residual Heat Removal System (RHRS) during mid-loop operation because only a short time is available for the operator to take any action due to the low primary coolant inventory (especially at the beginning of the outage).

Other particular sequences initiated by a spurious primary coolant boron dilution were also identified.

Although the PSAs completion was not a regulatory requirement, following the results presentation, the Safety Authority required from EdF to propose plant modifications in order to reduce the frequency of these sequences.

The Safety Authority (considering in particular that during shutdown, containment integrity is not guaranteed) for the CMF related to shutdown conditions set a probabilistic target of 10^{-6} /reactor x year.

These sequences were similar for 900 MWe, 1300 MWe and 1450 MWe series and led to modifications for all series.

Immediately following the publication of the results, EdF proposed preliminary measures (level measurement, technical specifications (TS) leading to avoid the most critical situations, training of operators...)

After a more complete safety reassessment, definitive measures were proposed.

The Safety Authorities have considered that the assessed CMF was significantly reduced by these measures which should be rapidly implemented on all the plants.

The proposals of EdF, and especially the corresponding PSA, were analysed by IRSN, especially on the basis of his own studies.

During this analysis, some new significant sequences leading to cold overpressurisation were identified. The most important sequence corresponds to an inappropriate isolation of the RHRS during cold shutdown when the primary circuit is closed : an inappropriate isolation of the circuit could lead to a rapid pressure increase with a low temperature in the primary circuit, and to a risk of reactor vessel rupture. This safety problem was due in particular to weaknesses in the Emergency Operating Procedures.

The Safety Authority required to improve the situation and to analyse in more detail the risk of overpressurisation. Following this analysis, the utility proposals are improvements of the EOPs for all the plants and, for the 900 MWe series, a modification of the PORVs setpoint during relevant plant states, in order to reduce significantly the risk of vessel rupture.

5.5.1.2 Periodic Safety Review

In the framework of the 900 MWe series Periodic Safety Review, each sequence of the PSA 900 with a CMF $> 10^{-7}$ per year was analysed, in order to investigate the interest and the feasibility of plant improvements. Particular attention was paid to, sequences resulting in early containment failure. Several plants improvements were defined by this approach.

For the 1300 MWe series, a more complete methodology is proposed by EdF and is under discussion.

5.5.1.3 Probabilistic analysis of Operating Events

The probabilistic analysis of operating events which occur in the plants is on going, by EdF for all the events, and by IRSN for some representative examples.

An operating event is considered as a Precursor when the conditional CMF due to this event is higher than 10^{-6} . Moreover the Safety Authority has required from EdF, for the most important events (conditional CMF higher than 10^{-4}), to define in the short term corrective measures and to assess the corresponding risk reduction.

EdF has been performing a systematic PSA-based precursor event analysis program since 1993. This analysis consists firstly in using deterministic methods in order to select main events to be analysed (54 outstanding events have been selected in 1998). Secondly, the outstanding events are analysed using PSA models in order to imagine and assess degradation scenarios. It led to 22 Precursor events in 1998. A special application has been led for a deep analysis of BLAYAIS incident at the end of 1999 (flooding of installation and loss of some outside power both caused by an exceptional storm).

Some other recent examples of interesting precursor events are the following:

- Degradation of pipe components on recirculation sections of containment spray system (Fessenheim - 11/12/98).
- Result : $4,2 \cdot 10^{-3}$
- Loss of train A essential service water pumps and train A charging pump (Nogent - 24/01/99).
- Result : $2,2 \cdot 10^{-4}$.
- Faulty connection of RHRS pumps (Chinon -13/07/98)
- Result : $1,1 \cdot 10^{-5}$

Moreover PSA is often used in every day safety analysis as a complement to deterministic analysis for decision making, for example in case of technical specifications waiver authorisations.

5.5.1.4 Other PSA applications for existing plants

Reliability data

EdF has set up a specific organisation on site and at corporate level in order to update PSA data at regular intervals. The aim is to support not only living-PSA programs but also to support maintenance and safety management activities. The reliability data of the most critical components (according to PSA results) are updated in priority.

Reliability Centred Maintenance

The Reliability Centred Maintenance Program (in France, this program called “OMF” is using a PSA-based selection of critical components) is completed for active systems and components. A new program aims to analyse operating experience to check the efficiency of RCM-based maintenance programs. Application to passive components (mainly pipes) is in progress.

Emergency operating procedures improvements

For the completion of the PSAs, an important effort was devoted to human factors analysis, especially by means of simulator experiments. This analysis and the high contribution of human factors in the overall results led to several modifications and improvements of operating and emergency procedures. PSA results are also used for operator training.

Technical Specifications

A review of technical specifications for the main electric sources is on progress. Its aim is to review the whole requirements, shutdown states and allowed outage times for step-down transformers, auxiliary transformers and diesel generators.

5.5.2 Future plant EPR (European Pressurised water Reactor)

For the French-German future plant EPR (European Pressurised water Reactor), the French and German Safety Authorities agreed on a wide use of PSA as a complement to the Basic Deterministic Design during all stages of the design.

A preliminary EPR PSA (level1, internal events) has been performed by the designers and included in the Basic Design Report. This preliminary PSA has been analysed by IRSN and GRS for the French and German Safety Authorities.

The PSA indicated the significant contribution of CCF (due to the high redundancy of the EPR systems) and of support systems, and led to several design improvements.

The general conclusions at the end of the Basic Design are that :

- the PSA is a positive approach
- it does not indicate particular weak points

- it has to be completed during the detailed design with a particular attention to the modelling of CCF, support systems, I&C and Human Factors.

Moreover the analysis of external hazards and the level 2 have to be undertaken.

Besides, a comparison between the EPR PSA and the existing French plants PSAs was conducted by IRSN. Although the comparison of absolute values is not realistic, the PSA comparison indicates several safety strengths of the EPR design.

5.6 Germany

PSAs for German NPPs in operation are performed in the context of the “Periodic Safety Review” or as R&D projects. Although mandatory quantitative safety criteria have not been formally fixed, PSA results are used to decide on backfitting measures concerning system design and plant operation (e.g. modifications of the operators manual, optimisation of test intervals). For this purpose the results of former PSAs for German and foreign plants are used as “yardsticks”. Modifications have been performed by the utilities by a (more or less) voluntary basis.

Up to now there are no formalised approaches for the application of PSAs by the utilities for optimisation of plant operation. It depends largely on the engagement of the respective plant managers how far insights from PSAs are applied for decision on plant operation.

Results of Precursor Analyses are used, together with results of the deterministic evaluation, for the assessment of reportable events and for the formulation of recommendations concerning safety optimisation.

The PSA application for the European Pressurised Reactor is described in the French contribution (section 5.5.2).

5.7 Hungary

In addition to quantitative evaluation of plant safety surveyed in Appendix A, the PSA models and results for Paks NPP have been used in a number of PSA applications ever since the completion of the first level 1 PSA study for unit 3. Both utility and regulatory activities have been supported by these applications. Up to date the most important PSA applications have been as follows:

- development of recommendations for safety improvement
- Prioritisation of measures for safety improvement included in the safety upgrading program for Paks (Most modifications have been scheduled in accordance with priority.)
- use of PSA during design and implementation of plant modifications
- case study demonstration of PSA based revision of technical specifications
- PSA based review of operator training at Paks simulator
- development of unit specific risk monitors and tools for analysis of precursor events to severe accidents

- development of a special risk monitor to be used for risk prediction by the nuclear regulatory authority in an emergency.

As an example of the listed applications, the precursor event analysis using probabilistic methods was initiated by the Nuclear Safety Directorate of the HAEA in 1995. The objectives of the precursor event analysis program are as follows:

- determination of the risk significance of the operational events on different levels of risk (e.g. core damage, system/component unavailability, etc.), identification of the most significant ones and their ranking,
- early signalisation of negative trends in performance,
- drawing conclusions based on the impact of the operational events,
- feedback to the PSA model and data used.

A computerised tool has been developed and used for the precursor event analysis. The Licensee Event Reports are evaluated quarterly and the summarised results are used as risk based indicators of operational safety at the Paks NPP.

Since the start-up our units undergo a continuous upgrade process. Another example of the applications is the systematic PSA based change analysis that supports this upgrade process. According to the regulatory approach, it should be proved that each modification preserves or increases the safety level. In order to gain the most complete insights not only deterministic principles but probabilistic evaluations are also systematically undertaken for any significant plant changes or any significant considerations of additional initiators or any significant considerations of other plant operational modes. In the justification of the plant modifications is a tendency to show that the calculated overall risk impact (in terms of core damage probability change) is negative or at least negligible. In many cases designs of the plant modifications have been optimised based on calculated risk characteristics.

The overall risk figure for internal events has been decreased by an order of magnitude during the last five years. Safety improvement has been achieved during full power operation and during low power and shutdown conditions as well. The PSA has quantitatively shown that this considerable risk reduction can be attributed to the safety enhancement measures that have been implemented at Paks up to now.

The other important application of PSA is supporting Periodical Safety Review required by our nuclear authority. These periodical reviews held after 10 years of operation offer the possibility – and obligation for the licensee – to perform a comprehensive assessment of the safety of the plant, to evaluate the integral effects of changes of circumstances happened during the review period. The goal of these reviews is to deal with cumulative effects of NPP ageing, modifications, operating experience and technical developments aimed at ensuring a high level of safety throughout plant service life.

5.8 Italy

In addition to the items included in PSA Programmes (see chapt.3), there are other activities in Italy that represent applications of the probabilistic treatment of risk in nuclear activities and installations different from nuclear reactors such as the reliability study of accelerators for ADS purposes (ref.7).

As far as nuclear fission safety is concerned, a joint effort sponsored by ENEA in collaboration with University of Pisa and Polytechnic of Milan, aimed at the reliability assessment of passive systems (1999-

2000) has been conducted: this topic is the content of a project financially supported by the European Community in the frame of the Fifth Framework Programme (partners are among others CEA and GRS). Related to research reactor, a failure analysis based on a fault tree approach was conducted in ANPA on the scram circuit of the safety system taking into account common cause failures that could occur among components.

5.9 Japan

5.9.1 Accident Management Strategies based on PSA

As described in 3.2, AM strategies have been implemented based on PSA. Level 1 and 2- PSAs have been conducted for individual NPPs to determine AM measures. Level 1 and 2- PSAs for plant conditions after AM implemented are being made to confirm the effect of AMs on CDF and CFF for eight kinds of typical NPPs. Effect of AMs in individual NPPs which introduce plant specific AMs different from the standard AMs will be confirmed in PSR by PSAs. NUPEC is making level 1 and 2- PSAs for the above eight kinds of NPPs to support METI AM review.

5.9.2 PSA in PSR

Since 1997 PSA for full power operation has been made in PSR and since 2000 for low power and shutdown operation, respectively. NUPEC has also made shutdown PSA to support PSR review of METI.

5.9.3 Evaluation of Allowed Outage Time (AOT) based on PSA

Specific PSA models have been developed for typical NPPs, which can evaluate the increase of CDF during the outage of systems taking into account surveillance test during outage in detail. AOTs have been estimated comparing ICCDP and ICLERP during outage with the safety criteria.

General view of risk increment during outage in various types of NPPs derived from risk achievement worth (RAW) in existing PSAs has been referred to establish the safety criteria for setting AOT.

Guideline for setting AOT through PSAs has been published.

5.9.4 Application of PSA to Inspection

Japan Power Engineering and Inspection Corporation (JAPEIC) has been looking into the feasibility of applying probabilistic inspection administration method to Japanese NPPs on the study of optimisation of periodical inspection ordered by MITI since 1995. The present conclusions are:

- a) quantitative calculation of risk significance is possible for systems and components in a plant using the results of PSA. It is also possible to evaluate the appropriateness of component selection for periodical inspection on the risk insight
- b) By using RI-ISI(Risk Informed In-service Inspection)/RI-IST(Risk Informed In-service Test) method, both the subjects and the frequencies of ISI/IST can be reduced without affecting the safety level of the plant.

JAPEIC is now studying the overall schedule until the realisation of RI-ISI/RI-IST, the methods suitable to the conditions in Japan and the guidelines for adopting those methods. In the guidelines the safety criteria and component failure data used in RI-ISI/RI-IST in Japan are also pursued.

5.10 Korea

PSA experience in Korea has grown considerably during the past 10 years. In Korea, most PSA are performed for NPPs under the design stage. Usually, KINS asked the utility to ensure the safety of the new plant by using the PSA technology. So, up to now, major PSA applications in Korea are the design changes of plants based on the PSA results. The details of such design changes will be explained in "Chapter 6 PSA Based Plant Modification."

In addition of research work, a preliminary program for establishing a framework of risk-informed regulatory implementation has been formed last year to enhance both safety and regulatory efficiency and effectiveness. Following activities for a preliminary program are ongoing by the end of 2001; (i) the investigation of domestic and world-wide trend, (ii) identification of key issues to be resolved for actual application, (iii) assessment of impact to current regulation, and (iv) development of regulatory guide. Currently, most of efforts are concentrated on the development of guides and methodologies for RI application. Trial application has been done in the area of regulatory inspection, which showed the effectiveness of the use of risk information in finding deficiencies of plant procedures and facilities.

As for actual applications, the risk insights have been used for the optimisation of technical specifications requirements, i.e. allowed outage time (AOT) and surveillance test interval (STI). For example, the utility has initiated two PSA application projects in the examples of RIR: the first one is on the relaxation of STI (Surveillance Test Interval) of some systems of Kori 1 Unit (WH 600 Mw Reactor) based on the PSA result, and the second one is on the RI-ISI (Risk-informed In-service Inspection). Additional research program is on going for the PSA applications: (1) the relaxation of STI of RPS/ESFAS of Ulchin 3&4 Units and (2) the optimisation of AOT (Allowed Outage Time) in plant Technical Specifications based on the PSA result.

As explained above, the PSA application is at the initial stage in Korea. However, all nuclear industries have a big interest in the PSA application since they expect this approach enables to enhance the safety and economic aspect of NPP simultaneously.

5.11 Mexico

As it was mentioned in previous chapters, there is not a formal PSA application program in which the PSA results and insights were systematically used to support decision making on safety related issues. However, there have been some applications in which the PSA insights played a complementary role to the deterministic analysis. These applications, submitted by the utility, have been evaluated to the full extent by the regulatory authority. The latter has also made some regulatory PSA applications using the agency's Internal Event Analysis.

A PSA application was submitted by the utility following the USNRC regulatory guide 1.174 to complement the deterministic analysis presented to support a plant modification request that involves the increase of the actual power in 5%. The calculated increase in core damage frequency was 2.87×10^{-6} per reactor year. This increase is in the range of 10^{-6} per reactor year to 10^{-5} per reactor year. The regulatory guide establishes, in this case, that the application can be considered if it can be reasonable shown that the total core damage frequency, considering internal events, external events, full power, low power and shutdown, is less than 10^{-4} . The IPE for Laguna Verde currently covers only internal events for full power operation. The contribution of the out-of-scope portions of the model was allowed to be addressed by bounding analysis, since significant margin exist between the calculated change in risk metrics and the acceptance guidelines. The application also covers the large early release frequency. The increase in this frequency was very small and was considered acceptable. The Mexican regulatory authority concluded that the application complies with the regulatory guide as well as with the key principles associated. These

principles establish that the proposed change meets the current regulation, that is consistent with the defence-in-depth philosophy, that maintains sufficient safety margins, that the risk increase associated are small, and finally the impact of the proposed change should be monitored using performance measurement strategies.

The utility has also submitted other PSA applications submitted to support temporary and permanent changes to the technical specifications of Laguna Verde NPP. Regarding the temporary changes to the technical specifications, most PSA applications have been intended to support a request to extend the Allowed Outage Time (AOT) conditions for some equipment during corrective maintenance. These applications have been reviewed in detail and approved when the risk increase was considered small. In connection with the permanent changes to the technical specifications for Laguna Verde, the PSA applications were intended to support changes to the Surveillance Test Interval in order to increase such interval for some equipment which did not contribute to the core damage frequency or large early release frequency according to the PSA. Based on the review, the risk increase associated with the test interval change came out small and therefore approved.

The regulatory authority own PSA results were used to prioritise inspection tasks. The use of risk-based information for inspection purposes started in the early 1995 with the development of plant specific risk inspection guides (RIGs). These RIGs provide the risk-based ranking of systems, components and operator actions. The RIGs along with the USNRC inspections and enforcement manual, the USNRC regulatory guides and the plant specific procedures are being used to set-up what it is referred to as improved inspection practices. The inspection teams have been trained in the efficient application of these practices in the field, and the RIGs are currently being used to focus the inspection effort to those aspects important from a risk point of view.

Although there is no formal ordinance to apply the PSA to the examination of operators by the regulatory authority, in the past the results of its Internal Event Analysis (Level 1 PSA), namely the main accident sequences, have been used to test the operators ability response at the plant simulator. From the experience gained so far it seems pleasing for the regulatory authority to use the PSA as input for operator training and later on for operator performance evaluations.

5.12 Netherlands

5.12.1 First Steps Towards Risk-Informed Regulation: A Feasibility Study

Because the regulatory body increasingly is confronted with design or operational changes which stem directly from, or are supported by arguments stemming from LPSA-applications at Borssele, which require approval of the KFD, the IAEA was asked to advice the KFD in order to support this process. Questions like: 'Are the LPSA-applications at the Borssele plant state-of-the-art and sufficient, or should Borssele do more?', 'How should the KFD respond to these applications, given a small regulatory staff and possible short remaining lifetime of the Borssele plant?', were the focal points of this review.

The main conclusions and recommendations were:

- Complete the implementation of the risk monitor with high priority in order for it to be used for maintenance scheduling, operating decisions and risk follow-up.
- Select those applications that can provide benefit to the plant in the near term. This selection could be based on criteria such as dose reduction, regulatory requirements, maintenance costs, refuelling outage duration, etc. Examples of such applications are risk-informed

improvement of technical specifications, risk-informed increment of on-line maintenance activities.

- KFD was suggested to develop a framework for the use of risk information in regulatory decisions. This should include the identification of objectives, description of the decision-making process and acceptance criteria, and clarification of how risk-informed decision-making is to be incorporated in the existing regulations. Since developing such a framework may take considerable effort, they were suggested to review existing risk-informed frameworks, bearing in mind that acceptance criteria need to be developed for the specific situation in The Netherlands.
- The resources required for accomplishing risk-informed regulation depend on how much use will be made of this approach, however, the IAEA team suggested that, as a minimum, KFD should continue to allocate one person, having in-depth knowledge of the Borssele PSA, for PSA-related activities, and that all decision-makers should have some training in PSA.
- The IAEA team felt that if applications are requested by the KFD to Borssele NPP, these should be discussed with the plant to maximise mutual benefit. Also, the discussions raised the idea that perhaps the KFD and Borssele NPP could develop a consensus document to conduct and assess PSA applications.
- Finally, the KFD was suggested to use PSA to focus the regulatory inspection program on the more significant systems, components, and plant practices.

As a follow-up of this advice, which took place in 1999, the KFD cautiously defined a follow-up program/feasibility study in order to proceed towards a more risk-informed regulation. It was decided to take a step-by-step approach. The first step is to familiarise with risk-informed regulatory approaches in West-European countries, whilst the next steps are centred on a particular application, such as Technical Specification optimisation.

5.12.2 Follow-up program

The objective of this program is to come to a situation in which regulatory attention is more consistent with the risk importance of the equipment, events, and procedures to which the requirements apply, so that regulatory and licensee resources can be used in a more efficient way when making decisions with respect to ensuring the health and safety of the public. This objective implies that the regulatory requirements be commensurate with the risk contributions (i.e., regulations should be more stringent for risk important contributors, and less stringent for risk unimportant contributors). Therefore, provided risk informed regulatory criteria are appropriately developed, a systematic and efficient expenditure of resources are to be expected, while, simultaneously, a balance in overall plant safety can be achieved.

Examples of typical regulatory actions where risk-informed methods and requirements are thought to be helpful and therefore being investigated in the project, include:

- evaluation of the design and procedural adequacy;
- performance of periodic safety reviews;
- assessment of changes to the licensing basis, e.g. Technical Specification optimisation: surveillance test intervals, allowed outage times, limiting conditions of operation;

- assessment of operational practices or strategies on safety such as: plant systems configuration management, preventive and corrective maintenance prioritisation;
- prioritisation of regulatory inspection activities;
- evaluation of inspection findings;
- investigation of ageing effects;
- assessment of risk-based safety indicators;
- the need for regulatory action in response to an event at a plant;
- one-time exemptions from Technical Specifications and other licensing requirements; and
- assessment of utility proposals for modifications of the design or operational practices.

The development of risk-informed regulation in The Netherlands is bounded by the present limited nuclear power programme: one NPP (Borssele) in operation, and shutdown of this NPP eventually foreseen by the end of 2003 (although shutdown at a later time cannot be excluded due to legal procedures and processes). No new reactors are planned as well.

Currently the focus of future activities/events for Borssele is governed by licence requirements or external circumstances. It concerns initiation/continuation of:

- new 10-year periodic safety review, formally started in 2001;
- two-year operational safety review;
- monitoring of the plant safety culture during the expected plant staff reduction;
- deregulation of the electricity market;
- preparation for decommissioning.

Under these boundary conditions, emphasis of the development of risk-informed regulation will be in the operational and not in the design area. Also QA is assumed to focus on operational items, in this respect. The design area, however, cannot be ignored, as the plant configuration determines much of the plant safety characteristics.

As the application domain is limited, as is the available manpower within the KFD, the development of Risk-informed Regulation should be based on existing approaches elsewhere; no separate 'Dutch' RiR development is to be foreseen. Main vehicle could be the USNRC development, plus useful parts of the approaches in Spain, Switzerland, Sweden, Finland, Belgium and the UK. Where the sources are diverse, special care must be exercised to obtain a coherent and consistent product.

'Deregulation' is meant as a support to the utility to be and remain competitive on the electricity market. In practice, it means that active support will be given to activities aimed to decrease costs, as long as they do not compromise safety.

The main objectives of the RiR are therefore:

- support the above mentioned (bulleted) activities;
- focus KFD and plant resources on items relevant for risk; and
- eliminate unnecessary ‘regulatory burden’.

As the available time frame is short, it is *not* the intention of the proposed RiR-project to generate formal revisions of the NVR-series Design, Operation and Quality Assurance, as would ultimately be the approach in a nuclear energy scenario of longer duration than just about 3 years. However, RiR-products will be documented and reviewed with industry.

Overall, the RiR products will be application-oriented. In some areas, fundamental aspects may be touched, where no written guidance can yet be formulated. In those cases, a conclusion must be reached how to proceed on a more ad-hoc basis.

A special aspect of this project is feasibility if the current oversight process can be transformed into a more risk-informed oversight process. This includes, the eventual use of safety significant performance indicators.

5.13 Spain

As it was mentioned before and described in previous yearly reports, the IP Edition 2 main objective is the promotion of PSA applications. Several points of the Program are devoted to it and described in the following.

There is a programmatic objective to shape the experience that have been acquired during the performance and application of the PSA in rules and/or guides. The guides for PSA performance have been shaped themselves, in fact, from PSA project to PSA project for each PSA task. This can be considered as a real standardisation process and, therefore, the latest PSA task procedures could be easily elevated, if decided, to the category of "standard".

In the field of the PSA applications, it was agreed in Spain that a consensus would be needed on specific methodologies for carrying out diverse types of applications which may become general. Examples are the PSA applications to technical specifications, in-service inspection or graded quality assurance. In that process of consensus, it would be possible to get experience with applications by means of using guides developed in the USA, as much on the part of the industry as on the part of the regulator of that country.

Advance has been done in Spain along this line, making use of a CSN and utilities joint working group on PSA, where these aspects are being discussed and pilot and R&D activities launched. On the other hand, it is also likely that a massive use of the concept of quantitative risk force the emission of regulatory quantitative targets for PSA applications. In this sense, PSA applications acceptance criteria are being explored as part of the PSA applications methodology development and testing. Absolute quantitative risk goals are not foreseen by this current edition of the Spanish PSA Integrated Programme.

Other IP point makes reference to a somewhat new aspect, but extremely necessary, since PSA can be systematically applied to the many fields where these techniques can contribute with their vision, simultaneously global and detailed, on the factors that affect safety. For it, training plans on PSA and dissemination of the knowledge and results of each PSA are needed as much within the CSN as within each NPP organisation. As for the CSN concerns, about one hundred persons of its staff have already taken

a basic course on PSA techniques and new courses and activities are also being carried out to promote the use of PSA inside the CSN, for instance, through the development of a CSN internal computerised PSA information system and other internal applications to the daily CSN work, like inspection and general planning. A new PSA methodology and applications course has been given, in this case directed specifically to the CSN executives.

A very significant point of the IP is related to Research and Development. This point is included in the IP part devoted to the promotion of the applications because all the aspects of PSA R&D focus, either direct or indirectly, towards the PSA applications.

Fourteen possible areas of activities of PSA R&D are described in this programmatic point, so that planning and specific projects be integrated in a "Plan on PSA R&D". This Plan is integrated in the "Five-Year Research Plan (1998-2002)" of the CSN. Main activities are discussed later on in this report.

In order to conclude the Edition 2 of the Integrated Program, a final part is included describing "Future Forecasts" on the use of PSA and the promotion of the technological progress that the same ones entail.

The first and main point of this IP part introduces the concept of "Living PSA" and the activities of permanent updating and constant applications that can be visualised in the future to fit and use the PSA systematically. As can be seen in the status of plant specific PSA, some plants are already more or less following this concept, keeping resources to maintain and update their PSA and to use them continuously to help decisions about safety and operation planning and analysis. In three plants (Cofrentes, Garona and Almaraz), risk monitors are already available and integrated in daily operations. Plans on that direction are being done by other two plants (Asco and Vandellos) and being considered by the other two.

CSN involvement in this kind of "living PSA process" is being discussed and activities to implement the models of the seven Spanish PSA in the CSN computers, for their use in CSN internal PSA applications and for the CSN review of applications proposed by the utilities, are planned to be completed in 2002.

Activities on this regard have been also launched in the above mentioned joint working group on PSA, where a general basic document on PSA maintenance and updating has been issued to be used as a basic reference for the future systematic maintenance and updating of PSA models and data by the utilities and for their systematic communication to the CSN for its own PSA maintenance and updating processes.

Finally, the last point of the IP is related to continue fomenting the technological development that the assimilation of the PSA technology has meant in the country and the extension of the use of these techniques to other fields where risk is induced by ionising radiations.

5.14 Sweden

The PSA program in Sweden has to follow the regulations in SKIFS 1998:1.

This regulation set the quality demands to be followed and full-filled. It says e.g., that all operating modes have to be analysed. Responsibility for researching and development lay at the licensees. See next section.

5.14.1 *General Safety Regulations*

From 1 July 1999 new general safety regulations are in force in Sweden. The regulations apply to all nuclear facilities with a Government permit to conduct nuclear activities. The new regulations replace a large number of individual licence conditions issued over a number of years.

Various usage of the PSA technique in Sweden in matters concerning:

- Verification of deterministic analyses in SARs: e.g., in level-1/2 studies, shut-down, fire, flooding studies, external hazard analyses.
- Identification of safety significant scenarios (e.g., functions, systems, components, human errors).
- Technical Specifications (AO, maintenance, testing, instructions).
- Impact and planning of plant modifications.
- Living PSA.
- Trend analyses.

5.14.2 *Applications of interest*

- Risk informed decision making
- Standard Technical Specification according to NUREG-1413 principles.
- Specific regulatory body activities, involving usage of PSA.
- Establishing of inspection, reviewing practices according to the results of the PSA-results and PSA-activities in Sweden.
- SKI site specific and annual safety assessment. PSA-results and PSA-activities are input to this SKI internal process.
- Trend analyses of impact of occurred event on safety barriers and on work & activities belonging to the defence-in-depth principles.
- Probabilistic & deterministic impact of introduced new technique (e.g., digital technique).

5.14.3 International projects, with coupling to PSA issues

Participation in e.g., OECD/NEA projects (pipe brake, fire, CCF)

5.14.4 Reliability data

The reliability data to be used for safety related components in Swedish PSA studies, are presented in the T-Book 5th edition (see chapter 4)

5.14.5 PSA Program

See chapter 4, for details.

The PSA studies and the results from them are a natural input for SKI to optimise, decide what kind of impact, observed weaknesses at the plants can give to our inspection plans.

The inspection philosophy of SKI and according to the regulations in SKIFS 198:1 is to concentrate on PSA activities at the licensees rather than on deep reviews of matters singled out in different analysis.

By PSA activities we mean, e.g.;

- Instructions to be used in probabilistic work
- Personal, resources
- Management
- Quality assurance of PSA products
- Operating experience of PSA matters

5.14.6 Periodic Safety Review

During the 80'ies and 90'ies the updating program for the domestic PSA studies was very closely linked to the ASAR program. Nowadays, and after that the SKI regulation 1998:1 went into force, PSA studies are separated from the ASAR program. In practice that means that PSAs have to be modified when necessary e.g., due to plant modifications.

5.14.7 Reliability Centred Maintenance

The RCM technique is used and practised at the Swedish NPPs.

5.14.8 Technical Specifications

The domestic licensees are requested to use PSA and to measure impact of changes in Technical Specifications, plant modifications. Risk informed approaches are used, when changes of AOT's are discussed.

5.14.9 Living PSA

The domestic licensees are in varying ways practising the LPSA applications in their follow-up of their safety work e.g., at evaluating operating experience, measuring impact of changes in Tech. Specs, plant modifications.

5.14.10 Safety monitor

In one of the Oskarshamn plants, safety monitor is planned to installed in the near future.

