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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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IMPROVING LOW POWER AND SHUTDOWN PSA METHODS AND DATA TO PERMIT BETTER RISK COMPARISON AND TRADE-OFF DECISION-MAKING

VOLUME 2: RESPONSES TO THE WGRISK SURVEY

Joint Report Produced by the Committee on the Safety of Nuclear Installations (CSNI) Working Group on Risk Assessment and the Cooperative Probabilistic Risk Assessment (COOPRA) program

CAUTION: It is important to note that the information contained in this report was gathered from two surveys, one by COOPRA and the other by WGRisk, which were performed over several years. Since this information is subject to changes, advancements, etc., the reader should take these types of occurrences into account.

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996), Korea (12th December 1996) and the Slovak Republic (14 December 2000). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of 28 OECD Member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Slovak Republic, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- 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.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

Committee on the Safety of Nuclear Installations (CSNI)

The 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 organizations, 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 organizations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups, and organization of conferences and specialist meeting.

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 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.

Cooperative Probabilistic Risk Assessment (COOPRA) Research Program

COOPRA is a U.S. Nuclear Regulatory Commission (USNRC) sponsored organization that includes member organizations from other countries. The goals of COOPRA are to improve probabilistic safety assessment (PSA) technology through the timely sharing of research information, and optimize use of members' resources through coordinated and cooperative research projects. COOPRA provides an international forum for technical experts to exchange information on safety assessments for commercial nuclear power plants.

The COOPRA organization consists of a Steering Committee and working groups in various technical areas of interest. The Steering Committee consists of representatives from each member organization, and meets annually. The first Steering Committee meeting was held in October 1997. Currently COOPRA has three working groups: fire-induced damage to electrical cables and circuits, low power and shutdown, and risk-informed decision-making. The working groups identify key technical/regulatory issues, formulate and execute collaborative research and development projects, report on work progress at Steering Committee meetings, and provide members with timely information, research results, and reports on working group activities.

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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. COOPRA objectives are to improve the sharing of Probabilistic Safety Assessment (PSA) information, and to facilitate the efficient development and use of needed PSA tools.

The mission of the CSNI Low Power and Shutdown (LPSD) Working Group (WG) on Risk Assessment is to advance the understanding and utilization of PSA in ensuring continued safety of nuclear installations in Member countries. In pursuing this goal, the WG shall recognize the different methodologies for identifying contributors to risk and assessing their importance. While the WG 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.

The COOPRA LPSD working group is charged with the responsibility to assess their Member country's plant operations at LPSD conditions. The sharing of information is expected to provide each of the Member country the means from which to render informed regulatory decisions for the benefit of public health and safety.

Each organization had developed a questionnaire to gather information from Member countries on LPSD PSAs experiences. The responses cover a broad spectrum of LPSD PSA topics, and identifies work for improving risk-informed trade-off decisions, using PSA techniques, between LPSD and full power operational states. Each organization recognized potential benefit for improving the state-of-the-art by combining the wealth of experiences from the questionnaire responses into a common report.

CAUTION: It is important to note that the information contained in this report was gathered from two surveys, one by COOPRA and the other by WGRisk, which were performed over several years. Since this information is subject to changes, advancements, etc., the reader should take these types of occurrences into account.

- Volume 1of this series contains the Summary from the CSNI/WGRisk and COOPRA Surveys
- Volume 3 of this series contains the responses from the COOPRA Survey

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INTRODUCTION

This report provides the responses to the CSNI LPSD questionnaire. The structure of this report is to present the responses for each PSA element in the questionnaire in an appendix. The content of each appendix are noted below.

Appendix A – Scope and Objective of the LPSD PSA

This appendix provides the overview of the international scope and objectives of the LPSD PSA studies. The scope and objectives influence the modelling of LPSD PSA elements such as plant operating states, which are also discussed in this chapter.

Appendix B – Initiating Events

This appendix discusses the identification as well as screening and grouping of initiating events.

Appendix C – Risk Metrics

The risk metrics used for the various applications are presented in this appendix.

Appendix D – Sequence Analysis

Accident sequence modelling and success criteria are presented in this appendix.

Appendix E – System Modelling

The various system modelling considerations are explored.

Appendix F – Data

This appendix presents the uses of data for LPSD conditions.

Appendix G – Human Reliability Analysis

Insights show the importance of human reliability analysis in LPSD PSAs. This appendix presents the various approaches to modelling human reliability.

Appendix H – Quantification / Sensitivity Studies / Uncertainties

This appendix presents the responses on the types of LPSD PSA quantification, sensitivity studies, and uncertainty considerations.

Appendix I – Plant Damage States

Plant damage states are presented and analyzed in this appendix.

Appendix J – Containment Performance

Containment performance, severe accident analysis, and Level 2 phenomenology responses are presented in this appendix.

Appendix K – Consequence Analysis

This appendix presents the consequence analysis for the LPSD studies performed.

Appendix L – Results of the LPSD PSA / Insights

This appendix provides insights gained from the LPSD studies, and examines the confidence in allowing risk trade-off studies.

Appendix M – Use of the LPSD PSA

The concise summary of the LPSD PSA uses was provided in chapter 2. This appendix presents the uses in more detail.

Appendix N– Areas for Research / Development

Areas for research and development are identified here as well as through-out the previous appendices.

APPENDIX A - SCOPE AND OBJECTIVES OF THE LPSD PSA

A.1 Scope and Objectives

PRELUDE AND QUESTIONS

The scope of the LPSD PSA is defined by the following characteristics:

- the degree of coverage of initiating events, either internal or external to the plant boundary, that results in off-normal conditions
- the level of risk characterization – that is, a Level 1 PSA that estimates the core damage frequency (CDF) (given an event that challenges plant operation); a Level 2 PSA that estimates the containment failure and radionuclide release frequencies (given a core damage state); and a Level 3 PSA that estimates the offsite consequences from a release (given a radionuclide release), and
- the sources of radioactive material addressed in that analysis.

The objectives relate to (current or planned) uses of the LPSD PSA in decision-making.

Please respond to the following questions.

1. What is the scope of the LPSD PSA?

a) What is the level of the PSA [Level 1, Level 2, Level 3]?

b) What categories of initiating events are included: internal events; internal hazards (fire, flood, etc); external events (earthquake, etc.)?

c) What sources of radioactive material are considered in the analysis (reactor core, fuel in the fuel storage pond, radioactive waste)?

2. What is the reason for carrying out the LPSD PSA?

a) What are the intended applications?

b) What future applications are being considered?

RESPONSES

BELGIUM

1. What is the scope of the LPSD PSA?

a) What is the level of the PSA [Level 1, Level 2, Level 3]

The level is Level 1 PSA only for LPSD.

b) What categories of initiating events are included: internal events; internal hazards (fire, flood, etc); external events (earthquake, etc.)?

Internal events only for LPSD are included.

c) What sources of radioactive material are considered in the analysis (reactor core, fuel in the fuel storage pond, radioactive waste)?

Reactor core only is considered.

2. What is the reason for carrying out the LPSD PSA?

The LPSD PSA is performed as part of the Periodic Safety Review to assess the level of safety of the plant during shutdown conditions.

a) What are the intended applications?

AVN (Regulatory organization) performs PSA Based Event Analysis in the framework of its experience feedback activities.

b) What future applications are being considered?

None at the moment are being considered.

GERMANY

1. What is the scope of the LPSD PSA?

a) What is the level of the PSA [Level 1, Level 2, Level 3]?

Level 1, mainly up to hazard states (i.e. without considering repair of components and accident management measures)

b) What categories of initiating events are included: internal events; internal hazards (fire, flood, etc); external events (earthquake, etc.)?

Internal events (transients, LOCAs and faulty injection of unborated coolant)

c) What sources of radioactive material are considered in the analysis (reactor core, fuel in the fuel storage pond, radioactive waste)?

The considered sources are the reactor core and the fuel storage pond

2. What is the reason for carrying out the LPSD PSA?

a) What are the intended applications?

The intended application is the evaluation of the PSA methodology for LP&SD-conditions

b) What future applications are being considered?

The application of the evaluated PSA-methodology in a LP&SD-PSA-guideline has been carried out.

HUNGARY

1. What is the scope of the LPSD PSA?

a) What is the level of the PSA [Level 1, Level 2, Level 3]?

The level is Level 1.

b) What categories of initiating events are included: internal events; internal hazards (fire, flood, etc); external events (earthquake, etc.)?

Internal events are included.

c) What sources of radioactive material are considered in the analysis (reactor core, fuel in the fuel storage pond, radioactive waste)?

Reactor core is considered. Ex-core situations have not been addressed.

2. What is the reason for carrying out the LPSD PSA?

a) What are the intended applications?

Safety upgrading measures have been implemented based on the LPSD PSA. Some further plant improvements are also expected.

The review of all of the current Technical Specifications is an ongoing activity in Hungary. One modification has been recently proposed based on the results of a systematic, PSA-based review of the boron dilution possibilities performed separately from the LPSD PSA study within a PHARE project. The proposed modification is concerned with the definition of a reserve loop.

The feasibility of configuration control/management during LPS conditions have not been studied yet in detail. However, selected configurations have been analyzed in each plant operational state for the purpose of relaxation of the maintenance program.

b) What future applications are being considered?

- feedback to safety improvement
- configuration control, risk monitor
- precursor event studies for events occurring at LPS conditions

JAPAN

1. What is the scope of the LPSD PSA?

a) What is the level of the PSA [Level 1, Level 2, Level 3]?

The level of the LPSD PSA carried out in NUPEC is the level 1 PSA.

NUPEC has started level 2PSA trial analysis for the PWR plant at the LPSD.

b) What categories of initiating events are included: internal events; internal hazards (fire, flood, etc); external events (earthquake, etc.)?

The initiating events are limited to the internal events in LPSD PSA currently carried out by NUPEC. However, flooding is not included.

On the other hand, NUPEC has started the trial analysis of Fire PSA for the PWR plant at the LPSD operation.

c) What sources of radioactive material are considered in the analysis (reactor core, fuel in the fuel storage pond, radioactive waste)?

It is carrying out only for discharge of the radioactive material from the reactor core in the present LPSD PSA.

2. What is the reason for carrying out the LPSD PSA?

a) What are the intended applications?

The Japanese utilities make LPSD PSA to secure the safety of NPP during LPSD operation as a part of PSR (Periodic Safety Review). NUPEC supports METI to review the PSR, carrying out LPSD PSA for the plant independently of utilities.

b) What future applications are being considered?

Application of fire PSA and level 2 PSA at the LPSD is considered.

KOREA

1. What is the scope of the LPSD PSA?

a) What is the level of the PSA [Level 1, Level 2, Level 3]?

Level 1 was performed.

b) What categories of initiating events are included: internal events; internal hazards (fire, flood, etc); external events (earthquake, etc.)?

The scope of APR-1400 PSA includes the evaluation of internal events and external events during shutdown modes. Since procedures and design details were not yet available, the events analyzed were performed using simplified analysis approach. In terms of evaluation of external events, the risks associated with internal fire and internal floods were quantified.

c) *What sources of radioactive material are considered in the analysis (reactor core, fuel in the fuel storage pond, radioactive waste)?*

Reactor core is considered only.

2. What is the reason for carrying out the LPSD PSA?

a) *What are the intended applications?*

The primary objective of the probabilistic assessment of risks during shutdown operation is to provide insights into potential plant vulnerabilities. Since a probabilistic assessment of risk necessarily involves quantification, it is important to emphasize the valid uses of the numerical results. Numerical results in shutdown risk assessment have two purposes;

- To point out areas where improvement is needed (schedule, procedures, or plant equipment) and prioritize as necessary,
- To decide between competing risks, when faces with alternate actions.

The objectives of the analysis do not require an accurate prediction of risk. And the results of the analysis should not be used to compare risk during shutdown with risk during full power operation.

MEXICO

1. What is the scope of the LPSD PSA?

As background, it is important to address the main activities performed until now in Mexico regarding LPSD PSA. The first activities in the field of LPSD PSA was performed by the regulatory authority (CNSNS) with the principal objective to obtain an understanding and preliminary estimation of their contribution in the overall plant risk. Therefore, the CNSNS started the review of the LPSD PSA methods and results used worldwide in order to define the convenience to adopt one or to develop proper methodologies to accomplish the project principal objective. The review process concluded that the methodology used in the NUREG/CR-6143 is suitable for the project principal objective and some specific tasks were identified to be required to be adapted to the Laguna Verde plant conditions. Therefore, the methodology described in the NUREG/CR 6143 was followed to identify and define LVNPP POS. The POSs obtained were analyzed by using modified full power models, when appropriate, or by adapting the NUREG models (basically event trees) to obtain an estimation of the CDF for each POS. Based on the POS risk contribution, candidates for a detail PSA analysis will be determined. Estimation of the CDF, using bounding calculation, for the first four POS is already finished. Adaptation of the NUREG models to reflect LVNPP characteristics for the remaining three POS have been planned in order to continue the tasks project. Therefore, the following answers should be understood in the context of the work performed in the first four POS and the process referenced.

a) What is the level of the PSA [Level 1, Level 2, Level 3]?

The level is Level 1.

b) *What categories of initiating events are included: internal events; internal hazards (fire, flood, etc); external events (earthquake, etc.)?*

Internal events are included.

c) *What sources of radioactive material are considered in the analysis (reactor core, fuel in the fuel storage pond, radioactive waste)?*

Reactor core is considered.

2. What is the reason for carrying out the LPSD PSA?

a) *What are the intended applications?*

To estimate their contribution on the overall plant risk.

b) *What future applications are being considered?*

PSA Applications to support evaluation of plant changes such as safety improvements, changes in licensee basis, etc.

THE NETHERLANDS

1. What is the scope of the LPSD PSA?

a) *What is the level of the PSA [Level 1, Level 2, Level 3]?*

b) *What categories of initiating events are included: internal events; internal hazards (fire, flood, etc); external events (earthquake, etc.)?*

c) *What sources of radioactive material are considered in the analysis (reactor core, fuel in the fuel storage pond, radioactive waste)?*

The scope of the Borssele PSA including LPSD part is a level 3 PSA comprising internal events and hazards and external events such as: airplane crash, toxic gas clouds or gas cloud explosions due to the heavy shipping on the Schelde river adjacent to the Borssele-site. Both core and fuel in fuel storage pool are included. In the following Figure the main safety figures of the Borssele NPP are shown.

2. What is the reason for carrying out the LPSD PSA?

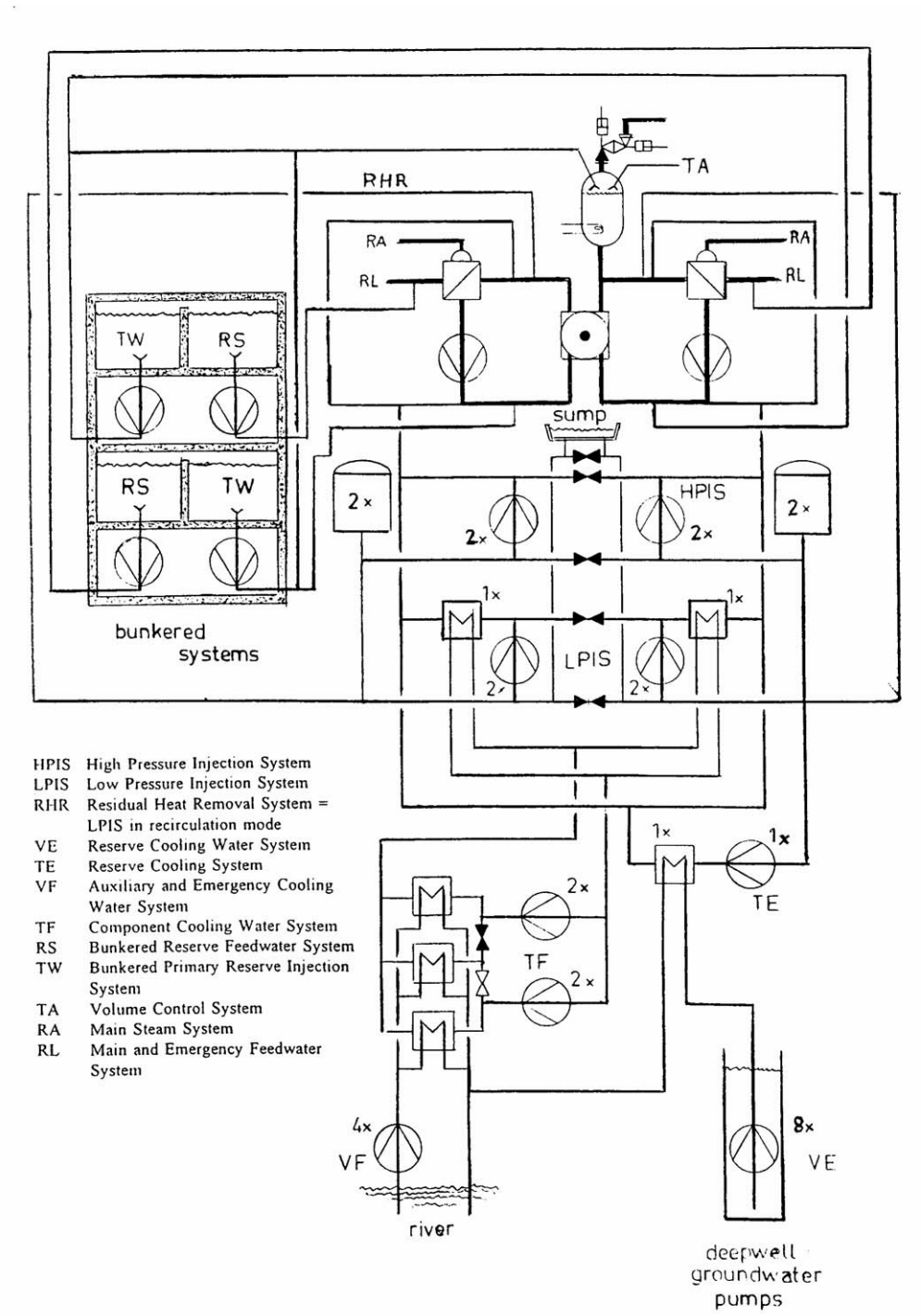
a) *What are the intended applications?*

b) *What future applications are being considered?*

The original objective of the LPSD PSA was to provide a complete picture of weaknesses in plant design and operation as an additional source of information in the ongoing modification/backfitting programme at that time.

Currently the PSA is used for risk monitoring (including configuration control during shut down), optimizing test and maintenance strategies (e.g., maintenance shifted from shut-down period to full power operational state), optimisation of Tech, Specs, etc.

Figure 1: Safety features of core injection & RHR systems at Borssele NPP



SPAIN

1. What is the scope of the LPSD PSA?

a) What is the level of the PSA [Level 1, Level 2, Level 3]?

All the Spanish LP&SD PSAs are Level 1.

b) What categories of initiating events are included: internal events; internal hazards (fire, flood, etc); external events (earthquake, etc.)?

Internal events only are included.

c) What sources of radioactive material are considered in the analysis (reactor core, fuel in the fuel storage pond, radioactive waste)?

Scenarios with fuel damage in spent fuel pools were initially considered but removed by screening later.

2. What is the reason for carrying out the LPSD PSA?

Spanish plants are fulfilling a CSN (Consejo de Seguridad Nuclear, Spanish Regulatory Body) program called "Integrated PSA performing and using program" that has as a goal to obtain a full scope PSA for each plant, including LP&SD PSA.

a) What are the intended applications?

Some plants that have already completed their LP&SD PSA are using it for RI-ISI purposes. The LP&SD PSA provides the analysis with qualitative insights of the shutdown risk. Currently, the scope of such studies is only ASME Class 1 pipes.

b) What future applications are being considered?

Many of the PSA applications require addressing the shutdown risk. The CSN accepts qualitative assessment while a LP&SD PSA is not available for a given plant, but in the future the shutdown risk insights for any application might be developed from LP&SD PSA.

On the other hand, the development of shutdown modes risk monitors is foreseen.

SWEDEN

The age of the Swedish LPSD studies differs a lot, as well as does the level of extent and scope of those.

The following table show when the studies were completed or the present status.

Plant	Year of finalisation
B2	1995
R1	Ongoing
R2	2001 R2 (R3/4) 2001. The most recent study was made for Ringhals 2 2001, and is considered to be applicable to Ringhals 3 and 4 as well. R2 1992, 1996. Previously, LPSD PSA has been performed for Ringhals 2 1992, updated 1996.
R3	2001 (results from R2)
R4	2001 (results from R2)
O1	1999
O2	Ongoing
O3	Ongoing
F1	1999
F2	1999
F3	1995. The Forsmark 3 study from 1995 is only a simplified pilot study for LPSD. In the near future both LPSD models will be extended and uniformed

1. What is the scope of the LPSD PSA?

a) What is the level of the PSA [Level 1, Level 2, Level 3]?

b) What categories of initiating events are included: internal events; internal hazards (fire, flood, etc); external events (earthquake, etc.)?

c) What sources of radioactive material are considered in the analysis (reactor core, fuel in the fuel storage pond, radioactive waste)?