5.15 Switzerland

5.15.1 Design and Operational Evaluation

To date, the main application of PSA has concerned the re-evaluation of various Swiss plants, in terms of identification of potential plant specific severe accident vulnerabilities, and to assess the upgraded status of the older Swiss plants. This has been the main focus of our PSA evaluation activities, within the framework of plant specific licensing actions and/or the periodic safety review, as a complementary tool to the deterministic safety analysis. The benefit of using PSA in this framework is illustrated in the following by means of some examples originating from the PSA review process:

- Many seismic backfits were performed as a result of the on-site inspections carried out within the PSA (review) process. In early 2001 HSK has required one plant to backfit a significant number of masonry walls of an electrical building, after an inspection had revealed that the risk-significant walls were not included in the PSA model.
- Due to shortcomings in the probabilistic seismic hazard analysis (PSHA), HSK has required the licensees in mid 1999 to perform a new comprehensive PSHA for all Swiss NPP sites. In the new PSHA, the use of experts and the assessment of epistemic and aleatory uncertainties will receive high priorities.
- The human reliability analysis (HRA) and the corresponding detailed review of the HRA initiated the improvement or the development of new procedures.

The PSA evaluation process continues to provide a valuable method to assess the balance in design, in terms of accident prevention and accident mitigation, and to identify important contributions to the core damage frequency and containment failure likelihood, and the PSAs have constituted useful tools to evaluate the effectiveness of the various plant modifications (especially, the older plants).

5.15.2 Accident Management

The plant specific level 2 studies are used as a part of the technical basis of the development of Severe Accident Management Guidance in order to provide information on the possible accident progressions and the possible plant states.

5.15.3 Evaluation of Technical Specifications

The plant-specific PSAs are increasingly used by the Swiss utilities in order to evaluate the risk-significance of Technical Specification changes proposed. Vice versa, risk-beneficial changes of Technical Specifications have been suggested and implemented based on PSA insight. HSK makes use of its own

PSA models in dealing with specific utility applications in this regard. This is an area where increased activity on the parts of both the utilities and HSK is anticipated over the next years.

5.15.4 PSA-Based Event Analysis

During 2001, HSK started a pilot project to assess the risk impact of various operational events at the Swiss nuclear power plants. This study has been very insightful, and it is expected to be continued in the coming years.

5.16 United Kingdom

5.16.1 Risk Monitors

Risk Monitors have been in operation at Heysham 2 and Torness since 1986. These were based on the Level 1 PSAs that were produced as part of the Pre-Operational Safety Case. Although these reactors were built to the same design, they were originally operated by different utilities and this resulted in different approaches being adopted for the two Risk Monitors.

The Essential Systems Status Monitor (ESSM) at Heysham 2 carries out a comparison with the deterministic requirements covering safety system availability and a probabilistic analysis which is done using a fault tree model of the Level 1 PSA that is solved for each plant configuration input into the Risk Monitor.

At Torness, compliance with the deterministic operating rules is assessed using the Essential Systems Outage Program (ESOP) and a separate probabilistic assessment is carried out using LINKITT which contains the pre-solved cut-sets from the Level 1 PSA model. As plant becomes unavailable, the basic events corresponding to those plant items are removed from the cut-sets. This revised list of cut-sets is minimalised and the overall risk is quantified. This produces an approximation to the exact solution but, even with a large amount of plant unavailable, it has been shown that the LINKITT results are acceptable.

The Risk Monitors at Heysham 2 and Torness are being updated to reflect the Level 2 PSAs that were produced in 1999 as part of the Periodic Safety Review and subsequently developed into four-quadrant models which correctly represents initiating events arising in each of the four quadrants (as required for a Risk Monitor application). This four-quadrant model is being maintained as a Living PSA.

It is now intended to introduce a common approach to the Risk Monitors at the two stations. The proposed new software, ESOP2000, will draw on the ESOP/ LINKITT approach already in use at Torness but will calculate risk using a full re-quantification of the four-quadrant PSA model. The pre-solved cut-set manipulation technique used with the present Level 1 PSA model has been found to be unsuitable when used with the new, more complex Level 2 four-quadrant model. Improvements in computer processing speeds allow re-quantification to be performed in a sufficiently short time to permit its use as an "on-line" Risk Monitor. This approach preserves a key requirement for the Risk Monitors, namely that updating of the software to reflect the Living PSA is both easy and quick.

5.16.2 Risk Informed Technical Specifications

The traditional approach in the UK is to define Operating Rules and a Maintenance, Inspection and Test Schedule which are derived from the safety case to define the operational envelope of the plant. Over time, these have become increasingly sophisticated and have made use of PSA insights, including the incorporation of Risk Monitors at Heysham 2 and Torness.

For Sizewell B, it was decided that this traditional approach would be replaced by developing Technical Specifications which was in line with normal practice for light water reactors. The Tech Specs which were produced were based on the Methodically Engineered, Restructured and Improved Tech Specs (MERITS). However, these needed to be developed to take account of the changes to the SNUPPS design that had been made for Sizewell B (see chapter 7).

The PSA was used to support the development of the Tech Specs. In particular, it was used to conform Allowed Outage Times (OATs) and Limiting Conditions of Operation (LCOs), and to justify the inclusion or omission of systems from the Tech Specs. The basic approach was to use the PSA model to determine acceptable completion times based on limiting both the average risk and the point-in-time risk. This provided the basic input into the Action Completion Times (ACT). For systems where the ACT was greater than 3 months for system restoration for either single train or all train failures, this provided the case for either the relaxation of the requirement to have all trains operable, in the former case, or for the omission of the system from the Tech Specs, in the latter case. In addition, the PSA was used to justify surveillance frequencies.

5.16.3 Use of PSA information for emergency planning

One of the licensees in the UK has used the results of a Level 2 PSA as a basis for the site emergency plan. Estimates have been made of the dose to a member of the public for the set of fault groups identified and these have been weighted by the fault group frequency to determine the net offsite consequences. This has been used to determine the area within which information on the potential radiation emergency from the site needs to be supplied to the public and the area within which replanning is required for countermeasures such as sheltering, the issue of potassium iodate tablets and evacuation.

5.16.4 PSA applications for nuclear chemical plants

The PSA is part of the safety case for nuclear chemical plant and has, along with other elements of the safety case, an input into operating rules and procedures, operator training and emergency planning. In modern plants PSA had an input to the design process. Risk Monitors are not a consideration for nuclear chemical plant.

5.17 United States

5.17.1 Reactor Oversight Program (ROP)

The NRC's operating reactor oversight process (ROP) provides a means to collect information about licensee performance, assess the information for its safety significance, and provide for appropriate licensee and NRC response. Because there are many aspects of facility operation and maintenance, the NRC inspects utility programs and processes on a risk-informed sampling basis to obtain representative information.

As part of ongoing work to improve the ROP, NRC's research program provides technical support in the following areas:

- Resolution of the technical issues associated with the implementation of the new unreliability PIs and enhanced unavailability PIs, which will be tested in a pilot program in July 2002.
- Development of performance indicators for the containment portion of the barrier integrity cornerstone in the ROP
- Technology transfer of the shutdown risk based performance indicators (RBPIs) to the Significance Determination Process (SDP) for shutdown.
- Development of PIs at higher levels, such as at the system level and function level.

For more information about the NRC's ROP visit the following website:
<http://www.nrc.gov/reactors/operating/oversight.html>.

5.17.2 Using PSA results to risk-inform the treatment of structures, systems, and components (SSC)

The Commission decided in 1998 to consider promulgating new regulations that would provide an alternative risk-informed approach for special treatment requirements in the current regulations for power reactors. Special treatment may be defined as current requirements imposed on structures, systems, and components that go beyond industry-established requirements for equipment classified as "commercial grade" that provide additional confidence that the equipment is capable of meeting its functional requirements under design basis conditions. These special treatment requirements include additional design considerations, qualification, change control, documentation, reporting, maintenance, testing, surveillance, and quality assurance requirements. In March 2000, the Commission invited comments, advice, and recommendations from interested parties on the contemplated approach for this rulemaking. Since September 2000, the staff has been working with industry and other interested stakeholders to resolve issues associated with industry-developed guidance intended to implement the rule. Additionally, the staff is currently working to develop the proposed rule language, supporting regulatory information, and interacting with industry on pilot activities intended to test the implementing guidance.

The licensee for the South Texas Project (STP) Nuclear Power Plant (NPP) submitted an exemption request that would allow them to apply the concepts underlying this rulemaking (categorisation, removal of special treatment requirements) at their facility, by receiving exemptions to certain existing requirements that would prevent them from otherwise undertaking such a program. The South Texas exemption was granted on August 3, 2001, and is considered to be a "proof-of-concept" prototype for the rulemaking. The exemption permits the licensee to implement an alternative treatment process that if effectively implemented by the licensee can result in safety-related low-risk significant (LRS) and non-risk significant (NRS) SSCs being capable of performing their safety functions under design-basis conditions throughout their service life. The staff has determined that the licensee's categorisation process provides a reasonable method for determining that safety-related LRS and NRS SSCs have a small contribution to overall safety. The experience from the licensee's efforts and the staff review are being co-ordinated with the rulemaking activities and guidance development.

The staff is currently working on developing rule language; working with industry on an implementation guidance; and interacting with industry on pilot activities. Challenges include translating the STP exemption lessons learned into the rule framework; addressing the issue of needed adequacy of the supporting PSA; and ensuring the framework can accommodate all facilities (existing, new, and renewed).

5.17.3 Using PSA results and perspectives to identify possible changes to NRC's reactor safety requirements

5.17.3.1 Changing 10 CFR 50.44 [ref. #8.2.17.2]

As part of the staff's program to risk-inform the technical requirements of 10 CFR Part 50 (Option 3 from SECY-98-300 [ref. #8.2.17.3]), the staff identified 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors," as a regulation that warrants prompt revision. Based upon current risk information and research results, the staff believes that little to no risk significance or benefit is associated with some of the combustible gas control requirements of this regulation, potentially resulting in unnecessary burden. Therefore, the staff has recommended the following changes to the requirements of 10 CFR 50.44:

- Delete the hydrogen recombiner requirement for all containment types (note, the thermal recombiners installed in currently licensed US reactors to meet design basis hydrogen control requirements are different than the passive autocatalytic recombiners for hydrogen control that are slated to be installed in all French PWRs)
- For facilities where the hydrogen monitors are only necessary for accident assessment purposes, the monitors would no longer be required to be safety grade

A proposed rule is scheduled to be issued for public comment in April 2002. Also as part of this effort, the staff has established Generic Issue 189 [ref. #8.2.17.4 and 5] (GI-189) to assess the costs and benefits of possible additional hydrogen control requirements for PWR ice condenser and BWR Mark III containment designs. Analyses indicate that these containments have a high conditional containment failure probability associated with station blackout sequences during which the AC powered igniters are not available. Therefore, removing the dependence on AC power for the combustible gas control systems could be of value for risk-significant accidents. In support of the resolution of GI-189, the staff is developing more realistic hydrogen source terms (with consideration of uncertainties) and assessing the implications of seismic and fire events on the risk from hydrogen combustion in BWR Mark III and PWR ice condenser facilities. This work is scheduled to be completed in parallel with completion of the rule change described above.

5.17.3.2 Changing 10 CFR 50.46 [ref. #8.2.17.6]

As part of the staff's program to risk-inform the technical requirements of 10 CFR Part 50 (Option 3 from SECY-98-300), the staff identified 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," Appendix K to 10 CFR Part 50, "ECCS Evaluation Models," [ref. #8.2.17.7] and General Design Criteria (GDC) 35, "Emergency Core Cooling," of Appendix A to 10 CFR Part 50 [ref. #8.2.17.8], as regulations that warrant revision. Based on the results of a feasibility assessment, in the near-term the staff has recommended to the Commission voluntary changes to the technical requirements of the current 50.46⁵ and Appendix K related to acceptance criteria and evaluation model, and voluntary risk-informed changes to the reliability requirements in GDC 35. While the Commission is considering these recommendations, the staff is continuing the technical work and proceeding with preparations for rulemaking. In the longer term, the staff believes that additional changes to 50.46 may also have merit and is continuing to assess the feasibility of making additional changes, potentially including redefinition of the limiting pipe-break size to be considered in design basis LOCA analyses.

The proposed near-term changes to the current 50.46 and Appendix K include (1) replacing the current prescriptive ECCS acceptance criteria in 50.46 with a performance-based requirement, and (2) revising the requirements for the ECCS evaluation model to optionally allow the model to be based on more realistic analyses. The proposed risk-informed changes to GDC 35 would include technical requirements to ensure an ECCS reliability that is commensurate with the frequency of challenge to systems. Two options are being considered to accomplish the ECCS system reliability in place of the single failure criterion: (1) a deterministic system reliability requirement based on risk information (e.g., and ECCS design requirement that only one train of ECCS is required for LOCAs larger than a specified size), and (2) an ECCS functional reliability requirement that is commensurate with the LOCA frequency (e.g., a requirement that ECCS design must be such that the core damage frequency [CDF] associated with a specified set of LOCAs is less than an NRC-specified CDF threshold).

5.17.3.3 Revising the Pressurised Thermal Shock rule (10 CFR 50.61) [ref. #8.2.17.9]

The staff is working to develop the technical basis to improve the realism of evaluations of reactor pressure vessel (RPV) integrity to support risk-informed modifications to the regulations associated with RPV integrity. The staff is evaluating the application of advanced fracture mechanics concepts to the revision of the regulatory framework for RPV integrity to provide analysis codes and techniques for evaluating licensee submittals pertaining to RPV integrity, particularly as related to pressurised thermal shock (PTS). The staff is also conducting the research and analyses needed to develop a statistically valid generic flaw density and size distribution for reactor vessel welds and plates for use by the staff and licensees in performing probabilistic fracture evaluations of reactor pressure vessels. In addition, the staff is performing an experimental program and computer analyses to support rulemaking for PTS and guidance for reactor vessel embrittlement. The results of these efforts will be reflected in review guidance documents and in modifications to the regulations addressing issues associated with reactor pressure vessel integrity such as setting operating pressure-temperature limits and low temperature over pressure (LTOP) setpoints, and in applying the 10 CFR 50.61 PTS screening criteria. Some specific staff activities include: mechanistic and statistical assessments of plant embrittlement data; a report on effects of heat treatment and chemistry unavailability on embrittlement trends; development of the technical bases for revision of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials;" [ref. #8.2.17.10] irradiation of high-Cu, high-Ni welds and validation of embrittlement trend curves; an expert elicitation to verify that a generalised flaw size and density distribution can be properly developed for the entire population of U.S. RPVs and to assist in developing a flaw distribution; and calculations to provide technical basis for revising 10 CFR 50.61.

5.17.4 Adequacy of PSA used in Risk-Informed Decision-making

The NRC staff is also developing a new regulatory guide (RG) that will provide guidance to licensees on how to use PSA standards and industry peer review programs to demonstrate that the risk metric input to a risk-informed decision is technically defensible. Accompanying this new RG will also be a Standard Review Plan (SRP) chapter. The main body of the RG will provide guidance on the use of PSA standards and industry guidance by licensees to determine the level of confidence that can be afforded PSA insights/results in support of decision-making. The staff's position on each PSA standard and industry guidance will be given in the appendices. For example, Appendix A will include the staff's position on the ASME standard. The intent of this new RG is to function as the "umbrella" document governing more limited-application RGs like RG 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis" [ref. #8.2.17.11]. Therefore, the new RG will be used to support a broader set of regulatory issues, including license amendments (the subject of RG 1.174).

5.17.5 *Operational Events Analysis*

To analyse operational events, the NRC uses probabilistic risk assessment techniques to provide estimates of operating event significance in terms of the potential for core damage. The types of events evaluated include initiators, degradation of plant conditions, and safety equipment failures that could increase the probability of postulated accident sequences. The accident sequence precursor (ASP) program systematically evaluates nuclear power plant operating experience using PSA methods to identify, document, and rank those operating events or conditions that were most significant in terms of the potential for inadequate core cooling and core damage. In addition, the program also does the following: (1) categorise the precursors for plant-specific and generic implications; (2) provide a measure that can be used to trend nuclear power plant core damage risk; and (3) provide a check on PSA-predicted core damage scenarios.

ASP analyses utilise information obtained from: (1) inspection reports and standardised plant analysis risk (SPAR) models; (2) industry-wide analyses reported via initiating event studies, component reliability studies, system reliability studies, common cause failure (CCF) studies, and special issue studies such as those addressing fire events and service water system events; and (3) operational data contained in the sequence coding and search system (SCSS) of the licensee event report (LER) database, reliability and availability data system (RADS), the CCF database, and the monthly operating report (MOR) database. The SPAR models will be discussed later in this section.

NRC uses comparisons between ASP analyses and significance determination process (SDP) assessments of inspection findings as part of their ROP self-assessment program. Trending information from the ASP program is part of the NRC's annual performance report to Congress. The ASP program provides the Commission with annual assessments of the significance of events/conditions occurring at commercial power plants and the trends in industry performance.

5.17.5.1 *Standardised Plant Analysis Risk (SPAR) Models*

RES is developing SPAR models to permit the NRC staff to independently analyse the risk significance of inspection findings and operational events and/or conditions. The SPAR models being developed include: (1) Level 1 models for full power, low power, and shutdown operations; (2) models for performing large early release frequency (LERF) calculations; (3) detailed models for Level 2 calculations; and (4) analysis tools for external event initiators.

The SPAR models use data obtained from: (1) industry-wide analyses reported via initiating event studies, component reliability studies, system reliability studies, CCF studies, and special issue studies such as those addressing fire events and service water system events; and (2) operating experience data contained in the SCSS LER database, RADS, the CCF database, the MOR database, and the ASP Events database. In addition, SPAR models use information about plant design that is found in final safety analysis reports (FSARs), plant information books, and licensee's updated plant PSAs.

SPAR models will be used by the NRC to:

- Determine the risk significance of inspection findings or events so that risk-informed decision scan be made regarding responsive actions;
- Support risk-informed decisions on plant-specific changes to the licensing basis proposed by licensees;

- Perform various studies in support of regulatory decisions; and
- Support the resolution of generic and other safety issues:
 1. Screen and analyse operating experience data in a systematic manner to identify those events or conditions that are precursors to severe accident sequences as part of the ASP program;
 2. Assist in the identification of threshold values for risk-based performance indicators (RBPIs) and in the development of an integrated RBPI;
 3. Analyse operating experience data to determine which risk-significant conditions need more or less regulatory attention, and which regulatory or licensee programs have had an impact on risk and to what degree; and
 4. Provide rigorous and peer reviewed evaluations of operating experience data to enhance the technical credibility of the NRC's ability to analyse operating experience data independently of licensee's risk assessments.

5.17.5.2 Risk-Based Performance Indicators (RBPIs)

The reactor oversight process (ROP) uses objective performance indicators and risk informed analyses of inspection findings to assess licensee performance. Risk-Based Performance Indicators (RBPIs) are being developed to improve performance indicators for potential implementation in the ROP. The potential RBPIs include:

- Reliability and availability indicators at the component, train, and system level;
- Indicators for shutdown modes and fire events that are consistent with current models, data, and methods;
- Plant-specific threshold values to reflect risk-significant differences in plant designs; and

An overall-plant-performance indicator that will include the risk significance of individual RBPIs and inspection findings for a plant.

RBPIs will support the ROP assessment activities by providing direct measurements of the performance of risk-important safety features to determine whether safety is improving, deteriorating, or remaining constant. The supporting analyses and data systems needed to develop RBPIs will also be used by NRC's inspection staff in developing risk-informed inspection guidance and significance determination process (SDP) evaluations, and by its research staff that use risk-important information to identify ways to improve the effectiveness of NRC regulatory requirements, guidance, and processes.

5.17.6 Materials and Waste Arena Support

5.17.6.1 Dry Cask Risk Analysis

NRC is performing a pilot PSA of a spent fuel dry cask storage system, the Holtec International HI-STORM 100. This cask is being studied at a specific BWR site where the operations can be observed and modelled. (Although developed for a specific cask at a specific site, the analytical models developed for this preliminary study can be modified and applied to other dry cask systems at other reactor sites.) During

its service life, the cask has three operational modes - handling in the reactor building, transfer to the storage pad, and storage for 20 years. In each of these modes, accidents that could result in mechanical and thermal challenges to the cask and that have the potential to cause the release of radioactive material, are postulated. Event tree/fault tree methods are used to develop logic models of plausible accident sequences. Engineering analyses are used to determine the stresses that would be imposed by the postulated events. Fracture mechanics and other engineering disciplines are used to determine the probability of a cask failing when subjected to postulated accident conditions. A human reliability analysis is used to determine the probability of accidents caused by incorrectly performed procedures, such as when the cask is moved while inside the reactor building or while being monitored during storage.

The preliminary results of the PSA suggest that the risk of the HI-STORM cask at the BWR plant is low compared to the risk of accidents involving the core of operating nuclear power plants. Events that have a high conditional probability of failing the cask have a low frequency (on the order of 10^{-6} per year or less). Conversely, events that occur with a high frequency have a low conditional probability (on the order of 10^{-6} or less) of failing the cask. Furthermore, the consequences of most of the postulated events that fracture the cask and the fuel are low because the energy driving the radionuclides from the fuel pellets is low and the inventory of radionuclides in the fuel pellets is relatively low compared to the reactor inventory. Accordingly, the risk, defined as the product of the frequency and consequences of the events, appears to be low. A draft report is expected to be completed in mid 2002.

5.17.6.2 Risk-Informed Initiatives for Nuclear Materials and Waste

NRC is now engaged in activities that move towards increasing the use of risk insights and information in its regulatory applications, where appropriate, in the nuclear materials safety and nuclear waste safety arenas. Regulatory applications include, but are not limited to, rulemaking, guidance development, licensing and certification, and inspection activities for fuel cycle facilities, industrial and medical licensees, site decommissioning, transportation, and waste management and disposal.

The staff proposed a four-part framework for using risk assessment in nuclear materials waste regulation:

Part 1 - Define regulatory application areas in which risk assessment methods can play a role in NRC's decision-making process. Group the areas by regulated use (e.g., fuel fabrication) and within each use by regulatory application (e.g., graded quality assurance).

Part 2 - Evaluate the current considerations underlying the application area to ensure that the existing approach is altered only after careful consideration. Factors to be considered include: deterministic considerations (hazard, relative importance of human vs. equipment error, defence-in-depth, codes and standards); current risk considerations (e.g., use of performance assessment in geologic repository licensing); and institutional considerations (existing statutory requirements, Agreement State issues, and licensee circumstances).

Part 3 - Evaluate new risk considerations in support of the proposed regulatory action. Elements of this evaluation include: scope and level of detail of the risk assessment, sensitivity and uncertainty analyses, and assurance of technical quality.

Part 4 - Integrate the current considerations and new risk considerations to ensure a consistent and scrutable decision-making process and to ensure that the underlying bases for rules, regulations, regulatory guides, and staff review guidance are maintained or modified to the extent supported by the conclusions of Parts two and three.

NRC is following a general, three-phase plan to implementing the framework described above. The first two phases address the first step in the framework implementation process. The first phase focuses on developing a systematic approach for identifying candidate regulatory applications that may be amenable for increased use of risk information. The second phase focuses on applying the systematic approach, developed through the first phase, to identify the candidate regulatory applications. Finally, the third phase addresses steps two through five of the framework implementation process described in SECY-99-100 [ref. #8.2.17.12]. The third phase focuses on the actual modification of the identified regulatory applications to make them more risk-informed.

CHAPTER 6 – PSA RELATED RESEARCH AND DEVELOPMENT

While much progress has been made, limitations exist in the methodologies. Weaknesses, such as large uncertainties need to be further studied. Obtaining more and improved data for use in quantification is another area of increased focus. This chapter provides input from the Member countries on current and proposed area of PSA research activities.

6.1 Belgium

In the period 1985 – 1992, a Belgian team participated to a series of reliability benchmark exercises, organised and co-ordinated by JRC-Ispra (EC). Those benchmark exercises focused on common cause failure modelling, human reliability analysis and sequence quantification. More recently, AVN participated in benchmark exercise on expert judgement methodology, with application in the level 2 PS field.

In follow-up of an international project on the development of guidelines for PSA base event analysis [1], AVN organises since 1998 a yearly technical meeting on Risk Based Precursor Analysis. It covers typically status reports on risk-based precursor studies, case studies illustrating particular aspects (e.g. the treatment of potential initiators, CCF, etc.), use of risk-based precursor studies, and a discussion on further activities. The minutes are communicated to the WGRisk members. A fifth meeting is foreseen on November 7 and 8, 2002.

At the end of 1998, a project was launched in the framework of the Franco Belgian working group on nuclear safety to perform an in-depth comparison of the PSA of Tihange 1 (B) and the EPS 900 MWe (F). From both the French and the Belgian side, it was found that the benefits of such a comparison exercise are multiple. A paper has been presented at PSAM 5 [2] and the latest results will be presented at PSAM 6 [3]. A second phase of this Franco Belgian exercise has recently started and focuses on a comparison of the PSA methodologies and results for non-power and shutdown states.

Further the PSA-comparison exercise for the power states has been extended through participation of NNR, the South African safety authority, to include the Koeberg PSA. Initiating events not covered in the first Franco-Belgian comparison will be investigated.

6.2 Canada

CNSC is a member of COOPRA and it is involved in the activities of the special interest groups on Risk-Informed Decision-Making and Uncertainty Analysis.

CNSC participates in the International Common-Cause data Exchange (ICDE) project. In this respect, CNSC initiated a research contract with an external contractor for collecting data on centrifugal pumps, MOVs, and safety/relief valves. The contract is in progress.

6.3 Czech Republic

There are several scientific projects in the Czech Republic supported by Government, which are related to NPPs safety improvements and cover also a probabilistic safety assessment issues. Improvement of

knowledge base, which is necessary for assessment and decision making process in safety controlled NPPs operation is a main goal of the PSA part in which current methodology problems, uncertainties and PSA limits are discussed. There are several areas of interest as modelling issues (PTS, LOCA frequencies, CCFs, errors of commissions and other HRA issues etc.), development of PSA applications as far as discussion of quantitative guidelines and decision criteria to be used and overall PSA quality aspects.

6.4 Finland

STUK is in the process of moving towards Risk-Informed Regulation and is therefore extending the scope of PSA based regulatory activities (i.e. ISI, IST, Tech Specs and Graded QA).

Pilot project dealing with PSA support to regulatory activities has recently been completed by STUK. The aim of the project was to explore on how the plant specific PSAs available can best be used effecting specific regulatory audits and safety management tasks such as ISI, IST, preventive and corrective maintenance and Tech Specs activities. The study was to clarify the merits of PSA in order to consolidate the present regulatory process, to make the regulatory audits more cost-effective in weighting more and more the risk reduction potential of the objects to be inspected and if well argued, to lessen the unnecessary burden of the licensee. The strength of PSA in support of regulatory audits and In-Service Inspections and Testing is recognised laying in the skills to address the importance of various components, systems and safety functions. PSA has turned out effective in ranking importance of components independent of the complexity of the accident sequences they are included in. In this context it is worth checking the validity of safety classification because the PSA studies have addressed that many of the support systems are of high importance. Based on these features of PSA the aim of the project is to integrate the deterministic and probabilistic regulatory approach in common procedure where appropriate.

The STUK's risk-informed procedure on ISI combines both the plant specific PSA information and the deterministic insights in support of the system specific, detailed ISI program selection. Piping of all systems important to safety are exposed to the selection procedure irrespective of the ASME class (1,2,3 or even non-code piping). The selection procedure includes several steps such as selection of systems and identification of the evaluation boundaries and functions, consequence evaluation and qualitative degradation mechanism evaluation of piping and division of the segments into different inspection categories. Division of pipe segments into various degradation categories is to be based mainly on qualitative identification of the mechanism which the pipe segment is exposed (such as erosion corrosion, vibration fatigue, water hammer thermal fatigue, stress corrosion cracking and others). Division of pipe segments into various consequence categories is based on conditional core damage probability estimated by PSA applications. Finally the expert panel containing all affecting engineering disciplines combines the deterministic and probabilistic information as emphasised by the EPRI's approach and the NCR's regulatory guides. The pipe segments are divided into different inspection categories containing high, medium and low risk segments, respectively.

6.5 France

6.5.1. Methodology evolutions (EdF)

6.5.1.1 Computer tools

The KB3 software (automatic tool for fault-tree generation in systems analysis) is available, so as its interface tool with RISK SPECTRUM. Both are used for the 1300 and 1450 MWe series PSAs.

6.5.1.2 Human factors

After experimental use and full documentation implementation, the new MERMOS methodology has been applied on 1450 MWe series PSAs. It should lead to its final industrial validation.

6.5.1.3 I&C

The new modelling method for digital I&C dependability is being used on EPR, 1300 MWe and 1450 MWe PSAs.

6.5.1.4 General principles

A document describing the basic principles for the use of PSA in EdF has been sent in 1997 and analysed by the Safety Authority. A new Fundamental Safety Rule (RFS) dedicated to the development and use of PSAs in France is under development with the Safety Authority.

6.5.2 Supporting studies (EdF and IRSN)

Several physical studies are carried out as supporting studies for PSA level 1 and level 2. In fact PSA has identified significant sequences for which the physical phenomena are not well known, and in order to reduce the corresponding uncertainties, several studies and calculations have been undertaken or are planned.

6.5.2.1 Thermalhydraulics

Several thermalhydraulics calculations were carried out in order to evaluate precise some systems success criteria, the activation of signals and the delays available for operator actions.

Examples are a series of calculations relating to LOCA during shutdown states, with different break size and different systems configuration.

The computer code used for these calculations is CATHARE.

6.5.2.2 Mechanics

For PSA level 1, studies are necessary for analysing the behaviour of the primary circuit and of the reactor vessel in case of overpressurisation

For PSA level 2, the important mechanical studies relate to SG tubes leak tightness in case of severe accidents and containment behaviour (very important 3D calculations are in progress). French codes CASTEM and MC3D are used.

6.5.2.3 Fire

In order to reduce uncertainty associated with fire modelling:

- Fire tests have been carried out about electrical damage. These tests consisted of studying the behaviour of CCI and power cables under thermal load
- Fire tests concerning combustion of electrical cabinets have begun and preparation of tests of fire propagation from a compartment to the adjacent ones is continuing.