In the "SKI REMARK column" we try to give the opinion of the authority or comments on the CSNI questions or questionnaire.

CSNI Question number	**Answer from Swedish plant**	**Answer**
Q1	Forsmark1/2	Analysis of internal events, fire and flooding.
	R1	To make an assessment of the risk during low power and shutdown operations. (LPSD)
	R2-4	To make an assessment of the risk during low power and shutdown operations. (LPSD)
	B2	The study focus on a typical outage. It covers internal events, fire, flooding and to some extent external events.
	O1	Only internal events such as Transients (CCI, spurious isolation signals, loss of off-site power etc.), Pipe Break and Human Errors (as initiating events) are included in current version. Answer only valid for low power PSA, not outage period.
	O2	Internal events (LOCA, Transients and CCI)
	O3	Internal events (LOCA, Transients and CCI) Note: At present the

		LPSD PSA for O3 only covers shutdown and start-up. A PSA for the refuelling outage period is planned.
SKI REMARK	The latest and ongoing studies focus on internal events	
Q1 a	F1/2	Level 1 and Level 2
	R1	Level 1 and level 2.
	R2-4	R2 2001 - Level 1 R2 1996 - Level 2
	B2	Level 2
	O1	PSA Level 1 Answer only valid for low power PSA, not outage period.
	O2	Level 1 and 2
	O3	Level 1 and 2
SKI REMARK	It is a demand from SKI to make level 1 and level 2 LPSD studies. According to the Swedish regulation SKIFS 1998:1, all operating modes (= normal operation) have to be analysed with PSA.	
Q1 b	F1/2	In the LP analysis the events are related to shut down phase (decrease of power phase Tp) and the start up phase (increase of power phase Tu). The initiating events for LP-analysis are the same as fore the full power analysis: LOCA's, LOOP, fire and flooding. For the SD-analysis LOCA's are divided into the categories below and under and over upper core level. Also loss of RHR is an initiating event. Fire and flooding are treated in a limited way with basis from fire and full power analysis.
	R1	Internal events and internal hazards in the low power analysis are the same as for power operation. For SD-analysis LOCAs and loss of RHR are considered.
	R2-4	R2 2001: Loss of residual heat removal, inadvertent boron dilution, total loss of heat sink, total loss of power supply, LOCAs, separate studies of heavy lifts have been done for Ringhals 3 and 4. R2 PSA from 1995 covered incidents during fuel movements, cold pressurisation of reactor vessel and SGs.
	B2	Drainage of the reactor pressure vessel, loss of decay heat removal, heavy lifts, fuel movements, criticality events, fire, explosion, cold pressurisation, pressure test of containment, flooding, external events (loss of offsite power)
	O1	Only internal events such as Transients (CCI, spurious isolation signals, loss of off-site power etc.), Pipe Break and Human Errors (as initiating events) are included in current version. As long as no increased frequency for fire, flooding etc can be identified during shutdown and start-up conditions these events (hazards) can be excluded. No external events have been included. Answer only valid for low power PSA, not outage period.
	O2	Internal events
	O3	Internal events
SKI REMARK	The latest and ongoing Swedish LPSD studies consider LOCAs and loss of RHR	
Q1 c	F1/2	Reactor core.
	R1	Reactor core, fuel in the fuel storage pond.
	R2-4	R2, reactor core

	B2	Reactor core, fuel in the fuel storage pond.
	O1	Reactor core Answer only valid for low power PSA, not outage period.
	O2	Reactor core and fuel in fuel storage pond
	O3	Reactor core
SKI REMARK		Reactor core and fuel in fuel storage pond

2. What is the reason for carrying out the LPSD PSA?

a) What are the intended applications?

b) What future applications are being considered?

CSNI Question number	Answer from Swedish plant	Answer
Q2	F1/2	Too calculate the risk and to find out possible weaknesses in the station. The outer aim is to get a balanced design
	R1	To assess the safety during none full-power operations.
	R2-4	R2, To assess the safety during none full-power operations. Update 2001 in connection with introduction of the EdF shutdown instruction package.
	B2	A requirement from the regulatory authority in connection with the periodical safety review
	O1	The PSA serves the following purposes: Give a measurement of the plants safety level. Identify weaknesses in the plant. Identify those manual actions that are of most importance to plant safety. Be a basis for instantaneous risk applications. Answer only valid for low power PSA, not outage period.
	O2	To make the PSA cover all POS
	O3	To make the PSA cover all POS
SKI REMARK	According to the Swedish regulation SKIFS 1998:1, all operating modes (= normal operation) have to be analysed with PSA.	
Q2 a	F1/2	Mapping of risk and identification the strength and weakness. Analysis of plant changes and risk follow up as a part of Forsmark safety index.
	R1	Safety improvements
	R2-4	R2, Safety improvements
	B2	Safety improvements
	O1	Give a measurement off the plants safety level. Identify weaknesses in the plant. Identify those manual actions that are of most importance to plant safety. Answer only valid for low power PSA, not outage period.
	O2	Assess complete CDF and compare to safety goals. Modernisation alternative selection. Risk follow up.

	O3	Assess complete CDF and compare to safety goals. Modernisation alternative selection. Risk follow up.
SKI REMARK	The main purpose for all the studies is to give a measurement of the plant´s safety level and to identify weaknesses in the plant	
Q2 b	F1/2	TS optimisation
	R1	Part of the FSAR, to analyse different mode of none full-power operations and assessments of major modifications.
	R2-4	R2, No further applications are considered until an updated analysis of LPSD is performed.
	B2	No further applications are considered until an updated analysis of LPSD PSA will be performed
	O1	Be a basis for instantaneous risk applications. Answer only valid for low power PSA, not outage period.
	O2	Test optimisation. Risk monitoring.
	O3	Test optimisation. Risk monitoring.
SKI REMARK	A joint Nordic project to harmonise LPSD method descriptions will start in the near future	

UNITED KINGDOM

The reply relates to the PSA for Low Power and Shutdown that has been carried out for the Sizewell B PWR.

1. What is the scope of the LPSD PSA?

a) What is the level of the PSA [Level 1, Level 2, Level 3]?

The PSA for the Sizewell B PWR is a full scope Level 3 PSA.

b) What categories of initiating events are included: internal events; internal hazards fire, lood, etc); external events (earthquake, etc.)?

It includes all initiating events – that is, all internal events, all internal hazards and all external hazards (including seismic events).

c) What sources of radioactive material are considered in the analysis (reactor core, fuel in the fuel storage pond, radioactive waste)?

All sources of radioactive material on the site are included. This includes the reactor core, fuel in the fuel storage pond and radioactive waste.

2. What is the reason for carrying out the LPSD PSA?

a) What are the intended applications?

To demonstrate that the Safety Case is complete. Aid station in decision making during shutdown. Support of modifications to systems and procedures during shutdown

b) What future applications are being considered?

Possible use for risk informed decision making.

United States

BWR

1. What is the scope of the LPSD PSA?

a) What is the level of the PSA [Level 1, Level 2, Level 3]?

The level is Level 3.

b) What categories of initiating events are included: internal events; internal hazards (fire, flood, etc); external events (earthquake, etc.)?

Internal events, internal fire and flood events, and seismic (earthquake) events are included.

c) What sources of radioactive material are considered in the analysis (reactor core, fuel in the fuel storage pond, radioactive waste)?

Reactor core is considered.

2. What is the reason for carrying out the LPSD PSA?

a) What are the intended applications?

The objectives of the analysis included:

1) Estimate the frequencies of severe accidents initiated during important plant operational modes other than full power operation for the Grand Gulf nuclear plant, specifically POS 5 (essentially cold shutdown) during a refuelling analysis.

2) Compare the estimated core damage frequencies, important accident sequences, and other qualitative and quantitative results of this study with those of accidents initiated during full power operation (assessed in NUREG-1150).

3) Demonstrate methodologies for accident sequence analysis for plants in other than full power modes of operation (i.e., POS 5).

4) Develop methods to compute the risk, utilizing to the extent possible the component analyses developed as part of the NUREG-1150 program.

5) Identify those factors that have the most impact on the risk estimates and highlight unique features of the risk analysis performed.

6) Compare results against the risk of full power operation as evaluated in the NUREG-1150 study.

b) What future applications are being considered?

LPSD PSAs may be used to support risk-informed regulatory activities, for example, making changes to the licensing basis using Regulatory Guide 1.174, events assessment, inspection and enforcement, maintenance rule, and PSA standards. LPSD PSAs have been used in evaluating the design of advanced reactors, developing tools for evaluating risk significance of inspection findings, and for proposed rulemaking.

PWR

1. What is the scope of the LPSD PSA?

a) What is the level of the PSA [Level 1, Level 2, Level 3]?

1) NUREG/CR-6144- Level 3

2) Screening analysis of NUREG/CR-6144- Level 1

3) NUREG/CR-6616 and 5718- Level 3

b) What categories of initiating events are included: internal events; internal hazards (fire, flood, etc); external events (earthquake, etc.)?

1) NUREG/CR-6144 - Internal events (level 3), internal floods (level 1), internal fires (level 1), seismic (level 1)

2) Screening analysis of NUREG/CR-6144 - Internal events, internal floods, internal fires

3) NUREG/CR-6616 and 5718 - Internal events

c) What sources of radioactive material are considered in the analysis (reactor core, fuel in the fuel storage pond, radioactive waste)?

Reactor core is considered.

2. What is the reason for carrying out the LPSD PSA?

a) What are the intended applications?

1) NUREG/CR-6144- The level-1 objectives of the study are: (1) Estimate the frequencies of severe accidents that might be initiated during mid-loop operation, (2) compare the estimated core-damage frequencies, important accident sequences, and other qualitative and quantitative results of this study with those of accidents initiated during full power operation (as assessed in NUREG-1150), and (3) demonstrate methodologies for accident sequence analysis for plants in modes of operation other than full power. The objectives of the level 2/3 study are: (1) Develop methods to compute the risk of the Surry plant during mid-loop operation and perform the study, utilizing to the extent possible the component analyses developed as part of the NUREG-1150 program, (2) identify those factors that have the most impact on the risk estimates and highlights unique features of the risk analysis performed, and (3) compare the results of the study against the risk of full power operation as evaluated in the NUREG-1150 study of Surry and the NRC safety goals.

2) Screening analysis of NUREG/CR-6144- The objective of the study was to identify potential vulnerable plant configurations, to characterize (on a high, medium, or low basis) the potential core damage accident scenarios, and to provide a foundation for a detailed analysis.

3) NUREG/CR-6616- The objective of this study is to compare the risks of scheduled maintenance during power operation vs. shutdown for a PWR and develop insights on on-line maintenance practices for PWRs. NUREG/CR-5718 (draft) will extend the level-1 cold shutdown PRA model of NUREG/CR-6616 to include in the core damage frequency calculation, the fraction of time the plant spent in different time windows of a cold shutdown conditions, and the maintenance unavailabilities, such that the CDF could be added to the CDFs of other plant conditions.

b) What future applications are being considered?

LPSD PSAs may be used to support risk-informed regulatory activities, for example, making changes to licensing basis using Regulatory Guide 1.174, events assessment, inspection and enforcement, maintenance rule, and PSA standards. LPSD PSAs have been used in evaluating the design of advanced reactors, developing tools for evaluating risk significance of inspection findings, and for proposed rulemaking.

A.2 Plant Operating States (POSs)

PRELUDE AND QUESTIONS

POSs are used to subdivide the plant operating cycle into unique states such that the plant response can be assumed to be the same for all subsequent accident initiating events. Operational characteristics (such as reactor power level; in-vessel temperature, pressure, and coolant level; equipment operability; and changes in decay heat load or plant conditions that allow new success criteria) are examined to identify those important to defining plant operational states. A LPSD PSA, unlike a full power PSA, deals with a whole spectrum of POSs, each representing a particular plant state as the plant goes from full to low power, to shutdown, and back to full power. Therefore, in order to reduce the level of effort needed to perform a LPSD PSA, the analysts typically identify the different the POSs a plant may undergo and screen them, i.e., identifies those that appear to be more important to risk and eliminate those that are not.

It appears that there is less than ideal uniformity or consistency in the definition of a POS; different analysts use different parameters to define a POS. The extent that the selection is based on the parameters of plant design and operations versus the analyst(s) input is not clear. Also, the POSs are not uniformly or consistently screened. In some PSAs, all POSs involved in LPSD are analyzed, but in most PSAs only a few are analyzed (cold shutdown and refuelling). However, it has been shown that POSs (such as hot shutdown and reactor startup) are not typically considered and could be important in the PSA results.

The explicit modelling of unplanned outages and of the “transition” from one state to another is typically not included in a LPSD PSA. The term “transition” refers to the plant conditions between any two “steady state” POSs modelled in the analysis. An example of a transition is the time between intermediate shutdown and cold shutdown in a PWR in which heat removal is transferred from the steam generators to the RHR system, or transition between two states with significantly different water inventories such as, during mid-loop operations or during refuelling. Experts have expressed the opinion that unplanned outages could be important since they may introduce different challenges to the plant from those encountered during planned outages. Also, experts have expressed the opinion that transitions should be explicitly modelled in LPSD PSAs because they represent plant evolutions of higher risk.

A plant undergoes different configurations in terms of equipment alignment within a POS. Some configurations put the plant in a more vulnerable state than others because of the limited amount of equipment available to cool the reactor in case of an initiating event. In order to reduce the level of effort

needed for a LPSD PSA, analysts identify and screen configurations. Similarly, there is little uniformity or consistency in the definition, identification, and screening of plant configurations within a POS. Furthermore, plant configurations vary from outage-to-outage and therefore, the risk associated with these configurations may also vary.

Inconsistencies in the definition, identification, and screening of POSs and plant configurations may result in producing erroneous results with respect to associated LPSD risk. In addition, it makes it difficult to compare results and derive insights needed for meaningful comparisons and tradeoffs. Please respond to the following questions.

Please respond to the following questions.

3. What POSs have been modelled explicitly in the LPSD PSA? [Please provide details of the set of POSs that have been analyzed and how they have been defined in terms of the decay heat level, the activities being carried out during the period of the POS, the condition in the reactor coolant system, the availability of safety systems, plant configurations, etc.]

4. How has this set of POSs been identified? [Please provide details of how the factors which affect the risk have been identified and taken into account in identifying the POSs.]

5. What screening, grouping or other process has been used to select the POSs explicitly addressed in the LPSD PSA?

6. How have you been able to determine that this set of POSs is adequate? [How have you been able to determine that no further subdivisions in the set of POSs addressed in the analysis are required?]

7. Does the set of POSs include unplanned as well as planned modes of operation?

8. Does the set of POSs modelled in the LPSD PSA include transition modes?

9. What areas of your analysis are, in your opinion, most in need of improvement?

10. What are the areas that would most benefit from further research?

RESPONSES

BELGIUM

3. What POSs have been modelled explicitly in the LPSD PSA? [Please provide details of the set of POSs that have been analyzed and how they have been defined in terms of the decay heat level, the activities being carried out during the period of the POS, the condition in the reactor coolant system, the availability of safety systems, plant configurations etc.]

The LPSD PSA considers the following POS:

1) intermediate shutdown and normal cold shutdown with RHR connected

- RCS pressure < 30 bar RCS completely filled
- Unavailabilities based on plant specific data

- SG are available

cold shutdown for intervention

- RCS pressure is atmospheric and RCS is vented
- SG are not available

two sub POS are considered

- RCS inventory complete decay heat level corresponds to 2.5 days after shutdown (40°C in RCS) reduced availability of safety systems due to preventive maintenance
- RCS inventory partially filled : mid-loop to reactor vessel flange level decay heat level corresponds to 2.5 days after shutdown (40°C in RCS)

4. How has this set of POSs been identified? [Please provide details of how the factors which affect the risk have been identified and taken into account in identifying the POSs.]

The following criteria were used:

- Starting point: operational states as defined in Technical Specifications
- In addition: Possibility of heat transfer by SG in case of loss of RHR
- In some cases: Time delay available for operator response based on RCS inventory

5. What screening, grouping or other process has been used to select the POSs explicitly addressed in the LPSD PSA?

The POS refuelling with filled reactor pool has been screened from the PSA because of extended time available before unacceptable conditions (boiling in the pool)

6. How have you been able to determine that this set of POSs is adequate? [How have you been able to determine that no further subdivisions in the set of POSs addressed in the analysis are required?]

The POS modelled are conservative regarding decay heat level. The shutdown risk profile of the plant therefore is bounding. Following the first analysis, the POS cold shutdown for intervention has been split into two sub-POSs.

7. Does the set of POSs include unplanned as well as planned modes of operation?

Only planned modes of operation are included.

8. Does the set of POSs modelled in the LPSD PSA include transition modes?

No.

9. What areas of your analysis are, in your opinion, most in need of improvement?

To be determined.

10. What are the areas that would most benefit from further research?

To be determined.

GERMANY

3. What POSs have been modelled explicitly in the LPSD PSA? [Please provide details of the set of POSs that have been analysed and how they have been defined in terms of the decay heat level, the activities being carried out during the period of the POS, the condition in the reactor coolant system, the availability of safety systems, plant configurations etc.]

The modelled POSs refer to a standard two week refuelling outage in the reference plant (see Table below)

Thirteen (13) POSs have been defined, the definition includes the status of the primary system, the availability of operating and safety systems and the decay heat level.

PLANT OPERATION STATES OF A TWO-WEEK OUTAGE IN THE REFERENCE NPP

No.	Changes in physical condition / *Systems features*
(1)A0	Power reduction to the condition subcritical hot / *Reactor protection signals and availability of the safety systems as during power operation*
(1)A1	Shutdown via steam generators down to primary system pressure 3,1 MPa and primary system temperature 120 °C / *All reactor protection systems still available*
(1)B1	Primary system cooldown to depressurised cold / *Start-up of the residual-heat removal (RHR) system at 120 °C; accumulators and HP-pumps are disconnected*
(1)B2	Level lowering to mid-loop, mid-loop operation / *Core within the reactor pressure vessel (RPV), primary system pressure-tight closed*
(1)C	Opening the RPV head, mid-loop operation / *Core within the RPV, primary system not pressure-tight closed, refuelling hatch between setdown- and fuel pool closed*
(1)D	Flooding of reactor cavity, unloading of the fuel elements / *Core wholly or in part within the RPV, refuelling hatch open*
E	Emptying of reactor cavity and RPV / *Core fully unloaded, refuelling hatch closed, work performed at lower-edge loop level*
(2)D	Refilling of reactor cavity, loading of fuel elements / *Core wholly or in part within the RPV, refuelling hatch open*
(2)C	Level lowering to mid-loop, closing of the RPV head / *Core within the RPV, primary system not pressure-tight closed, refuelling hatch closed*
(2)B2	Evacuation and refilling of the primary system / *Core within the RPV, primary system pressure-tight closed*
(2)B1	Primary system heat-up with the main coolant pumps / *All reactor protection systems available*

PLANT OPERATION STATES OF A TWO-WEEK OUTAGE IN THE REFERENCE NPP (Continued)

No.	Changes in physical condition / *Systems features*
(2)A1	Deborating of the coolant and taking the reactor to critical condition / *Withdrawal of control rods or/and deboration*
(2)A0	Power increase up to specified level / *Reactor protection signals and availability of the safety systems as during power operation*

4. How has this set of POSs been identified? [Please provide details of how the factors which affect the risk have been identified and taken into account in identifying the POSs.]

The set of POSs was identified after an evaluation of plant specific procedures and outage plans for shutdown and for restart.

The main criteria for the classification of the POS was: "The boundary conditions within a POS should be as constant as possible in view of the event tree analysis".

A second criteria was: "Define POSs as fine as needed and as coarse as possible"

Factors which have been considered were:

- Remarkable changes in the physical conditions of the primary system,
- Changes in the availability of operational systems important to risk and of safety systems
- Measures which are risk-important.

5. What screening, grouping or other process has been used to select the POSs explicitly addressed in the LPSD PSA?

- The outage schedule has been screened to define the POSs.
- A grouping was performed implicitly. The phases with a transition of the coolant level were conservatively grouped together with the next or previous state.

6. How have you been able to determine that this set of POSs is adequate? [How have you been able to determine that no further subdivisions in the set of POSs addressed in the analysis are required?]

Because no remarkable problems raised during the analysis due to the definition of the POSs

7. Does the set of POSs include unplanned as well as planned modes of operation?

The set of POSs includes only planned modes of operation because of the analysed standard reference refuelling outage.

8. Does the set of POSs modelled in the LPSD PSA include transition modes?

The set of POSs includes also transition modes. Some POSs are completely transition POSs (e.g. parts of hot shut down or start up), in some POSs significant changes of the cooling level are included. (see Table of POSs under Q3)

9. What areas of your analysis are, in your opinion, most in need of improvement?

The identification and definition of POSs for outages other than refuelling outages and of POSs which do not occur at any outage like the POS with a pressure test of the primary system.

10. What are the areas that would most benefit from further research?

A detailed description and analysis of all human activities within the POSs.