- A model related to the propagation of thermal effects from one compartment to the adjacent one has been added in FLAMME-S computer code. This model has been qualified by fire test results already performed.

6.5.2.4 Severe accidents phenomena

The physical phenomena studies are related to the core uncovering and degradation, the hydrogen production and combustion, the temperature induced breaks of the primary circuit under high pressure situations, the corium flowing down into the vessel bottom head, the steam explosion, the vessel rupture and the containment direct heating, the corium-concrete interaction, the thermalhydraulics in the containment and the fission products behaviour in the primary circuit and in the containment.

In order to quantify the physical phenomena, French codes such as ESCADRE-ASTEC, CATHARE, CASTEM and MC3D have been used. When some physical phenomena have been judged inadequately modelled in available codes, some specific simplified parametric models have been developed within the framework of the project, especially in the field of advanced core degradation. Corresponding uncertainties have been assessed. Recent experimental results, in particular as regards fission product behaviour (PHEBUS PF program), have been also taken into account.

6.5.2.5 Behaviour of equipment during accidental conditions

An exhaustive study of ultimate means for recovering water injection into primary and secondary circuits has been performed; a study of the survivability of equipment under severe accident conditions, including small scale experiments, is under way (clogging of containment sumps, I&C behaviour during extreme conditions); an exhaustive assessment of containment leakage paths has been also conducted.

6.6 Germany

With limited funding, a number of activities are underway to develop PSA methodology and data base. These are related to:

- Methods to improve fire PSA (screening for the selection of relevant rooms, evaluation of fire consequences, integration of results into the PSA)
- Participation in the IFDE (fire data)
- Methods for the probabilistic evaluation of software-based I&C
- Development of an approach for a “dynamic simulation” (mainly related to the level 2 of the PSA)
- Participation in the ICDE (CCF data)
- Development of methods for evaluating errors of commission.

In the context of the level 2 PSA for Neckarwestheim 2 a number of methodological improvements have been realised (mainly comprehensive uncertainty analysis instead of calculating “point values”, transparent interface between level 1 and level 2).

6.7 Hungary

The PSA related R&D activity in Hungary can be considered as applied research, i.e. an activity which directly supports an ongoing PSA study or a PSA application. The current interests are as follows.

Much attention has been paid at human reliability analysis (HRA) since the beginning of PSA activities in Hungary. Efforts are made to develop HRA methods that can better represent human behaviour and the effect underlying situational characteristics for various types of safety related human interactions, including maintenance and operation as well as responses to plant transients. These methods try to integrate field experience, insights from event reports, results of simulator observations and expert opinion into a common framework to help HRA modelling and quantification. Also, VEIKI develops data collection and analysis systems in support of identifying strengths and weaknesses in human performance and providing input for use in HRA.

The safety level of the Paks NPP is currently being assessed with respect to the risk of large radioactivity release. This analysis includes a full scale level 2 PSA that requires specific methodological developments. Also, the study is to identify areas where further research and development will be needed to increase credibility and usefulness of level 2 PSA results and to implement accident management strategies for the Paks plant. At present, the major interests are as follows:

- treatment of specific design features of the VVER-440/213 plant during the development of containment event trees and during the associated supporting severe accident analyses
- evaluation of containment response and fragility for loads from various potential severe accident processes in consideration of the rectangular, multi-compartment containment design
- treatment of uncertainties in view of limited experimental data, analyses and knowledge.
- conceptualisation of accident management strategies and their incorporation into the PSA models.

Although the ongoing seismic PSA for Paks follows mostly the guidelines of IAEA and US NRC, some important issues require special considerations and/or developments. Among these issues are:

- fragility analysis for the complex rectangular containment structure
- methods for identification of I&C and electrical contact devices that might chatter during a strong seismic motion
- fragility analysis for I&C and electrical with limited data, experiments and seismic experience
- development of a post-processor to PSA software to properly quantify minimal cut sets with seismic failures and the associated uncertainties.

6.8 Italy

PSA uncertainty and sensitivity analysis methods, and more specifically neural networks and genetic algorithms, development is the major research theme addressed by Polytechnic of Milan (ref.8,9).

An effort about development of PSA application to NRNF (Ref. 5) is performed by ANPA.

6.9 Japan

6.9.1 Introduction of the State-of-the-Art Methodology

6.9.1.1 Reliability Analysis for Digital Safety Protection System

In Japan a few of NPPs, such as advanced BWR (ABWR) and APWR, are under commercial operation, construction and contemplation, which introduce digital control system to the safety system. Utilities and NUPEC have made level 1 PSA for ABWR, including reliability analysis of digital safety system. In the PSA of NUPEC both hardware and software failures of the digital safety system were taken into account mainly based on IAEA-IWG-NPPCI-94/8, IAEA-TECDOC-581 and so on.

6.9.1.2 Human Reliability Analysis

Japan Atomic Energy Research Institute (JAERI), NUPEC and Central Research Institute of Electric Industry (CRIEPI) have programs on various aspects such as human reliability analysis, man-machine interface research, operational management, training, utilisation of artificial intelligence for operational aid and collection and analysis of human reliability data. Utilities are collecting and analysing human behaviour data at their training centres.

NUPEC has made basic research on human factors for the purpose of minimising human errors of operators and maintenance persons in NPPs. Since 1997 a project using the human characteristics experimental facilities has been underway to collect basic human error data and verify human reliability analysis (HRA) methodologies, where integration of ATHEANA into level 1 PSA has been also investigated.

6.9.2 Level 2 PSA

In a program of “Development of Level 2 PSA Methodology”, JAERI developed the THALES/ART code and its advanced code THALES-2 for analysing progression of a core melt accident and fission product release and transport behaviour. Improvements of aerosol transport and iodine chemistry models of THALES-2 are on going based on experimental data such as the WIND experiments of JAERI.

In the utilities the MAAP code has been extensively applied to examine effects of various accident management countermeasures on mitigation of accident progressions for implementing of accident management measures.

NUPEC has also modified the models for predicting radionuclide behaviour in the MELCOR code under controlled release through hardened venting conditions as accident management.

6.9.3 Level 3 PSA

JAERI has developed the OSCAAR computer code package, which consists of inter-linked computer codes to predict (a) transport of radio nuclide through the environment to man, (b) subsequent dose distributions, and (c) health effects in the population. Level 3 PSA for a generic LWR is in progress for providing inputs to discussions on various safety related issues such as the safety goals and the effectiveness of emergency measures.

NUPEC has investigated models of MACCS to estimate the effectiveness of the AM on the off-site radiological consequence and has developed database in accordance with Japanese environmental characteristics. The MACCS-2 code has applied to estimate radiation dose profiles based on the Level 2 PSAs for the typical PWR and BWR plants.

6.9.4 Seismic Risk Analysis

JAERI and Japan Nuclear Cycle Development Institute (JNC) have established whole sets of methodologies for seismic risk analysis of LWR and FBR, respectively. JAERI published a report on its seismic PSA for a generic BWR in 1999. Current activities at JAERI are directed to application of the methodology to issues in seismic design and seismic risk management, including the studies on (a) the use of seismic hazard analysis for determining scenario earthquakes for seismic design, (b) the use of seismic PSA for nuclear power plants sited on Quaternary Deposits, and (c) risk management for seismic risk at existing plants.

NUPEC started the development of comprehensive methodologies for seismic risk analysis in 1994. The preliminary seismic PSA analysis of a Japanese typical BWR has been performed since 1997. In this analysis seismic hazard curves for typical NPP sites were evaluated with experts' opinion and through discussions, and Japanese specific seismic fragility data have been analytically pursued on the basis of both the structural analysis and Japanese seismic proving test data. Fragility data, especially uncertainties of their capacity of active components and electrical components, were re-evaluated with experts' opinion. A study on probabilistic scenario earthquakes and ground motions was started in August 2000. The preliminary seismic PSA for a typical PWR and another type of BWR started in April 2001.

6.9.5 Fire Risk Analysis

Since 1998 NUPEC has made fire severity factors to be applied in fire PSAs for Japanese NPPs, using fire simulation codes and fire experiments. NUPEC has prepared fire severity factor for every categorised fire source, taking into account specific circumstances of fire source components deployed.

6.9.6 Others

JAERI started in 1994 a preliminary study on PSA methodology for external events other than seismic and fire events. This study aims at proposing a screening methodology to identify external events for which detailed examinations of hazard and/or plant fragility are necessary. It has proposed a screening methodology for volcanic activities.

6.10 Korea

The PSA methodology used in Korea is not unique in its feature. The general procedure and quantification methods are utilised, therefore, while developing new methodologies such as those of low power and shutdown PSA, as described in this sub-sections. In usual, the government, the Ministry of Science and Technology (MOST), funds many PSA-related research programs. The major topics of the research programs are as follows:

- The importance measures, which plays an essential role in risk-informed applications such as RI-ISI, RI-IST, etc.,
- Fast running algorithms used in PSA quantification codes,
- Korean standard component reliability data base (DB),

- Korean standard reactor trip DB,
- Some research areas co-operated with OECD/NEA,
- Cable Ageing data, piping failure data, CCF risk analysis data collection, fire risk analysis data collection
- A maintenance framework including the concepts of plant life extension, maintenance rule, and periodic safety review,
- PSA modelling reflecting organisation factors,
- PSA modelling reflecting ageing effects.

6.10.1 Technical Basis for Risk-informed Regulation

In order to meet the utilities' desire for risk-informed applications, it is essential to prepare the technical basis used for regulatory decision making, where following issues have been discussed with the long-term research project:

- Risk criteria,
- PSA standards,
- Review guidance for PSA results with diverse work scope,
- Guidance for using reliability DB,
- Regulatory guidance for various risk-informed application programs.

In near future, the precursor study and the preparation of risk-informed performance indicator will be started.

6.10.2 Plant Specific Reliability Database for KSNP, KIND

In Korea, the foreign generic component reliability database has been used in the PSA for KSNPs, since no Korean specific component reliability database was unavailable at that time. As the number of operation year of Korean NPPs has increased, the necessity of the site-specific component reliability database has been spread. KAERI (Korea Atomic Energy Research Institute) has developed a domestic NPP component reliability database, KIND, that reflects the plant-specific characteristics of KSNP since 1998. The operation and failure/repair data for components for about 24 safety related systems of KSNPs have been collected, and analyzed. KIND can provide the unavailability data, and 3 types of failure rates based on the component operating time, demand number, and plant operating time, respectively.

The failure rates of KIND are compared with those of generic database used for YGN 5&6 PSA. In the case of YGN 4, the result of the comparison shows that most of the failure rates of the KIND are lower than those of generic database. And 60% of compared failure rates show no big differences between KIND and generic database. KIND will be used not only for PSA of new NPPs but also for PSR (Periodic Safety Review) and RIR being performed in Korea.

6.10.3 Computational Tools

The accident sequence quantification in Level 1 PSA is processed by many codes for several initiating events, such as KIRAP (Korean cut set generation code), CAFTA, NUPRA, and FORTE (Korean code) depending on the projects, respectively.

KAERI changed the operating environment of KIRAP from MS-DOS to MS-Windows since 1995. Efforts to improve its capability and user interface are continued. They include (1) development of fast cut set generation algorithm by Shannon decomposition theory, (2) solution of logical loops by the analytical method, and (3) implementation of rule-based recovery analysis techniques. It is expected that the development of algorithm generating cut set and improvement of user interface could be an essential part.

To simulate the accident progression from core damage to the containment failure, usually EPRI's MAAP code has been used. However, for Wolsong 2,3&4 PSA, ISSAC (Integrated Severe Accident Analysis Code) computer code has been developed based on MAAP4 code by KAERI. Accident progression analysis for Wolsong NPPs has been performed in detail utilising this code.

As a risk monitor for safe operation of nuclear power plants, a computer code called DynaRM has been developed by KAERI. DynaRM can update the plant risk continuously according to the change of system/component configuration. Whenever plant configuration changes, DynaRM re-evaluates the plant risk based on the PSA results. DynaRM shows new plant risk relative to the baseline risk level. The basic purpose of DynaRM is to trace changes of plant risk so as to identify the early signals of deteriorating plant safety. In the case of multiple components outage, DynaRM could suggest maintenance priorities according to the importance of each component from the plant safety point of view. The history of risk level change is stored, and the history is also used to retrieve the plant operation experience, and this can help the operator comprehend the operating history. Since DynaRM is based on the change of component status, it can be used as a tool for indicating the current risk level and an advisory system for the risk management and the component maintenance policy.

DynaRM will be installed at UCN 3& 4 site in 2002. This will be the first risk monitor installed at the site in Korea. At present, another risk monitor is being developed by KOPEC (Korea Power Engineering Co.) for Kori 1& 2.

6.10.4 Digital I&C Systems Reliability Analysis

The increased use of digital instrumentation and control systems in nuclear power plants raises some unique reliability and risk issues. In Korea, digital I&C will be introduced into the design of the APR 1400. Therefore, there is a need to develop a methodology to assess the safety and reliability of digital I&C systems. A number of programs are underway, in Korea, in the areas of software verification and validation using qualitative methods. Research was begun in 1999 to develop the methodology as part of the APR 1400 development project. The project will focus on developing quantitative and probabilistic methods in order to assess the safety and reliability of digital I&C systems, including hardware and software reliability and human-system interface issues.

Through the study and the insights from our PSA experiences, some critical factors of system unavailability, such as the coverage of a fault tolerant mechanism, the model of common cause failures and the software failure probability were identified. Further, the effect of each factor through sensitivity study was examined. The result of sensitivity study shows that the unreasonable assumptions on these factors would severely distort the analysis results. Such results of this research program will be used as the basis of new research program that will be presented in Section 3.

6.10.5 Cognitive Human Reliability Analysis

The professional views for the conventional HRA can be summarised into two aspects. First, it has focused mainly on the quantification of human error probability. Second, it has focused only on the observable aspects of human behaviour. Accordingly, it has limitations in dealing with the cognitive errors, which are bound to be committed during the decision-making process. Currently, KAERI is going to develop a methodology for the human error analysis (HEA) to deal with the above-mentioned problems. The study will be focused on the cognitive errors that include potential errors in the process of situation judgement and decision-making. It also aims to provide a basis for performing HEA for a specific task required during an accident management situation involving a severe accident, and helps to understand errors more thoroughly and to develop error reduction strategies.

During the first year of research, KAERI tried to establish the requirements for the development of a new HEA method. To achieve this goal, case study was carried out through the following steps: (1) review of the existing HEA methods, (2) selection of methods which are considered appropriate for the analysis of operator's tasks in NPPs, (3) choice of tasks for the application, and (4) application of selected methods to the tasks. The following three HEA methods were selected for the case study:

- HRMS (Human Reliability Management System),
- PHECA (Potential Human Error Cause Analysis) and,
- CREAM (Cognitive Reliability and Error Analysis Method).

The research is in second stage. They are willing to introduce structured information analysis (SIA) as a task analysis method for error analysis, and finally to delineate the result of application on the emergency procedure of KSNPs.

6.10.6 Extension of AOT/STI of RPS/ESFAS for KSNP

This program has been performed for 3 years (1999.4 – 2002.4) by KAERI to provide technical basis for Technical Specification improvement of plant protection system (PPS) comprising RPS/ESFAS in KSNP. It contains the system performance and unavailability analysis, and sensitivity analyses for AOT/STI changes on risk like the core damage frequency and large early release frequency. In particular, the RPS/ESFAS performance analysis was based on the operating experience of KSNP. The total operating experience of 8.69 commercial reactor years at four units (YGN 3&4, UCN 3&4) during the period 1995 through 2000 was studied. The detailed fault tree models for the RPS/ESFAS were also developed, using plant specific data based on observed operational experience. Korean utility, KHNP (Korea Hydro & Nuclear Power Co. Ltd.) will submit the result of this analysis to KINS in 2003 in order to get the approval for the extension of AOT/SIT of ESFAS/RPS.

6.10.7 A New Long & Mid-Term Research Program

A new long & mid-term research program for PSA & RIR funded by MOST will start from April 2002. This research program will be performed by KAERI. The main topics of this program are as shown below:

- RIA technology
 - Development of PSA standard and model for KSNP
 - Enhancement of risk assessment techniques (Level 2, Fire, etc.)

- Reliability database
 - Development of plant specific reliability database (KIND) & Precursor Analysis
 - Development of PSA information DB
 - ICDE, OPDE
- LPS PSA and Digital I&C Reliability Assessment
 - Improvement of LPSD PSA methodology
 - Development of reliability assessment methodology for Digital I&C
 - Evaluation of the risk effect of digital I&C systems
 - Methodology development on the quantitative safety assessment of digital I&C systems
 - Evaluation of the coverage of a fault tolerant mechanism
 - Modelling and parameter estimation of the CCF of digital I&C systems
 - Methodology development on the quantitative evaluation of software failure probability
- Human factors
 - Development of HPDB & task complexity measure
 - Development of next generation HRA methodology (considering EOC)
- Life cycle risk assessment of various power sources

There is another PSA related research program performed by KINS that is also funded by MOST. This research program will start from April 2002 and it will cover the following items.

- Development of basic RIR guidelines (-2001)
- Development of regulation framework for RIR (2000-)
- Pilot implementation of RI-ISI (2000.2-)
- Development of independent review model (2002-)

6.11 Mexico

Risk Assessment is an important subject for both regulators and utilities, and in order to increase its use for risk informed regulations and decision making, improvements in methodology and data is needed in the following areas.

Efforts will need to continue on data collection and analysis in relation to plant specific data. It is necessary to promote the exchange of experience on reliability data collection of safety systems components. Common cause failures are incorporated into the PSAs but the associated quantification of such contributors does have large uncertainties. Due to the scarceness of relevant plant specific data, it is necessary to continue efforts to gather information originating from different sources, therefore, the International Common Cause Failure Data Exchange efforts will need to be maintained in order to overcome the above limitations.

Parameter uncertainties may be incorporated into the PSA models and propagated by means of Monte Carlo simulations. However, uncertainties caused by modelling and assumptions are rarely treated, so efforts should be made to systematically study, quantify and reduce such uncertainties.

Concerning ageing, data for components are being accumulated through plant specific data collection. However, the study of the effect that ageing has on structures and passive components in general has only just started, the methodology to incorporate ageing effects into PSA technology is also in its early stages, so efforts should be devoted to improve this knowledge.

Digital instrumentation and control is being introduced everywhere and must be included in the probabilistic safety assessment. So efforts must be maintained and expanded to develop the methods for the quantification of the reliability and failure modes of programmed systems including the problem of software reliability.

Severe accident research has significantly improved our knowledge in the field. Nevertheless, the results of R&D on severe accident have not been extensively incorporated into Level 2 PSAs. Accident management strategies will help the operator to mitigate the consequences of such accidents and such strategies also require assessment and PSA. Therefore, efforts should be devoted to the incorporation of results of R&D on severe accidents into PSA, in order to have an adequate model to evaluate the accident management strategies.

Only a few countries have experience on Level 3 PSA, so more discussion and international consensus is desirable on the applicability of it on safety related decisions, especially in the areas of cost and or risk analysis, site evaluation analysis and emergency preparedness.

6.12 Netherlands

6.12.1 PSA Supported Severe Accident Management Strategies for Borssele NPP

Following the backfitting programme at the Borssele NPP is finished, the next logical step is to enhance the safety of the plants is to pay a more structured attention to the development of plant specific Severe Accident Management (SAMs) strategies.

At the Borssele Nuclear Power Plant (SIEMENS PWR) developments are taking place to expand the existing symptom based emergency operating procedures (EOPs) including Functional Restoration Guidelines, to arrive at plant specific 'Severe Accident Management Guidelines' (SAMGs). Although, generic SAMGs (BWR and PWR Owners Groups, EdF, etc.) exist, and therefore can be used, plant specific features, weaknesses etc. will necessarily not be covered. Therefore, at least developments are needed to transform from these generic SAMGs into plant specific SAMGs and to adapt the existing EOP's in order to accommodate the SAMGs.

In the development of the SAMGs extensive use will be made of the PSA of the Borssele NPP. This PSA being a level3 PSA, includes both Power and Non-power Plant Operating States, both internal and external events, both Human Errors of Omission and Commission. In the level-2 part of the PSA 50 CETs, each consisting of 51 nodes, have been developed for the 111 summary Plant Damage States. Binning of the End States resulted in 16 Source Term Categories. A large number of MAAP4 runs was used for the quantification and timing of the corresponding accident sequences. The insights based on the PSA together with the insights originating from deterministic calculations enabled the Borssele staff to develop several Severe Accident Management Strategies. In this paper a more detailed treatment will be given of some of these developed strategies.

6.12.2 Sequence Selection

The Borssele NPP is using Westinghouse based Emergency Operating Procedure. Many Accident Management strategies to prevent core melt are included in these. Nowadays extra attention is given on the mitigation of the consequences of core melt sequences e.g. Severe Accident Management.

To focus the efforts of this study, priority is given to sequences that contribute most to the level 3 consequences. To perform this prioritisation, a level-3 consequence indicator had to be chosen, because most level 3 results are presented graphical, which is less suitable for prioritisation.

The individual Risk Measure at the site border turned out to be a good indicator, presented as a yearly frequency.

The NUCAP+ code, that was used for the Borssele PSA, makes it possible to go backward from the level 2 results to the level 1 accident sequences. This way the accident sequences can be prioritised against a level 3 indicator.

For the Borssele NPP the following indicators were found to be most important to the level 3 results:

1. Gas Cloud Explosions (industry or ships)
2. Steam Generator Tube Ruptures
3. Interfacing Systems LOCA
4. ATWS + SGTR Ranking against level 1 results gives different insights. (see table 1).

For each sequence in the top of the list, several mitigating strategies were suggested. These strategies were studied for their impact by performing MAAP 4-runs and in some cases RELAP 5- MOD3 was used. This is discussed in the following sections.

The Gas Cloud Explosion is not discussed any further in this paper, because no special SAM measures were indicated in this case.

Sequence abbr.	Sequence description	<i>L1 freq.</i>	<i>L3 rank.</i>	<i>L1 rank.</i>
ECD-03	ship explosion	<i>1.19 E-7</i>	<i>1</i>	<i>11</i>
LBO-P-11	SGTR: failure to isolate	<i>1.79 E-8</i>	<i>2</i>	<i>36</i>
LBO-P-08	SGTR: loss of cooling	<i>1.57 E-8</i>	<i>3</i>	<i>39</i>
LBS-P-13	SGTR + ATWS	<i>1.52 E-8</i>	<i>4</i>	<i>41</i>
LBO-P-07	SGTR: loss of late cooling	<i>1.01 E-8</i>	<i>5</i>	<i>44</i>
Li1-P-19	ISL: RHR: seal rupture	<i>1.18 E-8</i>	<i>6</i>	<i>43</i>
Li2-P-15	ISL: IOS pipe rupture	<i>9.62 E-9</i>	<i>7</i>	<i>45</i>
Li4ME-08	ISL- mid-loop: loss of cooling	<i>6.81 E-9</i>	<i>8</i>	<i>51</i>
Li1-P-22	ISL-mid-loop: 100 R pipe rupture	<i>4.60 E-9</i>	<i>9</i>	<i>55</i>
Li4RE-09	ISL-cold: no cooling	<i>2.30 E-9</i>	<i>10</i>	<i>64</i>

Table 6.12.1: L3-Sequence Ranking

6.12.3 Interfacing Systems LOCA

There are many possible AM-strategies to prevent core melt in case of an Interfacing Systems LOCA. One could mention:

- Isolation of the leak
- Depressurisation of the primary and blocking the leak (plugging, injection of plastics, squeezing the leak).
- Depressurisation of the primary and transferring the leak to inside containment.

When these strategies are for some reason not successful, the following strategies that mitigate the consequences of core melt were studied:

- Using long pathways of the auxiliary building.
- Fill the secondary side of the steam generators.

To model these possibilities, the PSA was used to select a likely (less unlikely) sequence that would lead to core melt. Since the Borssele plant is designed with two independent safety injection systems, interfacing systems LOCA combined with a loss of injection is very unlikely. Without preventive AM-measures core melt will occur when the injection sources are depleted. This will only take place after 20 hours. MAAP 4 was used to simulate this sequence.

The first ISL strategy is not a real AM-measure, but is passive in nature. In the original level-3 PSA for Borssele, no credit was given for the secondary containment and the auxiliary building. For this study these buildings were modelled in MAAP 4. The steam generators are assumed to be filled in this case.

For this run it is assumed, that the windows in the chemical laboratory in the auxiliary building will break due to overpressure, and so creating a leak to the environment. A major reduction of the source terms can be observed, caused by the retention in the secondary containment and auxiliary building.

The second suggested AM-measure seems a bit strange, because the secondary side of the steam generators is not involved in the sequence. But when the steam generators are kept filled on the secondary side, they represent a very large cold surface. When it comes to core melt and relocation of the core, fission products are deposited on the primary side of the steam generator tubes. The cold surface also prevents re-vaporisation, so this way a major part of the inventory can be kept in the primary system.

Because core melt only occurs after 20 hours after scram, the noble gases are less significant in the source term. The most important isotopes are Cesium and Iodine. For this isotopes a reduction with a factor of 5 could be achieved.

6.12.4 Steam Generator Tube Rupture

There are many AM strategies available to prevent core melt. Many of these are covered in the ECA-3 series of the Westinghouse EOP's.

A most promising mitigating strategy is investigated for this study, e.g.: scrubbing the source term through the water inventory in the steam generators on the secondary side. A factor of 14-source term reduction (CSI and CsOH) was achieved by this strategy.

A closer look at the results of the MAAP 4 runs showed that the major effect of this reduction was not caused by the scrubbing effect, but by the deposition of fission products on the primary side of the steam generator tubes. A similar effect as we have seen in the ISL runs.

6.12.4.1 Induced Steam Generator Tube Rupture I

When a core melt takes place at high primary pressure, failure of the steam generator tubes can occur. Relocation of radio isotopes to the steam generator tubes can cause mechanical creep, what could result in creep rupture.

In NUREG-1150 it is indicated that such a failure only can take place at pressures higher then 13.4 MPa.

One of the major objectives of the C1 procedure of the Westinghouse FRG's is to prevent core melt at high pressure. So most likely, core melt will not happen at higher pressure, so that failure of the steam generator tubes caused by creep rupture for most cases could be neglected.

Only in case of an ATWS scenario, the C1 (core cooling) will not be used, because the S1 (reactivity) is of higher priority.

When in case of ATWS core melt can not be prevented, as an AM-strategy opening the PORV's is suggested. Of course this way the primary inventory is lost much faster, but after a while the creation of steam bubbles will stop the fission process and the sequence will change into a "normal" accident.

Without an injection source core melt can not be arrested, but most probably it will never come to induced rupture of the steam generator tubes.

A RELAP 5 run is performed to show the potentials of the strategy. For this run opening of the 3 PORV's is performed at a core outlet temperature of 650°C as in the C1 procedure. A combination of Station Blackout and ATWS is assumed to come to a core melt sequence. So no feedwater and injection sources are available.

As can be seen in the figures 1,2 and 3 at core melt the primary pressure is well below 13.4 MPa, so that induced SGTR can be prevented.

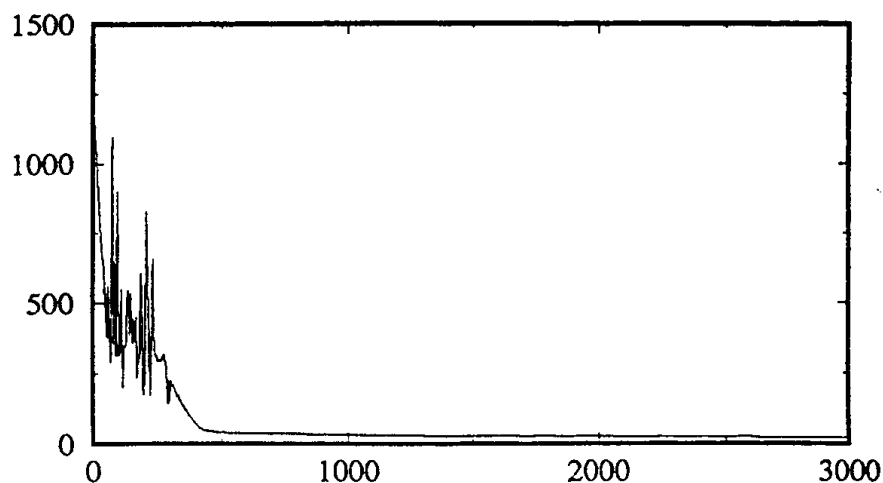


Figure 1: ATWS-prevention induced SGTR by bleeding using porv's - reactor power [MW_e]

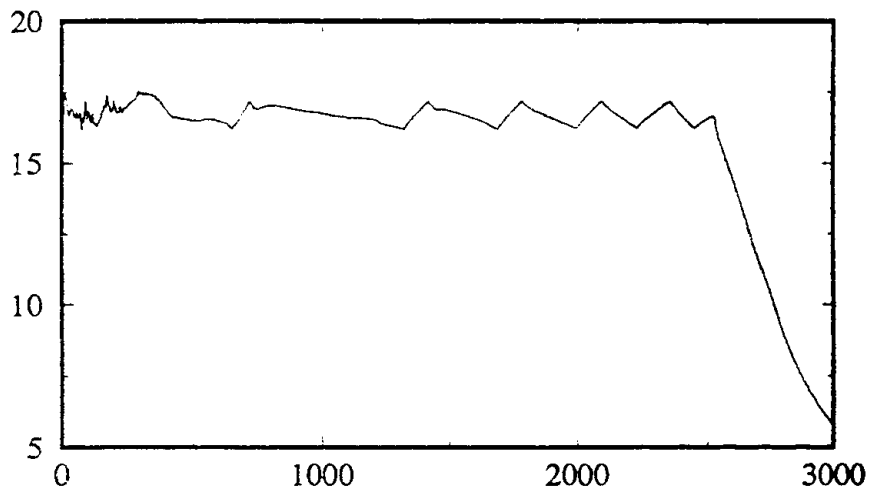


Figure 2: ATWS-prevention induced SGTR by bleeding using porv's - primary pressure [MPa]

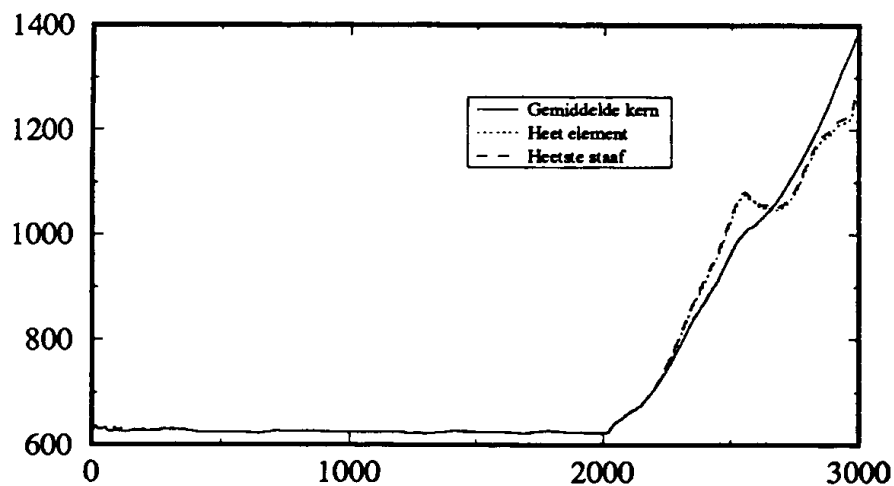


Figure 3: ATWS-prevention induced SGTR by bleeding using porv's - core outlet temperature [K]

6.12.4.2 Induced Steam Generator Tube Rupture II

When in the former case induced SGTR could not be prevented or when in case of a “normal” SGTR the scram function fails, reducing the primary pressure using the PORV's is still a good thing to do. This way it becomes possible to isolate the ruptured steam generator, before core melt occurs. Otherwise the primary pressure will stay raised at core melt and the source term will be released through the Main Steam Safety Valves.