Hungary

3. What POSs have been modelled explicitly in the LPSD PSA? [Please provide details of the set of POSs that have been analyzed and how they have been defined in terms of the decay heat level, the activities being carried out during the period of the POS, the condition in the reactor coolant system, the availability of safety systems, plant configurations etc.]

For the low power and shutdown operational mode of the Paks NPP 24 different plant operational states (POSs) have been defined, that can be separated in time. They were selected based on the availability and changing redundancy of systems that can mitigate potential initiating events. The scope of the available mitigating systems has been defined in each POS taking into account the list of all the potential initiating events in the given POS. Possible operational modes of these systems during the shutdown period have been identified on the basis of the Reactor Shutdown Procedure and a number of existing shutdown schedule plans. The operational modes of the systems determined mainly by the change in power and/or change in parameters were then adapted to the pre-defined plant operational states. Plant operational states analyzed were those that occur during a planned shutdown for refuelling. As concerns screening of the POSs identified it can be stated that no screening techniques have been generally used. All the plant operational states identified were evaluated. A short description of their characteristics is given in **Table 1.** The changes of the main primary and secondary parameters during cooldown and the heatup phases are illustrated on **Figure 1** and **Figure 2** respectively.

4. How has this set of POSs been identified? [Please provide details of how the factors which affect the risk have been identified and taken into account in identifying the POSs.]

See Q3.

5. What screening, grouping or other process has been used to select the POSs explicitly addressed in the LPSD PSA?

Neither screening techniques nor groupings have been generally used. All the plant operational states identified were evaluated.

6. How have you been able to determine that this set of POSs is adequate? [How have you been able to determine that no further subdivisions in the set of POSs addressed in the analysis are required?]

POSs have been defined for each different configuration occurring during a refuelling outage, and no grouping of those POSs has been performed.

7. Does the set of POSs include unplanned as well as planned modes of operation?

POSs have been defined for a planned refuelling outage.

8. Does the set of POSs modelled in the LPSD PSA include transition modes?

Yes, POSs include transition modes during both shutdown and startup phases.

9. What areas of your analysis are, in your opinion, most in need of improvement?

None.

10. What are the areas that would most benefit from further research?

None.

Table 1 - Plant Operational States during the Shutdown Period

Plant Operational State					
No.	**Description**	**Start**	**End**	**Estimated duration**	**Characteristics**
1	Low power operation with one turbine	First turbine out of service	Second turbine out of service	1 h	Only one turbine in operation
2	Boron addition to primary coolant to reach reactor subcriticality	Second turbine out of service	Start of cooldown	5 h	Both turbines stopped, emergency protection signal AZ1-9 disabled, feedwater provided by the EFW pumps, boron addition
3	Steam-water cooldown to 240 °C	Start of cooldown	Isolation of hydroaccumulators	6 h	Cooling down in progress at 30 °C/h, hydroaccumulators available
4	Steam-water cooldown to 150 °C	Isolation of hydroaccumulators	Primary temperature reaches 150 °C	8 h	Cooling down in progress at 30 °C/h, hydroaccumulators not available, “LSL-1” protection available
5	Water-water cooldown to 60 °C	Primary temperature reaches 150 °C	RCPs stopped, start of natural circulation	16 h	Cooling down in progress at 30 °C/h, “LSL-2” protection available, one safety system train not available
6	Natural circulation	RCPs stopped, start of natural circulation	Draining of 4 SGs and one half of main steam collector	3 h	RCPs, their primary support systems and make-up water pump out of service, natural circulation
7	Natural circulation in 2 loops	Draining of 4 SGs and one half of main steam collector	Primary circuit depressurization	6 - 8 h	Natural circulation in 2 loops, primary circuit not yet depressurized
8	Opening of the reactor	Control rod drive upper main flange disjoined	Removal of reactor upper head	2 days	Natural circulation in 2 loops, primary circuit depressurized
9	Reactor open for unloading/refuelling, reactor water level RVHF - 300 mm	Removal of upper head	Start of unloading	2-3 days	Natural circulation in 2 loops, reactor water level RVHF - 300 mm
10	Unloading/refuelling, natural circulation in 1 loop	Start of unloading	End of unloading	3.5/4.5 days	Natural circulation in 1 loop, 1 loop in reserve, spent fuel pond and refuelling pond level 21.27 m

Table 1 - Plant Operational States during the Shutdown Period (continued)

Plant Operational State					
No.	**Description**	**Start**	**End**	**Estimated duration**	**Characteristics**
11	Fuel in the spent fuel pond	End of unloading	Start of reloading	25 days	Core unloaded
12	Reloading, natural circulation in 1 loop	Start of reloading	End of reloading	3.5 days	Natural circulation in 1 loop, 1 loop in reserve, refuelling pond level 21.27 m
13	Reactor open after unloading/refuelling, reactor water level RVHF - 300 mm	End of reloading	Upper head reinstalled	3 days	Natural circulation in 1 loop, 1 loop in reserve, reactor water level RVHF -300 mm
14	Closing of the reactor	Upper head reinstalled	Control rod drive upper main flange reinstalled	4 days	Natural circulation in 1 loop, 1 loop in reserve, level CRDUMF -1 m
15	Pressurization of primary circuit	Control rod drive upper main flange reinstalled	Primary pressure reaches 25 bar	4 - 5 h	Natural circulation in 1 loop, 1 loop in reserve
16	Primary pressure 25 bar, natural circulation	Primary pressure reaches 25 bar	Start of containment leaktight test	24 h	Test of protection and interlock operations, systems fill-up
17	Leaktight test of containment	Start of containment leaktight test	End of containment leaktight test	12 h	Leaktight test
18	Primary circuit pressure 25 bar after leaktight test of containment	End of containment leaktight test	Start-up of RCPs	12 h	Plant conditions in accordance to end of POS 16.
19	Heatup to 120 °C, 5 RCPs in operation	Start of RCPs	Start of primary circuit leaktight test	11 h	5 RCPs in operation, LSL-2 tests, 56 bar leaktight test of SGs
20	137/164 bar leaktight test of primary circuit	Start of primary circuit leaktight test	End of primary circuit leaktight, pressure test	4/12 h	Leaktight/pressure test
21	Heatup to 150 °C	End of primary circuit leaktight, pressure test	Primary temperature reaches 150 °C	6 h	5 RCPs in operation, SGs level adjustments, closing of containment

Table 1 - Plant Operational States during the Shutdown Period (continued)

Plant Operational State					
No.	**Description**	**Start**	**End**	**Estimated duration**	**Characteristics**
22	Heatup till reconnection of hydroaccumulators	Primary temperature reaches 150 °C	Hydroaccumulators connected to primary circuit	4 h	LSL-1 protection available, steam blanket in pressuriser, primary pressure 123 bar, 5 RCPs in operation
23	Reaching reactor criticality	Hydroaccumulators connected to primary circuit	Reactor operates at MCPL	16 h	Control rods withdrawn, 6 RCPs in operation, hydroaccumulators connected to primary circuit, boron dilution
24	Reactor power increase	Reactor operates at MCPL	Second turbine in operation	1 day	Criticality tests, turbine start-up, SG safety valve tests, reactor power increased

Figure 1 Cooldown curve

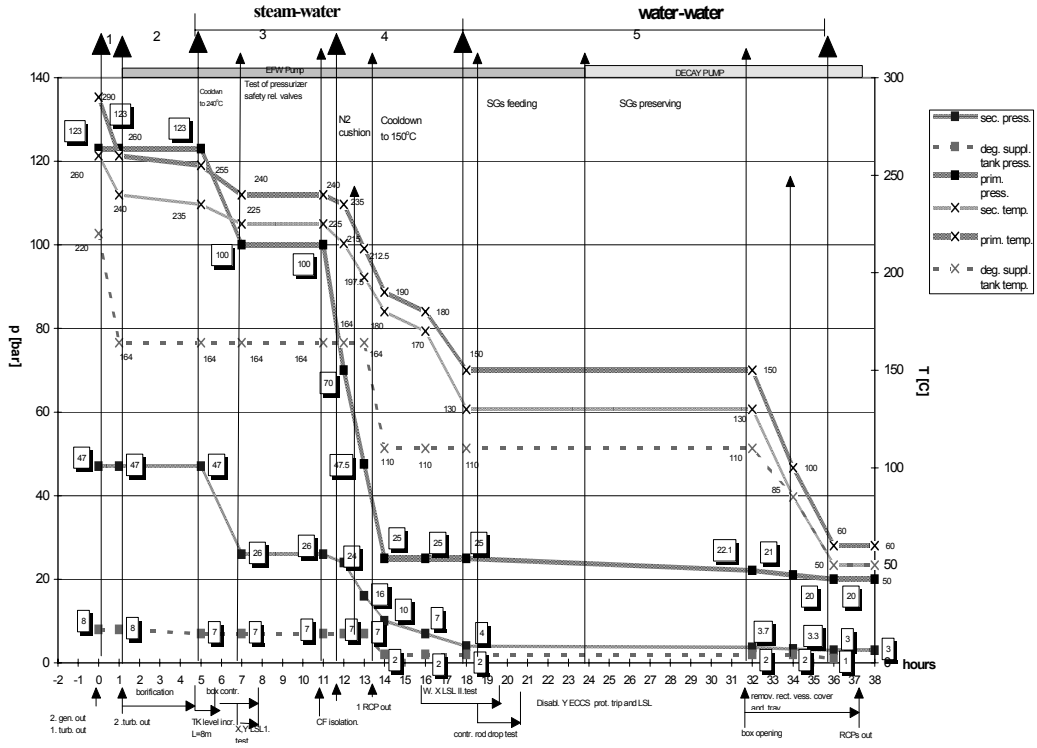
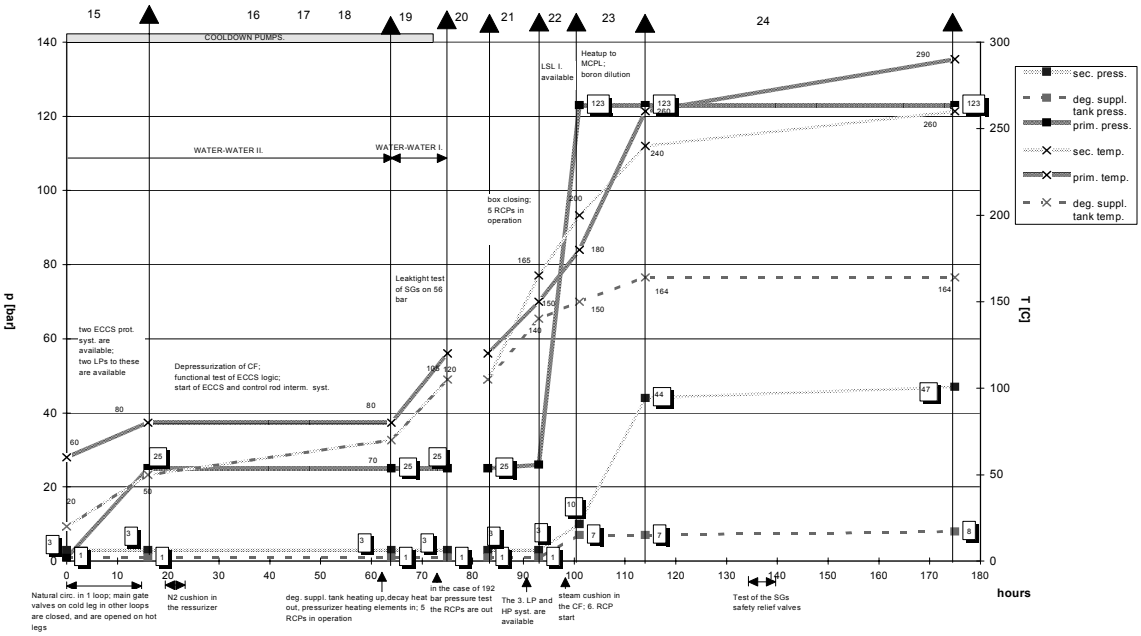


Figure 2 - Heatup curve



JAPAN

3. What POSs have been modelled explicitly in the LPSD PSA? [Please provide details of the set of POSs that have been analyzed and how they have been defined in terms of the decay heat level, the activities being carried out during the period of the POS, the condition in the reactor coolant system, the availability of safety systems, plant configurations etc.]

The period for evaluation

LPSD PSA in NUPEC has been carried out according to "A Procedures Guide for Probabilistic Safety Assessments of Nuclear Power Plants during Shutdown Conditions" (AESJ-SC-P001:2002), which is written by the Atomic Energy Society of Japan. The procedure guide suggests that the period for evaluation in Shutdown PSA be set up to cover the period which is not evaluated in the Full Power PSA.

BWR: The period from the time of the break of condenser's vacuum in plant shutdown process to the time of the control rod drawing-out in plant starting process corresponds to the period for evaluation.

PWR: The period from the time of the automatic starting signal block of ECCS in plant shutdown process to the time of the automatic starting signal release of ECCS in plant starting process corresponds to the period for evaluation.

POS classification: POS classification is carried out in consideration of plant parameters, such as mitigation's alignment, mitigation's ability, reactor coolant inventory, maintenance schedule, success criteria, time margin, and so on.

4. How has this set of POSs been identified? [Please provide details of how the factors which affect the risk have been identified and taken into account in identifying the POSs.]

(1) The items taken into consideration in the classification of POS are the following:

(2) plant operation state

(3) the state of primary system

(4) nuclear reactor coolant inventory (water level)

(5) decay heat

(6) the alignment of mitigation equipment

(7) the alignment of support system

(8) temperature and pressure of reactor coolant.

5. What screening, grouping or other process has been used to select the POSs explicitly addressed in the LPSD PSA?

Attention is paid to decay heat and the capability of decay heat removal.

It is necessary to secure a decay heat removal function and a core injection function in order to avoid reactor core damage. For decay heat removal, success criteria may be different during shutdown operation due to the differences of the relative ratio of the capability of heat removal of the equipment concerned to the decay heat. Moreover, if one POS becomes long, it should be subdivided further since the time margin to core damage within the state may change a lot.

In evaluating the initiating events frequency of the human error under maintenance work, the timing of maintenance work is taken into account in the classification of the POS.

Although the mitigation system that can be used is different according to the state of primary system, the state of core inventory, and maintenance work, it is acceptable to identify the state as the same POS when these differences seldom affect the result of the evaluation.

6. How have you been able to determine that this set of POSs is adequate? [How have you been able to determine that no further subdivisions in the set of POSs addressed in the analysis are required?]

NUPEC mainly groups POSs in light of system/equipment availability and water level and groups POSs as close to the real plant condition as possible. The plant response to the initiating event depends on several parameters, such as the decay heat level, the water level above the TAF (i.e., the top of active fuel), the configuration of the mitigation system, and so on. We confirmed that these parameters are practically constant during each POS interval, in viewpoint of the plant response to the initiating event. We think that the set of POSs is neither excessively coarse nor excessively fine in the analyses.

7. Does the set of POSs include unplanned as well as planned modes of operation?

The view of the above-mentioned POS classification is usually applicable not only to the planned outage but also unplanned outage. However, in addition to the planned shutdown, the following points should be considered in unplanned shutdown.

- The failure leading to the unplanned outage may have affected the mitigation system of Shutdown PSA, and the maintenance can be done about that system.
- When in the unplanned outage, the periodic inspection in normal refuelling outage is not done. The other mitigation systems except for the degraded system can be on stand-by.
- It is not necessary to assume the possibility of the loss of coolant accidents cause by the inspection and maintenance at the plant outage.

8. Does the set of POSs modelled in the LPSD PSA include transition modes?

The above-mentioned POS classification must be set up after also taking “transition modes” into consideration.

9. What areas of your analysis are, in your opinion, most in need of improvement?

It is unclear what evaluation about the POS from the low power at plant shutdown process to the brake of condenser’s vacuum or the S signal block, from the control rod drawing-out start at the reactor starting process or S signal block release to the low power at the reactor starting process should be carried out.

It is necessary to consider the view of the POS classification in LPSD PSA at the unplanned shutdown.

10. What are the areas that would most benefit from further research?

It is necessary to clarify the POS classification in the following cases:

- In the case of using LPSD PSA as a risk monitor during Plant Shutdown
- In the case where outage period is much longer than typical refuelling outage.
- E.g. the replacement of reactor internals
- In the case where the time-window analysis is systematically introduced.

KOREA

3. What POSs have been modelled explicitly in the LPSD PSA? [Please provide details of the set of POSs that have been analyzed and how they have been defined in terms of the decay heat level, the activities being carried out during the period of the POS, the condition in the reactor coolant system, the availability of safety systems, plant configurations etc.]

Due to the continuously changing plant configuration in any outage, the POS is defined and characterized by outage type. Each POS represents a unique set of operating conditions (e.g., temperature, pressure, and configuration). See Table 1.

4. How has this set of POSs been identified? [Please provide details of how the factors which affect the risk have been identified and taken into account in identifying the POSs.]

For the APR-1400, the major 15 POSs (totally 23) were defined using Overhaul Experience Data Book of Yonggwang Units 3&4. Table 1 represents the POSs for APR-1400. The first 11 POSs show the progression of shutdown, and the remaining last 11 POSs shows the startup operation modes.

5. What screening, grouping or other process has been used to select the POSs explicitly addressed in the LPSD PSA?

After identifying the POSs, grouping analysis was not considered.

6. How have you been able to determine that this set of POSs is adequate? [How have you been able to determine that no further subdivisions in the set of POSs addressed in the analysis are required?]

The POSs are classified in detail in order to identify the characteristics due to the configuration changes. There is no absolute criterion to determine the adequacy of POS subdivision.

7. Does the set of POSs include unplanned as well as planned modes of operation?

No, unplanned modes of operation are not included in the preparation of the set of POSs..

8. Does the set of POSs modelled in the LPSD PSA include transition modes?

No, transition modes are not considered.

9. What areas of your analysis are, in your opinion, most in need of improvement?

Estimation of thermo-hydraulic parameter and initiating event frequencies for each POS is very difficult.

Table 1. POS Characteristics (APR-1400)

POS No.	T/S Operation Mode	Plant Operation Status
1	Mode 1, 2	Power reduction to 1%, Reactor Trip
2	Mode 3	Depressurize and Cooldown the RCS using Steam Generator (to 300F)
3	Mode 4	Depressurize and Cooldown the RCS using SCS (to 210F)
4	Mode 5	Depressurize and Cooldown the RCS using SCS (to 140F)
5A	Mode 5	Drain RCS (before opening of Pressuriser M/W)
5B	Mode 5	Drain RCS (after opening of Pressuriser M/W)
6	Mode 5R	Mid-Loop Operation to Install Nozzle Dam, Refill RCS
7A	Mode 5	Maintain Refill RCS
7B	Mode 6F	Rx. Head Detention, Stud Bolt Turn-out
7C	Mode 6I	Rx. Head & UGS/CEA Pulling-up, Refill operation
8A	Mode 6E	Remove Fuel
8B	Mode 6E	Inspect Fuel
8C	Mode 6E	Install Fuel
9A	Mode 6I	Place Reactor Head to Startup
9B	Mode 6F	Finish Refuelling
9C	Mode 5	Drain RCS to Mid-loop
10	Mode 5R	Mid-Loop Operation to Remove Nozzle Dam
11A	Mode 5	Refill RCS (before closing of Pressuriser M/W)
11B	Mode 5	Refill RCS(after closing of Pressuriser M/W)
12	Mode 5	Start RCS Heatup (to 200F)
13	Mode 4	Maintain RCS Heating (to 350F), Remove SCS
14	Mode 3	Pressurize and Heatup the RCS using S/G
15	Mode 2, 1	Prepare Operation and Synchronize Plant

(Note - R:Reduced inventory, F:IRWST full, E:IRWST empty, I:Structure in-position)

10. What are the areas that would most benefit from further research?

Dynamic modelling for LPSD operational mode changes and/or realistic operator actions is needed in further research.

MEXICO

3. What POSs have been modelled explicitly in the LPSD PSA? [Please provide details of the set of POSs that have been analyzed and how they have been defined in terms of the decay heat level, the activities being carried out during the period of the POS, the condition in the reactor coolant system, the availability of safety systems, plant configurations etc.]

POS 1: Operating Condition (OC)1 and 2 as defined in Technical Specification for BWR, with reactor power not grater than 15%, reactor pressure from rated condition to 500 psig.

POS 2: OC 3, Hot Shutdown, reactor pressure 500 psig.

POS 3: OC 3, Hot Shutdown before the operation of the RHR in shutdown mode of operation (RHR/SDCM), reactor pressure 100 psig.

POS 4: OC 3, Hot Shutdown with the RHR/SDCM operating.

POS 5: OC 4 and 5, Cold Shutdown, reactor water temperature less than 200°F and level until main steam lines.

POS 6: OC 5, Refuelling, vessel head off, reactor water level until main steam lines.

POS 7: OC 5, Refuelling, reactor water level until spent fuel pool and refuelling transfer tube open.

4. How has this set of POSs been identified? [Please provide details of how the factors which affect the risk have been identified and taken into account in identifying the POSs.]

An operation condition table was developed to assess plant configuration changes as the power (electrical and thermal) decrease or increase. The main variables considered to define POSs were: reactor pressure, water level control, emergency core coolant system and residual heat removal systems availability, forced and natural recirculation.

POS 1,2 and 3, were intended to reflect the balance of plant systems availability changes (for instance, turbine bypass open at 20%, reactor recirculation pump trip, control rod insertion, etc).

POS 4 and 5, were selected to reflect the way in which the residual heat removal is accomplished.

POS 6 and 7, were intended to include refuelling challenges.

5. What screening, grouping or other process has been used to select the POSs explicitly addressed in the LPSD PSA?

Nine POSs were selected in the first assessment of plant operating conditions. A grouping process based on the system diversity, plant response and time available to mitigate the potential accidents reduced the number to seven POS.

6. How have you been able to determine that this set of POSs is adequate? [How have you been able to determine that no further subdivisions in the set of POSs addressed in the analysis are required?]

Determination of further subdivisions, if necessary, will be based on the detailed PSA developed for the dominant POS. For the time being, the set of POSs selected is adequate for the principal objective due to the coverage as the following explains:

In POS 1, 2 and 3, the state of the plant is similar to full power except that there is low power and pressure/temperature is decreasing. Therefore, initiating events and system availability are quite similar and their quantification was accomplished by looking for the applicability of the full power initiating events and models.

In POS 4 and 5, the residual heat removal is accomplished by the RHR/SDCM. At LVNNP there is no ADHR system, and the maximum expected decay heat is about 1% of full power.

In POS 6 and 7, the vessel head is removed, and one of their main differences is the amount of time needed to reach the core in case of an accident.

7. Does the set of POSs include unplanned as well as planned modes of operation?

Planned modes of operation only were considered.

8. Does the set of POSs modelled in the LPSD PSA include transition modes?

No, it does not.

9. What areas of your analysis are, in your opinion, most in need of improvement?

Exhaustive search of initiating events (internal and external operational experience, as well as conduct FMEA), plant specific data collection.

10. What are the areas that would most benefit from further research?

Transition risk assessment would benefit from further research.

THE NETHERLANDS

3. What POSs have been modelled explicitly in the LPSD PSA? [Please provide details of the set of POSs that have been analyzed and how they have been defined in terms of the decay heat level, the activities being carried out during the period of the POS, the condition in the reactor coolant system, the availability of safety systems, plant configurations, etc.]

4. How has this set of POSs been identified? [Please provide details of how the factors which affect the risk have been identified and taken into account in identifying the POSs.]

5. What screening, grouping or other process has been used to select the POSs explicitly addressed in the LPSD PSA?

In the Borssele PSA use has been made of so-called Operation Types (OTs), Plant Cycle Types (PCTs) and Plant Operating States (POSs). Operation types are governed based on the intentions and objectives of the plant management. If the objective is to perform a refuelling outage than the operation type is

refuelling. Each operation type consists of one or more Plant Operational States (POSs). The following OT definitions were used for the Borssele PSA:

OT1 Power

OT2 Non-drained maintenance outage without RHR. This OT is characterised as hot standby. (reactor tripped, coolant temp about 290 ^{o}C, coolant pressure 154 bar). This OT is used if only short outage times are expected, or to maintain/repair components which do not require cooldown. The end state of OT2 is an intermediate state when transitioning to OTs 3, 4, 5, or 6.

OT3 Drained maintenance outage with RHR. This OT is characterised as cold shutdown, cooled down below 50 ^{o}C, RPV head closed, heat transfer via RHR. The plant state will be reached if long outage times are to be expected (> 1 week), or if there is a loss of main heat sink, or for maintenance/repair, which requires cooldown. The end state of OT3 is an intermediate state when transitioning to OTs 4, 5 or 6.

OT4 Drained maintenance outage. This OT is midloop operation, which is used if maintenance requires low level of the coolant system without unloading of the core. Starting from OT3, the reactor coolant system is drained to midloop with RPV head closed. Heat transfer via RHR.

OT5 Short refuelling outage.

OT6 Long refuelling outage

OT 5 and 6 are the refuelling outage types. Starting with OT4 conditions, filling the reactor basin is required before the fuel assemblies can be handled. During the steady state portion of the OT, all fuel assemblies are transferred to the fuel pool. While the fuel is in the pool many maintenance activities and surveillance tests are performed. Decay heat removal via RHR and/or the fuel pool cooling system, depending on the location of the spent fuel assemblies. The length of time that the fuel is in the pool determines whether the operation is type 5 or 6. Taking into account these outage types in the Borssele PSA covers nearly all possible outages. Outages due to repair or replacement of major components such as Steam Generators are not yet planned. If necessary, than a separate risk evaluation of these events.

During a typical year there are no planned outages for operation types 2, 3 or 4. These are only used as needed in response to emergent plant conditions or problems. The Borssele PSA evaluates the safety over the course of a typical year for two cases.

The POSs represent different possible plant configurations within the different operation types. POSs represent the set of plant response boundary conditions, defining a unique set of physical plant parameters such as temperature and pressure, and safety systems. The Borssele POSs are mainly differentiated by their impact on the PSA success criteria, which is affected by the following terms: reactor states, states of safety systems and containment states.

There are five general steps in the definition of POS:

1. Definition of the critical safety functions necessary to control or mitigate possible accidents during power operation or reactor shutdown. The critical safety functions as defined for the power PSA were adapted for the shut-down and low power states with additional functions

identified as needed for refuelling, such as: Subcriticality, RCS integrity, RCS/fuel pool inventory make-up, core and containment heat removal and containment integrity

2. Fixing the initial conditions of the POSs. The initial conditions are defined as significant changes in plant characteristics that would alter post-accident plant response. They are summarized in next table (Table 1):

3. From the initial conditions, the NPP Borssele shutdown and start-up procedures were examined step-by-step to develop draft POSs, called pre-POSs. Also the test and maintenance procedures for planned activities during shutdown and startup have been reviewed. The Borssele operating manual has only one procedure for reactor shutdown and start-up and does not discriminate between different types of outages. The actions in each step of the procedure have been examined to determine if any safety related changes in boundary conditions result from changes in the following:

- Reactor power
- Reactivity
- Physical conditions of the RCS affecting the thermal-hydraulics: temperature, pressure, levels
- Availability/ redundancy of safety systems
- Available systems for decay heat removal
- Energy supply to safety systems (motive and control power)
- Actuation of safety systems (automatic/manual)
- State of reactor protection system (e.g., jumpering of signals)
- Location of fuel assemblies

Table 1

Physical Conditions	Decay Heat Removal Mechanism	Comments on Critical Safety Functions/Systems and Maintenance
Power operations, hot and pressurized RCS	Secondary heat removal (preferred) or primary feed and bleed (alternate)	Rods, boron, and temperature affect subcriticality. Full set of safety system (ECCS and AFW) available
Non-power operations, hot and pressurized RCS	Secondary heat removal (preferred) or primary feed and bleed (altern.)	Boron and temp. affect subcriticality. Full set of safety system (ECCS and AFW) available
Non-power operations, cold and pressurized RCS	Low pressure RHR is the main decay heat removal mechanism, secondary heat removal available as alternate if plant heats up	Boron and temp. affect subcriticality. Maintenance allowed on some systems, mostly secondary heat removal
Non-power operations, RCS cold, depressurised, and drained to midloop	Low pressure RHR provides decay heat removal, feed and bleed using bunkered injection system is alternate.	Boron and temp. affect subcriticality. Minimum initial primary inventory Maintenance allowed
Non-power operations, fuel in spent fuel pool	Spent fuel pool cooling system provides decay heat removal, with various pool fill sources available as alternate	Boron and temp. affect subcriticality.

- State of containment penetrations (e.g., air locks, ventilation ducts, pipes)
- Monitoring systems for safety parameters

General guidelines used for pre-POS definition are the following:

- A change in boundary conditions for a critical safety function results in definition of a new pre-POS
- Specific pre-POSs are defined for states with the potential for specific accidents, e.g., fuel handling accidents during core unloading and reloading
- Transitions (from one steady-state POS to another) which are initiated or completed by manual operation, and which have the potential for a severe accident are also defined as a specific pre-POS.

Twenty eight (28) pre-POS were identified. See next table (Table 2).

Table 2

POS Name	POS DES.	Pre-POS	POS Sub-group	POS Characteristics	RCS Pres.	Trans. or steady state
Power	P	0	High power	Power > 21%, in–house 6 kV ac from turbine, which also feeds the offsite power grid	Y	S
		1	Low power	21 % power, 6 kV supplied by turbine, which is removed from offsite grid. Response bounded by power operations	Y	T
Hot Early	HE	2	Low power	Electrical power shifted to offsite, response similar to power operations for systems other than 6 kV ac	Y	T
	HE	3	Low power	Turbine is tripped, turbine bypass used for decay heat removal	Y	T
	HE	4	Hot standby	Reactor shutdown, T=290 C, P=154 bar, Boron =CR	Y	S
	HE	5	SG cooldown	Secondary systems preferred decay heat removal, T= 290-100 C, P= 154-29.4 bar, Boron =CK	Y	T
Cold Shutdown Early	RE	6	RHR cooldown	RHR cools, T<100C, p=29,4 bar, Boron= CK, first RCP tripped, high head ECCS disabled (discharge valves chained closed)	Y	T
	RE	7	RHR cooldown	RHR cools, T=100-50 C, P=29.4 bar, boron = CK	Y	T
	RE	8	RHR cooldown	RCS depressurized, t<50C, P=29.4 – 4.9 bar, Boron =CK, second RCP tripped	Y	T
	RE	9	RHR cooldown	RCS depressurized, t<50C, P=4.9 - 0 bar, boron = CK, ECCS sump valves disabled	Y	T
	RE	10	Cold shutdown	T<50C, P=0 bar, boron = 2200 ppm, Prz-level = 3m	N	S
Midloop early	ME	11	RPV closed	Drain to midloop, bunkered injection system shifted to 80 C actuation, RCS vents open	N	T
	ME	12	RPV closed	Steady state	N	S
	ME	13	RPV open	RPV open start when first bolt is loosened	N	T
	ME	14	RPV open	RPV and reactor cavity above RPV flange and skirt (REBA) filled by ECCS	N	T
Core unload	CU	15	RPV open	REBA full, ECCS tanks empty, reactor internals removed	N	T
	CU	16	RPV open	Fuel transfer, REBA full, ECCS tanks empty, some fuel in fuel pool and some in RPV	N	T

POS Name	POS DES.	Pre-POS	POS Sub-group	POS Characteristics	RCS Pres.	Trans. or steady state
	N/a	17	Core empty	Core empty, all fuel in fuel pool. This is a POS only for fuel pool operations	N	S
Core load	CL	18	RPV open	Fuel transfer, REBA full, ECCS tanks empty, some fuel in pool and some in RPV	N	T
Midloop late	ML	19	RPV open	RCS drained with ECCS pumps	N	T
	ML	20	RPV closed	RPV head bolted	N	T
	ML	21	RPV closed	RCS vented, other systems start-up (SG secondary side filled)	N	T
	ML	22	RPV closed	Refill RCS with make-up system, bunkered injection system shifts to L/P	N	S
Cold shutdown Late	RL	23	Cold shutdown	RCS pressurized, P=0-28.4 bar, T<50C, vented, ECCS and PORVs enabled	Y	S
	RL	24	RCS heatup by RCPs	RCP heatup, T=50-100C, P=28.4 bar	Y	T
Hot late	HL	25	RCS heats up by RCPs	RCP heatup, T=100-270C, P=28.4 – 154 bar, boron 2200 ppm, turbine bypass operable	Y	T
	HL	26	Hot standby	Reactor startup and heatup, T = 270-303C, P=154 bar	Y	S
	HL	27	Low power	Turbine running, turbine bypass used for decay heat removal, 6kV ac shift from offsite to in-house	Y	T
Power	P	28	Low power	Power at or below 21%, 6 kV ac from in-house	Y	T
	P	0	High power	Power > 21%, 6 kV from in-house, turbine feeding the grid	Y	S
Fuel pool early	FE	16	Core unload	Fuel transfer, REBA full, ECCS tanks empty, some fuel in fuel pool and some in RPV	N	T
	FE	17	Core empty	Core empty, all fuel in pool. This is a POS only for fuel pool operations and not for the RCS	N	S
	FE	18	Core load	Fuel transfer, REBA full, ECCS tanks empty, some fuel in fuel pool and some in RPV	N	T
Fuel pool late	FL	0-15; 19-28	Fuel pool during RCS operation	Fuel pool operations at power, hot standby, cold shutdown and midloop. Fuel pool temp. <50C. Decay heat level FL < decay heat level FE	N	T

4. Next, the impacts of timing and allowed maintenance on safety system components on the condition of the POSs are indicated. The parameters used to define the POSs allow the same definitions for POSs in different operation types. However, these parameters are not sufficient to uniquely define the plant conditions without additional determinations of timing (such as POS duration) and changes in the allowed maintenance. As the time durations of different operation types can vary, the same POS can be reached within different time spans after reactor trip. Therefore, different decay heat levels may cause timing differences in plant response, even if the initiating event occurs in an identical POS. For example, midloop operation with the RPV closed has been reached for non-refuelling (planned) maintenance outages in about 35 hours after reactor trip, while it took twice as much time to reach the same POS in refuelling outages. These variations are encountered in three ways. First, the scope of the base case PSA is only for refuelling outages, so variations due to other operation types are considered separately as a potential sensitivity analysis. Next, when the success criteria thermal-hydraulic calculations have been made, conservative (high decay heat) levels have been analyzed. Finally, a distinction was made between the "early" and "late" POSs due to the change in decay heat levels.

5. Finally, sets of pre-POSs with similar plant physical conditions, decay heat removal mechanisms, and maintenance restrictions have been grouped together to form the POSs used in the PSA. The first column in previous table represents the selected POSs.

6. How have you been able to determine that this set of POSs is adequate? [How have you been able to determine that no further subdivisions in the set of POSs addressed in the analysis are required?]

Prior to the conduct of the current LPSD PSA a LPSD scoping study was made. In this study the 28 pre-POSs, mentioned in the answer to quest. 3 ,4 and 5, were assessed. It was shown that the plant responses were for several sets of these POSs practically the same. It could be concluded that a grouping of these POSs was warranted, and therefore could be considered as pre-POSs.

7. Does the set of POSs include unplanned as well as planned modes of operation?

In principle are only planned modes of operation analyzed as base case. For unplanned outages, adaptations can be made very easily. See point 4 of answer to questions 3,4 and 5.

8. Does the set of POSs modelled in the LPSD PSA include transition modes?

Yes, see table of POSs and pre-POSs. Several pre-POSs are transition states.

SPAIN

3. What POSs have been modelled explicitly in the LPSD PSA? [Please provide details of the set of POSs that have been analyzed and how they have been defined in terms of the decay heat level, the activities being carried out during the period of the POS, the condition in the reactor coolant system, the availability of safety systems, plant configurations etc.]

Given that the **Asco NPP** LP&SD PRA is only intended to refuelling outages, most of the POS definitions refer to processes instead of stationary states.

POS-1 Low Power Operation and Reactor Shutdown

It is defined as the process to lead the plant from a situation of 25 % of nuclear power rate to hot standby. By the end of this POS, feedwater is transferred from Principal to

Auxiliary System, the Steam Generators level is manually controlled and control banks are manually inserted (excepted 6 steps).

The end state is: P_{th}=0% ; K_{eff}< 0.99 ; T≅292 ºC ; P_{RCS}=157 kg/cm^2

POS-2 Cooldown with SGs (up to 175 ºC)

It is defined as the process to lead the plant from a situation of hot standby to hot shutdown. During this POS the RCS is borated, and RHR system is put on service and ready to cool primary.

The end state is: P_{th}=0% ; K_{eff}< 0.99 ; T ≅ 93 ºC ; P_{RCS} ≅ 27 kg/cm^2

POS-3 Cooldown with RHR (up to 93 ºC)

It is defined as the process to lead the plant from a situation of hot shutdown to cold shutdown. During this POS, RHR cools the RCS, AFW is put out of service and MSIVs are closed.

The end state is: P_{th}=0% ; K_{eff}< 0.99 ; T ≅ 93ºC ; P_{RCS} ≅ 27 kg/cm^2

POS-4 Cooldown with RHR (up to 60 ºC)

It is defined as the process to lead the plant from a situation of cold shutdown to an average temperature in the RCS of 60 ºC. During this POS, RCPs are stopped, pressuriser is vented and stop and control banks are fully inserted.

The end state is: P_{th}=0% ; K_{eff}< 0.95 ; T ≅ 60ºC ; P_{RCS} ≅ 2 kg/cm^2

POS-5 Drain RCS to mid-loop

It is defined as the process to lead the plant to the mid-loop conditions. During this POS pressuriser manhole is open. At the end of the POS, SGs manholes are open.

The end state is: P_{th}=0% ; K_{eff}< 0.95 ; T < 60ºC ; P_{RCS} = atm.

POS-6 Mid-loop operation

The plant is at mid-loop operation and the former end state conditions are maintained. During this POS the nozzle dams are installed.

POS-7 Fill for refuelling

The refuelling cavity is filled up with water from the refuelling water storage tank. During this POS the RPV head is disassembled and removed. The filling rate follows the RPV head hoisting speed.

The end state is: P_{th}=0% ; K_{eff}< 0.95 ; T < 60ºC ; P_{RCS} = atm.

POS-8 Refuelling

It is defined as the POS in which with the vessel head already disassembled and removed, the spent fuel assemblies are carried to the pool and replaced by fresh fuel. The cavity and the spent fuel are communicated.

POS-9 Cavity empty out and RCS drain to midloop after refuelling

It is defined as the POS in which, being initially the vessel head removed and the refuelling cavity full, the cavity is drained up to 22 cm above the middle nozzle plane. During this POS the RPV head is installed.

POS-10 Midloop operation (after refuelling)

It is defined as that in which the NPP is in mid-loop operation in order to complete maintenance, test and inspection tasks. During this POS nozzle dams are removed and all of the manholes are closed.

POS-11 Refill RCS completely

It is defined as the process to fill from minimum inventory in the RCS to solid RCS. During this POS the RCS is degasified.

The end state is: P_{th}=0% ; K_{eff}< 0.95 ; T < 60 ºC ; $P_{RCS} \cong$ 2 kg/cm^2

POS-12 RCS heatup (up to 93 ºC)

It is defined as the process to lead the plant from a situation of RCS solid to hot shutdown. During this POS stop banks and 6 steps of control banks are withdrawn, automatic safety injection is unblocked and containment integrity is restored.

The end state is: P_{th}=0% ; K_{eff}< 0.99 ; T ≅ 93 ºC ; $P_{RCS} \cong$ 27 kg/cm^2

POS-13 RCS heatup with RCP's (up to 175 ºC)

It is defined as the process to lead the plant from a situation of hot shutdown to hot standby. During this POS, RCPs are started and RHR is put out of service.

The end state is: P_{th}=0% ; K_{eff}< 0.99 ; T ≅ 175ºC ; $P_{RCS} \cong$ 27 kg/cm^2

POS-14 RCS heatup with SGs

It is defined as the process to lead the plant from a situation of hot standby to reactor start-up. During this POS the SG level is manually controlled and the steam is dumped to the main condenser.

The end state is: P_{th}=0%; $K_{eff} \geq$ 0,99 ; T ≅ 292 ºC ; P_{RCS} = 157 kg/cm^2

POS-15 Reactor start-up and low power operation

It is defined as the process to lead the plant from reactor start-up to minimum load and then to increase load to 25%. During this POS, MSIVs are open, Feedwater system put on service, boron concentration adjusted, turbine and generator put on service and coupled to grid.

The end state is: P_{th}=25% ; $K_{eff} \geq$ 0.99; T ≅ 294.5 ºC ; P_{RCS} = 157 kg/cm^2.

For **Santa Maria de Garonna NPP**, the following table identifies the POS attributes:

POS	Description
0	Reactor power higher than 25%. Situation contemplated in the At power PSA. Out of scope.
1	Reactor power lower that 25%. Decreasing power. Systems availability very similar to At power situation, no differences in the Tech Specs with At Power requirements.
2	Reactor mode switch in "Shutdown" with the feed water system removing the residual heat. No reactivity control needed, given that all the control rods are inserted.
3	Reactor mode switch in "Shutdown". Shutdown heat removal system removing residual heat. Coolant temperature higher than 100ºC. Vessel head on
4	Reactor mode switch in "Shutdown". Shutdown heat removal system removing residual heat. Coolant temperature lower than 100ºC. Vessel head on
5	Reactor mode switch in "Refuelling". Shutdown heat removal system removing residual heat. Reactor cavity not totally filled. Vessel head off
6	Reactor mode switch in "Refuelling". Shutdown heat removal system evacuating residual heat. Reactor cavity totally filled and communicated with the fuel pool.
7	Core offloaded. All the fuel elements in the fuel pool.
8	Reactor mode switch in "Refuelling". Shutdown heat removal system removing residual heat. Reactor cavity totally filled and communicated with the fuel pool. Reloading the core. Similar situation than POS # 6 but with less residual heat.
9	Reactor mode switch in "Refuelling". Shutdown heat removal system evacuating residual heat. Reactor cavity level decreasing. Similar to POS # 5 but with less residual heat.
10	Reactor mode switch in "Shutdown". Increasing coolant temperature. Residual heat removed with residual heat removal system or with feedwater system. Equivalent to POS 3 and 4, but with less residual heat.
11	Reactor mode switch in "Start up" or "Run". Increasing Power.