So this strategy re-establishes the containment function. A RELAP 5 run is performed to know the potentials of this strategy. An ATWS run is simulated, where after 10 minutes a double ended tube rupture takes place (caused by the transient).

Again when the core outlet temperature reaches 650°C all 3 PORV's are opened. As the figures show, at core melt the secondary pressure is well below the set points of the Main Steam Safety Valves. So isolation of the steam generators becomes possible.

Doing this the source term reduction effect is tremendous caused by the washing effect, the retention on the primary side of the tubes and because of retention of radioactive materials in the rest of the primary system (because of relative low temperatures).

6.12.5 Overview of Results

In table 2 the source term consequences of the suggested SAM-measures are put together.

The results of these reductions on level 3 can be seen in the figures 4 (individual risk) and 5 (group risk).

An overall reduction of a factor 5 is achieved.

Another possible SAM-measure in case of a SGTR+ATWS-combination is once again filling the secondary side of the broken steam generator.

nr.	Suggested Measure	Cs I without	Cs I with	Reduction factor
3.1.1	ISL: full secondary SG	52%	10%	5
3.1.2	ISL: retention in Aux. Building	10%	2%	5
3.2	SGTR: full secondary SG	70%	5%	14
3.3	ATWS: prevent induced SGTR	19%	0.007%	2700
3.4.1	ATWS+SGTR: isolate SG	19%	0,007%	2700
3.4.2	ATWS+SGTR: full SG	19%	0,3%	60

Table 2: Source Term Reduction of suggested SAM-measures.

6.12.6 Conclusions

In situations where core melt can not be prevented, there are still possible strategies, that have the potential to reduce the source terms dramatically. This is the field of Severe Accident Management. In this paper very rare accidents, that are however important to risk, were discussed. They show an important reduction of source terms and level 3 results.

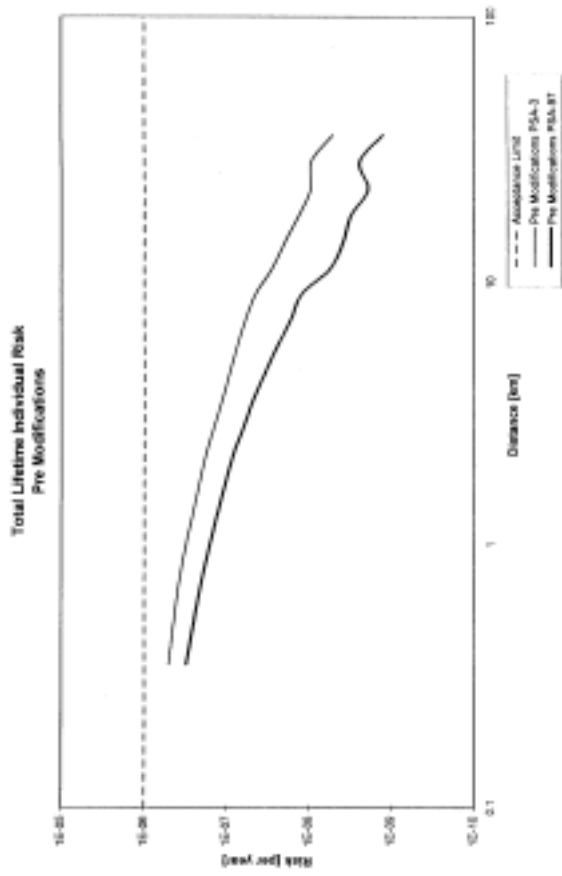
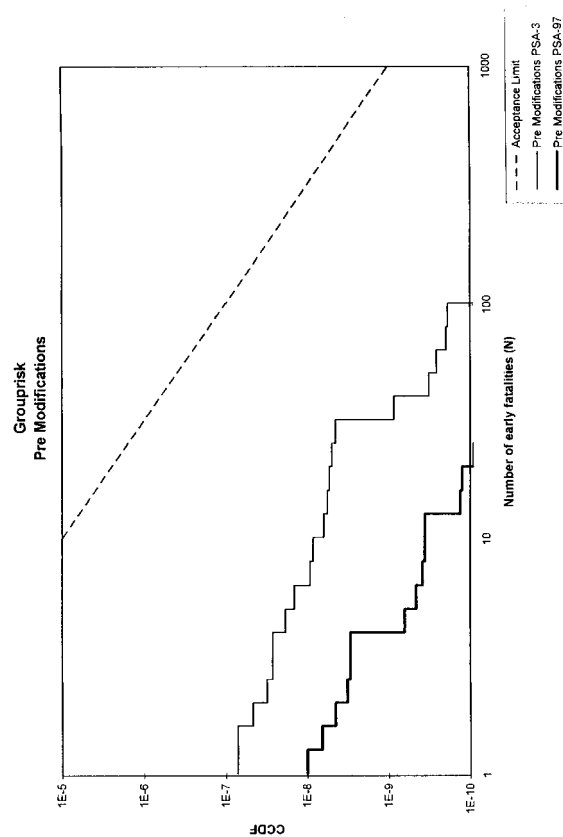


Figure 4: Total Lifetime Individual Risk

Figure 5: Societal Risk



6.13 Spain

The research needs identified by the Second Edition of the Integrated Program can be divided into two categories:

- 1) R&D to improve or develop methodologies to do the PSA.
- 2) R&D to develop and agree on the methodologies to use the PSA.

Category 1 has also an indirect impact on PSA applications, since more solid qualitative and quantitative PSA results, that might be achieved by means of improved PSA methods, would strengthen also the use of PSA models and results. Hence, more sound and generalised PSA applications are the motor which is perceived in Spain as impulsive of R&D activities.

The IP Edition 2 discusses and identifies the following research subjects within category 1:

- Some severe accident issues, as they impact the Level 2 PSA models (fuel-coolant interaction, molten core coolability, core-concrete interactions, hydrogen transport and combustion, ...).
- Some issues associated with Low Power and Shutdown PSA. The IP emphasises at this point on supporting thermal-hydraulic analysis.
- Some issues of external events risk analysis. Improvement needed in the estimation of external event frequency and analysis of the plant vulnerabilities are discussed, but the need for analysis of all modes of plant operation from this risk viewpoint is highlighted.
- Issues on human reliability analysis. Many aspects are mentioned and several discussed, but two are underlined: 1) organisation and management impact on safety and on PSA models, and 2) errors of commission, in particular for analysis of all modes of plant operation.
- Methods for inclusion of Expert Judgement in a PSA.
- Dependent failures.
- PSA of others types of nuclear installations and radioactivity sources.
- Other subjects: seismic risk, software reliability, structural (including piping) reliability, and passive system reliability.

Research activities have been carried out in Spain, or are planned for short or medium term, regarding all the subjects not included in the last group “other subjects”, where planning is not discussed yet for most of them.

All-high-priority PSA research areas included into category 2 are:

- Standard methods for PSA applications. Not only a need for development, but for consensus on acceptance criteria and on how to submit and discuss PSA reasoning on an application, as well.
- PSA as a technique for decision-making support. Cost-benefit or other types of decision analysis techniques are being analysed. PSA would play an important role in the associated decision making.
- Computer tools for PSA applications. An appropriate tool was needed at the CSN to maintain models and data of all the Spanish PSA, once they have been reviewed and accepted by the CSN. This is judged as basic for own applications and for proposal reviews. In 2001, the installation of the PSA models and data in the CSN computers is continuing, making use of a licence acquired for using the computer program Risk Spectrum. Six out of the seven Spanish PSA models are already available for CSN use.
- Improvements in the CSN-required National Data Bank. There are needs to improve some aspects of the structure and quality of data being supplied to make better use of them for PSA applications.
- Need to better include dynamic considerations into PSA. Not only to improve PSA models, but to have appropriate methodologies for some applications as well.

Along the line of R&D promotion in general and in the PSA field in particular, the CSN and UNESA (association of Spanish utilities) signed an agreement which allows for wide collaboration, between regulator and utilities, on nuclear safety research.

Many current and potential areas of collaboration in PSA and PSA-related research are included in that agreement. The summarised status of the R&D activities in those areas is as follows:

- Pilot experience on the PSA application to optimise ISI. Completed CSN-UNESA project.
- Pilot experience on the PSA application to graded QA. Completed CSN-UNESA project.
- Pilot experience on the PSA application to improve Technical Specifications. Completed CSN-UNESA.
- Human errors of commission. Ongoing CSN-UNESA project to be completed in 2001.
- Organisation and management impact on safety. Ongoing CSN-UNESA project to be completed in 2003, that includes an exploration of a potential incorporation of the O&M aspects in the PSA.
- Severe accidents. CSN and CSN-UNESA participation in several international (PHEBUS-FP, SR5CAP, RASPLAV-II, MACE, STORM, FARO-KROTOS) and national (pool scrubbing, iodine chemistry, hydrogen behaviour, accident management) projects.
- Operating experience on dependent failures. CSN-UNESA participation in the OECD/NEA International Common-Cause Failure Data Exchange (ICDE) project.
- Expert Judgement in PSA. Completed CSN project.
- Development of methods and tools for accident precursor analysis. Ongoing CSN project.
- Development of methods and tools for analysis of plant operation during accidents. Ongoing CSN project.
- Pilot experience of PSA for non-reactor radioactive sources in a NPP (focused on spent fuel pools). Ongoing CSN-UNESA project to be completed in 2001.
- Methodology development and testing of the use of PSA in cost-benefit analysis. Ongoing CSN-UNESA project to be completed in 2001. A continuation project with several pilot cases being planned.
- Pilot case about the NRC “Option 2” for Risk-Informed Regulation. New CSN-UNESA project being planned.
- Improvement of the National Data Bank for better use in PSA. Ongoing UNESA project to be completed in 2002.
- Operating experience on fire events. CSN-UNESA participation in the OECD/NEA Fire Incident Records Exchange (OECD-FIRE) project being explored and planned.

- Level 2 analysis for external events. Selected project to be planned in the next future within the CSN-UNESA agreement.
- External events risk analysis (priority given to external events during shutdown operations). Selected project to be planned in the next future within the CSN-UNESA agreement.
- Level 2 analysis for Low Power and Shutdown PSA. Selected project to be planned in the next future within the CSN-UNESA agreement.

6.14 Sweden

6.14.1 PSA Related Research and Development in Sweden, as of October 2001

According to the principles of SKIFS 1998:1, it is a responsibility of the licensees to initiate, run PSA related research activities in Sweden. SKI finances via governmental research means research and development project, that is considered as important for the future development of safety at SKI and at licensees in Sweden.

The research program at the licensees:

- Each domestic NPP has its own research plan. The research projects that are decided and those under discussion are presented at each plant in a report called – the yearly safety programme.

The research program at SKI is divided into two main steps:

- Research plans for the short-term (3-5 years)
- Research plans for the long-term (6-10 years)

The whole SKI research program is huge, it covers project proposals from all kind of fields SKI has supervision for. Some of the these areas is presented mentioned here; Swedish East projects, Inspection,

Reactor safety, Thermal hydraulic, Fuel, Structural mechanics, Man-Technique-Organisation, Nuclear waste, Nuclear non-proliferation.

The SKI personnel working at the different offices are responsible for the reactor safety issues of the department and for the own specialised research fields and knowledge areas.

To better understand how SKI is working, an organisation chart is viewed in this section:



SKI supports needed domestic as well as international researching in the field of e.g., reactor safety. The main aim with the SKI research program is to deliver knowledge and experience to administrators at the regulatory body and to be used in our supervision of the licensees.

Identified fears about coming reactor safety issues and issues we are lacking knowledge and information about, are topics for the short-term research program. Areas in which we are lacking tools (e.g., models, methods, software) to be used in the work of supervision, can also be developed with research money.

PSA related research and development projects that are common for SKI and the licensees, are nowadays discussed in a common Nordic PSA Group (NPSAG). Representatives from the Nordic NPP operators and SKI are members of this group. See earlier presentation of the NPSAG in chapter 4.

6.14.2 Active PSA related projects

Sweden participates in the OECD/NEA, ICDE project	International CCF data exchange and CCF databases are created.
Sweden participates in the OECD/NEA, OECD-FIRE project	International fire data exchange and fire database is created.
Sweden participates in the OECD/NEA, OPDE project	International Pipe Data Exchange.
Nordic CCF working group	Modern and domestic CCF parameter values are established, from the ICDE CCF databases.

6.14.3 Planned projects

Shutdown analyses	Method development
End states of analyses in PSA	Method development
External events	Method development
High redundant CCF	Update of old CCF parameter values
Living PSA	In the planning process
Norms and rules applicable for PSA	In the planning process
Quality demands on PSA in risk based applications	In the planning process

6.14.4 Under discussion

How to use the high schools better in the safety work Early planning process

6.15 Switzerland

Switzerland is an active member of various international programs on risk assessment. For example, HSK participates in some OECD working groups and in a European Union Concerted Action program on the development of risk-based safety performance indicators. Several European regulators and utilities participate in this program, co-ordinated by GRS, Germany.

In the area of HRA, the utilities fund research at the Paul Scherrer Institut (PSI). At present, this research focuses on errors of commission and the assessment and quantification of human performance related to decision-making. In recent work, the CESA method addressing errors of commission was developed and applied in a pilot study performed with a Swiss nuclear power plant. This pilot application led to the identification of several scenarios centred on errors of commission (EOC). The preliminary results show that the EOC scenarios identified with CESA deserve careful consideration. Although safety insights were attained, these results underscore the problems with human reliability quantification and motivate further efforts dedicated to this challenge.

For EOC scenarios as well as the classical PSA scenarios involving the omission of operator actions, the dynamics of the plant and operator responses and the interplay between these responses call for a tighter integration of the human performance analysis within the PSA. Accordingly, a second area of PSI's research is aimed at simulation-based, dynamic safety assessment tools and methodology. As human performance modelling, dynamic event trees, as well as system (plant transient) simulation have to be integrated, co-operation is central in this work, in which the PSI contribution emphasises the human side of the problem.

Looking forward, fire PSA, uncertainty calculation, and the reliability analysis of digital safety protection systems are additional areas of particular interest.

6.16 United Kingdom

In the UK, there is a Health and Safety Commission Co-ordinated Programme of Nuclear Safety Research which addresses generic nuclear safety issues identified by NII. This programme of work is described below.

6.16.1 *Validation of PSA*

Fault tree analysis consistency: The aim was to investigate the variability that analysts could introduce into the fault trees that are produced as part of the PSAs for the gas cooled reactor and form a view on their significance. The work included a benchmark comparison of three independent fault tree analyses, a literature review of world-wide research and a questionnaire to analysts on their approach. Consideration is now being given to whether there are sufficient similarities in systems across all of the reactors to allow a standardised approach to be used and to investigate how guidelines for fault tree modelling could be developed. [Ongoing]

PSA data validation: The work focused on the additional validation requirements introduced by different PSA applications. The applications studied were determination of minimum safety related plant, allowable outage times, retrospective calculation of outage risk and maintenance optimisation. The final report provides a clear example of how to identify the relevant validation requirements.

As a result of this work, it was recognised that there was no overall strategy or specifications for what was needed to validate various applications of PSA. Further work was carried out to identify the validation needs for various applications of PSA. The project developed a list of PSA applications under three headings: safety assessment; operational monitoring; and operator training. It was concluded that the licensees had procedures to monitor and record operational events and to monitor and record component failures and maintenance activities but not to provide a controlled version of the PSA model and data, and for comparing recorded data with the assumptions and data used in the PSA.

Accident sequence precursor analysis: An approach to precursor analysis has been developed and applied successfully. The benefits are that the analysis allowed the events to be ranked so that attention could be focused on those events which are probabilistically more significant and allowed the major contributors to the significance of these events to be determined so that recurrent problems can be identified. However, it is not clear that there are any additional benefits at the present time over the existing UK operational feedback processes, which already take advantage of knowledge of the plant design and operation and of the results of the PSA.

6.16.2 PSA Methods

Comparison of PSA techniques: Two approaches to PSA exist: i) the use of FMEA to identify initiating events and event tree/ fault tree analysis to identify the resulting fault sequences and ii) HAZOP to identify the hazards followed by a HAZAN in which a fault tree is constructed to identify the likelihood that a particular hazard consequence would arise. The former approach is usually used for reactors and the latter for nuclear chemical plant. The aim of the work was to identify the pros and cons of the two approaches. The conclusion drawn is that the major differences in the approaches are in the identification stage and guidance is given on the preferred methodology for particular circumstances.

Sensitivity studies: Sensitivity studies are often performed by looking at changes in the risk resulting from changes in the failure probability of components one at a time. However, it is also important to identify the changes in the risk that could occur if several component failure probabilities were changed at the same time. To address this issue, an algorithm has been developed for combining the importance and uncertainty associated with fault tree basic events both individually and in combinations. In addition, acceptance measures have been developed and guidance provided on the choice of data distributions.

Worker risk: NII's Safety Assessment Principles require that an estimate is made of the risk to workers from accidents. Although this has been done for nuclear chemical plant, it has not been the case for reactors. A specific methodology for reactors has been developed and a pilot study carried out. This approach is now being routinely used for assessments at other gas cooled reactor stations.

Capability of methods for specified applications: A review was carried out to consider those aspects of PSA methods which it might be appropriate to develop further. This was based on a knowledge of the current state of risk analysis in the nuclear industry and other fields. This identified that further work was required in the following areas: the format of the PSA to ensure the maximum benefit to stakeholders, the significance to the PSA of compliance with site procedures, and the consistency of plant specific initiating event and failure rate data.

Common cause failure across systems: There is the potential for CCF to occur for similar components in different systems – for example, motor operated valves, relays etc. which could be the same in the main and emergency feed systems. The aim was to develop a way of considering these effects across systems. The approach considered dependent failure environments and the conditional probability of failing in that environment.

Significance to the risk of non-compliance with site procedure: The aim was to determine the significance of non-compliance with site procedures which could lead to operator errors of commission; incorrect setting or calibration of instrumentation, control or protection systems; and failure to comply with Operating Rules and other operating restrictions. This used operating experience as the basis for determining the possible means of lack of compliance with site procedures and the associated frequency. The impact of these types of problem on the initiating event assumptions or protection system performance and reliability can then be determined and hence the significance to the risk as a whole.

Reliability of repair: Historically, safety cases/ PSAs do not make any claims on the repair of safety systems which may have failed. Hence, if a component fails on demand, the reliance is placed on redundant or diverse components to fulfil the required safety functions. However, for the UK gas cooled reactors, provided the reactor has successfully tripped and shutdown, there are long time scales available before fuel would be damaged and large radiological releases occur. This means that there is scope for being able to repair failed component, as has occurred during actual plant operation. In addition, there is also the possibility of carrying out repairs to components which have failed as part of the initiating event,

where such repair can mitigate the radiological release – for example, blocking up a breach in the pressure circuit. The aim of the work is to explore the possibility of quantifying the benefits that can be claimed for repair, primarily in terms of the confidence which can be placed on claims for repair, and including this in the PSA thus giving more realistic estimates of the risk. [Ongoing]

Review of international standards for PSA: A review of the national and international standards for the production and maintenance of PSAs is being carried out to identify whether they contain any methods or data which could impact on PSAs carried out in the UK. [Ongoing]

6.16.3 *PSA Applications*

Reliability/ conditioned centred maintenance: There is a movement away from fixed interval maintenance towards reliability or conditioned centred maintenance. Although this has clear economic advantages to the operator, it is unclear what effect it will have on component reliability. The aim of the work was to determine how maintenance practices are changing, identify the economic and safety benefits and disbenefits, and suggest how reliability models may need to change to better reflect current maintenance practices. A key issue is to identify circumstances under which undergoing maintenance actually harms safety, and how such harms can be quantified. In parallel with the research, Nuclear Electric were developing their own approach to maintenance optimisation and a streamlined form of RCM has now been adopted.

Extension of the scope of PSA to include all hazards: The aim was to consider the various approaches to quantifying the risk from hazards and to recommend the appropriate method for use with regard to the safety benefit and the cost associated with the technique. The major part of the work focussed on the seismic hazard since the PSAs carried out in many countries world-wide address this to a more detailed level than is currently done in the UK. The use of the methods recommended by the contractor are the subject of formal discussions between NII and industry.

Feasibility of Event Based Seismic Risk Criteria: The aim was to consider the feasibility of setting seismic risk criteria on a “per event” basis rather than on a frequency, “per reactor year”, basis. The way that the criteria are currently specified disregards the fact that if the seismic event is severe enough to affect the nuclear plant it will also present other (individual and societal) risks to the public which could be much greater than the nuclear risk and the aim of the work was to investigate the feasibility of specifying criteria for nuclear seismic risk based on a comparison with the direct non-nuclear consequences of earthquakes. The conclusion was that, although this was feasible and practicable, it is of questionable effectiveness. This is because nuclear consequences will predominate for severe earthquakes close to the nuclear power plant site. Thus it would not be possible to adopt a purely event-based approach in determining design features, so that nuclear consequences would be insignificant compared with non-nuclear effects, without imposing very restrictive requirements. The contractors concluded after the first year's work that the event-based approach is not recommended as an alternative to the current frequency-based approach.

Comparison of deterministic and probabilistic based Operating Rules: The work compared the effectiveness of deterministic Operating Rules and Identified Operating Instructions in controlling the combination of equipment outages that could be allowable at any time. This concluded was that, although there was a degree of consistency between the two, it was possible that the deterministic operating rules could allow combinations of equipment to be taken out of service for maintenance at the same time leading to a high point-in-time risk. This has raised the concern that deterministic criteria alone may not be adequate to control the risk and to ensure that it is ALARP.

ALARP - non-risk based factors: The legal requirement in the UK is that risks should be reduced to a level which is as low as reasonably practicable (ALARP). In recent years, the availability of PSAs has led to a greater use of quantitative ALARP arguments and cost-benefit analysis (CBA). However, qualitative factors such as good engineering practice and deterministic safety considerations also need to be taken into account. Guidance has been prepared which has proposed a systematic framework for qualitative ALARP decisions.

Time at risk - acceptance criteria: There may be periods during plant operation when the point-in-time risk is high. The aim of the work was to investigate suitable acceptance criteria for deviations from the baseline risk.

New approaches to decision making in conditions of uncertainty: There are a number of methods available for representing and solving the risk decision problem including new methods such as belief networks and influence diagrams, and more established methodologies such as decision tree analysis and dynamic programming. Methods are also available for eliciting uncertainty and building it into the decision making process. The aim of the work was to consider how these approaches could be used to improve the applications of PSA results. The conclusion reached was that they did not offer any advantages over methods currently in use.

Seismic PSA - application to gas cooled reactors: In the UK, the seismic assessment of nuclear plant has been performed on a deterministic basis - that is, it has been shown that the design of the plant is sufficiently robust to survive a defined event and that essential systems survive with an integrity adequate to maintain the safety of the plant. However, the recent trend world-wide has been to supplement this deterministic assessment by carrying out seismic PSAs. The aim is to consider the current state of the art world-wide on seismic PSA and on the feasibility of carrying out such analyses for gas cooled reactors. One of the inputs to the work is the "State-of-the-Art Report on the Current Status of Methodologies for Seismic PSA" produced by Bob Budnitz under contract from NEA.

Phase 1 of the work concluded that it is feasible to generate fragility relationships even for the most complex structures and components in gas cooled reactors and worked examples of how this can be done are presented and Phase 2 proposed a methodology for seismic PSA that could be used for gas cooled reactors. This issue is now being progressed in the context of the Periodic Safety Reviews.

Fire PSA: PSAs which have been carried out world-wide have shown that internal fire can give a significant contribution to the risk of a core melt or of a release of radioactivity from a nuclear plant. In the UK, the internal fire safety case has been based primarily on meeting deterministic guidance on fire protection best practice backed up by limited probabilistic arguments based on initiating event statistics, assumed fire consequences which were generally considered conservative, and reliance on survival of sufficient reactor protection plant to satisfy probabilistic targets

The aim of the work was to determine whether there may be safety benefit in improving the approach to analysis of the internal fire hazard. A comparison was made between the current state of the art of fire PSA to the approach taken in the UK and consideration was given to the benefits and practicability of adopting the state of the art approach for the UK gas cooled reactors. The conclusion drawn was that there were a number of practicable improvements which could be made in terms of the UK fire PSA but recognised that because the state of the art fire PSA was subject to gaps, weaknesses and large uncertainties, application of the fire PSA should be limited to assessment of the relative benefits of improvements in the robustness and completeness of the safety case.

Review of risk management arrangements: The aim of the work was to document and categorise the approaches to the management of risk at UK nuclear power stations. For each category there would be a more detailed review for each station to identify good practices and to consider whether any improvements could be made. The review would extend to consideration of the arrangements for monitoring and assessing plant availability, and those that ensure that actions are taken to keep risks as low as reasonably practicable.

ALARP decisions database development and maintenance: Nuclear Electric has developed guidance on the interpretation of the ALARP principle, in terms of qualitative factors and on cost-benefit analysis, to assist in decision making on whether risks are as low as reasonably practicable and whether any proposals for enhancement will result in cost-effective improvements in safety. Since, the approach relies on engineering judgement, there can be disagreement between stakeholders on the relative weightings that should be given to the various qualitative factors for a particular problem. The availability of information on previous ALARP cases is considered to be potentially helpful in assisting judgements on new ALARP problems. The aim of the work is to construct a database which contains ALARP assessments for all gas cooled reactors. This will allow safety case authors and assessors to consider the results of any previous ALARP assessments which may have considered similar issues, and use this to support decision making. [Ongoing]

6.16.4 *Derivation of Data for PSA*

Zero failure data: One of the problems in PSA is how to derive a component failure rate from a data set with no failure events. Existing PSA had had used failure rates of $1/T$ and $0.33/T$ (where T is the number of component years of operation covered by the database). The aim of the work was to review the various models that have been used to date and to recommend an acceptable method. The conclusion is that the 'best estimate' failure rate is $0.55/T$ and this has been recommended as the standard for use in the UK nuclear industry.

Effects of ageing: As components age, the effects of wear-out may affect their reliability. These effects have always been considered for short term operating cycles but rarely for longer periods (of the order of several years). World-wide there are now many reactors which are over 30 years old and approaching the end of their design life and an understanding of ageing effects is required so decisions can be made, both by regulators and utilities, regarding life extension. The aim was to develop classification systems and analysis methods to cover these longer term ageing effects.

The work carried out has addressed two main areas. Firstly, a data field framework has been developed for ageing classification, which requires detail beyond that currently collected by the utilities. Secondly, the results of the data analysis on a sample of existing data found no evidence of significant ageing effects, though this is not conclusive and may be simply a consequence of the data available. It is concluded that further research in this area is not worthwhile at present, but due to the strong international interest in this area, the position should be reviewed in the future. The recommendations regarding implementation of data collection systems should be considered by the licensees.

Data for standby systems: The failure rate for standby plant does not obviously depend on time as a time based failure mode is not always identifiable. Hence the use of a simple relationship: (probability of failure) = (component failure rate) x (time) is inappropriate in many cases and is pessimistic. The use of this relation could lead to unnecessarily high levels of testing for standby plant. The aim was to consider whether an alternative relationship would be more appropriate and to provide analysis for fitting other models from data. The work has considered several aspects of the derivation of data for standby plant and quantified the degree of pessimism in using the above relationship for a number of probability distributions. The contractor has also developed equations which can be used in data analysis.

Methods of combining generic and specific data: The aim was to investigate methods of combining generic and plant specific data - in particular, the possibility of using Bayesian techniques. This also considered the uncertainties which are introduced due to the use of generic data, which may be greater than the statistical uncertainties alone. This was overtaken by recent advances in data collection activities and analysis at reactor sites and hence is now being considered as a regulatory issue.