4. How has this set of POSs been identified? [Please provide details of how the factors which affect the risk have been identified and taken into account in identifying the POSs.]

POS have been defined in a constructive way that means, by starting from the Tech Specs modes of operations and main situations or manoeuvres within each of them. By this way, 15 POSs have been defined in **Asco NPP**.

The parameters considered to define POS at **Santa Maria de Garonna** plant are the following:

- Reactor Power
- Reactor mode
- Shutdown Heat removing system used
- Coolant temperature
- Reactor cavity level/ Gate separating upper pool and fuel pool installed or removed
- Fuel location

5. What screening, grouping or other process has been used to select the POSs explicitly addressed in the LPSD PSA?

Asco NPP: All the defined POS were evaluated in the LP&SD PRA. The screening took place after the definition of scenarios. No grouping of POS was a priori done but, for a given IE, several situations that only depend on the POS were grouped in a single scenario, for instance, POS shutting down and starting up systematically fell in the same scenario.

Santa Maria de Garonna NPP: The following POS were eliminated of the analysis:

POS 0: In fact, this POS corresponds to the At Power PSA.

POS 1: Power lower than 25%, but same requirements about systems unavailability. At Power PSA is still valid.

POS 2: All the control rods are inserted, but the residual heat is still removed by the Feed Water System. All the considerations of the At Power PSA, except those related with the ATWS accident, are still valid.

POS 7: When all the fuel elements are out of the vessel, it is considered out of the scope because the end state analyzed is "Core Damage" or "Core uncovered".

POS 11: Starting up. Equivalent considerations to POS numbers 1 and 2.

Besides this, some other Scenarios (not POSs) were screened out of the analysis:

1) Large recirculation line breaks LOCA in POS 5, 6, 8 and 9 are driven directly to core damage.

2) Steam line break LOCAs are screened in POS 3, 4, 5, 9 and 10 because the steam generation is considered negligible.

3) Other LOCAs in POS 3 are considered equivalent to At Power situations.

4) Out of the Drywell LOCAs (Interface system LOCAs) are screened in POS 3-10 because there is not enough pressure in the lines.

5) Vessel drains: All the possible special maintenance operations on standby systems that may drain the vessel are screened out because of their low frequency and short duration, and because all maintenance activities require a post-maintenance test. Neither are considered those that require a pipe break and a failure to remain closed of a valve, or the failure of more than one valves to remain closed. Vessel drains considered are just related with systems used to normally fill or drain the reactor cavity, or to extract the residual heat, or with the control rod drive system, or the safety and relief valves.

6) In POS 6 and 8, with the reactor cavity filled, transients are not considered that would cause the loss of the running residual heat remove system (Loss of Off Site Power, loss of Service Water system, loss of the running SHC system, etc.) because it is considered there would be enough time available to adopt compensating measures.

6. How have you been able to determine that this set of POSs is adequate? [How have you been able to determine that no further subdivisions in the set of POSs addressed in the analysis are required?]

The CSN contractors for reviewing both LP&SD PSAs and the CSN, themselves, have felt that the final set of POS assessed is adequate in both cases. Even though the POS is based in engineering judgment, the Spanish PSAs in general follow a strict process to guarantee their technical adequacy. An independent engineer reviews in detail all the work performed and states his comments by filling out a form, a written response is given by the author. The whole process is under QA.

7. Does the set of POSs include unplanned as well as planned modes of operation?

Initially, the submitted reports analyze only planned outages. In principle, most of the unplanned outages can be covered by the existing analyses by only adjusting the duration time of the POS.

8. Does the set of POSs modelled in the LPSD PSA include transition modes?

See answer to question Q3.

9. What areas of your analysis are, in your opinion, most in need of improvement?

See answer to question Q68.

10. What are the areas that would most benefit from further research?

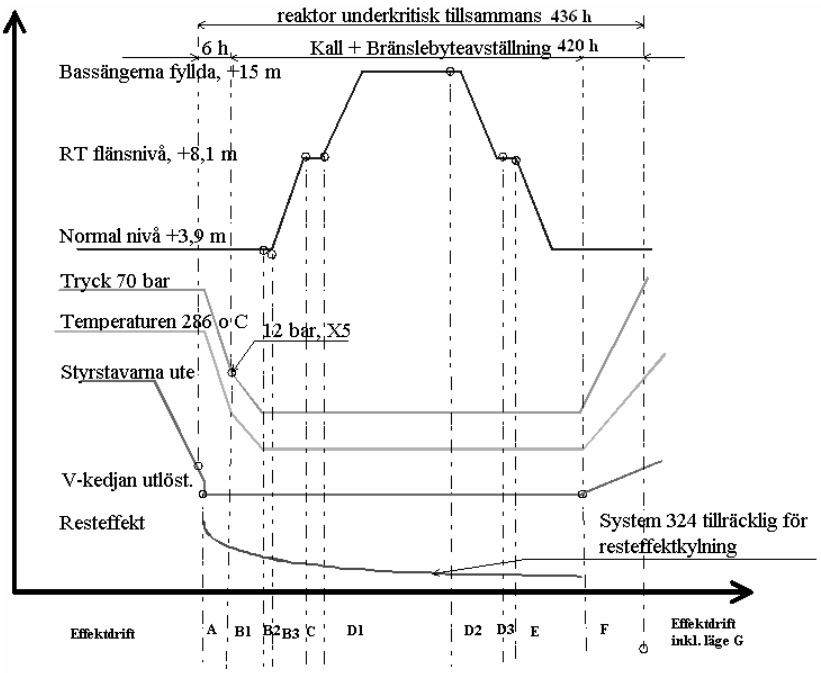
See answer to question Q69.

SWEDEN

3. What POSs have been modelled explicitly in the LPSD PSA? [Please provide details of the set of POSs that have been analyzed and how they have been defined in terms of the decay heat level, the activities being carried out during the period of the POS, the condition in the reactor coolant system, the availability of safety systems, plant configurations, etc.]

CSNI Question number	Answer from Swedish plant	Answer
Q3	F1/2	Outage phases. The graph below shows grouping in different phases of the LPSD analysis.

Graph # 1 - Forsmark NPP



*A - Hot shutdown. Via motor driven shutdown system (V-kedja). Residual heat RHR cooled via warm sinks:

- Steam condenser system/414/415
- Condensations pool system/314/327
- 321/331/713 (classified as shutdown phase)

B - Hot shutdown. Reactor pressure lower than 12 bar down too cool shutdown with pressureless reactor. The vessel head removed. RHR with system 321/324.

B1 - Hot shut down. Reactor pressure < 12 bar. RHR with system 321/711/715. (classified as shut down phase.)

B2 -Cold shutdown. Pressure 1 bar. The vessel head removed.

B3 - Cold shutdown. Top filling. The vessel head mounted.

C - Cold shutdown. RCPB open. Pressureless reactor. Vessel head dismounted.

D -Refuelling. Condensation pools filled up. Cooling of pools in the initiating phase with system 321 afterwards cooling via system 324.

D1 -Refuelling. Pools filled. Cooling via system 321/324.

D2 -Like D1 but with cooling via system 324.

D3 -Cold shutdown. C-pool drained. The vessel head is dismounted.

E - Cold shutdown. Pressureless reactor. The vessel head is mounted.

F - Nuclear heating. Reactor pressure < 70 bar. Control rods driven out. (Belong to start up phase.)

G – Hot stand by phase. Reactor pressure at 70 bar. One turbine operable. (Belong to start up phase.)

Analyzed in shutdown- and start up phase.

The limit for LPSD is 8% reactor power.

CSNI Question number	Answer from Swedish plant	Answer
Q3	F1/2	Outage phases. The graph below shows grouping in different phases of the LPSD analysis.
Q3	R1	For LP: Five POS in the power decrease phase and five POS in the power increase phase. Difference in the time scale of the power decrease phase, 6 hours each POS and 20 hours each POS power increase. For SD: Vessel head on and vessel head off and refuelling. Different cooling phases.
	R2-4	R2 2001: Hot shutdown state, cold shutdown state and static shutdown state (subcritical reactor, pressure below 29 bar, residual heat water system connected, going to temperature <60 C and open reactor coolant system). R2 1996: In addition to the states mentioned above, refuelling was also included
	B2	The modelling of POSs is not strict. Shutdown to cold reactor, dismantling of the reactor pressure vessel head, unloading and loading fuel, system tests, mounting of internal parts and RPV head, further system and components tests, inerting the containment, warming, further tests, and power operation.
	O1	The different POSs (phases) that are being considered for NPP OKG1 is given in table below. See OKG1 table 1 Answer only valid for low power PSA, not outage period.
	O2	No answer (See R1, SKI remark)
	O3	Specific POS's during shutdown are: Hot shutdown reactor, turbine accessible. Hot shutdown reactor, cooling with 331, 314-depressurization. Hot shutdown reactor, residual heat removal with 321. Start-up at almost full pressure, reactor coolant make-up with system 328 at less than 10 kg/sec. Hot standby, start-up of turbine system, 312 in automatic control mode. Turbine by-pass operation, reactor pressure at 7Mpa, increase of power until power operation is reached.
SKI REMARK	One of the main tasks in the coming joint Nordic PSA projects is to harmonise LPSD method descriptions, e.g., to define POSs.	

OKG1 Table 1				
Phase	Main activities	Reactor condition		
		Power [%]	Pressure [bar]	Temp. [° C]
Shutdown I	Power set point adjusted Start venting containment Switch over to speed control of MCP - 3000 kg/s Reduce power with control rods	100 – 25	70	286
Shutdown II	One feed water pump is shut down V-chain (control rod screw insertion) TS-chain (turbine shutdown) All control rods inserted Operation mode switch in position 0	25 – 0	70	286
Shutdown III	Engagement of Auxiliary Condenser for pressure control Closure of all man steam line isolation valves (MSIV)	Residual heat only	70 – 65	286 – 280
Hot shutdown condition				
Shutdown IV	All dump regulation valves (DRV) are closed. Start cooling with Auxiliary Condenser Stop remaining feed water pumps	Residual heat only	65 – 15	280 – 200
Shutdown V	Engagement of one heat exchanger in Shutdown Cooling System. Auxiliary Condenser taken out of service	Residual heat only	15 – 1	200 – 100
Shutdown VI	Start of Auxiliary Feed water System for reactor vessel injection Start of remaining pump in Shutdown Cooling System. Engagement of other heat exch. in Shutdown Cooling System Start to rise water level in Reactor Pressure Vessel	Residual heat only	1	100 – 85

Cold shutdown condition				
Start-up I	Start of Auxiliary Feed water System if requested Shutdown Cooling System is taken out of service Operation mode switch to SS Resetting of V-chain (control rod screw insertion)	Residual heat only	1	40 – 100
Start-up II	Start to increase power using control rods Start of one Condensate pumps Reactor vessel injection with Feed water System is possible. Vacuum in condenser Pressurisation of main steam lines Opening of MSIV	Residual heat only	1 – 15	100 – 200
Start-up III	Start of another Condensate Pump Start of first pump in Feed water System Stop of system Auxiliary Feed water System D-chain is reset (possible to use Condenser) Increase reactor pressure to 70 bar Auxiliary Condenser is switched over to stand-by TS-chain (Turbine Shutdown) is reset	Residual heat only	15 – 70	200 – 286
Start-up IV	Dump Regulation Valves are regulating reactor pressure at 70 bar Remaining feed water pumps are started.	0 – 100	70	286

4. How has this set of POSs been identified? [Please provide details of how the factors which affect the risk have been identified and taken into account in identifying the POSs.]

CSNI Question number	Answer from Swedish plant	Answer
Q4	F1/2	This POS's have been identified via FSAR/TS. It is provided that all POS's are considered. Full power operation >8% reactor power Shutdown phase (including warm SD) Cold Shutdown phase/outage (RA) The start up phase
	R1	The factors taken in account are power level, temperature and pressure.
	R2-4	R2, The main factors taken in account are power level, temperature and pressure. Sub states where water levels in RCS and important activities have been taking into account when screening for initiating events, and

		choosing event sequences.
	B2	The modelling follow the main activities during an outage.
	O1	The POSs have been identified by examination of those written instructions that control the shutdown and start-up period. Another important source of information that has been used is interviews with operating personnel. The POSs above (phases) have been divided according to three criteria: *System configuration:* In each phase the system configuration is the same and therefore also the system availability. *Initiating events:* In each phase the same set of initiating events must be valid. For example if the turbine condenser has been taken out of service the initiating event "loss of turbine condenser" is no longer applicable. *Consequences:* In the same phase each initiating event must lead to the same consequence, i.e. either OK or a core damage consequence (CD). Answer only valid for low power PSA, not outage period.
	O2	No answer (See R1, SKI remark)
	O3	The identification of different POS's was based on which systems are in operation at different stages of the LPSD period.
SKI REMARK	Since the latest and ongoing studies are plant specific the POSs have been identified by putting a lot of effort on studying instructions and interviewing the operating personnel. This approach has been relatively similar for the studies.	

5. What screening, grouping or other process has been used to select the POSs explicitly addressed in the LPSD PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q5		See Q4
	F1/2	This POS's have been identified via FSAR/TS. It is provided that all POS's are considered. Full power operation >8% reactor power Shutdown phase (including warm SD) Cold Shutdown phase/outage (RA) The start up phase
	R1	Guideline and engineering judgement.
	R2-4	R2, Engineering judgement, results from EdF 900 MW LPSD PSA and previous PSA from 1992.
	B2	No detailed work was done to select POSs.
	O1	See answer to question Q4. Another important factor to the selection of the POSs has been if a system has been included in the PSA model or not. Basically the same systems as in the full power PSA have been included in the low power PSA. Answer only valid for low power PSA, not outage period.
	O2	No answer. (See Q4, SKI remark.)
	O3	See Q4
SKI REMARK	See Q4	

6. How have you been able to determine that this set of POSs is adequate? [How have you been able to determine that no further subdivisions in the set of POSs addressed in the analysis are required?]

CSNI Question number	Answer from Swedish plant	Answer
Q6	F1/2	There is no need of splitting the defined phases in more phases. This is because they are relatively detailed (see diagram in Q3). Analysis team also follows the outage (RA), shutdown and start up period on site (included shut down and start up) and discusses all possible initiating events with the staff.
	R1	By screening and comparing with other Nordic LPSD (POS). We are able to handle further subdivisions if it will be required in the future.
	R2-4	R2, Engineering judgement, EdF experience
	B2	No
	O1	The above mentioned division into phases (POSs) does cover the entire transaction from full power operating mode to outage period (shutdown) and back from outage period to full power operation (start-up). Answer only valid for low power PSA, not outage period.
	O2	No answer (See F1/2. SKI remark.)
	O3	The adequacy has not been verified, see Q4 and Q8.
SKI REMARK	The joint Nordic project to harmonise LPSD method descriptions will be a good help to compare the POSs.	

7. Does the set of POSs include unplanned as well as planned modes of operation?

CSNI Question number	Answer from Swedish plant	Answer
Q7	F1/2	Just planned (see Q6) There is no need of splitting the defined phases in more phases. This is because they are relatively detailed (see diagram in Q3). Analysis team also follows the outage (RA), shutdown and start up period on site (included shut down and start up) and discusses all possible initiating events with the staff.
	R1	Yes.
	R2-4	R2, No
	B2	No
	O1	Yes it does. Answer only valid for low power PSA, not outage period.
	O2	Yes
	O3	Yes. Both planned and unplanned shutdowns were considered.
SKI REMARK	When following a specific shutdown, also the unplanned modes of operation during the LPSD modes will be analysed.	

8. Does the set of POSs modelled in the LPSD PSA include transition modes?

CSNI Question number	Answer from Swedish plant	Answer
Q8	F1/2	No. Initiating event (IH) identification for respective phase (including the different transition modes)
	R1	Yes.
	R2-4	R2, To some extent
	B2	To some limited extent
	O1	The model developed for low power PSA does cover the entire time between transition from full power to cold shutdown condition and back to full power again (outage period not included here). Also any important difference in system configuration is included. Transition modes are thereby included. Answer only valid for low power PSA, not outage period.
	O2	No
	O3	No
SKI REMARK	Slightly difference exist	

9. What areas of your analysis are, in your opinion, most in need of improvement?

CSNI Question number	Answer from Swedish plant	Answer
Q9	F1/2	Identification of: "Man made" initiating events Area events External events (hazards) Level 2
	R1	No firm issue identified but there will always be a need for improvements.
	R2-4	R2, Will be evaluated when current full power PSA update is finished.
	B2	Due to new knowledge in this field the whole study has to be done from the very beginning
	O1	System requirements – the same requirements as in full power PSA have been used. Pipe break frequency – what pipe break frequency should be used at different system configurations, temperatures and pressures? Answer only valid for low power PSA, not outage period.
	O2	Shutdown period risk assessments
	O3	Refuelling outage risk assessments.
SKI REMARK	The SKI opinion is that more effort has to be put on man-made initiating events, area events, external events and on level 2	

10. What are the areas that would most benefit from further research?

CSNI Question number	Answer from Swedish plant	Answer
Q10	F1/2	See Q9 "Man made" initiating events Area events External events (hazards) Level 2
	R1	N. A.
	R2-4	R2, Will be evaluated when current full power PSA update is finished
	B2	There are methodologies developed and applied in Finland and Sweden for LPSD PSA. This type of questionnaire, e.g. a survey of what has been done world-wide in LPSD has been discussed
	O1	See answer to question Q9. Answer only valid for low power PSA, not outage period
	O2	Shutdown period risk assessments
	O3	Refuelling risk assessments
SKI REMARK	See Q9	

UNITED KINGDOM

3. What POSs have been modelled explicitly in the LPSD PSA? [Please provide details of the set of POSs that have been analysed and how they have been defined in terms of the decay heat level, the activities being carried out during the period of the POS, the condition in the reactor coolant system, the availability of safety systems, plant configurations etc.]

The potential POSs that have been identified and the various states that they cover would take pages of information to describe. These potential POSs have been condensed into a more manageable set of POSs that can be analysed and are bounding for the faults analysed.

Modes 2 and 3, Start up and Hot shutdown respectively (with heat removed via the steam generators) are bounded by the 'at power' faults.

Mode 4, Intermediate shutdown, temperatures in the range $177^{O}C > T_{avg} > 93^{O}C$; split into:

- steam generators used for decay heat removal
- residual heat removal system used for decay heat removal

Mode 5, Cold shutdown, temperatures $T_{avg} \leq 93^{O}C$, RCS intact and RHR operational; split into:

- RCS intact with steam bubble present
- RCS intact and water solid
- RCP in operation/not in operation (for boron dilution faults)

Mode 5, Cold shutdown, temperatures $T_{avg} \leq 93^{O}C$, RCS not intact, or Mode 6 (vessel head de-tensioned) not flooded up; split into:

- RCS drained down with small vents (can still recover using steam generators)
- RCS with large vents (RCS at mid-loop)

Mode 6, Refuelling, vessel head de-tensioned; split into:

- flange level head on/ head off
- refuelling pool filled
- mid-loop

4. How has this set of POSs been identified? [Please provide details of how the factors which affect the risk have been identified and taken into account in identifying the POSs.]

A very detailed assessment of the Station Operating Instructions, Technical Specifications, the application/ removal of protection systems vetoes and changes to the way the plant is controlled was undertaken. Plant status changes were identified (pressure, etc.) and equipment availability, changes/ realignments were also identified. This resulted in tables being produced which contained potential operating states from Mode 1 to No Mode and back to Mode 1 (i.e. Power mode). Note these defined 'potential' operating states were not linked to any particular type of outage. The tables of these potential operating states were then reviewed with regard to the following 5 internal initiating fault types:

- LOCA
- Loss of decay heat faults
- Boron dilution faults
- Overpressure faults
- Cooldown faults

5. What screening, grouping or other process has been used to select the POSs explicitly addressed in the LPSD PSA?

Using the information produced in the tables, the analysis of the relevant initiating faults was reviewed to determine whether the analysis was 'explicit' or 'bounding' and whether the 'bounding' was still valid. This also helped to identify where additional supporting arguments and analysis were required

6. How have you been able to determine that this set of POSs is adequate? [How have you been able to determine that no further subdivisions in the set of POSs addressed in the analysis are required?]

The comprehensive nature of the identification of 'possible' states and the review of the faults applicable and whether they are bounding or not determines the boundaries of the POSs.

7. Does the set of POSs include unplanned as well as planned modes of operation?

Yes.

8. Does the set of POSs modelled in the LPSD PSA include transition modes?

The PSA addresses the POSs which occur from full power operation down to refuelling and back again to full power operation. This includes some of the transitions that occur – for example, using the SGs to cooldown from hot/ intermediate shutdown to cold shutdown. However, the analysis does not address the transition from power operation to shutdown when a reactor trip occurs or the transition from heat removal via the SGs to heat removal by the residual heat removal system (RHRS).

9. What areas of your analysis are, in your opinion, most in need of improvement?

No currently identified need given recent review of shutdown safety case.

10. What are the areas that would most benefit from further research?

Requirement for containment integrity (i.e. isolation) when RCS is not intact (currently need modified containment integrity ensured through Tech Specs at Sizewell B)

UNITED STATES

BWR

3. What POSs have been modelled explicitly in the LPSD PSA? [Please provide details of the set of POSs that have been analyzed and how they have been defined in terms of the decay heat level, the activities being carried out during the period of the POS, the condition in the reactor coolant system, the availability of safety systems, plant configurations etc.]

Table 1 (included at the end of this discussion) identifies the POSs examined in Phase 1 of the analysis. These POSs were identified/defined after review of plant-specific procedures and operational history. Phase 2 of the analysis examined POS 5 in detail for a refuelling outage. POS 5 is rigorously defined as: Cold Shutdown , i.e., Operating Condition (OC) 4, and Refuelling (OC 5) only to the point where the vessel head is off .

POS 5 covers plant conditions during Cold Shutdown, and during Refuelling while the head is being removed. In performing the detailed analysis of POS 5 it was assumed that the initial condition was Cold Shutdown with the vessel head on.

In contrast to the situation at full power, the plant can be in POS 5 with a variety of known, pre-existing unavailabilities of equipment. At full power, essentially no equipment important for mitigating an accident is known to be unavailable, because the technical specifications do not allow the plant to remain at full power with important equipment known to be inoperable for extended periods of time. The plant can, and does, enter POSs such as POS 5, with known, pre-existing unavailabilities. For example, in response to detecting inoperability of certain equipment while at full power, the technical specifications require transitioning of the plant to POS 5.

POS 5 can be entered in coming down from power or in going back up to power. In terms of decay heat loads and known, pre-existing unavailabilities of equipment, coming down from full power is the worst case for analysis. Therefore, the initial approach to modelling POS 5 in detail, was to assume that the POS

was entered in coming down from full power. However, in a detailed review of the plant conditions during cold shutdown, it was noted that following a refuelling outage while going up in power, the vessel is hydro tested in a water solid condition in cold shutdown, using the Reactor Water Cleanup System (RWCU) in a recirculation mode for cooling the core. This hydro condition was included in the detailed model of POS 5.

The plant can be in POS 5 with a variety of conditions that affect the availability of mitigating equipment should an accident occur. In the Phase 1screening study, conservative assumptions about these conditions were made. The detailed analysis has modelled various possibilities for those conditions having significant impact on accident progression. The varieties of conditions considered in the detailed analysis for POS 5 are as follows:

1) Cold Shutdown, not in Hydro (200°F, 0 psig, decay heat 0.9% of full power)

Recirculation: Forced, or Natural

Shutdown Cooling: Train B of Residual Heat Removal (RHR), or Auxiliary Decay Heat Removal System (ADHRS) if plant has been shutdown for at least 24 hours

Main Steam Isolation Valves (MSIVs): closed, or open

Suppression Pool (SP) Inventory

Containment: closed, open only above grade, or open below grade

Safety Relief Valves (SRVs): two operable

Cold Shutdown, in Hydro (200°F, 1000 psig, decay heat 0.16% of full power)

Recirculation: forced

Shutdown Cooling: RWCU

MSIVs: closed

Suppression Pool Inventory

Containment: same as for (a5).

SRVs: all operable

Based on a review of the technical specifications and procedures for Grand Gulf, augmented with information from staff at Grand Gulf, the analysis specified the initial availability of certain systems and components in POS 5 to be fixed as follows:

Suppression Pool Makeup (SPMU): requires manual initiation (auto actuation not operable)

Isolation of RHR Shutdown Cooling: auto isolation on high pressure (135 psig) operable, and auto isolation on low level 3 operable.

As discussed previously, in POS 5, certain equipment can be known to be inoperable, and impose known, pre-existing unavailabilities of equipment prior to the occurrence of an accident initiating event. For

example, entry into cold shutdown from full power is required by the technical specifications in response to inoperability of specified equipment. Known, pre-existing unavailabilities of equipment can also happen due to maintenance outages of equipment during cold shutdown as allowed by the technical specifications. In this detailed screening study, we have not evaluated every possible unique combination of known, pre-existing unavailabilities, because the number of possible cases to consider is essentially endless. To conservatively consider the impact of these known, pre-existing unavailabilities, we made the following assumption:

Train A of the ECCS systems and the associated diesel generator are unavailable in POS 5 when an accident initiating event occurs.

Items (a) through (e) as previously described, specify the initial conditions considered for POS 5 prior to the occurrence of an accident initiating event.

Table 1 Grand Gulf POS Descriptions

POS	Description
1	Consists of: Power operation, Mode Switch in run, Any temperature Pressure at rated conditions (about 1000 psig), Thermal power no greater than 15%
2	Consists of: Hot shutdown Mode switch in shutdown, Temperature greater than 200°F, Pressure from rated pressure to 500 psig
3	Consists of: Hot shutdown Mode switch in shutdown, Temperature greater than 200°F Pressure from 500 psig to where RHR/SDC is initiated (about 100 psig)
4	Consists of: Hot shutdown Mode switch in shutdown, Temperature greater than 200°F, Unit on RHR/SDC
5	Consists of: Cold shutdown Mode switch in shutdown, Temperature 200°F or lower, Fuel in vessel with head detensioned or Removed, Mode switch in shutdown or refuel, Temperature 140°F or lower
6	Consists of: Refuelling,

Table 1 Grand Gulf POS Descriptions	
POS	**Description**
	Fuel in vessel with head removed, Mode switch in refuel, Temperature 140°F or lower, Water level raised to the steam lines
7	Consists of: Refuelling, Fuel in vessel with head removed, Mode switch in refuel, Temperature 140°F or lower, Upper pool filled, Refuelling transfer tube open.

4. How has this set of POSs been identified? [Please provide details of how the factors which affect the risk have been identified and taken into account in identifying the POSs.]

As stated previously, the screening analysis in Phase 1 examined all operational states between low power operation and refuelling back up to low power operation. These POSs were identified/ defined after review of plant-specific procedures and operational history. Phase 2, the detailed analysis, was to examine the "most important" POS identified during the screening analysis. Given that one POS was to be analyzed, POS 5 was selected. The selection process is described below.

From an examination of the results of the screening study, two figures (Figure 1 and Figure 2) were constructed. As can be seen from Figure 1, approximately 60 percent of the total core damage frequency occurs in POS 5. Therefore, from a frequency point of view POS 5 was the most logical choice.

However, frequency is not always the most important discriminator for offsite risk. In an attempt to identify the more important sequences from a risk perspective, Figure 2 was constructed. This figure provided a breakdown of the sequences classified as having a potentially high frequency with regard to an open containment and early core damage—important characteristics from the limited plant damage state analysis performed during the screening study. From Figure 2 we see that out of a total of 303 potentially high core damage sequences, 178 are from POS 5 with an open containment and early core damage. This information lent additional support to the choice of POS 5 for detailed analysis.

In addition to the numerical results, engineering insights were used to support the selection of POS 5 for detailed study. The insights for selecting POS 5 for detailed analysis were as follows.

In POSs 1, 2, and 3, the state of the plant is essentially the same as for full power except that the power is lower and pressure/temperature can be lower. Therefore, the initiating events and configuration of mitigating systems are essentially the same as for full power. Since the plant is in these POSs less often than it is at full power, the risk in these POSs is less than at full power, by a factor approximately equal to the fraction of time in these POSs divided by the fraction of time at full power. Based on this rationale, neither POSs 1, 2, nor 3 would be selected for detailed analysis.

In POSs 6 and 7, the vessel head is off, thus alleviating concerns over overpressurisation of shutdown cooling systems components. Also, in POSs 6 and 7, the water level is raised, thereby providing more time for mitigation of accident initiating events than in POSs 4 or 5.

POS 4 and POS 5 both are shutdown states. The plant is in the Hot Shutdown mode during POS 4, and it is in the Cold Shutdown mode during POS 5 (except for that part of POS 5 associated with removing the vessel head, for which the plant is in the Refuelling mode.) The vessel head is on in POS 4, and it is assumed to be on in POS 5. The core is cooled with the Shutdown Cooling (SDC) system in both POS 4 and POS 5 and with the Alternate Decay Heat Removal system in parts of POS 5. These systems are not designed for high-pressure service. If an uncontrolled pressurization transient occurs, failure of low pressure shutdown cooling systems components is possible in these POSs, if the systems/components are not isolated. Such a scenario leads to an interfacing system Loss of Coolant Accident (LOCA) outside containment which cannot be mitigated with Emergency Core Cooling Systems (ECCS), in the long term, since the Suppression Pool (SP) inventory will be lost through the break. In POS 4, shutdown cooling is with the Residual Heat Removal (RHR) system. In POS 5, shutdown cooling can be provided with either RHR, or with the Auxiliary Decay Heat Removal (ADHR) system. The maximum expected decay heat in POSs 4 and 5 is almost identical: 1.0% of full power for POS 4 and 0.9% for POS 5.

When the selection of the POS for detailed analysis was made, in August 1991, it was understood that in POS 5 at Grand Gulf, auto-isolation of SDC on high pressure was inactive, while in POS 4 it would be active. This understanding was based on information that was received during a plant visit in January 1991.

Isolation of low-pressure shutdown cooling components during pressurization transients is less likely if the auto-isolation function on high pressure is isolated; this behaviour was indicated in the screening study, which considered this function to be inoperable in POS 5. (Isolation on low level is active in POS 5, as well as in POS 4, thus providing for the ability to isolate an interfacing system LOCA in the shutdown cooling system(s) after the break occurs.) Because of the inoperability of auto-isolation on high pressure in POS 5, and because of the possible use of ADHR during POS 5, POS 5 should be chosen over POS 4 for detailed analysis.

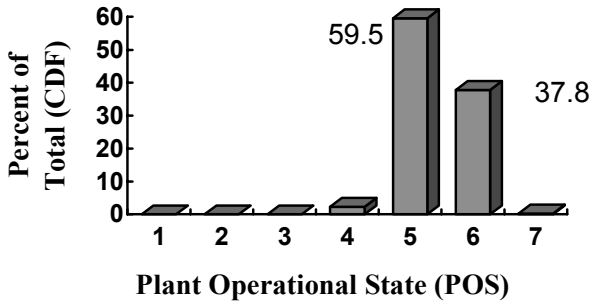


Figure 1: POS vs Percent CDF

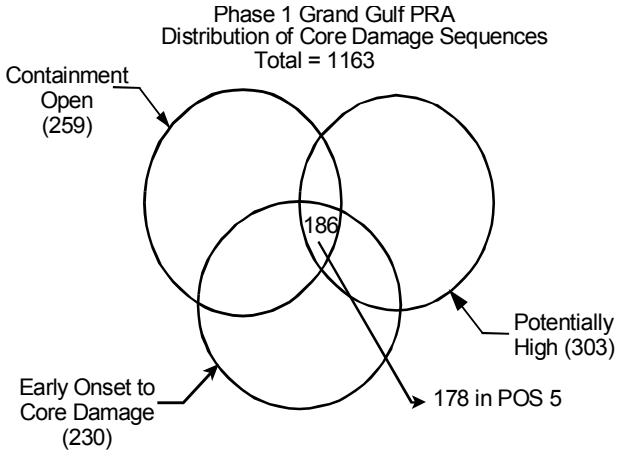


Figure 2: Open Containment and Early Core Damage Sequencesin POS 5 with a Potentially High Frequency

During a subsequent visit to the site in June 1992, it was learned that auto-isolation of SDC would be active in cold shutdown, POS 5. This change meant that one engineering reason for selecting POS 5 over POS 4 was negated; however, it was believed that POS 5 was still more of an engineering concern than POS 4 for the following reasons:

- The time for which the plant is in POS 5 is greater than the time for which the plant is in POS 4, by about a factor of 12.
- The technical specifications allow for more equipment to be inoperable in POS 5 during Cold Shutdown, than in POS 4 in Hot Shutdown

Given both the sequence insights obtained from the screening study and the engineering rationale described above, POS 5 was the logical choice for the detailed study.

5. What screening, grouping or other process has been used to select the POSs explicitly addressed in the LPSD PSA?

See above discussion.

6. How have you been able to determine that this set of POSs is adequate? [How have you been able to determine that no further subdivisions in the set of POSs addressed in the analysis are required?]

Only one POS (for one type of outage) has been examined in detail; thus, no claim is made about the adequacy of the detailed analysis with regard to completeness of coverage. However, the screening analysis did cover the spectrum of plant operation from low power down to refuelling, and back up to low power operation; thus, providing complete coverage of the possible POSs. (Note: this does not imply that all POS from all types of outages have been examined, only that the high level POS framework has been identified.)

7. Does the set of POSs include unplanned as well as planned modes of operation?

No. Analysis only dealt with planned refuelling outages.

8. Does the set of POSs modelled in the LPSD PSA include transition modes?

No. Only conditions "within" a POS were analyzed.

9. What areas of your analysis are, in your opinion, most in need of improvement?

The detailed analysis could be improved by inclusion of other POSs, other types of outages, and the inclusion of the complete set of known pre-existing conditions relevant for all types of outages. The detailed analysis might be improved by more subdivisions of the POSs and by modelling the transition between POSs.

10. What are the areas that would most benefit from further research?

Both modelling of the transition between POSs and development of a method to deal with the complete set of known pre-existing conditions relevant for all types of outages would benefit from further research.

PWR

3. What POSs have been modelled explicitly in the LPSD PSA? [Please provide details of the set of POSs that have been analyzed and how they have been defined in terms of the decay heat level, the activities being carried out during the period of the POS, the condition in the reactor coolant system, the availability of safety systems, plant configurations etc.]

1) NUREG/CR-6144- Three mid-loop POSs were modelled. Two occur in a refuelling outage, before and after refuelling, and one in a drained maintenance outage. In order to account for decay heat level when the accident occurs, four time windows (TWs) were defined.

   - TW1: ≤ 75 hours, representative decay heat 13.23 MW (2 days)
   - TW2: > 75 hours and ≤ 240 hours, representative decay heat 10 MW (5 days)
   - TW3 > 240 hours and ≤ 32 days, representative decay heat 7 MW (12 days)
   - TW4 > 32 days

   Some time windows are not applicable to all 3 POSs.

2) Screening analysis of NUREG/CR-6144- In the screening analysis, four outage types were considered, refuelling outage, drained maintenance outage, non-drained maintenance outage with use of RHR, and non-drained outage without use of RHR. For refuelling outage, 15 POSs were defined. They are listed in Table 1, along with information characterizing the conditions of the plant. The POSs of other outages were defined in a similar way.

3) NUREG/CR-6616 and 5718- The cold shutdown PSA considers the cold shutdown POSs and adopted the time windows defined in 1).

4. How has this set of POSs been identified? [Please provide details of how the factors which affect the risk have been identified and taken into account in identifying the POSs.]

The outage types and POSs were identified by reviewing the Graybook database (NUREG-0020) on outages, operating procedures during shutdown, and shift supervisor's log books.

5. What screening, grouping or other process has been used to select the POSs explicitly addressed in the LPSD PSA?

The screening analysis of NUREG/CR-6144 is to screen the POSs based on their risk contributions. Based on the screening model, the POS in which refuelling is taking place may be screened based on the large inventory of coolant that is available.

6. How have you been able to determine that this set of POSs is adequate? [How have you been able to determine that no further subdivisions in the set of POSs addressed in the analysis are required?]

The time window approach was used to account for the decay heat level in a POS. It was necessary because of the long time the plant stays in a POS. It effectively subdivided the POSs. With the recent trend of the US nuclear industry of shortened outages, the time window approach may no longer be necessary.

7. Does the set of POSs include unplanned as well as planned modes of operation?

Yes, see answer to Q3.

8. Does the set of POSs modelled in the LPSD PSA include transition modes?

No. Only conditions "within" a POS were analyzed.

9. What areas of your analysis are, in your opinion, most in need of improvement?

For this study, the numbers of outage types and POSs are believed to be about right.

10. What are the areas that would most benefit from further research?

- Assessment of risk contribution of outages that are not refuelling outages and do not necessarily involve going to midloop.
- Development of methods for screening POSs.

Table 1: Plant Operational States for Surry Unit 1 - Low Power and Shutdown Outage Activities (10/86 Outage)

		Activities	Level	K	T (°F)	P (psig)	Surry T.S.	Standard T.S. Mode	Operating Procedures[2]
1	Low Power Operation & RX Shutdown	Transfer to manual RX control Transfer to manual SG level control	In Przr.	=1.0 (<15% Power)	547	2235		1,2	GOP-2.2; GOP-2.3; OP-58.2.1
				# 0.9823TS[2]	547	2235	HSD		
2	Cooldown With SG	Borate RCS Stop 2 RCPs Establish LTOP <350EF		<1.0				3	GOP-2.4; OP-8.3.3
			In Przr.	< .96Adm[3]	345	345	ISD		
3	Cooldown With RHR	Initiate RHR						4	GOP- 2.5; OP-14.1
			In Przr.	<0.96Adm[3]	200	345	CSD		
4	Cooldown With RHR	Secure SGs						5	GOP- 2.6; OP-14.6; OP-31.2,-31.3,-63.1
			In Przr.	<0.96Adm[3]	140	345	CSD		
5	Drain RCS to Mid-loop	Deenergize Przr heaters Stop last RCP Cooldown Przr by fillup						5,6	OP-RC-004; OP-RC-005; OP-5.1.3; OP-8.4
			Mid-loop	<0.96Adm[3]	<140	0	CSD		
6	Mid-loop Operation	Possible loop isolation						5,6	MOP-5.3; 5.4; 5.5
			Mid-loop	<0.96Adm[3]	<140	0	CSD		
7	Fill for Refuelling	Remove vessel head & upper internals structure Establish Refuelling Ctmt Integrity						6	OP-4.1; 5.8.1; 5.8.2; OP-7.1.2
			Refuel Cavity Full	<0.95TS[1]	<140	0	CSD		
8	Refuelling	Move fuel assemblies per refuelling sequence						6	OP-4.1
			Refuel Cavity	<0.95TS[2]	<140	0	CSD		

			Full						
9	Drain RCS to Mid-loop after Refuelling	Install vessel upper internals structure & head No longer need Refuelling Ctmt. Integrity						6	OP-RC-004; OP-RC-005; OP-4.1; OP-4.4; OP-14.3; -15.2
			Mid-loop	0.96Adm(3)	<140	0	CSD		
10	Mid-loop Operation after Refuelling	Possible further loop isolation						5,6	MOP-5.3; 5.4; 5.5
			Mid-loop	<0.96Adm(3)	<140	0	CSD		
11	Refill RCS completely	Pressurize RCS & jog RCP for RCS vent & Fill Unisolate any and all loops						5,6	GOP-1.1; OP-5.8.2; 5.1.1; 5.2.1; MOP-5.6,-5.7, or -5.8
			RCS Full	<0.96Adm(3)	<140	<125	CSD		
12	RCS Heatup Solid and Draw Bubble	Pressurize RCS, start RCP & energize Przr. htrs Establish Ctmt. Integrity and Ctmt. vacuum						5	GOP-1.1; OP-5.2.1; OP-5.12
			In Przr.	<0.98Adm(3)	200	345	CSD		
13	RCS Heatup	Start other RCPs Enable Engineered Safeguards Secure RHR						4	GOP-1.1; GOP-1.2; GOP-1.3; OP-14.2
			In Przr.	<0.9823TS(2)	350	.345	ISD		
14	RCS Heatup with SGs	Flow test MDAFWs Disable LTOP protection >350EF Unblock accumulators Verify auto SI re-established Dilute RCS		<1.0				3	GOP-1.3; OP-14.2
			In Przr.	#0.9823TS(2)	547	2235	HSD		
15	Rx Startup & Low Power Operation	Flow test TDAFW Transfer to auto SG level Transfer to auto RX control		=l.0 (<15% power)	547	2235		1,2	GOP-1.4; GOP-1.5; OP-8.3.2

1. The operating procedures listed are based on the most updated information as of yet. Some of these procedures' numberings are different from those prevailing in October 1986.

2. Shutdown Margin Requirements - Per Technical Specifications (1.0 Definitions)

CSD(Cold Shutdown) Condition - RX sub-critical by at least 1%. K/K (□ K˜ 0.99) and Tavg is .200□F.

ISD(Intermediate Shutdown) Condition - RX sub-critical by. 1.77%. K/K (□ K˜ 0.9823) and Tavg is >200□F but <547□F.

HSD(Hot Shutdown) Condition - RX subcritical by. 1.77%. K/K (□Ḳ0.9823) and Tavg is >547□F.