Relationship between testing and reliability data: It is recognised that the reliability of components may be a function of the test regime. In particular, there is the possibility that the test may cause unnecessary wear if the failure modes/ mechanisms are not properly understood or that the component may not have been correctly restored to its operational mode. In addition, the test may not fully model the duty required of the component in accident situations either in terms of the required performance or the conditions of operation, hence data derived from test may not be directly appropriate. The aim is to investigate, both through data considerations and theoretical development, how a strategy for the derivation of reliability data which can be used with high confidence in PSAs.

A report on the testing interval optimisation strategy was produced. The safety and risk of the overall plant relates to the best availability and reliability of sub-components. The report investigates the identification of a method of reviewing and optimising the testing regime of standby systems. A mathematical expression is derived for the unavailability, which distinguishes time-based and event-based failures. The report develops a corresponding test optimisation strategy, using gas turbine data as the example. The report concludes, *inter alia*, that reduction in testing can be justified if the failures are shown to be time-dependent rather than demand based; but that for single-train (non-redundant) systems with short test intervals, existing PSA results could be optimistic if unavailability associated with testing is ignored. The methodology developed still requires further development, verification and validation.

Loss of grid frequency: The frequency of loss of grid as an initiating event has been derived from the station operating experience. However the probability of coincident and consequential loss of grid, and the probability distribution of the time for which the grid is unavailable are also required in the PSA and it is unlikely that there are sufficient events to enable values to be obtained from normal operational experience. An analysis has been carried out which involved consideration of the grid configuration in the location of individual stations and local operating experience, as well as national statistics. But since they involve specialist analysis methods and data from a single external source (the National Grid), it is necessary to review this area generically and update the analysis to ensure that figures for the different stations have been derived on a consistent basis

The conclusion reached was that theoretical studies of loss of off-site power (LOSP) have tended to under-predict in comparison with historical frequency because the effect of different weather states and salt pollution have not been adequately represented. In addition, the effects of human error at some sites needs to be monitored as it can be a significant factor. Long duration LOSP events are considered usually to be caused by either collapse of the transmission system or permanent damage to transmission equipment local to a power station, probably caused by unusually severe weather. The most likely duration of a LOSP event under these conditions is considered to be between 12 and 24 hours though a duration in excess of 24 hours cannot be ruled out. Conclusions were also drawn on upper bound figures for loss of grid frequency for UK reactors both of short duration (up to 12 hours) and longer duration (more than 12 hours) and as a consequence of reactor trip. These results were supportive of data used in the existing PSAs and will be used in future updates.

Update of accident costs: The costs associated with accidents specified in the quantitative ALARP/ cost benefit analysis methodologies for gas cooled reactors were based on work carried out by the National Radiological Protection Board in 1992. The aim was to revise these estimates so that it takes into account such factors as changes in population, land use and costs. The scope of the work was to identify the input parameters and methods which have changed, update the data and methodologies to be used in deriving cost of nuclear accident consequences, and recalculate the costs of accident consequences. [Ongoing]

6.16.5 Human Action Representation

Quantification of human actions over long timescales: The aim of the work was to provide support for claims of recovery by the operator for situations in which there are long timescales available for action and the means of achieving the tasks are not well defined by operating instructions. An approach based on six influencing factors and using a back-up table has been developed and tested in actual situations. This approach is considered to be applicable to screening and initial evaluation but not for operator error qualification.

Representation of human error within PSA: Although the incorporation of the failure of human actions within PSA has become general practice within the UK, a variety of approaches have been used to identify the human errors and decide how to incorporate this into the PSA. Human error has been represented at a variety of levels i.e. task level, diagnosis and action levels etc. Different methods have also been used to take account of human error dependency.

The aim was to review current UK Nuclear Power Plant PSAs, identify the advantages and disadvantages of the approaches adopted, and develop guidance for assessors on the best practice for conducting the Human Reliability Assessment (HRA) aspects of the PSA. Benefits are seen in terms of establishing a fair comparison of results, achieving completeness of consideration, and gaining efficiency in the preparation of the PSA.

The approach adopted was to review existing PSAs to determine the strengths and the weaknesses of the HRA approaches adopted. In addition, interviews were conducted with the assessors involved in the development of the PSA. Reference was also made to existing guidance on HRA in PSA. This therefore was effectively the raw data pool describing HRA best practice in PSA and this information was translated into a format to enable contrasts to be made between the different HRA approaches for different PSAs. The contrast analysis identified those techniques and approaches that worked and were indicative of best practice across all PSAs.

The outcome is a set of guidance tables for each stage of the HRA as it feeds into the PSA. The focus of this guidance is on how human error is ultimately represented and used in the PSA, and the impact this human error representation can have on the PSA results. The guidance is therefore 'HRA-in-PSA' focused rather than purely 'HRA-focused', since the results of the HRA, to be effective, must operate 'through' the PSA. The guidance is correspondingly and necessarily aimed at the integration of all the human error considerations in the PSA, and how to render this human error information most useful for the PSA, to ensure that the PSA obtains accurate results, and derives sound risk reduction measures.

6.17 United States

6.17.1 *PSA Information Exchange - Co-operative Probabilistic Risk Assessment Research (COOPRA)*

COOPRA is an international organisation formed for the purpose of sharing PSA information and to facilitate the efficient development and use of needed PSA tools. The members of COOPRA are the U.S. Nuclear Regulatory Commission (NRC) and those organisations, which have entered into a bilateral, co-operative PSA research agreement with the NRC. The COOPRA organisation involves a Steering Committee, a Secretariat, and multiple issue-oriented working groups.

The steering committee consists of the COOPRA Principals (representatives from each member country) plus COOPRA Working Group Chairs. The committee:

- Decides on R&D needs to be addressed by COOPRA.
- Facilitates the co-ordination of PSA R&D activities.
- Initiates working groups and oversees interactions among and within working groups.
- Decides on the scheduling of future COOPRA general meetings.

Steering committee members are responsible for the dissemination and collection of information from and for interested parties in their own countries. For example, in the U.S., the Office of Nuclear Regulatory Research will collect information from other NRC offices (NRR, NMSS) and from industry research groups (e.g., EPRI), and will disseminate information to these same bodies.

The responsibilities of the secretariat, currently sponsored by the NRC, include:

- Facilitation of the establishment of working groups.
- Organisation of COOPRA general meetings.
- Management of communications between the steering committee and working groups.
- Maintenance of the COOPRA information base.

The COOPRA “information base” is a repository of information and associated processing tools needed to support COOPRA activities.

The working groups are chartered to work on specific PSA research issues. General working group responsibilities include:

- Develop recommendations for specific R&D collaborations including objectives, approaches, milestones, and responsibilities.
- Facilitates collaborative R&D activities.
- Report on progress and results at COOPRA general meetings.
- Provide input to COOPRA information base.

The following countries are currently members of COOPRA: Argentina, Brazil, Canada, Croatia, Czech Republic, Finland, France, Germany, Hungary, Italy, Japan, Lithuania, Mexico, Russia, Slovenia, South Africa, South Korea, Spain, Switzerland, Taiwan, United Kingdom, United States.

For more information about COOPRA visit: <http://coopra.inel.gov>

6.17.2 Developing and demonstrating methods that improve existing techniques or fill gaps in the current state of PSA technology

6.17.2.1 Human Reliability Analysis

Recent research and development work conducted in the area of human reliability analysis (HRA) has focused on the development of A Technique for Human Event Analysis (ATHEANA). ATHEANA is an HRA method aimed at addressing the issue of scenario-specific context and a particularly challenging topic in HRA: the treatment of errors of commission. ATHEANA's underlying premise is that significant human errors occur as a result of a combination of influences associated with plant conditions and specific human-centred factors that trigger error mechanisms in the plant personnel. This premise requires the identification of these combinations of influences, called the "error-forcing contexts," and the assessment of their influence. Recognising that the ATHEANA development process has made sufficient progress to allow its application in actual regulatory applications (e.g., the analysis of pressurised thermal shock scenarios in support of a potential change to 10 CFR 50.61 [ref. #8.2.17.13]), and that the NRC has a broad range of HRA research and application needs which need to be addressed, the NRC staff has developed an HRA Research Program Plan to guide its efforts in the next few years.

Continuing research and development work in the HRA program focuses on finalising the ATHEANA quantification process (including uncertainty); collection and analysis of human reliability data sources; development of HRA guidance for reviewers; the treatment of latent errors in PSA; extension of advanced HRA concepts to include ex-control room activities, low-power and shutdown conditions, long-term recovery actions, and severe accident conditions; and development of formalised HRA methods for screening, cognitive modelling, and crew modelling. Application-oriented tasks in the program provide integration of advanced HRA methods into topics such as pressurised thermal shock, fire risk analysis, steam generator tube failure under severe accident conditions, ageing cables in nuclear power plants, nuclear materials and waste PSA, synergistic effects on safety, and upgraded/advanced control room layouts.

6.17.2.2 Fire Risk Analysis

The results of numerous fire risk assessment (FRA) studies and the experience gained from actual fire events show that, depending on the design and operational characteristics of a particular nuclear power plant (NPP), a fire can be a significant or even dominant contributor to plant risk. To support its initiatives to increase the use of risk information in regulatory decision-making, the U.S. Nuclear Regulatory Commission (NRC) therefore needs to ensure that this risk contribution is well characterised and understood. In particular, a number of ongoing NRC fire protection-related activities, including the development of a risk-informed, performance-based alternative to the current fire protection regulations in 10 CFR Part 50; support of the revised reactor oversight program; and identification of fire protection vulnerabilities requires FRAs of sufficient quality and accuracy to allow the confident use of their results and insights in the regulatory decision making process.

When used in a risk-informed decision making framework, FRA provides a systematic, integrated method for evaluating the importance of fire protection issues. However, the current FRA state-of-the-art is not as

mature as that for assessing the risk contributions of many other important accident initiators in NPPs. To address the need for improved FRA, the Office of Nuclear Regulatory Research (RES) initiated a fire risk research program in FY 1998. The initial plan for this research program was issued in June 1999 and then updated for the 2001-2002 fiscal year [ref. #8.2.17.14].

Since the issuance of the plan, NRC has identified a number of areas (including the reactor oversight program) requiring additional fire research (including FRA research) support, the nuclear industry has indicated a number of regulatory applications where increased use of FRA (and associated staff review) is expected, and the Advisory Committee on Reactor Safeguards (ACRS) expressed concern with the breadth and depth of the planned research activities (in terms of their ability to support the NRC's move towards risk-informed regulation). The FRA research program supports the NRC's Risk-Informed Regulation Implementation Plan (RIRIP), primarily by providing improved methods, tools, and data for calculating fire risk in support of risk-informed regulatory decision making. The research program also supports the RIRIP through the support of fire-protection related activities (e.g., the development of alternative fire protection standards for NPPs).

6.17.2.3 Considering Ageing Effects in PSAs

The incorporation of the effects of the ageing of structures, systems, and components (SSCs) is one of the activities in the NRC research program in PSA. The emphasis in this work is on passive SSCs because the ageing of these SSCs is expected to dominate the risk from the ageing of SSCs. Active components are assumed to be replaced or overhauled before their ageing becomes an important contributor to risk. The need for the use of reliability physics models to estimate the failure probability of SSCs arises because of the lack of failure data. A feasibility assessment using reliability physics models has been completed [ref. #8.2.17.15]. In this feasibility assessment, the effect of flow accelerated corrosion of piping on the core damage frequency of a nuclear power plant was evaluated. The piping being considered was that of the main feedwater system of a PWR, and pre-heater steam generator piping. Current work involves estimating the risk from in-containment instrumentation and control cables failures in the harsh environment after a loss of coolant accident (LOCA), and how this risk varies with the ageing of the cables.

Because the use of reliability physics models is resource-intensive, an important part of the method is the selection of the components whose failure probabilities are to be estimated by reliability physics models. Importance measures are used here. In-containment instrumentation and control cables in redundant trains of a system are exposed to the same or similar environments during normal operation, and may be exposed to the same or similar harsh environments after a LOCA. As a result, the likelihood of failure of the cables in one train of a system after a LOCA, given failure of the cables in another train of the system, may be high. Consequently, it is appropriate to use importance measures for the sets of cables associated with the redundant trains, rather than single component type importance measures. An importance measure consisting of an approximate estimate of the conditional frequency of core damage, given failure of the cables, times an approximate estimate of the probability of failure of the cables, seems appropriate. The reliability physics models for estimating the failure probabilities of the cables are based on existing models of cable ageing such as the Arrhenius model for ageing due to temperature. Parameters in the model are assigned distributions in order to estimate the failure probability of the cable. The model also includes the random variation in the containment environment after a LOCA, due to randomness in the size and location of the LOCA. Once the failure probabilities as a function of plant age are obtained, they are input into a PSA calculation. Condition monitoring techniques and ageing management strategies used throughout the lifetime of the reactor could modify the PSA calculations.

6.18 European Commission Joint Research Centre

The European Commission funds and performs research in the Nuclear Safety within the multi-annual Euratom Framework Programme. In particular of EC Directorate General for Research is responsible for defining, funding and co-ordinating the so-called indirect research actions, which are performed by the member states of the European Union. The research programmes are open also to many non EU states on the basis of special arrangements and conditions.

The European Commission Directorate General Joint Research centre is responsible for the definition and implementation of the so-called direct research actions, which are directly performed on its research establishments and directorates. In addition JRC is allowed to participate in indirect research projects obeying to the same rules of applicable national laboratories and institutions.

Information of the EC funded projects can be found on the CORDIS web site (<http://www.cordis.lu/en/home.html>), which includes information for both direct and indirect research activities, in all domains of application including Nuclear Safety. The advancements and results of EC indirect research projects in nuclear safety are regularly presented at the EC FISA conferences [1].

Hereafter the JRC research activities on Probabilistic Safety Assessment in the Nuclear field, or methodologically related are shortly summarised.

The research line “Harmonisation of safety methodologies and procedures” mainly deals with harmonisation of methods for safety assessment, focusing on methodological issues. Benchmark exercises on critical aspects of Probabilistic Risk/Safety Assessment (PRA/PSA) have been the main activity in the recent last years. Other activities relate to code development and maintenance. Participation and follow-up of international activities and community programmes on reactor safety are also ensured. Other methodological work in the area of risk assessment, including methodological aspects, is carried out in non-nuclear domains of applications.

EC JRC Ispra has organised and co-ordinated several successful international benchmark studies in the last 15 years (Systems Reliability, Common Cause Failure, Human Factors, Event Sequence), more recently on expert judgement techniques in level 2 PSA [2-4] and, currently on safety evaluation of computer based systems [5].

The Benchmark Exercise on Expert Judgement Techniques in PSA level 2 (BE-EJTs) has been an international project aiming to collect information about the use of structured expert judgement techniques among level-2 PSA researchers and practitioners and to compare methods and results. A network of ten different institutions (EC-JRC-ISIS (I), UPM (E), GRS (D), STUK (SF), NNC Ltd., (UK), AVN (B), ENEL (I), UNESA (E), NCSR DEMOKRITOS (GR), HSK (SW)) was partially supported by a EC DG Research concerted action and tackled, from 1995 to 1999, the probabilistic assessment of two prototypical severe accident issues, with different structured expert judgement methods. The problems considered were fuel coolant interactions (Phase 1) and the event of hydrogen deflagration/detonation in a pressurised water reactor system of evolutionary design (Phase 2) [2-4]. In addition to the co-ordination of the project, JRC-ISIS developed a new approach, KEEJAM, to expert judgement based on knowledge engineering techniques. The JRC approach KEEJAM was applied in both phases of the benchmark project.

The objectives of the Benchmark Exercise on Safety Evaluation of Safety Critical Computer Based Systems (BE-SECBS) are concerned with the development and implementation of a comparative evaluation of existing methodologies in use in the nuclear field among EU regulators and technical support organisations, tackling the problem of assessing safety-critical computer-based systems, with particular attention to the software part [5]. In this project, Framatome ANP (D) provides the reference case study.

To this aim FANP provides the requirements and functional specification of a limited number of safety functions and performs the design and implementation. FANP proprietary tools for automatic code generation and documentation will be employed. The source code and the documentation concerning all the software lifecycle phases will be made available to the assessor partners, namely STUK (SF), VTT (SF), ISTec (D) and IPSN (F), who will be involved in an independent assessment activity. It will be the role of JRC to design proper metrics to compare the assessment methodologies proposed and applied by the assessor partners and to actually perform the comparison between the proposed assessment methodologies.

Another activity in which JRC is currently involved is the project RMPS [6], Reliability Methods for Passive Safety Functions, aiming to develop reliability and PSA methods for the assessment of passive systems. The project is co-ordinated by CEA, other partners are CIRTEN (networking Italian universities), ENEA (see also section 5.8), GRS and Technicatome.

The further development and documentation of the ASTRA code [7] for reliability analysis, based on a Binary Decision Diagram Technique, is carried out in synergy with other research activities of JRC. A module for Event Tree analysis and a module for Common Case Failure have been recently completed and are currently under testing. These modules add to the existing ones for fault tree analysis, for time dependent analysis and for sensitivity analysis. The DYLAM-3 [8] code for dynamic reliability analysis, developed by JRC in the past, has been maintained and has been distributed to various institutions in the frame contract or collaboration agreements.

A Prototype database for supporting level 2 PSA studies have been developed in the recent past in collaboration with NNC (UK), which was the project co-ordinator, GRS (D), EdF (F) and Vattenfal (S), with reference to hydrogen generation, distribution and combustion phenomena. The pilot PSA database on hydrogen that has been produced within the project includes some 270 references related to hydrogen production, distribution and ignition and reduces the risk that major pieces of work have been missed and ensures that obsolete information has been identified as such [9]. The data base is distributed on a case by case request.

A reliability data collection has been also produced on the basis of EdF operational data and of EdF and JRC expertise in data collection and treatment. The data have been collected in the data-book EIReDA 1998 (European Industry Reliability Data Bank), jointly published by EdF and JRC. The book presents reliability parameters for 133 equipment types such as pumps, valves motors, sensors etc [10]. The estimates derive from operational and failure data collected from 1975 to 1995 in French NPP and are compared with other data collected, among the other, within ESReDA projects. The Data book can be ordered to the Foundation For Research & Technology, Crete University Press, P.O. Box 1527, 71110 Iraklion Crete, Greece. The book also contains theoretical methodological appendices about the Bayesian estimation procedures. The data bank is also available on a PC version.

Work on risk-informed in service inspections is also being done by JRC, also in connection with the activity of the European Network for Inspection Qualification (ENIQ): EURIS project etc.

Other research activities are being carried out on Human and Organisational factors (previously focused on nuclear safety, now mainly focused in the transport and aviation safety domain), uncertainty analysis, decision support systems etc.. PSA activity is also done in other domain of application such as in the areas of technological and natural hazards. Summaries of activities can be found in the annual reports of the JRC institutes.

CHAPTER 7 – PSA PLANT BASED MODIFICATIONS

Following up on the report produced in 1997, this chapter presents information on insights that have been gained and the role PSA has had in safety decision-making.

7.1 Belgium

For all NPPs, results indicate that the risk during non-power states is considerable compared to the risk during power operation.

Modifications (both hardware and procedural changes) have been proposed and implemented in the framework of the periodic safety reviews or in the framework of major modifications to the installation.

7.2 Canada

7.3 Czech Republic

Risk management is understood as an important part of operational decision making policy in which Probabilistic Safety Assessment is playing a key role. A lot of safety improvements have been made in Dukovany NPP since 1991 with aim to decrease units risk level. PSA insights have helped significantly in this still ongoing systematic process, which helps in Dukovany NPP achieve a comparable risk level of WWER unit with “western” design (CDF for FP operation decreased by factor of 15 within last 6 years).

Relocation of emergency feedwater collector and feeding heads, and implementation of new EOPs, including all new human post accident interventions, previously identified by PSA, are the plant modifications with the most significant influence on unit risk level. Implementation of additional PORV to be used for F&B and cold overpressure protection, improvements of ECCS, I&C, electric power systems etc. represent some other important safety measures PSA insights have been used for in decision making process.

7.4 Finland

7.4.1 Major risk informed plant and procedural changes at Loviisa 1 and 2

7.4.1.1 Internal Initiators

The original results of the Loviisa level 1 PSA (internal initiators) submitted to STUK in 1989 resulted in immediate measures at the plant, because one initiating event caused 73 % of the total core melt frequency (1.7×10^{-3} 1/a).

The dominating event was loss of cooling of electrical and control instrumentation room. The ventilation system of this room had only one train equipped with cooling unit. The assumption, made in PSA, that the control of whole plant is lost if the temperature exceeds the design limit of control instrumentation led to the aforementioned high core damage frequency. A quick demonstration, however, addressed that the air-cooling is needed only during the hottest summer days, which are infrequent in Finland. Most of the year the cooling could be managed by blowing the air also by two standby fans without cooling unit. A rapid review of the accident sequence assured as well that auxiliary feedwater system can be manually managed even though the automatic control would be lost. These corrections updated the core melt probability of the respective initiating event to 3.3×10^{-4} 1/a and the total core melt probability to 9×10^{-4} 1/a .

Instead of further analysis immediate actions were tackled to redesign the air cooling system and to install two additional 100 % air cooling units in each vital room. The redesigned system decreased the core damage frequency resulted from the loss of instrument room cooling to 1.2×10^{-5} 1/a and the total core damage estimate to 6×10^{-4} 1/a.

Improvements have been made in several other systems causing high probability core damage frequencies such as

- primary circulation pump seal system
- service water system
- minimum circulation of ECC system.

All the aforementioned systems suffered from design errors which could be eliminated by proportionally moderate efforts.

- The redesigned back-rotation prevention system and a new stop signal activated by too small seal cooling flow in primary circulation pumps and improved operator instructions for avoiding seal LOCA decreased the frequency of the respective accident sequence from 2×10^{-4} 1/a to about 10^{-5} 1/a.
- The redundancy of the service water system was improved by changing base states of a few valves. This change eliminated the total loss of service water system in case of pipe break and decreased the core damage frequency caused by the loss of service water from 1.3×10^{-4} to 1.9×10^{-5} 1/a.
- An important design error was found in ECC system leading to high frequency accident sequence. If the closing valves in minimum circulation lines fail to close on demand, it makes the sump line valves and suction line valves hunting back and forth due to the suction cycling between water tank and sump. The closing valves in the minimum circulation lines are replaced by more reliable type of valves in order to prevent the ECC water backflow to ECC tank. This change reduced the core damage risk from 5.4×10^{-5} /a to 1.4×10^{-5} /a in LOCA cases.
- Back up of the PCP seal cooling outlet valves by batteries reduced the seal LOCA contribution to the core melt in case of the loss of off-site power.

Several improvements have been made in emergency operating procedures such as refilling of the ECCS tank in case of multiple steam generator tube ruptures, primary circuit pressure reduction with pressuriser and ATWS management.

Related to the steam generator tube and collector rupture the isolation of the steam generator can be made both at primary and secondary side. The isolation made at primary side interrupts the leakage with certainty but the reliability of the main isolation valves is questioned, due to sparse data.

In order to reduce the risks resulted from the tube and collector ruptures in steam generator the following backfits have been implemented:

- The reliability of pressuriser sprays are improved by installing new pipelines from ECC system to pressuriser sprays to back-up the normal pressuriser sprays from main coolant pumps.
- New protection signal activated by high water level in the steam generator (steam generator tube rupture) will close the main steam line and the main feedwater line and to stop the respective main cooling pump.
- Additional ECC water tank has been build up to maintain the volume of primary circuit in case of steam generator tube rupture.
- New protection system to control the level of radioactive substances in the secondary circuit has been assembled. This system is to alarm in case of tube ruptures in steam generator.

The aforementioned changes lowered the risks of steam generator tube ruptures from $1.6 \times 10^{-4}/a$ to $1.4 \times 10^{-6}/a$.

To improve the reliability of ECC system, minimum flow lines from the pumps forcing side to the ECC injection water tank have been replaced by new lines with heat exchangers leading from the pumps forcing side directly to the suction side. The failure of former minimum flow line valves could lead to refilling of the ECC injection water tank, alternating of the line-up of the ECCS suction between the tank and the sump, and possible additional valve failures.

7.4.1.2 Fires

- sprinkler protection of several safety significant cables have been improved.
- physical fire protection of several safety significant pressure air pipes has been improved.
- physical protection and sprinkler protection of hydraulic oil stations of turbine by-pass
- valves have been improved aiming to prevent the spreading of high-pressure oil leaks to the surroundings.
- Based on the results of Loviisa fire risk analysis, several cables of important safety systems have been re-protected. The several cable protections and the protection of turbines dumping valves' oil station lowered the fire risk roughly from $2 \times 10^{-3}/a$ to $1 \times 10^{-4}/a$.

7.4.1.3 Internal floods

- a wall was built to prevent the floods from spreading from turbine building to the lower rooms of reactor building where they can damage the cooling system of main cooling pumps and ECC system function.
- the drainage of cable room beneath the control room was improved to prevent the flooding water depositing on the floor and to exceed the acceptable loading of the floor. The corresponding improvement of drainage was made for the feed water storage tank level.
- ventilation system cooling pipes routing in control building has been have been moved to a safer location in order to protect the control room from flooding in case a break of the respective piping.
- High-pressure water jets from the breaks in the main feed water system could fail the floor on top of the control building. The pipelines routing on top of the control building have been replaced with pipes from better material and protected.

7.4.1.4 Harsh weather conditions

Sea vegetation can cause a blockage of chain basket filters in sea water channel

- To reduce the risk of filter breaks due to high pressure difference over filters and to prevent the consequent access of algae to the main circulating and service water system an automatic power and flow reduction system has been installed

Blockage of diesel generator air in-take by snow or freezing rain during a storm can result in a loss of emergency diesel system.

- To upgrade the reliability of DG system, dampers opening automatically on pressure difference were installed to enable the taking of the combustion air directly from DG rooms

7.4.2 Major risk informed plant and procedural changes at OL I and II

7.4.2.1 Internal Initiators

The original results of the OL level 1 PSA (internal initiators) submitted to STUK in 1989 resulted in some plant changes:

- TVO has improved the water level measurement system to prevent the water from boiling in the reference piping and to ease the surveillance test of the system, because for example the function of auxiliary feed water system is controlled by water level in reactor vessel.
- Mussels capture strainers were installed into the sea water cooling channels in order to prevent mussels of blocking the intermediate cooling and diesel generator cooling heat exchangers (weak link to risk assessment)

- In 1994 STUK required that the lower air lock will be kept closed during the refuelling outages when the maintenance of the main coolant pumps is underway, because the maintenance work can result in large bottom LOCA in the reactor tank. If large bottom LOCA takes place and the lower air lock remains unlocked, the coolant escapes out of the containment and prevents adequate core cooling function which leads to core uncover and core damage within few hours.
- the connections of the plant to the outside grid are upgraded by installing a new additional start transformer and improving the plant connections to the Hydro power plant

New EOPs have been made as follows

- refilling of the EFW tank and condenser
- cross connection of the diesel generators of neighbouring plant units
- manual depressurisation of the reactor tank from the relay room.

7.4.2.2 Harsh weather conditions

In the spring 1995 TVO PSA was revised due to two weather related phenomena which took place at Olkiluoto.

In February 1995 snowstorm blocked the air intake filter of air suction channel to diesel generators that stopped two diesels of running in surveillance test.

- To upgrade the reliability of DG system, dampers opening automatically on pressure difference were installed to enable the taking of the combustion air directly from DG rooms.

In January 1995 sub-cooled seawater blocked coarse bar screen in the inlet channel of service water system, which is vital to emergency core cooling systems.

- To reduce the risk coming from the crystal ice, a system circulating warm water to the intake of sea water channel has been installed. The system is to prevent crystal ice formation in the coarse bar screen and its blocking.

Modelling of these two CCF type of phenomena contributed to TVO PSA core damage frequency an increment no less than $1.9 \times 10^{-5}/a$. The total core damage frequency including all identified initiating events and changes made due to the regulatory review was $3.34 \times 10^{-5}/a$. The defence against the aforementioned type of external initiators has been introduced with respective plant changes which lowered the core damage frequency back to the almost preceding level.

7.4.2.3 Seismic

Seismic risk analysis resulted in some plant changes. The major contributions to seismic risk came from loose anchoring of diesel generator battery system and of some electronics cabinets.

- To reduce the risk, the battery system will be supported by surrounding frame that prevents the batteries falling down from their foundation. The electronics cabinets will be adequately anchored to solid structures

After the implementation of these plant changes the contribution of seismic events to the core damage probability is about $4 \times 10^{-6}/a$.

TVO submitted to STUK also the level 2PSA which showed that the average probability of large release (atmospheric release of Cesium-137 is more than 100 Tbq) is about 4×10^{-6} /a. The majority of the risk comes from early high pressure transients and the remainder mainly from low pressure transients and the shut down mode initiators.

7.5 France

7.5.1 Probabilistic studies related to the loss of redundant safety systems

These studies showed the need of complementary measures to achieve a satisfactory safety level. Specific procedures, called “H” procedures (H for “hors dimensionnement”, i.e. beyond design basis), if necessary implementing supplementary equipment, have been established :

- H1 procedure “total failure of the heat sink” ,
- H2 procedure “total failure of the steam generator water supply” (includes primary “Feed and Bleed”),
- H3 procedure “total loss of electrical supply” (led to the implementation of a gas turbine on the sites. Moreover a turbo-alternator was added with the aim of providing supply for water injection to the primary pump seals and power to instrumentation and control),
- H4 procedure “total loss of Low Pressure Safety Injection (LPSI) or Containment Spray System (CSS) during long term post-LOCA situations” (including interconnection between LPSI and CSS for mutual back up).

The H procedures are implemented on all the French PWRs (900 MWe - 1300 MWe and 1450 MWe series).

For the latest series of French PWRs (the 1450 MWe series - also called N4), the Safety Authorities required at the early stage of the design to study a more complete list of multiple failures events and to perform a probabilistic demonstration. A result of this requirement is that, in order to fulfil the safety objectives for anticipated transients without scram (ATWS), due to difficulties for quantifying software reliability, a diversified scram system was deemed necessary.