3. Shutdown Margin Administrative Limits - Per 1-OP-1C, Shutdown Margin Calculation

   A further requirement states that except for during the approach to criticality, the minimum boron concentration allowed in a shutdown core must be the greater of the following:

   Critical Boron for the 0% power insertion limit OR

   The boron required to maintain the administratively (Adm*) required shutdown margin > 5500 pcm (K=0.945) from BOL (beginning of life)

   to 9000 MWD/MTU and >4000 pcm (K≲0.960) after 9000 MWD/MTU.

APPENDIX B – INITIATING EVENTS

B.1 Identification of Initiating Events

PRELUDE AND QUESTIONS

Initiating events are the events that have the ability to challenge the condition of the plant. These events include failure of equipment from either "internal plant causes" such as hardware faults, operator actions, floods or fires, or "external plant causes" such as seismic or high winds. For most PSA uses, the risk perspective should be based on the total risk associated with the operation of the plant which includes: all operational modes (full power, low power, planned outages, and unplanned outages); events from both internal and external sources; and events resulting in radioactive releases from the reactor or other sources (e.g., fuel pool).

Most LPSD PSAs are limited to the evaluation of the risk associated with internal events and to radioactive releases from the reactor core. However, some studies have shown that fire, flood, and seismic events can be important contributors to risk. Furthermore, although spent fuel pool (SFP) misloadings are not typically included in the analysis, it has been identified as an area of concern. (SFP loading and reracking of the SFP inventory are becoming increasingly challenging due to the increase in SFP inventories. The concern is that the potential for a fuel criticality accident during the SFP reracking activity increases with the increase in SFP inventory.) In addition, some analysts believe that crane failures during heavy lifts inside containment present the potential of an initiating event with potentially severe consequences.

Determining the initiating events that should be included in the analyses is very important in analyzing the risk associated with LPSD operations. Please respond to the following questions.

11. What process used to identify initiating events in the LPSD PSA?

12. How has this process differed from the one used for the full power PSA?

13. How do the set initiating events identified reflect the greater level of the activity on the plant (particularly during shutdown)?

14. Have you identified initiating events that are unique to LPSD conditions? If so, what are they?

15. How have you justified that the process used to identify the set of initiating events addressed is adequate and complete

16. How does the process used to identify the set of initiating events relate to the objectives of the PSA and to the actual or (foreseeable) future use of the PSA?

RESPONSES

BELGIUM

11. What process used to identify initiating events in the LPSD PSA?

The process involved screening from list of IE for power states, and deleting those that are not applicable for shutdown. Specific IEs for shutdown were added (e.g. loss of RHRS)

12. How has this process differed from the one used for the full power PSA?

There is no difference.

13. How do the set initiating events identified reflect the greater level of the activity on the plant (particularly during shutdown)?

The IEs do not reflect the greater level of activity during shutdown.

14. Have you identified initiating events that are unique to LPSD conditions? If so, what are they?

Inadvertent loss of RCS level during mid-loop conditions, and loss of RHRS in all POSs were identified.

15. How have you justified that the process used to identify the set of initiating events addressed are adequate and complete

No other potential sources of IEs were identified.

16. How does the process used to identify the set of initiating events relate to the objectives of the PSA and to the actual or (foreseeable) future use of the PSA?

The objective of the PSA is to have a probabilistic assessment of the level of safety of the plant in the Periodic Safety Review, which is part of the licensing conditions of the plant. For the future, the list of IEs will be re-evaluated when defining the scope of PSA updates.

GERMANY

11. What process used to identify initiating events in the LPSD PSA?

The possible IE within any POS have been investigated on the basis of

- The experience from PSA for power operation,
- The operating experience of German PWR during LP&SD,
- International experience with LP&SD analysis and
- Engineering expertise.

12. How has this process differed from the one used for the full power PSA?

The initiating events during LP&SD have a different definition compared to FP, this leads to differences in the process of identification.

An important source for initiating events (IE) was the operating experience in terms of notified incidents. The classification of IE on that basis under the use of the definition of the IE for LP&SD was different from the procedure for FP-IE.

13. How do the set initiating events identified reflect the greater level of the activity on the plant (particularly during shutdown)?

Some IE are only possible during special POS like mid loop operation and some IE are caused by maintenance faults (loss of preferred power due to maintenance, unintended activation of ECCS signals while lowering the coolant level to mid-loop, unexpected injection of unborated coolant)

14. Have you identified initiating events that are unique to LPSD conditions? If so, what are they?

Loss of residual heat removal due to faulty level lowering or due to operational failure of RHR trains during mid-loop-conditions.

Unintended activation of ECCS signals.

IE concerning unexpected injection of unborated coolant.

15. How have you justified that the process used to identify the set of initiating events addressed are adequate and complete

It is the understanding of the PSA methodology and the safety culture that this process will never be completed.

The set of IE seems to be adequate as the main contributors to risk according to the evaluated sources and with respect to our engineering judgement have been analyzed.

The contribution of the not analyzed IE has been screened and found to be negligible.

16. How does the process used to identify the set of initiating events relate to the objectives of the PSA and to the actual or (foreseeable) future use of the PSA?

Since the objective was the determination of the safety level and the identification of the main risk contributors, the main contributors to risk (according to other PSA and a screening of the identified IE) were analysed.

As to the future use for the development of a PSA-LP&SD guideline, IE have been analyzed which can only occur during shut down states, particularly during mid-loop operation and deboration accidents.

HUNGARY

11. What process used to identify initiating events in the LPSD PSA?

The initiating event list of the PSA study for nominal power operational mode was considered as a starting list of the initiating events for low power and shutdown states. This list was partly reduced, and partly supplemented taking into account the specifics of the different plant operational states. Reduction of the list was based on a review of the validity and the possibility of occurrence of the original initiating events in the defined plant operational states. During this, events having no chance to occur in a POS due to its technological or operational characteristics were excluded, while others, having the same consequences were combined. After that, shutdown-specific events that have the potential to endanger the safety of the unit and not analyzed before were studied. The above list was supplemented by these shutdown-specific events. The latter included events that have the potential to lead to (1) inadvertent boron dilution, (2) cold overpressurisation of the reactor vessel, (3) termination of the natural circulation or initiated by (4) heavy load drop. The final list of initiating events is given in **Table 1**, where the validity of the initiating events in the different plant operational states is also indicated.

12. How has this process differed from the one used for the full power PSA?

The starting list of initiating events for the full power PSA was compiled from generic sources and other PSAs, and then it was translated into a plant-specific initiating event list using plant experience and expert opinion. Since this was used as a starting list for the LPSD PSA, it was already a plant-specific one. Except for this difference, the process was basically the same.

13. How do the set initiating events identified reflect the greater level of the activity on the plant (particularly during shutdown)?

Initiating events caused by potential erroneous interaction of the plant staff or maintenance personnel were taken into account to a great extent.

14. Have you identified initiating events that are unique to LPSD conditions? If so, what are they?

Yes, a wide range of initiators have been considered that have the potential to lead to (1) inadvertent boron dilution, (2) cold overpressurisation of the reactor vessel, (3) termination of the natural circulation or initiated by (4) heavy load drop.

Initiating Event		Plant Operational State																							
ID	Description	0. Nominal Power	1. Operation with one turbine	2. Boron addition to subcrit.	3. Cooldown to 240 °C	4. Cooldown to 150 °C	5. Cooldown to 60 °C	6. Natural circulation	7. Nat. circulation in 2 loops	8. Opening of the reactor	9. Reactor open, RVHF -300	10. Unloading	12. Reloading	13. Reactor open, RVHF -300	14. Closing of the reactor	15. Pressurization	16. Primary pressure 25 bar	17. Containment leaktight test	18. Primary pressure 25 bar	19. Heatup to 120 °C	20. 137/164 bar leaktight tests	21. Heatup to 150 °C	22. Heatup to hydroacc.	23. Reaching reactor criticality	24. Reactor power increase
A1	Gross Reactor Vessel Rupture	+	+	+	+	+	+	+	+	+	+			+	+	+	+	+	+	+	+	+	+	+	+
A2	Control Rod Ejection	+	+	+	+	+	+														+	+	+	+	+
A3	Overpressurization during 25 bar Leaktight Test																+								
A4	Inadvertent Cooldown during Pressure Test																				+				
B1	Large LOCA: Loops 2, 3, 5 Cold Leg	+	+	+	+	+																	+	+	+
B2	Large LOCA: Loops 1, 6 Cold Leg	+	+	+	+																			+	+
B3	Large LOCA: Loop 4 Cold Leg	+	+	+	+																			+	+
B4	Large LOCA: Loops 1, 2, 3, 5, 6 Hot Leg	+	+	+	+																			+	+
B5	Large LOCA: Loop 4 Hot Leg	+	+	+	+																			+	+
B6	Large LOCA: Loops 1, 6 or Loops 2, 3, 5 Hot Leg					+																	+		
B7	Large LOCA: Loop 4					+																	+		
B8	Large LOCA						+	+	+	+	+	+	+	+	+	+	+	+	+	+	+	+			
C1	Medium LOCA not Affecting ECCS Operation (L5)	+	+	+	+																			+	+
C2	Medium LOCA Affecting Low Pressure ECCS Operation (L5)	+	+	+	+																			+	+
C3	Medium LOCA Affecting High Pressure ECCS Operation (L5)	+	+	+	+																			+	+
C4	Rupture of Hydroaccumulator Pipeline (L5)	+	+	+	+																			+	+
C5	Medium LOCA not Affecting ECCS Operation (L4)	+	+	+	+	+																	+	+	+
C6	Medium LOCA Affecting High Pressure ECCS Operation (L4)	+	+	+	+	+																	+	+	+
C7	Medium LOCA not Affecting ECCS Operation (L3)	+	+	+	+																			+	+
C8	Medium LOCA Affecting High Pressure ECCS Operation (L3)	+	+	+	+																			+	+
C9	Inadvertent Opening of Pressurizer Safety Relief Valve	+	+	+	+	+																	+	+	+
C10	Medium LOCA not Affecting ECCS Operation (L2)	+	+	+	+																			+	+
C11	Medium LOCA Affecting High Pressure ECCS Operation (L2)	+	+	+	+																			+	+
C12	Medium LOCA not Affecting ECCS Operation (L3+L2)					+																	+		
C13	Medium LOCA Affecting High Pressure ECCS Operation (L3+L2)					+																	+		

Table 1 - Initiating events in different plant operational states

Initiating Event		Plant Operational State																							
ID	Description	0. Nominal Power	1. Operation with one turbine	2. Boron addition to subcrit.	3. Cooldown to 240 °C	4. Cooldown to 150 °C	5. Cooldown to 60 °C	6. Natural circulation	7. Nat. circulation in 2 loops	8. Opening of the reactor	9. Reactor open, RVHF -300	10. Unloading	12. Reloading	13. Reactor open, RVHF -300	14. Closing of the reactor	15. Pressurization	16. Primary pressure 25 bar	17. Containment leaktight test	18. Primary pressure 25 bar	19. Heatup to 120 °C	20. 137/164 bar leaktight tests	21. Heatup to 150 °C	22. Heatup to hydroacc.	23. Reaching reactor criticalit	24. Reactor power increase
D1	Small LOCA Initiating ECCS Operation	+	+	+	+	+																	+	+	+
D2	Small LOCA not Initiating ECCS Operation	+	+	+	+	+																	+	+	+
D3	Small LOCA						+	+	+	+	+	+	+	+	+	+	+	+	+	+	+	+			
E1	Primary Water Flow to Secondary Side in Steam Generator	+	+	+	+	+																	+	+	+
E2	Interface LOCA Initiating ECCS Operation	+	+	+	+	+																	+	+	+
E3	Interface LOCA not Initiating ECCS Operation	+	+	+	+	+																	+	+	+
E4	Interface LOCA						+	+	+	+	+	+	+	+	+	+	+	+	+	+	+	+			
E5	Inadvertent Overdraining of Primary Coolant									+				+											
F1	Trip of One or Two Reactor Coolant Pumps	+																							
F2	Simultaneous Trip of Three Reactor Coolant Pumps	+																							
F3	Inadvertent Closure of One Main Gate Valve	+												+	+	+	+	+	+						
G1	Loss of One Feedwater Pump	+																							
G2	Loss of All Feedwater Pumps	+	+	+	+	+																			+
G3	Feedwater Collector Rupture	+	+	+	+	+	+	+	+	+	+			+	+	+	+	+	+	+	+	+	+	+	+
G4	Feedwater Line Rupture Outside Containment/Feedwater Line Rupture	+	+	+	+	+	+	+	+	+	+			+	+	+	+	+	+	+			+	+	+
G5	Rupture of Feedwater Pump Discharge Line before Check Valve	+	+	+	+	+																	+	+	+
G6	Rupture of Feedwater Pump Suction Line	+	+	+	+	+																	+	+	+
G7	Feedwater Line Rupture Inside Containment	+	+	+	+	+	+																+	+	+
G8	Pipeline Rupture in Secondary Residual Heat Removel System				+	+	+	+	+	+	+			+	+	+	+	+	+						
G10	Loss of Secondary Residual Heat Removal Pumps						+	+	+	+	+			+	+	+	+	+	+						
H1	Inadvertent Closure of Steam Generator Isolation Valve	+	+	+																					+
H2	Inadvertent Closure of Secondary Residual Heat Removal Line				+	+	+	+	+	+	+			+	+	+	+	+	+						
I1	Inadvertent Opening of Steam Generator Safety Relief Valve	+	+	+	+																				+
I2	Inadvertent Opening of Main Steam Atmospheric Relief Valve	+	+	+	+																			+	+
I3	Steam Line Rupture Inside Containment	+	+	+	+	+	+																+	+	+
I4	Steam Line Rupture Outside Containment/Steam Line Rupture	+	+	+	+	+	+	+	+	+	+			+	+	+	+	+	+	+			+	+	+
I5	Main Steam Collector Rupture	+	+	+	+	+	+	+	+	+	+			+	+	+	+	+	+	+	+	+	+	+	+

Table 1 - Initiating events in different plant operational states (continued)

Initiating Event		Plant Operational State																							
ID	Description	0. Nominal Power	1. Operation with one turbine	2. Boron addition to subcrit.	3. Cooldown to 240 °C	4. Cooldown to 150 °C	5. Cooldown to 60 °C	6. Natural circulation	7. Nat. circulation in 2 loops	8. Opening of the reactor	9. Reactor open, RVHF -300	10. Unloading	12. Reloading	13. Reactor open, RVHF -300	14. Closing of the reactor	15. Pressurization	16. Primary pressure 25 bar	17. Containment leaktight test	18. Primary pressure 25 bar	19. Heatup to 120 °C	20. 137/164 bar leaktight tests	21. Heatup to 150 °C	22. Heatup to hydroacc.	23. Reaching reactor criticality	24. Reactor power increase
J1	Trip of One Turbine	+																							
J2	Trip of Both Turbines	+																							
J3	Loss of Electric Load Down to Selfconsumption	+																							
J4	Total Loss of Electric Load	+																							
J5	Loss of Second (Last) Turbine		+																						+
K1	Loss of All 6 kV Busbars (31BA&31BB&32BA&32BB)	+	+	+	+	+	+	+	+	+	+	+	+	+	+	+	+	+	+						+
K2	Loss of One 6 kV Busbar (31BA+31BB+32BA+32BB)	+	+	+	+	+																			+
K3	Loss of EV Busbar	+	+																						+
K4	Spurious "Large LOCA" Signal	+	+	+																					+
K5	Spurious "Main Steam Collector Rupture" Signal	+	+	+	+	+																			+
L1	Loss of Intermediate Cooling to Reactor Coolant Pumps	+	+	+	+	+																	+	+	+
L2	Loss of Intermediate Cooling to Control Rods	+	+	+																					+
L3	Loss of Make-up Water Pump (Pump in Reserve Fails to Start)	+	+	+	+	+																	+	+	+
L4	Loss of Service Water System							+							+										
M1	Spurious Reactor Trip	+	+	+																					+
N1	Uncontrolled Control Rod Withdrawal	+																							
N2	Uncontrolled Control Rod Group Withdrawal	+																							
N3	Inadvertent Dilution in Primary Circuit	+																							
NX	External Dilution				+	+	+													+		+	+		

Table 1 - Initiating events in different plant operational states (continued)

15. How have you justified that the process used to identify the set of initiating events addressed are adequate and complete

Completeness of the initiating event list was ensured by the application of the following methods:

- use of operational experience
- expert evaluation
- comparison of the initiating event list with initiating event lists of earlier PSA studies and with other reference lists
- master logic diagram.

Application of the above methods enabled the review of the initiating event list from different points of view in order to ensure that no important initiating event remained outside the scope of the analyses.

16. How does the process used to identify the set of initiating events relate to the objectives of the PSA and to the actual or (foreseeable) future use of the PSA?

The objectives of the LPSD PSA were to quantify core damage risk when the plant is operated at low power or is shut down in a planned manner, identify dominant contributors (initiating events, accident sequences, malfunctions, equipment and human failures) to risk and develop recommendations for improving safety in low power and shutdown modes. Since the initiating event identification process has resulted in a plant-specific and complete list of initiating events for such operational states, it is considered that it served the purposes of the PSA. With regards to the use of PSA it is considered that the process also served that purposes until the use of PSA is concerned with the operational states studied and the internal initiating events. It may be necessary later to extend the scope of the LPSD PSA with both internal and external hazards.

JAPAN

11. What initiating events have been addressed in the LPSD PSA? [Please provide details of the initiating events addressed in the LPSD PSA for each of the POSs.]

NUPEC addresses the following initiating events in LPSD PSA for PWRs:

1) LOCA1

Loss of RCS inventory due to the events such as stuck open of RHR relief valves during the period of RCS filled, spurious opening of drainage valves of RCS during drainage and/or refilling RCS and so on.

LOCA2

Loss of RHRS during Mid-Loop Operation due to over drain of RCS inventory or failure to maintain water level.

RHRT

Loss of one train of RHRS including failure of RHR pump, Loss of CCWS, spurious closure of MOV at the suction of RHR pump due to failure of inter-lock and recoverable failure of RHRS.

LOSP

Loss of Offsite Power including All AC Power Supply.

RIA

For example, in 4 Loop PWRs, reactivity initiation due to inducement of demineralised water slag to reactor core in case of loss of offsite power and its recovery during plant starting up, when the boron dilution is continued during loss of offsite power.

In the LPSD PSA for Japanese PWRs at NUPEC, the initiating events are considered depending on POS (see the following Table 2.1).

NUPEC addresses the following initiating events in LPSD PSA for BWRs:

1) Loss of decay heat removal

Loss of decay heat removal due to the failure of SDCS (Shutdown Cooling System) in operation

Following events are considered:

i. Failure of SDCS-A

ii. Failure of SDCS-B

iii. Failure of common suction of SDCS-A and SDCS-B

Loss of Off Site Power

Loss of Offsite Power including All AC Power Supply

Pipe Break LOCA

Large / Intermediate / Small Break LOCA at the re-circulation system in conservative manner

Table 2.1 Initiating Events considered in LPSD PSA for PWRs in NUPEC

POS	Brief Description of POS	Initiating Events				
		LOCA1	LOCA2	RHRT1	LOSP	RIA
1	Low Power Operation Reactor Shut down				○	
2	Cooling down via SGs				○	
3	Cooling down via RHRS		○	○	○	
4	Cooling down via RHRs (RCS filled)		○	○	○	
5	RCS drain to Mid-Loop		○	○	○	
6A	Mid-Loop Operation w/o nozzle dam	○	○	○	○	
6AB	Mid-Loop Operation with partial nozzledam	○	○	○	○	
6B	Mid-Loop Operation with complete nozzle dam	○	○	○	○	
7	Refilling Cavity for fuel re-load		○	○	○	
8	Re-fuelling				○	
9	Drain Cavity after Re-fuel		○	○	○	
10A	Mid-Loop Operation w/o nozzle dam	○	○	○	○	
10AB	Mid-Loop Operation with partial nozzle dam	○	○	○	○	
10B	Mid-Loop Operation with complete nozzle dam	○	○	○	○	
11	RCS refill		○	○	○	
12	RCS heat up operation with RHRS (RCS filled)		○	○	○	
13	RCS heat up operation with RHRS with 25% water level in pressurisers		○	○	○	
14	RCS heat up operation with SGs				○	
15	Reactor start up Low Power Operation				○	○

○: The initiating event is considered in the POS

RHR LOCA

The following two types of RHR are considered:

LOCA during changing configuration of RHR

LOCA from valves in SDCS due to the failure of inter-lock or operators' omission error in case of SDCS start up

LOCA during RHR operation

LOCA from valves in SDCS operating

In the LPSD PSA for Japanese BWRs at NUPEC, the initiating events are considered depending on POS (see the following Table 2.2)

Table 2.2 Initiating Events considered in LPSD PSA for BWRs in NUPEC

POS	Brief Description of POS	a-(1)	a-(2)	a-(3)	b	c	d-(1)	d-(2)
S	Vacuum break of main	○	-	○	○	○	○	○
A	Outage of HPCS / LPCI	○	-	○	○	○	-	○
B1	Well Filled up	○	-	○	○	○	-	○
B2	Outage of CUWS	○	-	○	○	○	-	○
B3	RHR switch over	-	○	○	○	○	○	○
C	RPV / PCV recovery	-	○	○	○	○	-	○
D	All system stand-by	○	-	○	○	○	○	○

(The response's Q12 How has this set of initiating events been identified?)