7.5.2 Dominant shutdown sequences

In the 1990 PSAs, the CMF was particularly high for a loss of Residual Heat Removal System (RHRS) during mid-loop operation because only a short time is available for the operator to take any action. Other particular sequences initiated by a spurious primary coolant boron dilution were also identified.

These sequences were similar for 900 MWe, 1300 MWe and 1450 MWe series and led to modifications for all series.

7.5.2.1 Loss of RHRS during mid-loop operation

Immediately following the publication of the results, EdF proposed preliminary measures (level measurement, technical specifications (TS) leading to avoid the most critical situations, training of operators...).

After a more complete safety reassessment, definitive measures were proposed:

- improved level monitoring
- improved TS, Emergency Operating Procedures (EOP) and training,
- implementation of a vortex alarm,
- implementation of an automatic water make-up.

The Safety Authorities have considered that the assessed CMF was significantly reduced by these measures which should be rapidly implemented on all the plants.

7.5.2.2 Rapid boron dilution

Due to the potentially high consequences of this type of accident sequence (reactivity accident), an immediate corrective action has been taken by EdF: implementation of an automatic suction of the Chemical and Volume Control System (CVCS) pumps from the borated Refuelling Water Storage Tank (RWST) in case of reactor trip, in order to avoid the formation of an unborated water slug.

To perform an in-depth assessment of this kind of sequences, complementary studies have been undertaken including physical calculations, experiments and more complete identification of boron dilution sources. These studies led to complementary improvements of plant operation (operating procedures, technical specifications...).

7.5.2.3 Other shutdown sequences

The most important modifications decided recently on the French plants are related to the risk of cold overpressurisation during shutdown.

A threat of violent cooling of the vessel followed by a strong repressurisation of RCS has been identified in case of RHR break (Residual Heat Removal system) on French PWRs (their effect on the vessel being similar to Pressurised Thermal Shock). PSA –based analysis of such sequences have been done both on 900 and 1300 MWe series : the most important sequence corresponds to an inappropriate isolation of the RHRS during cold shutdown when the primary circuit is closed, which could lead to a rapid pressure increase with a low temperature in the primary circuit, and to a risk of reactor vessel rupture. This safety problem was due especially to weaknesses in the Emergency Operating Procedures.

The Safety Authority required to improve the situation and to analyse in more detail the risk of overpressurisation. Following this analysis, the utility proposals are, for all the plants, improvements of the Emergency Operating Procedures concerned. In particular the Emergency Procedures will require to align the circuits in order to avoid a complete RHRS isolation.

Moreover, for the 900 MWe series, a modification of the PORVs setpoint during relevant plant states, in order to reduce significantly the risk of vessel rupture, is proposed.

7.5.3 900 MWe series Periodic Safety Review

In the framework of the 900 MWe series Periodic Safety Review, the main following backfits are to be implemented:

- functional redundancy of the auxiliary feedwater system for all modes of operation (by the main feedwater system or the RHRS)
- improvement of the ventilation system (cooling of LPSI and CSS)
- diversification of the reactor scram function
- modifications which could mitigate the consequences of 6.6 kV switchboards common mode failures

7.6 Germany

In the past numerous plant modifications have been based on PSA insights. Already Phase B of the German Risk Study (1989) initiated the implementation of plant internal accident management measures in all German NPPs, including secondary-side and primary-side bleed and feed and containment venting. Plant-specific PSAs led to numerous backfitting measures in the respective plant (resp. plant types). In the following table typical examples for selected plants are compiled.

Main measures in German NPPs derived from PSA insights

Plant	Modification (Examples for measures)
Neckarwestheim 1	Secondary System: – Improvement of steam release for 100 K/h cool down Nuclear service water system: – Improvements to prevent / limit internal floods (automatic switch-off of pumps)
Brunsbüttel	Secondary System – Backfitting of main steam safety relief valve system (reduction of CCF probability by adding 4 motor driven valves to the already installed 7 steam controlled valves with 2 solenoid pilot valves for each main steam valve and by introducing a diverse trip signal using self-actuating spring loaded pilot valves with reduced “pneumatic load”)

Plant	Modification (Examples for measures)
Isar 1	<p>ECCS:</p> <ul style="list-style-type: none"> - Technical modification of ECCS to increase system reliability <p>Feedwater supply system:</p> <ul style="list-style-type: none"> - Technical modification of connected lines to improve mechanical resistance <p>Containment:</p> <ul style="list-style-type: none"> - Improvements of containment penetration assemblies in order to prevent impermissible pressure rise and to assure system integrity <p>I&C:</p> <ul style="list-style-type: none"> - Measures to adapt reactor protection system and I&C-system to current standards (KTA) <p>Plant Protection Systems:</p> <ul style="list-style-type: none"> - Technical and administrative improvements and additional tests on systems and components for fire protection, internal flooding (turbine building), lightning protection and earth quake protection
Philippsburg 1	<p>Feedwater Supply:</p> <ul style="list-style-type: none"> - final evaluation and comprehensive exchange of feedwater lines in the turbine building in order to fulfil material quality requirements and “basic safety criteria” (e.g. leak-before-break-criterion) <p>Secondary System:</p> <ul style="list-style-type: none"> - exchange of safety relief valves <p>I&C:</p> <ul style="list-style-type: none"> - Improvements of the reactor protection system (RPS) and the independent sabotage and accident protection system to protect from flash-over - Additional technical measures to improve air-cooling of RPS and I&C - Improvements of the quality assurance documentation of I&C-facilities - Improvements for detectability and limitation of neutron flux oscillations
Gundremmingen B/C	<p>ECCS:</p> <ul style="list-style-type: none"> - Installation of a diverse residual heat removal (RHR) and injection system (ARHR-system) - Additional shutdown line at the level of the feedwater line nozzles in a RHR train - Operation of HP injection pump without low pressure stage (demeshing of I&C actuation and installation of separate cooling for HP pump) - Reduction of interlocking time from 30 to 5 minutes to shut down the residual heat removal pumps and maintain the supply of coolant within the containment <p>Pressure suppression pool (PSP):</p> <ul style="list-style-type: none"> - Actuation of operational PSP cooling by all subsystem controls <p>Feedwater supply:</p> <ul style="list-style-type: none"> - Assurance of feedwater supply (RL system) after loss of the main heat sink (shut-off of RL system feeding after failure of PSP water level control only if at least one of the residual heat removal systems is working) <p>RPV depressurisation:</p> <ul style="list-style-type: none"> - Manual actuation of RPV depressurisation at high PSP temperature (60 deg C) only if RPV feeding is available <p>RPV feeding:</p> <ul style="list-style-type: none"> - Possibility of RPV feeding with control rod purge water, pump seal water, fire fighting system, raw water from river with service water system (AM) - Cross-ties of AC buses within the same unit and between units (AM)

7.7 Hungary

See other chapters.

7.8 Italy

Work in this area was performed in the nineties in the context of European Commission programs of technical assistance to eastern countries for nuclear safety: based on cost estimate and simplified PSA model for risk estimation, a prioritisation of modifications proposed for Smolensk plant was performed (ref. 10). More recently a reliability study was performed to support the licensing process performed by ANPA in regard of modification of research reactor (already mentioned in Chap. 4.8)

7.9 Japan

7.9.1 Accident Management Measures based on PSA

Utilities have implemented AM measures for all LWRs in Japan. An example of standard AM measures implemented in Japanese typical BWR and PWR is shown in Table 1 and 2. The effectiveness of the AM will be confirmed by PSA as a part of AM review by METI.

Table 1 Standard Accident Management Measures for BWR-5 implemented since 1996

Accident management function	Accident management measures
Shutdown of reactor	Alternate reactivity control
Water injection to reactor and containment vessel	<ul style="list-style-type: none"> - Alternate water injection (water injection to reactor and containment vessel using pumps associated with condensate makeup water system or fire fighting system) - Automatic reactor de-pressurisation (ADS logic addition)
Heat removal from containment vessel	<ul style="list-style-type: none"> - Alternate heat removal using heat exchangers of non-safety systems (drywell cooler or cleanup water system) - Recovery of failed components of RHR system - Hard vent
Supply of electric power	<ul style="list-style-type: none"> - Cross-tie of power supplies (480 V from adjacent plant) - Recovery of failed components of emergency diesel generators

Table 2 Standard Accident Management Measures for four loop PWR with dry type containment implemented since 1996

Accident management function	Accident management measures
Shutdown of reactor	Diversity of emergency cooling of secondary loop
Cooling of reactor core	<ul style="list-style-type: none"> - Use of turbine bypass system - Alternate re-circulation - Natural convection inside containment vessel - Alternate component cooling
Confinement of radioactive material	<ul style="list-style-type: none"> - Natural convection cooling in containment vessel - Water injection in containment vessel - Forced de-pressurisation of primary loop
Support function for safety function	<ul style="list-style-type: none"> - Alternate cooling of component - Cross-tie of electric power between units

7.9.2 PSAs as part of PSR

In the process of PSA for shutdown operation in PSR some modifications have been made, such as reinforcement of operating manuals at accident during shutdown operations, installation of ultra-sonic RCS level gauge, extension of AM measures (use of water for fire fighting and cross-tie of power supplies) to shutdown operation.

7.9.3 Trial Application of PSA to Allowed Outage Time (AOT)

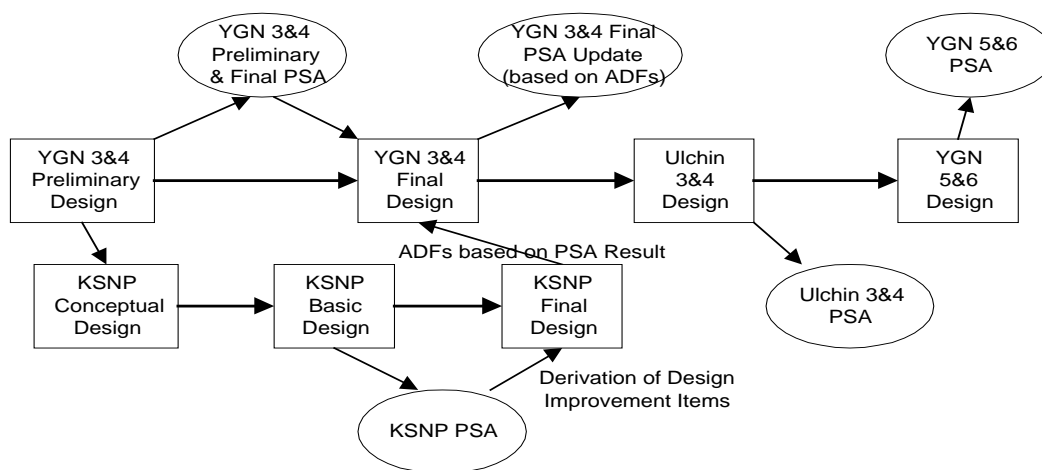
In the technical specifications revised in January 2001, AOT for main safety systems with redundancy has been set to be ten days, in principle. These have not always been determined based on PSA but NUPEC has confirmed the AOT does not lead to excessive risk increase through PSA as long as the current operating situation may be maintained.

7.9.4 Improvement in JNC Tokai Reprocessing Plant using HAZOP

In JNC Tokai Reprocessing Plant, the incident prevention system of each process has been investigated by using HAZOP, Failure Mode and Effects Analysis (FMEA) and Event Tree method. The incident mitigation system of each system has been also investigated by evaluating some design-based accidents. Based on these results, JNC has improved some plant equipment and instrumentation, and modified operation manuals and procedures.

7.10 Korea

As explained in Chapter 4, a number of PSA have performed in Korea including the plant in the design stage and in operation. Based on the PSA results, several items for plant modifications are recommended for the plant in the design stage and in operation. The major activities for plant modifications based on PSA results are focused on the design improvement of KSNPs. The design evolution process of KSNP based on the related PSA is shown in the following Figure.



(ADFs = Advanced Design Features)

Figure 1. Design Evolution Process of KSNP and Related PSAs

In addition, there are several plant modifications based on PSA results. These are summarised in the following table. The minor procedure changes are not included in the table.

As shown in Table 1, during the design review of YGN-3&4, we found that the conceptual design without primary bleeding has a weak function to remove the residual decay heat. The sensitivity analyses based on the PSA model were performed to provide the information for the decision making of installing safety depressurisation system (SDS) and third emergency diesel generator (Alternate AC: AAC). In conclusion, it was strongly recommended to add these new systems in order to mitigate severe accidents. The SDS is designed to be equipped with 2 trains and two 4-inch nozzles on the pressuriser. AAC, also, has a function to cope with a station blackout (SBO) accident, which will share between 3 and 4 units.

Recently, the improved KSNP (KSNP+) has been being developed to enhance the competitiveness of nuclear energy. In this project, PSA is used to reduce the unnecessary over-design of the plant while assuring the plant safety. Based on the PSA results, several systems have been optimised considering the construction costs.

PSA	Items	Contents	Basis	Status
YGN 3&4 Final PSA	AFWS Design Change	It is recommended that the design change of AFWS from cycling operation to flow modulation control.	Unavailability change: 5.13E-4 → 3.25E-6	Implemented (* described in detail)
	AAC*	It is recommended to install AAC to the site.		
	SDS*	It is recommended to install SDS to the primary system.		
Kori 3&4, YGN 1&2	RPS/ESFAS	Changes in AOTs/STIs	Slightly increase in CDF	Accepted
Ulchin 3&4 Final PSA	Aggressive Cool down	The clarification of EOP is recommended for the case of small LOCA with loss of HPSI.	Δ CDF = 38%	Implemented
Wolsong 2/3/4 Final PSA	SCS Test Interval	Reduction of the test interval of SCS from 1 year to 3 months is recommended.	Δ CDF = 30%	Implemented
	EWS Building	The reinforcement of EWS building is recommended.	Δ CDF = 22.5%	
YGN 5&6 Pre. PSA	Shutdown Alarm Panel	Improvement of Shutdown Alarm Panel is recommended. (Measurement and display)		Implemented (from YGN 3&4)

7.11 Mexico

There have been not many plant modifications in which the PSA played a significant role. Notwithstanding, it is important to note that PSA is not the only basis for deciding whether or not to perform a modification or backfit. The regulatory guide which established an approved methodology for changes to the licensing basis notes that the risk results and insights should be used in a complementary way with the deterministic analysis and evaluations in order to have an integral decision making process.

During the development of the Laguna Verde PSA level 1 analysis and as a result of the high contribution of the station blackout scenarios (loss of off-site power plus the failure of the emergency diesel generators division I and II), it was decided and implemented a cross connection between the diesel driven pump of the fire protection system with the reactor heat removal system (RHR). This connection provides an alternative way to inject water into the reactor vessel or to spray the containment during this kind of accident.

Another application in which the PSA results and insights played a role in the safety decision making was the approval of the utility request to increase 5% in the thermal power. Along with the deterministic analysis that supported the request, the utility was called for to perform an evaluation of the risk increase associated with the power increase, following the guidelines of the USNRC regulatory guide 1.174.

7.12 Netherlands

Major modification and backfitting programmes were announced around 1989, partly as a result of the accident at Chernobyl. A backfitting requirement was formulated for the existing NPPs. Although backfitting primarily addresses the design basis area, the beyond-design basis area and associated severe accident issues are also taken into account. The 'backfitting rule' also requires ten-yearly safety reviews. This requirement is included in the operating licences issued for both plants. At that time an important part of these ten-yearly safety reviews was a level-1 'plus' PSA (level 1⁺).

It became clear at a later stage that the plants needed to have new licences in order to put the major modification programmes into effect. As part of the licensing procedure, both plants were required to submit an Environmental Impact Assessment. A substantial part of this Environmental Impact Assessment was taken up by a 'full scope' level-3 PSA, including an assessment of the influence of the proposed modifications. This meant expanding the scope of the ongoing studies. These studies were completed early in 1994. Their findings were also communicated to the Dutch Parliament.

The scope of the PSAs was also extended in the light of review processes, interim findings of the PSA, changes in the state-of-the-art (e.g. assessment of the risks associated with low-power and shut-down states) and the broadening of the objectives.

In the early 1990s, these level-1⁺ PSAs were expanded to full-scope level-3 PSAs, including internal and external events, power and non-power plant operating states, human errors of omission and commission. The PSAs were expanded partly in order to comply with the requirement that the studies should be 'state-of-the-art' (i.e. non-power plant operating states and human errors of commission), and partly because of the licensing requirements associated with the ongoing modification programmes (i.e. an Environmental Impact Assessment had to include a level-3 PSA).

Because the PSAs were intended primarily to identify weak spots in the operation and design of both Dutch NPPs, they could be used to support the modification programmes and to alter them if necessary. As the NPP Dodewaard is closed now, only the Borssele PSA is discussed here.

In Table 1 an overview is given of the influences of the modification programme of the Borssele NPP on the TCDF and the contributing accident sequences in terms of initiating events.

In the current plant situation, 53% of the TCDF is due to internal events. Spatially dependent events (internal flood & fire) contribute 33% and External events contribute 13%. Internal events during Power Plant Operational States (POS), and spatially dependent events occurring in the Mid-loop POS dominate the level-1 results; both ca. 26%. In the old plant situation the internal events were still 76%, the spatially dependent contributed only 21% and the external events contributed only ca. 3%. These figures demonstrate clearly that the modification programme was quite effective for the internal events but less effective for the spatially dependent events and external events.

In both the old and current plant configuration a large percentage of the TCDF is contributed by a small number of cut sets. In both cases about 40 cut sets are responsible for 60% to 70% of the TCDF.

A good demonstration of the influence of the modifications on the level-1 outcomes is the reduction in the contribution to the TCDF of the very small break LOCAs (from 1.41E-5 to 1.18E-7). Before the modifications the dominant accident sequence was a very small LOCA followed by a success of reactor trip, high pressure injection via the volume control system, power available from off-site sources or diesels, feedwater to the steam generators and successful secondary cooldown. Late in the progression of events, failure of the low pressure residual heat removal system leads to failure to remove decay heat from

the core and eventually results in core damage. With a sequence frequency of $8.9E-6$ per year, it contributed 15.9 % of the total CDF (rank 1 sequence). After the modifications the frequency was reduced to $1.1E-7$ per year, which is a contribution of 4.2% to the TCDF (rank 7 sequence). In the old case the top four cut sets (rank 3, 4, 7 and 8 of the total TCDF cut set list (cut set list involving all initiators and all POS)) involved failure to isolate the inundation tanks from the suction of the low pressure pumps. Failure to isolate the inundation tanks after switchover to recirculation leads to failure of the low pressure pumps. The frequency of the sum of these four cut sets was $8.3E-6$. Due to the installation of the check valves in the inundation lines, which prevent backflow from the sump to the inundation, tanks and failure of the low pressure pumps these cut sets almost disappeared from the sequence cut set list. On the other hand, these extra check valves slightly increased the system unavailability. Therefore, this is a good example that system unavailability's of the safety systems don't provide the complete story of TCDF improvement.

In the non-power situation, the mid-loop POS dominates because of the reduced inventory and because there is only one manually actuated single system to act as a redundancy for the low pressure RHR/ LPIS, namely the Reserve Cooling System TE. Automatic actuation of the bunkered primary reserve injection system TW has a large impact on the accident sequence because it extends the time window for operator action and recovery.

In the current post-modification plant state, the total core melt frequency of $2.83 E-6$ per year is governed by sequences with the containment initially intact (92%). The remaining sequences are classified as bypass sequences and include interfacing system LOCAs (1.3%), SGTR (1.4%), containment isolation failure sequences (0.1%), and external sequences which directly fail the containment (4.8%) of the non-bypass sequences. Transients (69%) and small LOCA (19%) are the dominant sequence types. Transients most likely lead to low pressure (< 9 bar) or intermediate pressure (9 - 134 bar) core melt (62% of transients - low, and 8% of transients - intermediate) whereas for small leaks low pressure core melt is dominant (72% of small LOCA sequences) followed by intermediate pressure (27%). Station blackout contribution to the core melt frequency is minor (1.2 %). For 72% of the high-pressure transients, systems would become available to inject water in the core with depressurisation. This percentage is even higher for LOCA sequences (approximately 100%). The major contributors (in terms of plant damage states, and not in terms of initiating events as in table 1) based on frequency are as follows:

- 36.2% Low pressure transient with the reactor vessel open, where injection is possible but containment heat removal is not available (Mid-loop operation).
- 9.4% Small LOCA with low system pressure, where injection is possible but containment heat removal is not available
- 6.5% High pressure transient, where injection is possible but containment heat removal is not available.
- 5.9% Transient in the fuel storage pool with no injection and no containment heat removal
- 4.7% Containment failed by initiating event at or near time of reactor shutdown
- 4.5% Small LOCA with intermediate system pressure, where injection is possible and containment heat removal is available.

A major difference with the 'old' plant situation is significant reduction of the medium pressure core melt transient type scenarios, from 37% to 8%). At these intermediate pressures, low pressure injection is not possible. The main reason for this reduction is the improved capability for secondary cooldown and improved high pressure primary injection capability.

Within the framework of the Borssele PSA, a qualitative assessment was made of the Errors of Commission (EOCs) with potential serious consequences. The assessment of the EOCs during power states is based on the 'HITLINE' method, which was developed at the University of Maryland. The method which was used for the analysis of the EOCs during the low-power and shut-down plant operational states closely resembles the methods which form the basis of current developments in the ATHEANA project (A Technique for Human Error Analysis; NUREG/CR-6265, NUREG/CR-6093 and NUREG/CR-6350), which has been developed for the USNRC.

Table 1: Results of PSA - Borssele (All Plant Operating States, Internal & External Events)

Event	contribution to TCDF (Old Plant)	contribution to TCDF (Current Modified Plant)
INTERNAL EVENTS DURING POWER STATES	3.6 E-5 64.1 %	9.4 E-7 33.2 %
- LOCA	2.4 E-5 43.0 %	6.1 E-7 21.6 %
- Large LOCA (6"-29")	3.68 E-6 6.6 %	8.2 E-8 2.9 %
- Medium-Large LOCA (4"-6")	1.30 E-7 0.2 %	2.0 E-8 0.7 %
- Small-Medium LOCA (2"-4")	5.99 E-7 1.1 %	9.6 E-8 3.4 %
- Small LOCA (½"-2")	5.44 E-6 9.7 %	2.3 E-7 8.1 %
- Very Small LOCA (<½")	1.41 E-5 25.2 %	1.2 E-7 4.2 %
- Interfacing System LOCAs,	2.90 E-7 0.5 %	1.7 E-8 0.6 %
- Steam Generator Tube Rupture)	1.96 E-8 0.0 %	3.7 E-8 1.3 %
- Internal Flood/Fire	5.4 E-6 9.7 %	1.8 E-7 6.2 %
- Loss of Support System	4.3 E-6 7.7 %	1.0 E-7 3.7 %
Loss of Main & Auxiliary Cooling Water,	3.30 E-6 5.9 %	5.7 E-9 0.2 %
Loss of Closed Cooling Water (Component Cooling)	9.43 E-7 1.7 %	1.0 E-9 0.0 %
Catastrophic Feedwater Tank Rupture	8.61 E-8 0.2 %	9.9 E-8 3.5 %
- ATWS with Main Feedwater available	6.34 E-7 1.1 %	2.3 E-8 0.8 %
- ATWS with Loss of Main Feedwater	1.35 E-6 2.4 %	1.1 E-8 0.4 %
- Transient Losses of Feedwater (Steam/Feedwater Line Break Outside Containment & Ringroom)	1.1 E-7 0.2 %	2.8 E-9 0.1 %
EXTERNAL EVENTS DURING POWER STATES	6.7 E-7 1.2 %	2.3 E-7 9.8 %
- Vapour Cloud Explosions (Shipping LPG)	2.8 E-7 0.5 %	1.1 E-7 2.9 %
- External Flooding	1.7 E-7 0.3 %	2.0 E-8 0.7 %
- Toxic Gas Releases	1.1 E-7 0.2 %	6.8 E-8 1.6 %
- Long-term Loss of Off-site Power due to external hazards	5.6 E-8 0.1 %	1.3 E-8 0.3 %
- Other	5.6 E-8 0.1 %	8.6 E-9 0.2 %
INTERNAL & EXTERNAL EVENTS DURING (EARLY + LATE) HOT STEAMING POS (LOCAs dominate; added Reactivity Addition Accident during Start-up; Increased Loss of Off-site Power, due to potential problems in shifting Off-site Power	5.6 E-7 0.9 %	1.66 E-9 0.0 %
INTERNAL & EXTERNAL EVENTS DURING (EARLY + LATE) COLD SHUTDOWN	1.1 E-6 1.9 %	2.0 E-7 7.0 %
- LOCA inside containment		
- IS-LOCA	9.0 E-7 1.6 %	1.3 E-7 4.7 %
- Other	< 5.6 E-8 <0.1 %	5.9 E-9 0.3 %
	< 2.8 E-7 <0.5 %	5.9 E-8 2.0 %
INTERNAL EVENTS DURING MIDLOOP POS	1.6 E-5 28.2 %	1.1 E-6 40.2 %
- LOCA	7.9 E-6 14.3 %	2.4 E-7 8.5 %
- Fire	6.3 E-6 11.2 %	7.3 E-7 25.9 %
- Loss of RHR	6.2 E-7 1.1 %	4.2 E-8 1.5 %
- Loss of 6 kV ac bus BU	3.4 E-7 0.6 %	8.5 E-8 3.0 %
- Loss of component cooling	3.4 E-7 0.6 %	3.1 E-8 1.1 %
- Other	3.4 E-7 0.6 %	5.7 E-9 0.2 %
EXTERNAL EVENTS DURING MIDLOOP POS	5.0 E-7 0.9 %	4.8 E-8 1.7 %
- Vapour Cloud Explosions (Shipping LPG)	2.2 E-7 0.4 %	1.4 E-8 0.5 %
- External Flooding	2.2 E-7 0.4 %	1.0 E-8 0.3 %
- Other	5.6 E-8 0.1 %	2.4 E-8 0.9 %
INTERNAL & EXTERNAL EVENTS DURING CORE UNLOADING/LOADING POS	< 1.0 E-9	< 1.0 E-9
INTERNAL & EXTERNAL EVENTS FUEL POOL POS (All Fuel in In- containment Fuel-Pool)	1.7 E-6 2.8 %	2.0 E-7 7.0 %
- Loss of Support Systems (Station Blackout)	- 1.0 E-6 1.8 %	2.8 E-8 1.0 %
- Loss of Fuel-Pool Cooling	- 5.3 E-7 0.9 %	1.7 E-7 5.9 %
TOTAL	5.6 E-5 100 %	2.83 E-6 100 %

7.13 Spain

7.13.1 Introduction

All Spanish PSA have identified design and operating procedure modifications that, when implemented, allowed for significant improvements of risk results. These modifications use to be plant specific and they can not be extrapolated to other plants. Nevertheless, changes must always be analysed for similar plants, to conclude whether they respond to a generic problem in reactors of the same type. One example of this type, found also in the Spanish PSA, may be the design of the support systems to the PWR coolant pump seals.

Typical classes of modifications identified are:

- Design changes in component control and/or power supply, like duplication of equipment (e.g. relays) or electric power supply or instruments.
- Design changes to cope with transients involving losses of power or other support systems (e.g. ventilation).
- Design changes to protect against identified fire or flood scenarios.
- Design changes in some actuation system logic or some actuation system components (e.g. type or state of relays).
- Design or procedure changes to allow for the use of alternative systems during accident sequences identified as important.
- Changes in testing procedures to test equipment not included in the testing method or to change testing frequencies.
- Changes in operating procedures to make easier or clearer to the operator some actions in dominant sequences that were not considered relevant before the PSA or to include new equipment verification.
- Inclusion of new aspects, or validation of others, in emergency procedures.
- Changes in operator training programmes to include PSA findings.

Another type of PSA use related to plant modifications is its role in backfitting decision making. This use of PSA had not been frequent in Spain until the Jose Cabrera PSA case. Nevertheless, some examples could be found, like, for instance, consideration of PSA insights for decisions taken about applying new standards to earlier plants in the field of fire analysis or remote shutdown control panel design. This consideration of PSA insights was not only to justify some exceptions in earlier plants, but to identify new modifications, not required by the new standards, as well.

Nevertheless, more frequent use of PSA insights for backfitting was foreseen in the case of the oldest plant in Spain, the Jose Cabrera NPP, a unique plant where PSA applications of this type were very likely, once the final version of the PSA was completed. This recent case is more deeply described in the following of this report.

7.13.2 *Jose Cabrera PSA use in the Periodic Safety Review process*

Jose Cabrera NPP is a 160 MWe Westinghouse PWR in operation since 1968. The NSSS design is unique among commercial PWR all over the world, since it has only one reactor coolant loop, with its corresponding only one steam generator. Other plant systems also present unique characteristics for operating reactors, like some main control room, containment building and other system design features.

The Jose Cabrera PSA was the fifth of the Spanish NPP PSA and was started in 1991, when required by the CSN within the PSA Integrated Programme framework. According to the IP philosophy, the scope of this PSA was extended from the preceding one and it was the first Level 2 PSA to be done in Spain. Level 1, fire and internal and external flood analyses were also included in the PSA project scope, like in the preceding fourth PSA. CSN review of the project was done mostly in parallel to the PSA project and CSN comments were being incorporated to the PSA final report, which was revised twice until it was finally accepted by the CSN in 1999.

Also in 1999 this plant was to be reviewed by the CSN within the Periodic Safety Review (PSR) process for the Spanish NPP, which was started in Spain in 1998. The PSR process gives a role to the PSA for assessment of the PSR findings significance. The PSR and the accepted PSA became thus an opportunity to assess the safety importance of many of this plant unique design features. The PSR process is linked to the licensing process and, normally, PSR is the point for the plants to obtain operating licences for the period between successive PSR, that is to say, ten years. In the case of Jose Cabrera NPP, the utility was applying just for a nine-year licence, since the forty-year term foreseen for its operating life expires in 2008.

In this PSA, like in the rest of Spanish PSA, some design and procedure modifications that impact risk results were identified during the PSA project. Nevertheless, in this case, only a few (i.e., 10 design and several significant procedure modifications) were implemented during the PSA project, because a final balance was expected, after the CSN final acceptance of the PSA, to assess which of the possible identified improvements were going to be really safety and cost effective.

Examples of modifications implemented during the PSA project include:

- Improvements in emergency operating procedures (EOP), especially for steam generator tube ruptures (SGTR).
- New procedures related to the operation of the emergency power supply system train, which is fed by an adjacent Hydro power station.
- New procedure to refill the refuelling water storage tank during SGTR sequences.
- Improvements in administrative controls of some doors especially significant to preclude possible internal flood propagations.
- New double electrical power supply to the solenoid valves in the air control circuit of the pressuriser relief valves, for increasing the Feed & Bleed function reliability.
- Improvements in instrumentation design and testing procedures to prevent an inadvertent interfacing LOCA if the operator were to open a second isolation valve for periodic testing after the first isolation valve would have failed open.
- New tests to demonstrate and survey the feasibility of Feed & Bleed operation without being supported by the instrument air system.

- Environmental qualification of the pressuriser relief valves and their associated isolation valves.

However, the most important PSA findings and the already known unique plant design features were held to be assessed during the PSR process.

The Jose Cabrera PSA was used in the PSR process to assess the safety importance of a total of 21 issues. Many of them were already known before the PSR process, which was used to get a real knowledge about their significance and the safety impact of potential improvements.

The 21 issues analysed were the following:

1. Main control panels inadequate location. Safety systems control panel is behind the normal operation panel, which is impeding the visibility and hindering the access to the safety panel during emergency operation. This design deficiency is augmented by the lack of a full scope replica simulator which could be used for operator training and for an optimal dynamic validation from the human factors engineering point of view.
2. There are two important issues related to fire risk scenarios. The first one is regarding the loss of information in the control room originated by fire-caused instrumentation losses in other plant areas, though the safety systems themselves be still available.
3. The second issue on fire risk is related to main control room fires, due to the lack of a remote shutdown control panel.
4. An additional man machine interface design deficiency for the operator action to change the safety injection system operating mode from injection to the recirculation. Several valves had to be locally actuated outside the control room.
5. Dependency of human actions to control auxiliary feedwater system and to start Feed & Bleed. It is originated by the EOP writing and by the common steam generator instrumentation used to guide the operator. Special importance of this dependency in station blackout scenarios.
6. Low reliability of the auxiliary feedwater turbine driven pump, given by failure statistics along the plant specific operating experience of this component.
7. Room ventilation failures impact on components located in the room. Analysis on the need of PSA models of ventilation systems for additional components, like inverters or motor driven auxiliary feedwater pumps.
8. Need of an expanded spectrum of steam generator tube rupture sizes and number of ruptured tubes for the SGTR thermal-hydraulic analysis. Need to check the applicability of SGTR EOP to larger and smaller sizes and higher tube numbers. Need to check PSA success criteria and human reliability analyses for those cases.
9. Need to develop extended analyses and additional emergency measures for conditions considered currently in the PSA as steady after 24 hours. Some PSA final states are not stable at medium term and will need additional human actions and system availabilities. This is the case of states where the reactor core is being cooled by a system from an externally refilled tank and the cooling loop is open, discharging to the external atmosphere (e.g.: some SGTR sequences).

10. There are two issues about possible additional or different containment failures modes. The first one is due to the fact that the CSN structural analysis of the containment overpressure failure mode gives a different result for the most likely failure mode than which is included by the utility in the PSA Level 2. In the CSN analysis the dominant failure mode is the steel dome structure breakdown, while utility's is the breaking of the structural junction between the basemat and the cylindrical concrete shield.
11. The second issue on containment failure modes is that the CSN review identified a possible failure mode due to hydrogen detonation when the upper containment hydrogen concentration gets higher because of steam condensation near the dome, since the containment external spray system discharges cold water from the service water system on the dome outer surface.
12. Lack of an AMSAC system, that is, lack of an alternative for reactor protection system (RPS) actuation, that enhances reactor scram function reliability to prevent ATWS.
13. Lack of RPS bypass breakers, that allow for the RPS breakers testing while reactor power operation.
14. Analysis of the applicability of several Westinghouse emergency operating guidelines (ECA) to the Jose Cabrera NPP specific case.
15. Assessment of the emergency safeguard features actuation system (ESFAS) testing policy, regarding testing frequencies.
16. Non-compliance of some containment penetrations with general design criteria.
17. Lack of redundancy for some piping segments in several safety systems, like some segments in the safety injection, auxiliary feedwater, component cooling water, essential service water and residual heat removal systems.
18. Potential risk impacts of different operating modes of the NPP-dedicated emergency power supply (EPS) line from the adjacent Hydro power plant. This line constitutes the second EPS train and it is diverse from the train of the only diesel generator.
19. Assessment of the testing of the whole logic electrical circuits of safety related systems, according to a NRC generic letter. It is the case of ESFAS, RPS and diesel load sequencer systems.
20. Assessment of some design features in the direct current system, related to the lack of alarms in the main control room to announce to the operator the loss of a DC train or an inadvertent disconnection of a battery or a charger.
21. Assessment of the utility proposal about the elaboration of severe accident management guides. Findings of the PSA Level 2 are an important information source for this elaboration.

The PSA and the knowledge of the CSN experts on the Jose Cabrera PSA were used to assess all the preceding issues. Some of them were closed after this assessment, but others were considered open issues and the utility given a six-month term to prepare a Safety Improvement Plan (SIP) to cope with those issues and which had to be evaluated and accepted by the CSN. In this Plan they were also included some more issues that were identified during the PSR process and could not be assessed by means of the PSA.

At the same time, a three-year operating licence was granted, until October 2002. At the end of this licence, all the improvements identified in the Safety Improvement Plan and approved by the CSN shall have to be implemented to issue a new operating licence for the additional six years initially requested by the utility.

In 2000, the Safety Improvement Plan was presented by the utility and evaluated by the CSN. Most of the issues were then adequately considered in the Plan, but there still remained some which had to be reconsidered by the utility. By the beginning of 2001, the utility submitted to the CSN a revision of the Plan sections related to those open issues. After reviewing the new proposal, the CSN decided to accept the Safety Improvement Plan. This plan is being implemented and all the modifications identified shall have to be done at the plant before October 2002.

In summary, the assessment results and final situation of the above described issues is as follows:

1. This issue was kept open after the first utility proposal. The modifications proposed by the utility in the SIP were not considered sufficient by the CSN or, at least, it was not demonstrated that they were sufficient by means of a rigorous process to identify and select them. The utility had to proceed to a rigorous project to identify the human factors engineering problems caused by the safety systems control panel location and to identify and propose the solutions for each of the problems. A guide for this project was, as required by the CSN, the NRC-Brookhaven model for human factors engineering reviews contained in NUREG-0711. Authors of this model were the technical directors of the project. A decision by the CSN, regarding this issue and the proposed modifications of the control room derived from the project results, was taken in March 2001 after a very close follow-up of the project. Modifications shall be validated and implemented in the control room in October 2002, when current operating licence expires. A total of about thirty control room design modifications will be done to cope with the original design problem. Maybe the most significant concept for the accepted solution was to duplicate instrumentation from the front to the back panel and to have, in a very short term after an emergency declaration, a new shift available at the control room. In this way, one reactor operator will be always at the back panel during emergencies and the operation and surveillance of the reactor front panel will be performed by the other shift reactor operator. This concept had to be complemented with modifications to improve communications, indications, procedures and many other aspects of the ergonomic design.
2. Regarding the fire-caused loss of instrumentation problem, the utility included in the SIP the commitment to: a) protect many additional cables, b) analyse additional fire zones with a more thorough human reliability method, c) list the reliable instrumentation that will be available to the operators for EOP follow-up in case of fires, and d) include this subject in the operator training program. All these actions are planned to be implemented in October 2002.
3. Regarding the control room fire risk problem, the utility proposed to install in a different room a so-called Temporary Control Room Leave Panel which, together with a local panel currently available in the auxiliary feedwater system room, will allow for plant control during some time from outside the main control room. Additional risk analysis of potential control room scenarios not covered by this proposed situation were required by the CSN for a decision in March 2001, like in the issue about control room panels location. Finally, additional detection and protection measures shall have to be implemented at some control room panels to cope with some control room fire scenarios where proper operation outside the control room is unlikely.

4. The utility included in the SIP the commitment to install motor actuators for many of the valves affected by this issue. Those valves will be remotely actuated from the control room safety systems panel. Although this issue is really improved in this way, an additional analysis of the risk impact of the remaining unchanged valves, which might need to be actuated in some accident sequences, was required by the CSN and finally accepted in March 2001, after including a new valve to be modified.
5. The utility presented a new human reliability analysis for those human actions considering the affected steam generator instrumentation and concluding that no modifications are necessary. CSN review is planned to be completed by the end of 2001.
6. The utility included in the SIP the commitment to carry out a very detailed analysis of the auxiliary feedwater turbine driven pump system to identify all the potential failure mechanisms and to propose consequent design or maintenance procedure modifications, if necessary. The analysis was submitted to the CSN for review in the last quarter of 2000. The review is planned to be completed by the end of 2001.
7. The utility also included in the SIP the commitment to carry out detailed analysis of the temperature evolution in case of ventilation losses in the rooms of the mentioned safety equipment (i.e., inverters and auxiliary feedwater motor driven pumps) and to implement the consequent corrective measures if equipment damage temperatures are expected to be reached. The analysis was submitted to the CSN for review in the last quarter of 2000. The review is planned to be completed by the end of 2001.
8. The utility also included in the SIP the commitment to: a) compile all the thermal-hydraulic analyses already done related to SGTR, b) carry out additional analyses for multiple tube ruptures, c) contrast the less detailed code analyses with more detailed codes, and d) make the EOP modifications which might be identified as necessary. Revision 0 of this analysis was submitted to the CSN for review in the first quarter of 2001.
9. The CSN required that the new extended analysis to consider plant states which are not steady in 24 hours shall be submitted by the utility in October 2001, along with a new revision of the PSA. Including those states in the plant operational states analysis of the Low Power and Shutdown PSA to be done, was recommended, if the SPSA were to be available at that time. If not, explicit consideration must be done in that next PSA revision.
10. CSN identified containment overpressure failure mode must be used for future PSA applications to severe accident management guides or emergency preparedness.
11. Regarding the potential containment failure mode due to hydrogen detonation in the upper containment because of steam condensation, the CSN is performing itself new detailed analysis to verify that possibility. If final results showed a high likelihood for that phenomenon, it should have to be included in the PSA Level 2 model and new protective measures taken, if necessary.
12. PSA assessment of this issue showed that the current situation is acceptable at this plant.
13. PSA assessment of this issue also showed that the current situation is acceptable at this plant.
14. In a decision that was taken making use of the PSA assessment and other arguments, three new EOP were required by the CSN to be implemented, after being developed from three

generic Westinghouse ECA. They are: 1) an EOP to recover the safety injection recirculation capability, 2) an EOP to identify and isolate an interfacing system LOCA, and 3) an EOP to reduce safety injection flow in situations, like a SGTR, where a reactor coolant inventory loss can not be finalised until cold shutdown conditions are reached. These new EOP must be implemented by the end of 2001.

15. The utility included in the SIP the commitment to analyse the feasibility to test at power some of the identified ESFAS components, including contact pairs, where a more frequent testing would mean a significant risk reduction and to make the appropriate design modifications and procedure developments, if necessary. It also compromised to test annually some identified components that were not tested before. Tests not needed of design modifications were performed by October 2000, needed design modifications will be implemented by October 2001 and the whole new ESFAS testing policy after 2001.
16. PSA assessment of this issue showed that the current situation is acceptable at this plant.
17. The utility had already performed an analysis of the piping degradation mechanisms of the affected non-redundant piping in safety systems and had planned an associated analysis of maintenance practices. It is included in the SIP the commitment to incorporate all the new CSN PSA assessment identified segments, like a segment in the safety injection system, into these analyses. The revised study on degradation mechanisms and the associated maintenance practices evaluation and potential improvements was submitted to the CSN in the first quarter of 2001.
18. The PSA assessment of this issue showed that a less severe limitation of the operation modes of the EOP Hydro power line was acceptable. This relaxation was needed due to seasonal river water flow changes.
19. PSA systems analysis task was used as an information source to verify the degree of compliance with the NRC generic letter.
20. PSA assessment of this issue showed that the current situation is acceptable at this plant.
21. The utility developed severe accident management guides for Jose Cabrera NPP and submit them to the CSN by the end of 2000. PSA Level 2 was used extensively and will be used also for the corresponding CSN review. Some aspects have already been communicated to the utility, like the need to make radiological considerations when elaborating the guides for some actions, due to the lack of concrete shielding at the containment steel dome.

7.14 Sweden

One of the major aims with the PSA studies produced in Sweden is verify that the deterministic rules are full-filled, to identify weaknesses in plant design and instructions, also to discover dependencies. PSA studies are also used at measuring the impact of changes plant modifications, at changes of Technical Specifications.

Lot of studies has been conducted, and they are not referred to in this chapter.

7.15 Switzerland

PSA-based modifications and backfits have often been introduced in the Swiss NPPs and are listed below.

7.15.1. Beznau plant

Beznau I and II are Westinghouse PWRs. They are in commercial operation since 1969 and 1971, respectively. The original design consisted of two trains safety systems with relatively poor physical separation and seismic qualification. Backfits, which were performed based on PSA results, are given in the following:

- Automation of the switch in secondary side steam relief control from T-average-mode to SG-pressure-mode.
- Installation of new electrical transformers and improvement of the anchorage of existing transformers in the electric power system.
- Installation of a new instrument air compressor.
- Reinforcement of some electrical cabinet anchorage to the floor for seismic events.
- Reinforcement of a brick wall in the area of the main control room for seismic events.
- Reinforcement of cable trays for seismic events.
- Installation of a feed path from a third, existing battery, which is separated from the two existing paths, to the plant DC buses.
- Several changes in the plant emergency operational procedures.

By these cost effective measures the core damage frequency of the plant was reduced by about a factor of five.

Optimisation of a large backfitting project in using PSA results:

During the late 1970, the Swiss licensing authorities (HSK) required that the Beznau plant be upgraded to more recent safety standards. The aim of the backfit called NANO ("Nachrüsten Notstandssysteme") was to upgrade the safety systems of the plant with respect to redundancy, separation, qualification and protection against external events (bunkered decay heat removal system). In using the Beznau PSA model, the reductions to the core damage frequency of the following two configurations of NANO were analysed:

- a) a simple single train system, consisting of one train of SG feed and one of RCP seal injection, one ECCS low pressure injection pump, single train support systems and the rebuilding of the RWST so that it would be protected against external events.
- b) an expensive two trains system, consisting of two trains of SG feed and of additional component cooling water, one train of ECCS recirculation, one ECCS high head and two ECCS low pressure injection pumps, one charging pump, two trains of support systems and the rebuilding of the RWST so that it would be protected against external events.

As a result of this PSA investigation, the simple single train system was found more cost effective than the expensive two trains system. The configuration of NANO as finally realised was a combination of these two systems and included the following modifications per unit:

Front line systems:

- adding one train of emergency SG feedwater and one of emergency RCP seal injection
- adding one train of ECCS recirculation
- adding two accumulators to the ECCS system
- replacing one ECCS safety injection pump by a new one in the NANO bunker
- rebuilding the RWST protected against external events
- replacing the pressuriser safety and relief valves by three new tandems of combined safety and relief valves
- seismic requalification of the primary circuit and of some other components and structures

Support systems:

- adding one emergency diesel generator and one emergency cooling water pump, each with a cross-connection to the other unit
- adding a control system and a separate emergency control room for all NANO systems.

All systems added are located in a new and separate bunker protected against external events. As a result, the reduction to the core damage frequency obtained by the NANO upgrade is about a factor of 30.

Independently from PSA, a filtered containment venting system was installed.

During the last few years, the Beznau PSA was used to evaluate the optimal configuration for feedwater upgrade project. This investigation resulted in the decision to install one additional emergency feedwater train. Recently, a new, fourth battery train was implemented. Furthermore, passive recombiners will be installed in the next two years in order to avoid containment failure due to hydrogen explosions during a severe accident.

7.15.2 Gösgen plant

Gösgen nuclear power plant is a three-loop PWR built by Siemens-Kraftwerk Union AG (KWU). The design is four train safety and an additional two train special emergency systems with strict physical separation and seismic qualification. The plant began commercial operation in November 1979. Based on the PSA results the utility has performed a number of changes and taken courses of action to address the principal contributors to risk. They include:

Based on full power PSA results:

- An on-site inspection carried out by HSK within the PSA review process revealed that the masonry walls in the electrical building were not included in the PSA model. An improved PSA model showed later that a lot of these walls are risk-significant. Therefore, HSK required the plant to backfit the walls.
- Addition of seismic restraints for electronic cabinets on double floors in the electrical and special emergency buildings (Notstand buildings).
- Modifications to reduce service water intake blockage vulnerability and new Technical Specifications to restrict Notstand (special emergency system) and 220-kV systems maintenance during periods of high debris content in river.
- Larger diameter emergency diesel generator heat exchanger tubes to reduce vulnerability to debris plugging.
- New accident management procedures and documentation for RCS injection via Notstand equipment, for steam generator feed via external sources and for active steam generator cooldown on loss of Notstand buses.
- A change to keep containment sump lines isolated during normal operation except during controlled sump drain.
- Dedicated search for ways to enhance special emergency diesel reliability.

Based on low power and shutdown PSA results:

- By Technical Specifications of the plant it was possible to enter the outage state and go on RHR cooling even if only one RHR cooling train were available. There was an additional situation that was exacerbated by the technical specifications in which a two-train equipment outage may lead to an enforced plant shutdown and a need to go on RHR cooling when RHR and/or support systems are seriously degraded. Therefore, an additional train for spent fuel pool cooling that is capable of cooling the spent fuel pool when the core is unloaded into the pool during the refuelling outage was implemented. Plant practice and technical specifications were correspondingly modified to ensure that there is no initial degradation of the RHR cooling function at the beginning of an outage involving cold shutdown.
- Technical Specifications were further modified in 2001, in order to ensure that the single failure criterion is fulfilled during all outage configurations. These modifications reflect the practice introduced earlier as a result of the shutdown PSA.

7.15.3 Leibstadt plant

Leibstadt nuclear power plant is a General Electric BWR/6 with a Mark III containment. The plant has an additional two train special emergency heat removal system (SEHR). The plant began commercial operation in December 1984. Based on the PSA results the utility has implemented the following major plant and procedure modifications:

- Mitigation of the consequences of ATWS requires that the plant operator reduce reactor pressure vessel water level to lower core power generation. The reduction of the RPV water level below Level 1 will initiate the logic sequence for initiation of ADS, which is designed to provide automatic actions in support of the low pressure Emergency Core Cooling Systems following small to intermediate sized loss of coolant accidents. Automatic vessel depressurisation is not desirable following this postulated ATWS. Reducing RPV water level is desirable following the postulated ATWS because the lower water level will reduce power generation. This is therefore necessary to inhibit opening of the ADS Safety Relief Valves during an ATWS event. The plant change incorporates modifications to the plant logic to automatically inhibit ADS when the ATWS control logic determines that an ATWS event is underway.
- Containment isolation failure is an important aspect of the level 2 PSA. Even though the isolation failure do not result in high radiological consequences due to the nature of the failure (a long narrow path for release), the utility implemented instructions in the EOPs to isolate two manual valves in the equipment drain lines (which are not supported by DC power), outside the containment, when the suppression pool reaches a certain temperature.
- For the depressurisation of the reactor coolant system, accident management actions were implemented in the instructions. The instructions describe the use of alternative water source (the line-up of reservoir water) and the manual opening of SRVs.

The utility will use the level 2 PSA in order to systematically check the concept and the completeness of the SAMG (Severe Accident Management Guidance) which are currently developed.

7.15.4 Mühleberg plant

Mühleberg nuclear power plant is a BWR/4 Mark-I built by General Electric. It started commercial operation in November 1972. A major upgrading of plant redundancy and safety was done during the years 1985 -1989, when an additional and independent two train safety system was added, called SUSAN ("System zur unabhängigen and sicheren Nachwärmeabfuhr"). SUSAN was declared "ready for service" in September 1989 and consists of the following equipment:

- a bunkered, sabotage, airplane crash, earthquake and flooding resistant building containing:
- an emergency control room which has a priority logic overriding any commands from the main control room
- two specially separated 800 kVA diesel generators
- associated cooling equipment including independent river water intake
- filtered air ventilation equipment which can be operated to guarantee SUSAN-building habitability even during hypothetical core melt accidents and after the noble gas releases.

In the reactor building, which acts, as in all Swiss NPP, as a secondary containment, the following equipment has been made part of the SUSAN-ECCS:

- two low pressure ECCS-pump trains delivering 150 t/h each at about 17 bars (ALPS)
- two high pressure steam turbine driven ECCS-pumps delivering 50 t/h each at primary system operating pressure (RCIC)
- two 100 % RHR-systems providing cooling to the pressure suppression chamber water (TCS)
- associated I+C hardware
- for severe accident mitigation: containment pressure relief system and containment spray and flooding system.

In the containment, the most remarkable addition belonging to the SUSAN system are two electric motor driven pressure relieve valves (PRV), each of which is capable of discharging 50 t/h live steam into the pressure suppression chamber. This in addition to the standard ADS-function, which was also made part of SUSAN. Last, but not least, a totally independent SCRAM-function using SUSAN I+C has been added.

SUSAN is designed as an independent system, which is able to shut down the reactor and to assure RHR automatically, i.e. not requiring any operator interference. This effective safety system backfit has been planned and realised without making reference to any PSA analysis, which did not exist at that time. A level 1 and level 2 PSA for KKM, called MUSA (Mühleberg Safety Analysis) was started in the second half of 1988 and has been submitted to the Swiss Nuclear Safety Authorities (HSK) in 1990. The bottom line of this study was, that the plant, taking full credit of SUSAN, displays a risk profile which is comparable to new plants. Nevertheless, two modifications were made to the plant, which can be considered a direct result of the PSA results:

- a depressurisation logic and hardware was added which is triggered by low RPV water level only. (The original automatic depressurisation system, ADS, is activated by a low RPV level and high drywell pressure signal.) This backfit was the result of an accident sequence identified by MUSA, which starts with an RPV isolation, followed by the failure of both high pressure injection systems and an operator error by failing to manually de-pressurise the primary system. (Note that automatic depressurisation would not have functioned because of lack of the high drywell pressure signal.)

Transient analysis done for this sequence showed that the PRV's would de-pressurise the primary system after a 30 minute delay time but that the low pressure injection system would deliver too late to prevent massive fuel overheating and damage, though the RPV would most likely have remained intact. It was decided that this accident sequence is highly undesirable and that this fairly simple extension to the existing depressurisation logic would eliminate it.

- During the analysis of ATWS success criteria to be used in the PSA, it was realised that the 120 sec delay reset for automatic depressurisation might interfere in an undesirable way with the very high capabilities of the plant to ride out an ATWS. Therefore it was decided to add an "ATWS switch" which will, in this very rare case, eliminate any possibility to omit the repeated reset of the 120 sec delay.

7.16 United Kingdom

The PSAs which have been carried out either as part of the design process (for example, Heysham 2, Torness, Sizewell B and the naval nuclear facilities) or as part of the Periodic Safety Reviews for the other facilities. In all cases the PSA has identified areas where improvements have been made to the design and operation of the facility. This has ranged from major improvements – for example, the addition of a set of diverse safety systems which carry out the safety functions for frequent initiating events to relatively minor improvements to operating procedures and training as discussed below.

7.16.1 Modification to the Sizewell B PWR

The Sizewell B design is based on the Westinghouse Standardised Nuclear Power Plant System (SNUPPS). However, changes were required to meet the UK safety requirements which included deterministic requirements (for redundancy/ single failure criterion, diversity, etc.) and probabilistic/reliability targets.

PSA work was carried out throughout the design and construction phases of the plant and continued into operation which included the following:

- a Level 1 PSA for a range on internal initiators at power which was carried out during the design phase and which forms part of the Pre-Construction Safety Report (PCSR)
- a Level 3 PSA for a range of internal initiators at power which forms part of the evidence presented to the Public Inquiry, and
- a comprehensive Level 3 PSA for all initiating events and hazards, and addresses all modes of operation (including full power, low power and shutdown modes) which form part of the Pre-Operational Safety Report

The most important probabilistic target that influenced the design was the one related to the frequency for uncontrolled releases for single accidents. This requires that the frequency of fault sequences which could give rise to a large uncontrolled release of radioactivity should be less than 10^{-7} per year and the sum of all such fault sequences should be less than 10^{-6} per year.

In addition, it was recognised that common cause failure limited the reliability that could be claimed for a safety system that incorporated redundancy only. This limit would generally be in the range 10^{-3} to 10^{-5} failures per demand and a value of 10^{-4} failures per demand was chosen for design purposes for active safety systems such as pumping systems.

The requirement that:

$$[\text{initiating event frequency}] \times [\text{safety system failure probability}] < 10^{-7} \text{ per year}$$

along with the limit of 10^{-4} failures per demand on the reliability of redundant safety systems required that, for initiating events with a frequency of $>10^{-3}$ per year, diverse safety systems would need to be provided for each of the required safety functions. This led to the following safety systems being added to the SNUPPS design:

- a Secondary Protection System (SPS) using magnetic logic devices (LADDICS) which was diverse from the computer based Primary Protection System (PPS),

- an Emergency Boration System (EBS) which was a fast acting system to inject a highly concentrated boron solution into the reactor coolant system following failure of control rods to drop into the core,
- the auxiliary feedwater system was replaced by two diverse systems one using electrical motor driven pumps and one using steam turbine driven pumps,
- an Emergency Charging System (ECS) which used steam turbine driven pumps which was diverse from the Chemical and Volume Control System (CVCS) which used electrical motor driven pumps. This provided diverse protection for the reactor coolant pump seals and boration of the primary circuit, and
- a seismically qualified air-cooled Reserve Ultimate Heat Sink (RUHS) to provide diversity from the seawater cooling system.

Further design changes were made as a result of the PSAs carried out at the PCSR stage. As a result of the Level 1 PSA, further design changes were made which included:

- the provision of two battery charging diesels to give long term DC power for control and instrumentation following an extended loss of all AC power, and
- additional, diverse provisions for isolation of the containment mini-purge system,
- and as a result of the Level 3 PSA:
- additional isolation valves and interlocks were introduced to reduce the frequency of an interfacing-systems LOCA which would discharge outside the primary containment, and
- changes to provide better protection for the containment following a severe accident.

A living PSA has now been developed for Sizewell B and is being used to provide operational support. In particular, it was used to address the risk arising from increasing the enrichment of the fuel used in the reactor and the consequent increase in the time between refuelling outages, and in considering the options available for refuelling including the risk which would arise if the reactor coolant level was reduced to mid loop level.

7.16.2 Modifications to the AGRs

The PSAs carried out for the early AGRs led to, or supported, a number of significant design change proposals and the licensee is undertaking a programme of work to improve safety. Those which were

Changes have been made to the design or operation which are based (in part) on probabilistic considerations including:

- the change from 2 to 3 year outages. The design PSA for this AGRs was revised to reflect the new regime of three year statutory outages and was able to demonstrate that the move did not unduly increase the risk. This move has major financial implications for the operation of the plant,

- at one of the AGRs, the emergency feed system and the back-up cooling system shared common pipework and valves. The PSA identified that this limited the reliability of post trip cooling and was used to investigate the options for improvement, and
- whilst the newer AGRs have control rods and a rapid nitrogen injection system for reactor shutdown, the older AGRs only have control rods. The PSA was used to investigate a number of options for enhancing the reliability of the shutdown system. The option chosen was to separate a group of rods - the grey rods, which are used to trim the power of the reactor and hence are constantly in motion. These are now wound into the core on reactor trip as opposed to the other rods which fall in under gravity.

More recently, there have also been a large number of other modifications which have been supported by the PSA as follows:

- change to the Maintenance Schedule definition of a calendar month from 30 to 35 days to align with station shift rotas (Heysham 2),
- modifications to the functionality of the Vessel Overpressure Protection Equipment to reduce the level of water ingress following a boiler tube leak, together with a revision to the boiler tube leak faults safety case (Hinkley Point B),
- enhancement to the fire hydrant system to provide an alternative heat sink for the PVCW and installation of backup diesels for the PVCW pumps to provide diversity following loss of grid faults (Hinkley Point B),
- installation of an Additional Feed System to provide increased post-trip cooling reliability, particularly following loss of grid faults and hazards (Dungeness B),
- installation of an Electrical Overlay System to provide dedicated electrical supplies following hazards (Dungeness B),
- enhancement of CO₂ and N₂ injection systems to provide enhanced post-trip cooling and shutdown capability (Dungeness B), and
- enhancement of the LPBUCS (backup PVCW system) to provide increased protection against loss of pressure vessel cooling (Hartlepool and Heysham 1).

In addition, there are many other revisions to the station safety cases where hardware modifications have not been required where the PSAs have been used to support the revised safety cases - a prime example being the Gas Circulator Run-on safety cases which have now been completed for most of the AGRs.

7.16.3 Modifications to the Magnox reactors

The PSAs for the Magnox reactors were carried out as part of the safety cases produced for the Long Term Safety Reviews (LTSRs) which were carried out to consider whether it would be safe to operate the reactors beyond 30 years and some of these have been updated as part of the programme of Periodic Safety Reviews (PSRs).

During these reviews the safety of the Magnox reactors has been assessed against both engineering/deterministic and probabilistic criteria and changes have been made to the design and operation which included the following:

- a secondary shutdown system in which boron beads are blown into the reactor following failure of the control rods to enter the core. This system provides protection against earthquake and additional diversity,
- a secondary guardline which provides a diverse means of detecting that a fault condition has occurred and initiating a reactor trip. The primary and secondary guardlines both use relays but they are of a different design and manufacture,
- a tertiary feedwater system which is diverse from the existing main and back-up feedwater systems and provides feed to the boilers in fault conditions, and
- modifications to mitigate the consequences of a hot gas release.

For the reactors with steel pressure vessels, the secondary shutdown systems took the form of a boron ball injection system and, for the reactors with concrete pressure vessels, it took the form of articulated control rods. These provide protection for fault sequences such as earthquake where the geometry of the core could be changed and provides a diverse means of shutting down the reactor.

For the reactors with steel pressure vessels, modifications were made to ensure that the release of hot gas that would occur if one of the main coolant ducts should rupture would not lead to consequential failures which were unacceptable. For the reactors with concrete pressure vessels, there are no main coolant ducts outside the pressure vessel. However, the hot gas release safety case has addressed the failure of less significant pressure circuit components.

Cooling of the core by natural circulation has now been demonstrated for all the Magnox reactors although the concrete pressure vessel Magnox reactors have less inherent natural circulation cooling capability than the steel pressure vessel Magnox reactors due to the less favourable geometry of the gas circuit. Natural circulation cooling capability for fault sequences involving a gradual depressurisation has also been demonstrated for the reactors with steel pressure vessels but this has proved more challenging for the reactors with concrete pressure vessels. This has led to the significant development of the natural circulation cooling safety cases for the reactors with concrete pressure vessels.

The PSAs were revised to take account of these modifications and to bring them up to a better standard and are now sufficiently good that they can play a larger role in targeting potential weaknesses in the design or operation. At NII's request, licensees are reviewing their Operating Rules (equivalent to Technical Specifications) which govern the availability of safety related equipment to ensure that the increase in risk during periods of maintenance is kept as low as reasonably practicable.

Another key area identified for improvement by the PSA involves the important recovery actions which could be undertaken. This has led to the development of new procedures and additional operator training.

The PSA for one plant has indicated relative weaknesses in post trip cooling reliability for some infrequent accident scenarios (major depressurisations) and this has led to ongoing investigations into up-rating of the CO₂ injection system to provide diverse emergency cooling.

7.17 United States

The US Nuclear Regulatory Commission's efforts to risk-inform its regulatory culture encompasses a wide range of staff efforts to review, modify, or develop regulations, guidance documents, standards, processes and programs. Following are selected descriptions of the ways in which insights from plant-specific PSAs, issue-specific PSAs, or risk-informed processes and programs affect plant-level operations.

7.17.1 Graded Quality Assurance (QA) Pilot Program

The purpose of graded QA is to apply licensees' QA controls (such as reviews, inspections, and audits) in a manner that is consistent with plant equipment's importance to safety. Thus, graded QA allows both licensee and NRC staff to focus on more safety-significant equipment. Similarly, graded QA reduces the resources that must be allocated for QA activities for equipment of lesser safety significance. In general, existing licensee QA controls continue to apply to safety-significant equipment; less-rigorous licensee QA controls apply to equipment of lesser safety significance.

In November 1997, the NRC approved the implementation of a graded QA program for the South Texas Project facility. While implementing the graded QA pilot application, South Texas determined that it will derive much less benefit than it had anticipated from application of graded QA. In particular, according to the licensee, special requirements in other regulations require continued complex and costly controls on many structures, systems and components (SSCs) regardless of the reduced QA requirements. In July of 1999, STP submitted a request for exemptions from many of the special treatment requirements, including greater relief from quality assurance requirements than allowed by the GQA program. The staff is utilising this exemption request within the ongoing effort to risk-inform the special treatment requirements of 10 CFR Part 50.

To provide guidance in implementing graded QA practices, the licensees for Palo Verde, Grand Gulf, and South Texas worked with the NRC to develop a regulatory guide (Regulatory Guide 1.176, "An Approach for Plant-Specific, Risk-Informed Decision-making" [ref. #8.2.17.16]) issued in August, 1998. Because the regulatory guide (RG) was issued *after* the pilot application was approved, it reflects lessons learned from the pilot review. For example, some challenges described in the RG include:

- Establishing a baseline for the quality, scope, and rigor of the Probabilistic Risk Assessment upon which the determination of safety significance of plant equipment would be based
- Formulating a realistic quantitative estimate of the change in plant risk arising from implementing graded quality assurance
- Defining the scope, form, and content of the requisite performance monitoring program that would be used to judge the success and effectiveness of the graded quality assurance program
- Ascertaining the effect of graded quality assurance program on commitments to industry standards relied upon by licensees to satisfy Appendix B to 10 CFR Part 50 requirements
- Formulating the extent of and need for NRC staff-industry regulatory interface during implementation

7.17.2 Special Treatment Requirements

The Commission decided in 1998 to consider promulgating new regulations that would provide an alternative risk-informed approach for special treatment requirements in the current regulations for power reactors. Special treatment may be defined as current requirements imposed on structures, systems, and components that go beyond industry-established requirements for equipment classified as "commercial grade" that provide additional confidence that the equipment is capable of meeting its functional requirements under design basis conditions. These special treatment requirements include additional design considerations, qualification, change control, documentation, reporting, maintenance, testing, surveillance, and quality assurance requirements. Since September 2000, the staff has been working with industry and interested stakeholders to resolve issues associated with industry-developed guidance intended to implement the rule. The staff is currently working to develop the proposed rule language, supporting regulatory information, and interacting with industry on pilot activities to test the implementing guidance.

The licensee for South Texas submitted an exemption request that would allow it to apply the concepts underlying this rulemaking (categorisation, removal of special treatment requirements) at the facility, by receiving exemptions to certain existing requirements that would prevent it from otherwise undertaking such a program. The South Texas exemption was granted on August 3, 2001, and is considered to be a "proof-of-concept" prototype for the rulemaking. The exemption permits the licensee to implement an alternative treatment process that, if effectively implemented by the licensee, can result in safety-related low-risk significant (LRS) and non-risk significant (NRS) SSCs being capable of performing their safety functions under design-basis conditions throughout their service life. The staff has determined that the licensee's categorisation process provides a reasonable method for determining that safety-related LRS and NRS SSCs have a small contribution to overall safety. The experience from the licensee's efforts and the staff review are being co-ordinated with the rulemaking activities and guidance development.

7.17.3 Risk-Informed Technical Specifications

Consistent with the Commission's policy statements on technical specifications and the use of PSA, the NRC and the industry continue to develop risk-informed improvements to the current system of technical specifications. Proposals for risk-informed improvements to the STS are judged based on their ability to maintain or improve safety, the amount of unnecessary burden reduction they will likely produce, their ability to make NRC's regulation of plant operations more efficient and effective, the amount of industry interest in the proposal, and the complexity of the proposed change. The staff is re-evaluating the priorities for its review of risk-informed technical specification initiatives. The staff intends to follow the process described in NRC Regulatory Issue Summary 2000-06, "Consolidated Line Item Improvement Process For Adopting Standard Technical Specifications Changes for Power Reactors," [ref. #8.2.17.17] for reviewing and implementing these improvements to the STS.

The industry and the staff have identified eight initiatives to date for risk-informed improvements to the STS. They are: 1) define the preferred end state for technical specification actions (usually hot shutdown for PWRs); 2) increase the time allowed to delay entering required actions when a surveillance is missed; 3) modify existing mode restraint logic to allow greater flexibility (i.e., use risk assessments for entry into higher mode limiting conditions for operation (LCOs) based on low risk); 4) replace the current system of fixed completion times with reliance on a configuration risk management program (CRMP); 5) optimise surveillance frequencies; 6) modify LCO 3.0.3 actions to allow for a risk-informed evaluation to determine whether it is better to shut down or to continue to operate; 7) define actions to be taken when equipment is not operable but is still functional; and 8) risk-inform the scope of the TS rule.

7.17.4 Risk-Informed In-service Inspection

The NRC issued Regulatory Guide 1.178, “An Approach for Plant-Specific Risk-Informed Decision-making: In-service Inspection of Piping,”[ref. #8.2.17.18] and a companion standard review plan chapter in September 1998. The NRC has approved two industry topical reports describing in detail two different methods (produced by the Westinghouse Owners Group and the Electric Power Research Institute) to develop alternatives to the ASME Section XI In-service Inspection program. During its review of the topical reports, the NRC also approved three pilot applications. As of the end of May 2001, 39 plants had submitted their inspection program applications. The staff had approved 16 programs and the remaining 23 programs were still under review.

7.17.5 Risk-Informed In-service Testing

In August 1998, the NRC issued Regulatory Guide 1.175, “An Approach for Plant-Specific, Risk-Informed Decision-making: In-service Testing,”[ref. #8.2.17.19] which provides guidance regarding changes to the risk-informed in-service testing program. The agency subsequently completed a pilot application of risk-informed in-service testing in 1998, and has approved or is reviewing several other applications, generally of limited scope.

7.17.6 IPEEE

The Individual Plant Evaluation, External Events (IPEEE) program successfully resulted in the nuclear power industry identifying safety improvements that substantially reduced the risk of accidents. Over 80% of the licensees have identified and implemented or proposed plant improvements to address concerns revealed through the IPEEE program. These voluntary licensee improvements have led to enhanced plant capability to respond to external events (such as earthquakes and floods) which can be important contributors to total plant core damage frequency. The generic insights from this effort will be used to support development of PSA guidance and standards, while plant-specific risk information will support the risk-informed reactor oversight program.

CHAPTER 8 - REFERENCES

References are provided to establish a contact point for obtaining further information or details about the PSA Programmes within the contributing countries and for providing information on specific documents

(NOTE: Addresses, phone numbers and names provided in this section represents current information, *as of 1 April 2002*. Since this information is subject to changes due to re-organisations, advancements, etc., the reader should take these occurrences into account.)

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APPENDIX A – STATUS OF PSA PROGRAMMES IN NEA MEMBER COUNTRIES

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed
Belgium (all plants are PWRs)	Doel 1 and 2	Level 1 + 2 ^A	Expected 2002	
	Tihange 1	Level 1 + 2 ^A	Expected 2002	
	Doel 3	Level 1 + 2 ^B	1993	2000
	Tihange 2	Level 1 + 2 ^B	1993	2000
	Doel 4	Level 1	Expected 2002	
	Tihange 3	Level 1	Expected 2002	
Canada	Point Lepreau	Risk Baseline – Level 1	Expected 2002	
	Gentilly 2	Risk Baseline – Level 1	Expected 2002	
	CANDU	Generic	Expected 2002	
	CANDU6/CANDU9	Level 2	N/A	
	Bruce B	Level 3	1999	
	Bruce A	Level 3	Expected 8/2002	
	Pickering A	Level 3	1995	2004
	Pickering B	Level 3	Expected 2003	
Darlington A	Level 3	1987	Expected 3/2003	
Czech Republic	Dukovany NPP WWER 440/213	Level 1, internal events, floods and fires, all states	1995	since 1996 Living PSA programme
		Level 2	1998	Planned by 2002
	Temelin NPP WWER 1000/320	Level 2, internal and external events, all states	1996	Planned by 2001
		Level 2	1996	Planned by 2002

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed	
Finland	Loviisa VVER-500	Internal Ies	1989	regular updates	
		Fire	1994	regular updates	
		Flooding	1994	regular updates	
		Harsh weather	1994	regular updates	
		level 2 internal Ies	1996	regular updates	
		shutdown mode, internal IEs	1997	regular updates	
	Olkiluoto- BWR	internal Ies	1989		
		Fire	1992		
		Flooding	1994		
		harsh weather	1996		
		level 2 internal Ies	1996		
		shutdown mode, internal IEs	1997		
France	Generic 900 MWe PWR	Level 1			
		Internal events	April 1990	End 2001	
		Power and Shutdown states			
	Paluel 3, 1300 MWe PWR	Level 1			
		Internal events	April 1990	Rev 1 1996 Rev 2 1999	
		Power and Shutdown states			
France/Germany	EPR (PWR)	Level 1			
		Internal events	1997		
		Power and Shutdown states			

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed
Germany	Obrigheim, PWR	Level 1, full power, internal events	1999	-
	Stade, PWR	Level 1, full power, internal events	1997	2000
	Biblis A, PWR	Level 1, full power	1990	2001
	Biblis B, PWR	Level 2- ^(a) , full power	1989	2000
	Neckarwestheim 1, PWR	Level 1, full power	1997	2001
	Unterweser, PWR	Level 1, full power	1995	2001
	Grohnde, PWR	Level 1, full power	1999	2008 (planned)
	Grafenrheinfeld, PWR	Level 1, full power	1998	2007 (planned)
	Brokdorf, PWR	Level 1, full power	1996	2004 (planned)
	Philippsburg 2, PWR	Level 1, full power	1999	2008 (planned)
	Emsland, PWR	Level 1, full power	1999	2001
	Neckarwestheim 2, PWR	Level 2, full power + Level 1, off power	2001	2009 (planned)
	Isar 2, PWR	Level 1, full power	2000	2009 (planned)
	Brunsbüttel, BWR	Level 1, full power	1997	2001
	Isar 1, BWR	Level 1, full power	1995	2001
	Philippsburg 1, BWR	Level 1, full power Level 1, off power	1998 2001	2005 (planned)
	Krümmler, BWR	Level 1, full power	1997	2008 (planned)
	Gundremmingen B, BWR	Level 2, full power + level 1, off power (partial)	1998	2007 (planned)
	Gundremmingen C, BWR	Level 2, full power + level 1, off power (partial)	1998	2007 (planned)

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed
Hungary	Paks NPP, VVER 440/213, Unit 3	Level 1, full power, internal events	1994	1998 ¹
	Paks NPP, VVER 440/213, Units ½	Level 1, full power, internal events	1995	1998 ¹
	Paks NPP, VVER 440/213, Units ½	Level 1, off power, internal events	1997	1998 ¹
	Paks NPP, VVER 440/213, Unit 4	Level 1, full power, internal events	1998	1999 ¹
	Paks NPP, VVER 440/213, Unit 1	Level 1, full power fire + internal flood	1999	2000 ¹
Korea (Republic of)	Kori 3 and 4, & YGN 1 and 2 (PWR, 3 loop)	Level 1	1992	1999
	YGN 3 and 4 (PWR, 2 loop)	Level 2	1993	-
	Ulchin 3 and 4 (PWR, 2 loop)	Level 2	1997	-
	Wolsong 2, 3, and 4 (CANDU)	Level 2	1995	1997
	Yonggwang 5 and 6 (PWR, 2 loop)	Level 2	2001	-
	APR 1400 (PWR, 2 loop, Standard Design)	Level 3	2001	-
Mexico	Laguna Verde, BWR5, Mark II Containment	Level 1 and Level 2 Internal Events, including internal flooding Full Power	January, 1996	Updating is underway

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed
Netherlands	Borselle	Feasibility Study	1988/89	
		Internal Events Level 1 / 2 minus	1990 - 1993	
		External Events / LPSD / Level 2 and 3 for all operational states	1993 - 1997	
		Level 3 all events and POS for new modifications	1997 - 1998	
		Level 3 LPSA	1998 – current	
		Level 3 Risk Monitor	2001 – current	
Spain ²	Garona – BWR GE Mark I	IP scope	1985	2001
	Almaraz – PWR W 3 loop	IP scope	1989	1998
	Asco – PWR W 3 loop	IP scope	1992	2001
	Cofrentes – BWR GE Mark III	IP scope	1991	2000
	Jose Cabrera – PWR W 1 loop	IP scope	1993	2001
	Vandellos – PWR W 3 loop	IP scope	1994	2000
	Trillo – PWR Siemens 3 loop	IP scope	1998	2002 (planned)
Sweden	Barsebäck 1 & 2	Level-1. Transients & LOCA	1998	2002
		Level-2. Full power	1995	2002
		Fire	1998	2002
		Flooding	1998	2002
		Low power, refueling	1995	2003
		Start- and shutdown	1998	2003
		External events	-	2005

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed
Sweden (continued)	Forsmark 1 & 2	Level-1. Transients & LOCA	1995	2001
		Level-2. Full & low power	2001	2003-2004
		Fire	1997, 2001	2003-2004
		Flooding	1997, 2001	2003-2004
		Low power, refueling	2001	2003-2004
		Start- and shutdown	2001	2003-2004
		External events	2001	2003-2004
	Forsmark 3	Level-1. Transients & LOCA	1985,1995, 1998	2002-2003
		Level-2. Full & low power	1995, 1996, 1998	2002-2003
		Fire	1995	2002-2003
		Flooding	1995	2002-2003
		Low power, refueling	1995	2002
		Start- and shutdown	1989	2001
		External events	-	-
	Oskarshamn 1	Level-1. Transients & LOCA	1997	2001-2002
		Level-2. Full & low power	1998-1999	2002-2002
		Fire	1997	2002
		Flooding	1997	2002
		Low power, refueling	1999	2002
		Start- and shutdown	-	2002
		External events	-	2002-2003

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed
Sweden (continued)	Oskarshamn 2	Level-1. Transients & LOCA	1999	-
		Level-2. Full & low power	1999	-
		Fire	1999	2002
		Flooding	-	2002
		Low power, refueling	-	2002
		Start- and shutdown	-	-
		External events	-	-
	Oskarshamn 3	Level-1. Transients & LOCA	1998	2001
		Level-2. Full & low power	1998	2002-2002
		Fire	1998	2003
		Flooding	1998	2003
		Low power, refueling	2001-2002	-
		Start- and shutdown	2001-2002	-
		External events	-	-
	Ringhals 1	Level-1. Transients & LOCA	1992	1999
		Level-2. Full & low power	1996	2002-2003
		Fire	1997	2002-2003
		Flooding	1994	2002-2003
Low power, refueling		1996	2002-2003	
Start- and shutdown		-	2002-2003	
External events		-	-	

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed
Sweden (continued)	Ringhals 2	Level-1. Transients & LOCA	1992	2002
		Level-2. Full & low power	1994	2002
		Fire	1994	2002
		Flooding	1994	2002
		Low power, refueling	1995, 1998	2002
		Start- and shutdown	1992, 1995, 1999	2003
		External events	-	-
	Ringhals 3&4	Level-1. Transients & LOCA	1992	2002-2003
		Level-2. Full & low power	-	2003
		Fire	1997	2003
		Flooding	1997	2003
		Low power, refueling	-	2003
		Start- and shutdown	1990, 1999	2003
		External events	-	-

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed
Switzerland	Beznau (units 1&2) Westinghouse 2 Loop PWR	Level 1 & 2 All internal and external events for full power and shutdown mode No level 2 study for shutdown	1993 (full power level 1 & 2) 1998 shutdown	Periodic updates for full power PSA 2002: detailed review of shutdown PSA
	Gösgen Siemens-KWU 3 Loop PWR	Level 1 & 2 All internal and external events for full power and shutdown mode No level 2 study for shutdown	1994 (full power level 1 & 2) 1994 shutdown	Periodic updates for full power PSA 2005 detailed review of the updated shutdown PSA
	Leibstadt GE BWR/6, Mark-III	Level 1 & 2 All internal and external events for full power and shutdown mode No level 2 study for shutdown	1994 (full power level 1 & 2) 2001 shutdown	Periodic updates for full power PSA 2003: detailed review of shutdown PSA
	Mühleberg GE BWR/4, Mark-I	Level 1 & 2 All internal and external events for full power and shutdown mode No level 2 study for shutdown	1993 (full power level 1 & 2) 1997 shutdown	Periodic updates for full power PSA 2002: detailed review of shutdown PSA

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed	
United Kingdom	Magnox Reactors				
	Calder Hall	Level 1+ PSA	1991	1998	
	Chapel Cross	Level 1+ PSA	1990	1998	
	Bradwell	Level 1+ PSA	1986	1992	
	Hinkley Point A	Level 1+ PSA	1987	1994	
	Dungeness A	Level 1+ PSA	1992	1995	
	Sizewell A	Level 1+ PSA	1988	1994	
	Oldbury	Level 1+ PSA	1990	1997	
	Wylfa	Level 1+ PSA	1998	(2003)	
	AGRs				
		Hinkley Point B	Level 2 PSA	1995 ¹	
		Dungeness A	Level 2 PSA	1996	2001
		Hartlepool	Level 2 PSA	1997 ¹	
		Heysham 1	Level 2 PSA	1997 ¹	
		Heysham 2	Level 2 PSA	1986 L1 PSA)	1998 (L2 PSA)
		Torness			2002 (LPSA)
			Level 2 PSA	1991	1998
		Sizewell B (PWR)	Level 3 PSA	1992	1997 (LPSA)
		CONSORT (Research Rx)	Level 1 PSA	2000	2002
	Other				
		Naval Reactors	Now: Level 1 PSA	2001	
		Plan: Level 3 PSA	(2002)		

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed
United States (see notes 1 and 2)	ANO 1 [PWR, B&W]	Level 1, limited Level 2, internal flooding External Events (see US note 1)	April 1993 December 1991	
	ANO 2 [PWR, B&W]	Level 1, limited Level 2, internal flooding External Events	August 1992 December 1991	
	Beaver Valley 1 [PWR, West. 3-loop]	Level 1, Level 2, internal flooding External Events	October 1992 June 1995	
	Beaver Valley 2 [PWR, West. 3-loop]	Level 1, Level 2, internal flooding External Events	March 1994 September 1997	
	Braidwood [PWR, West. 4-loop]	Level 1, Level 2 External events	September 1992 June 1987	
	Browns Ferry [BWR, 3/4 (4)]	Level 1, Level 2 External events	August 1992 July 1995	
	Brunswick [BWR, 3/4 (4)]	Level 1 External events	August 1992 July 1995	
	Byron [PWR, West. 4-loop]	Level 1, Level 2 External events	April 1994 December 1996	
	Callaway [PWR, West. 4-loop]	Level 1, Level 2 External events	September 1992 June 1995	
	Calvert Cliffs [PWR, CE]	Level 1, Level 2, internal flooding External events	December 1993 August 1997	

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed
United States (see notes 1 and 2) (continued)	Catawba	Level 1, Level 2, Level 3	September 1992	
	[PWR, West. 4-loop]	External events	June 1994	
	Clinton	Level 1	September 1992	
	[BWR, 5/6 (6)]	External events	September 1995	
	Columbia Generating Station [BWR, 5/6 (6)]	Level 1, Level 2	August 1992	
		External events	February 1995	
	Commanche Peak	Level 1, Level 2	November 1992	
	[PWR, West. 4-loop]	External events	June 1995	
	Cooper	Level 1, Level 2	March 1993	
	[BWR, 3/4 (4)]	External events	October 1996	
	Crystal River	Level 1, Level 2, internal flooding	March 1993	
	[PWR, B&W]	External events	June 1996	
	D C Cook	Level 1, Level 2, Level 3 internal flooding	May 1992	
	[PWR, West. 4-loop]	External events	May 1992	
	Davis Besse	Level 1, Level 2	February 1993	
[PWR, B&W]	External events	November 1995		
Diablo Canyon	Level 1, Level 2, seismic	April 1992		
[PWR, West. 4-loop]	External events	June 1994		
Dresden	Level 1, Level 2	January 1993		
[BWR, 1/2 /3 (3)]	External events	December 1995		

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed
	Fort Calhoun [PWR, CE]	Level 1, Level 2, Level 3 External events	December 1993 December 1993	
	Ginna [PWR, West. 2 - loop]	Level 1, Level 2, internal flooding External events	March 1994 September 1992	
	Grand Gulf [BWR, 5/6 (6)]	Level 1, Level 2 External events	December 1992 December 1994	
	H B Robinson [PWR, West. 3 - loop]	Level 1, Level 2 External events	August 1992 June 1995	
	Hatch [BWR, 3/4 (4)]	Level 1, Level 2 External events	December 1992 January 1996	
	Hope Creek [BWR, 3/4 (4)]	Level 1, Level 2 External events	May 1994 July 1997	
	Indian Point 2 [PWR, West. 4 - loop]	Level 1, Level 2 External events	August 1992 December 1995	
	Indian Point 3 [PWR, West. 4 - loop]	Level 1, Level 2 External events	June 1994 September 1997	
	J A Fitzpatrick [BWR, 3/4 (4)]	Level 1, Level 2 External events	September 1991 June 1996	
	Joseph M Farley [PWR, West. 3 - loop]	Level 1, Level 2, internal flooding External events	June 1993 June 1995	

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed
	La Salle	Level 1, Level 2	April 1994	
	[BWR, 5/6 (6)]	External events	April 1994	
	Limerick	Level 1, Level 2	July 1992	
	[BWR, 3/4 (4)]	External events	June 1995	
	McGuire	Level 1, Level 2, Level 3, internal and external events	November 1991	
	[PWR, West. 4- loop]	External events	June 1994	
	Millstone 2	Level 1, Level 2	December 1993	
	[PWR, CE]	External events	December 1991	
	Millstone 3	Level 1, Level 2, Level 3, internal flooding and seismic	August 1990	
	[PWR, West. 4 – loop]	External events	December 1991	
	Monticello	Level 1, Level 2, internal flooding	February 1992	
	[BWR, 3/4 (3)]	External events	March 1995	
	Nine Mile Point 1	Level 1, Level 2, internal events (including flooding)	July 1993	
	[BWR, 1/2/3 (2)]	External events	August 1996	
	Nine Mile Point 2	Level 1, Level 2	July 1992	
	[BWR, 5/6 (5)]	External events	December 1995	

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed
	Oconee [PWR, B&W]	Level 1, Level 2, Level 3, internal and external events External events	November 1990 December 1995	
	Oyster Creek [BWR, 1/2/3 (2)]	Level 1, Level 2 External events	August 1992 December 1995	
	Palisades [PWR, CE]	Level 1, Level 2 External events	January 1993 June 1995	
	Palo Verde [PWR, CE]	Level 1, Level 2, internal flooding External events	April 1992 June 1995	
	Peach Bottom [BWR, 3/4 (4)]	Level 1, Level 2 External events	August 1992 May 1996	
	Perry [BWR, 5/6 (6)]	Level 1, Level 2 External events	July 1992 July 1996	
	Pilgrim [BWR, 3/4 (4)]	Level 1, Level 2 External events	September 1992 June 1994	
	Point Beach [PWR, West. 2- loop]	Level 1, Level 2 External events	June 1993 June 1995	
	Prairie Island [PWR, West. 2- loop]	Level 1, Level 2 External events	March 1994 December 1996	

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed
United States (see notes 1 and 2) (continued)	Quad Cities	Level 1, Level 2	December 1993	
	[BWR, 3/4 (3)]	External events	February 1997	
	River Bend	Level 1, Level 2	December 1993	
	[BWR, 5/6 (6)]	External events	June 1995	
	Salem	Level 1, Level 2	July 1993	
	[PWR, West. 4 loop]	External events	January 1996	
	San Onofre	Level 1, Level 2	April 1993	
	[PWR, CE	External events	December 1995	
	Seabrook	Level 1, Level 2	March 1991	
	[PWR, West. 4- loop]	External events	October 1992	
	Sequoyah	Level 1, Level 2	September 1992	
	[PWR, West. 4 loop]	External events	June 1995	
	South Texas Project	Level 1, Level 2	August 1992	
	[PWR, West. 4 loop]	External events	December 1991	
	Turkey Point	Level 1, Level 2	June 1991	
	[PWR, West. 3 loop]	External events	June 1994	
Saint Lucie	Level 1, Level 2, internal flooding	December 1993		
[PWR, CE]	External events	December 1994		
Summer	Level 1, Level 2, internal flooding	June 1993		
[PWR, West. 3- loop]	External events	June 1995		
Surry	Level 1, internal flooding	November 1991		
[PWR, West. 3- loop]	External events	November 1997		

Country	Plant and Type	PSA Type	Original PSA Date Completed	Revised PSA Date Completed
United States (see notes 1 and 2) (continued)	Susquehanna	Level 1, internal flooding	December 1991	
	[BWR, 3/4 (4)]	External events	June 1994	
	Three Mile Island	Level 1, Level 2	May 1993	
	[PWR, B&W]	External events	December 1994	
	Vermont Yankee	Level 1, Level 2	December 1993	
	[BWR, 3/4 (4)]	External events	June 1998	
	Vogtle	Level 1, Level 2	December 1992	
	[PWR, West. 4 loop]	External events	November 1998	
	Waterford	Level 1, Level 2	August 1992	
	[PWR, CE]	External events	December 1994	
Watts Bar	Level 1, Level 2	September 1992		
[PWR, West. 4 loop]	External events	February 1998		
Wolf Creek	Level 1, Level 2	September 1992		
[PWR, West. 4 loop]	External events	June 1995		

Notes to the Appendix

Belgium

- a) Level 2 limited to containment response; no source term analysis
- b) Level 2 limited to deterministic analysis of containment response for characteristic core melt scenarios.

Hungary

- Annual updating has been performed since the indicated date for the units of the Paks NPP within the framework of its systematic living PSA programme.

Spain

- “IP scope” is the PSA common scope, established by Edition 2 (1998) of the Spanish Integrated Program on PSA, for the seven Spanish NPP, that is, Levels 1 & 2 for full power operation and Level 1 for low power and shutdown operations and for all external events.
- “Original PSA date” is the first version of the PSA and the date of the first submission (partial scope) to the CSN.
- “Revised PSA date” is the currently latest version of the PSA and its date of submission to the CSN. Only Vandellos, Asco and Garona PSA have got already to the IP scope.

United Kingdom

- Further enhancements have been carried out to the original PSAs to support ongoing modification.

United States

- The NRC issued Supplement 4 to Generic Letter 88-20, “Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, 10 CFR 50.54(f) in June 1991. The IPEEE program includes analyses of the following external events: seismic events, internal fires, and high winds, floods, and other external events. The dates in this table for external events indicate the date that the licensee initially submitted their IPEEE report.

Note: The NRC does not require licensee to submit updated PSAs. Therefore, we do not have the revision dates of the individual PSAs.