NUPEC has identified this set of initiating events from the all-possible abnormal events relevant to core damage on the view points of over heating of fuels systematically, referring to a procedure guide for LPSD PSA (AESJ-SC-P001), the events in Japanese PWRs / BWRs and the previous LPSD PSA abroad (e.g. NUREG/CR-6144). The mechanical fuel damage is not considered because such event, which can be caused by the accident in fuel handling, can not cause the sever core damage.

For PWRs, NUPEC identified the set of initiating events on the following categories:

- Loss of RHR function
- Loss of Coolant

- Loss of Off Site Power
- Reactivity Insertion Accident

For BWRs, NUPEC identified the set of initiating events on the following categories:

- Loss of RHR function
- Loss of Coolant
- Loss of Off Site Power

The following figure 2.1 illustrates the example of identification of initiating events for PWRs during LPSD operation.

12. How has the process used to identify the set of initiating events differed from the one used for the full power PSA?

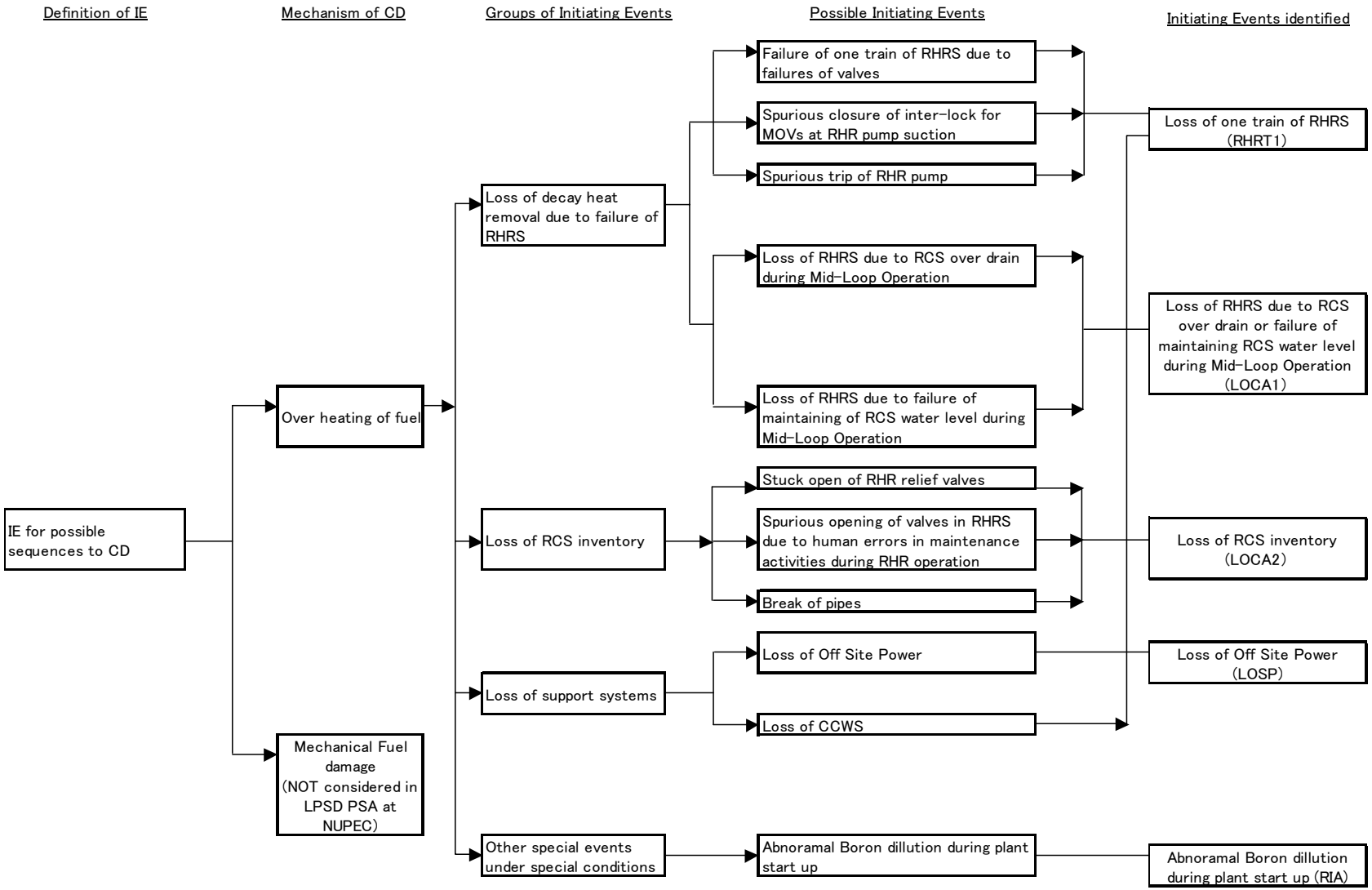
In full power PSA, NUPEC sets initiating events referring to the events taken place in Japanese PWRs / BWRs, design base events and possible severe events considered in previous studies (e.g. EPRI-NP-2310) in consideration with the plant design and operating condition of Japanese PWRs / BWRs.

In LPSD PSA, the process of identification of initiating events is basically NOT different from that used in full power PSA. In LPSD PSA, NUPEC has identified this set of initiating events from the all-possible abnormal events relevant to core damage on the view points of over heating of fuels systematically, referring to a procedure guide for LPSD PSA (AESJ-SC-P001), the events in Japanese PWRs / BWRs and the previous LPSD PSA abroad (e.g. NUREG/CR-6144). An example of the process for identification of initiating events for LPSD PSA at NUPEC is shown in A12.

13. How do the set initiating events identified reflect the greater level of the activity on the plant (particularly during shutdown)?

NUPEC identifies initiating events in consideration with possible human errors referring to maintenance procedures, T-Spec, plant operating conditions (configurations) and actual events taken place in Japanese and foreign countries during LPSD operation. Furthermore, NUPEC refers to the previous LPSD PSA studies performed in foreign countries (e.g.. NUREG/CR-6144).

Figure 2.1


Definition of IE
Mechanism of CD
Groups of Initiating Events
Possible Initiating Events
Initiating Events identified
IE for possible sequences to CD
Over heating of fuel
Mechanical Fuel damage (NOT considered in LPSD PSA at NUPEC)
Loss of decay heat removal due to failure of RHRS
Loss of RCS inventory
Loss of support systems
Other special events under special conditions
Failure of one train of RHRS due to failures of valves
Spurious closure of inter-lock for MOVs at RHR pump suction
Spurious trip of RHR pump
Loss of RHRS due to RCS over drain during Mid-Loop Operation
Loss of RHRS due to failure of maintaining of RCS water level during Mid-Loop Operation
Stuck open of RHR relief valves
Spurious opening of valves in RHRS due to human errors in maintenance activities during RHR operation
Break of pipes
Loss of Off Site Power
Loss of CCWS
Abnoramal Boron dillution during plant start up
Loss of one train of RHRS (RHRT1)
Loss of RHRS due to RCS over drain or failure of maintaining RCS water level during Mid-Loop Operation (LOCA1)
Loss of RCS inventory (LOCA2)
Loss of Off Site Power (LOSP)
Abnoramal Boron dillution during plant start up (RIA)

14. Have you identified initiating events that are unique to LPSD conditions? If so, what are they.

NUPEC identifies such initiators referring to maintenance procedures, T-Spec, plant operating conditions (configuration) and the previous LPSD PSA studies performed in foreign countries (e.g. NUREG/CR-6144). The initiating events unique to LPSD of PWRs are as follows:

1) Loss of RHRS during Mid-Loop Operation due to failure to maintain the water level in RCS or over drain

2) Loss of RHRS due to failure of RHR pump, Loss of CCWS, spurious closure of MOV at the suction of RHR pump due to failure of inter-lock and recoverable failure of RHRS

3) Loss of RCS inventory due to stuck open of RHR relief valves during the period of RCS filled, spurious open of drainage valves of RCS during drainage and/or refilling RCS and so on (Loss of RCS inventory due to pipe break is not considered explicitly)

4) Reactivity initiation due to inducement of water slag to reactor core in case of loss of offsite power and its recovery during plant starting up, when the boron dilution is continued during loss of offsite power (Spurious boron dilution due to spurious opening the valve isolating water is NOT currently considered)

For BWRs, loss of coolant inventory due to spurious configuration by human error is considered. This initiating event would cause the discharging of the coolant to suppression pool.

15. How have you justified that the process used to identify the set of initiating events addressed is adequate and complete

The process to identify the initiating events is justified in terms of the requirement / description of a procedure guide for LPSD PSA (AESJ-SC-P001). Also, NUPEC checks the initiating events for LPSD PSA in terms of operating procedures, plant design, T-Spec, plant operating conditions (configuration). Furthermore, NUPEC compares the initiating events to those in the previous procedures (e.g. NUREG/CR-6144).

16. How does the process used to identify the set of initiating events relate to the objectives of the PSA and to the actual or (foreseeable) future use of the PSA?

NUPEC conducts the LPSD PSA in order to review the LPSD PSA by licensees in the Periodic Safety Review (PSR) report, where the LPSD PSA is mainly used to confirm the safety level during the LPSD state. Therefore, the initiating events are identified for the purpose of this objective.

Two additional questions / responses are included here:

What areas of your analysis are, in your opinion, most in need of improvement?

1) To address explicitly Loss of RCS inventory due to pipe break LOCA (e.g. Possibility of pipe breaks and break size)

2) To address / investigation of the possibility on over pressure events at low temperature

3) To address the spurious boron dilution due to spurious opening of isolation valve of water by human error and so on (PWR)

4) More detail analyses the initiating events relevant to human error

5) Inclusion of the effect of external events, such as fire, flood and seismic, to identification of initiating events

What are the areas that would most benefit from further research?

1) The construction of an international database for events during LPSD and precursor analyses on the events taken place at Japanese and foreign PWRs / BWRs are beneficial.

2) Inclusion of the effect of external events, such as fire, flood and seismic, to identification of initiating events

3) The mechanism / possibility of LOCA during LPSD operation

KOREA

11. What process used to identify initiating events in the LPSD PSA?

The identification and selection of initiating event were performed as follows:

- Review the previous shutdown PSA experiences,
- Review the accidents during the low power and shutdown operation modes described in the APR-1400 standard safety analysis report,
- Review the initiating events considered in the full power PSA for APR-1400,
- Review the accidents occurred in the past, and
- Select appropriate initiating events for each POS during shutdown operation modes.

The initiating event analysis was mainly performed with the insight obtained from previous shutdown PSAs.

12. How has this process differed from the one used for the full power PSA?

There is no different process between the LPSD PSA and the full power PSA for the identification and selection of initiating events.

13. How do the set initiating events identified reflect the greater level of the activity on the plant (particularly during shutdown)?

See the response to Q11.

14. Have you identified initiating events that are unique to LPSD conditions? If so, what are they?

For example, the loss of decay heat removal (LDHR) is not considered as an initiating event in the full power PSA. The shutdown cooling system (SCS) is not a normal operating system. The purpose of the SCS is to cooldown the RCS after reactor trip. If SCS does operate correctly and if there is no appropriate

operator response to loss of SCS, the core damage can be occurred due to loss of decay heat removal. In shutdown PSA, the LDHR is a major concerned initiating event.

15. How have you justified that the process used to identify the set of initiating events addressed is adequate and complete?

No justification was made.

16. How does the process used to identify the set of initiating events relate to the objectives of the PSA and to the actual or (foreseeable) future use of the PSA?

Referring the maintenance practices, it meets the analysis objectives for the identification of plant vulnerabilities. From the evaluation results, we can prioritize all events for possible LPSD scenarios.

MEXICO

11. What process used to identify initiating events in the LPSD PSA?

The process involved review of the applicability of full power initiating events at low power conditions, reports and studies for plants of similar design (external operational experience), and NER/ LER (internal operational experience)

12. How has this process differed from the one used for the full power PSA?

The major difference was to decide, in some POSs, if the initiating event will demand proper attention due to the amount of time available to produce a significant challenge to systems or operator actions.

13. How do the set initiating events identified reflect the greater level of the activity on the plant (particularly during shutdown)?

There are some initiating events related directly with misunderstanding or errors due to the amount of activities performed during shutdown stage. For instance, small LOCA for diversion to suppression pool via RHR.

14. Have you identified initiating events that are unique to LPSD conditions? If so, what are they?

Yes, for instance loss or isolation of residual heat removal systems, loss of reactor water clean up system, and LOCA at low pressure.

15. How have you justified that the process used to identify the set of initiating events addressed is adequate and complete

The process to identify the initiating events have not been justified formally.

16. How does the process used to identify the set of initiating events relate to the objectives of the PSA and to the actual or (foreseeable) future use of the PSA?

The process used give us an understanding of the kind of accident that can be initiated during different stages of the low power and shutdown plant conditions. The understanding gained was part of the objective. The understanding will be and is been applied to support, in some extend, regulatory decisions.

THE NETHERLANDS

11. What process used to identify initiating events in the LPSD PSA?

12. How has this process differed from the one used for the full power PSA?

For power operations initiating events can generally be described as 1) an event resulting in a thermal-hydraulic plant perturbation that potentially challenges the thermal design criteria of the core, thus necessitating reactor runback, reactor shutdown, or the actuation of other standby safety systems; or 2) a perturbation resulting from the actuation of the reactor protection system due to spurious, manual, or other reactor trips not directly related to core protection.

For non-power conditions, an initiating event is defined as any event that requires an automatic or manual plant response in order to maintain critical safety functions. This definition differs from power operations. This is because during many of the non-power plant operational states (POSs) the reactor is already tripped and the plant conditions may differ significantly from the conditions when at power.

For the identification of initiating events a three-tier approach is used. The approach involves development of a master-logic diagram and a safety parameter review, completion of system load reviews for all important support systems, and a review of past probabilistic studies and documents and plant-specific data as well as the shutdown and startup procedures of the Borssele NPP.

The master logic diagram (MLD) is an analytical technique similar to fault tree analysis which can be used to identify initiating events For the non-power operation the MLD provides a starting point for the Systematic Safety Parameter Review. After reactor trip and reactor runback were modelled as the top events of the MLD, the following trip functions were modelled in the MLD.

- Low primary flow
- High neutron flux
- High primary pressure
- Low primary pressure
- High containment pressure
- Low pressuriser level and low primary pressure
- High pressuriser level
- Reactor flux/ Coolant temperature imbalance
- Low inlet temperature (overcooling events)
- Inadequate subcooling
- Low or High Steam Generator level
- High Main Steam Pressure

- Low Feedwater Flow and loss of Feedwater Tank
- Steam Generator Tube Rupture
- Manual Trip

In the following table (Table 1) an overview is given of the review of the Borssele NPP Safety parameters for non-power initiating event applicability.

For the assessment of the support system initiators first a candidate list of support systems was compiled. Each of these systems were reviewed to determine their ability to impact the critical safety functions during non-power operation

For the non-power initiating event identification process, the PSA/document review comprised a.o.: the industry categorization (EPRI NP-2230), NUREG/CR-3862, NUREG-1449, NUMARC-91-06, the NUREG/CR-4550 series, NUREG/CR-5015, NUREG/CR-1410 (Loss of AC power during midloop in Vogtle), The French LPSD-PSAs, as well as insights derived from LPSD-PSAs made by Scientech (Halliburton/NUS).

The historical record review has been conducted on three various types of records to identify potential plant-specific initiators. The records reviewed consisted of NPP Borssele Monthly reports, Shutdown Evaluation Reports (which summarize significant events during an outage) and Safety Event Review Group Evaluations (which describe events important to plant safety).

Table 1

Safety Parameter	Potential Cause	Applicable POS	Comments
Low Primary Flow -RCP trip -RHR trip -TG trip	pump failure loss of seal cooling loss of bearing cooling loss of motor cooling loss of power	RCP-trip: Low power POSs RHR: Cold shutdown, midloop and core load/unload TG: fuel pool and core load/unload	Treatment similar to power MLD for RCP trip during low power, otherwise screen out. Add loss of RHR and loss of TG for non-power.
High neutron flux	pump restart in an idle loop rod withdrawal rod ejection boron dilution	All POSs	Non-power: k/k period limit in intermediate flux channel included in T/TYD/TSA, other evaluate as reactivity addition accidents
High primary pressure	no single failure for Power MLD	All POSs	Non-power add cold overpressurization, LTOP
Low primary pressure	LOCA Main Steam Line Break SGTR	Steaming and cold shutdown POSs	Treatment similar to Power MLD
High Containment pressure	LOCA Loss of TL	Steaming POSs	Treatment similar to Power MLD (except

Safety Parameter	Potential Cause	Applicable POS	Comments
	MSLB inside containment Interfacing systems LOCA		loss of TL is not an IE)
Low pressuriser level & Low pressure	LOCA	Steaming, cold shutdown, midloop, and core load/unload	Treatment similar to Power MLD
Low fuel pool level	LOCA in fuel pool	Steaming, cold shutdown, midloop, and core load/unload	Consider for non-power
High pressuriser level	TA or other injection source failure	All POSs	Treatment similar to Power MLD when RPV closed, otherwise screen out
Reactor flux/ Coolant temperature imbalance	- non unique for power MLD or non-power	Steaming POSs	Treatment similar to power MLD
Low reactor inlet temperature/overcooling	MSLB Open SG relief valve EHU failure Feedwater control failure	Steaming POSs	Treatment similar to Power MLD
Inadequate subcooling/ pressuriser failure	pressuriser control failure LOCA Open SG relief valve	Steaming and cold shutdown POSs	Treatment similar to Power MLD
Low/High SG level/ Feedwater failures	Main Feedwater Line Break Feedwater control failure MSIV closure Low feedwater tank level Feedwater regulating valve failure SG water level instrumentation failure	Steaming POSs	Treatment similar to Power MLD
High Main Steamline pressure	MSIV closure EHU failure TUSA without bypass Loss of condenser vacuum	Steaming POSs	Treatment similar to Power MLD
Low feedwater flow/ Loss of feedwater tank	feedwater tank rupture RG001-S190 fails open RM pump failure	Steaming POSs	Treatment similar to Power MLD
SGTR/ N16 activity	SGTR Fuel element failure	All POSs	Treatment similar to Power MLD
Manual trip	Manual trip	Low power	Treatment similar to Power MLD

13. How do the set initiating events identified reflect the greater level of the activity on the plant (particularly during shutdown)?

Partly, this is done by the historical data review of the Borssele outages, and partly, by adjusting the power IE frequencies for non-power. An example is maintenance induced LOCAs.

14. Have you identified initiating events that are unique to LPSD conditions? If so, what are they?

A special initiating event, which became important for LPSD was PTS. A key parameter for the onset of PTS is the Reference Temperature for Nil Ductility Transition (RTNDT). This temperature for Borssele NPP is calculated to be no greater than 40 ^{o}C for the life of the plant, which is very low relative to other plants. Therefore, PTS was not an issue for the power PSA. PTS is, however, a concern during non-power operation since peak temperatures may be at or below 50 ^{o}C. This is addressed in a separate event tree developed for a spurious ECCS-injection signal from the reactor protection system during cold shutdown.

Maintenance induced LOCAs (KLOCA) were included in the small and very small LOCA calculations. This contribution was calculated assuming a Chi-square distribution with 0 events in 20 years. Maintenance induced LOCAs were postulated in the cold shutdown (RE, RL) and midloop (ME, ML) POSs.

Other IEs to mention are: Loss of RHR during LPSD POSs, Loss of spent fuel pool cooling, Loss of Midloop Level instr., Fire in several rooms containing safety related components due to maintenance, boron dilution via RHR, uncontrolled boron dilution from CVCS, dilution due to steam generator maintenance, etc. For the boron dilution accident scenarios, a document review had been conducted to identify potential boron dilution accident scenarios. This review identified 22 potential boron dilution/reactivity addition accidents based on historical events and theoretical situations. Slower developing events and sequences with mixing or smaller volumes of pure water had been qualitatively screened.

SPAIN

11. What process used to identify initiating events in the LPSD PSA?

The process used depends on the specific LP&SD PSA.

For Asco NPP: To identify Initiating Events there has been performed:

a) A qualitative analysis, screening the applicability of the initiating events groups identified in the at-power PRA, according to the operational state of the plant. As a result of this evaluation a list was obtained, in which each operational state is related to the possible initiators that may apply among those identified in the PRA at power.

b) An analysis of specific initiators at LP&SD, throughout:

   1) An analysis of events occurred in other PWR:

      – Selective query lists of NPE Data Base.

      – Selective query lists of NPDRS of INPO.

- Other plants operational experience analysis reports such as NSAC-183, NSAC-52, NSAC-156 and NUREG-1449.

- Analysis of other LP&SD PRA such as NSAC-84, NUREG/CR-5015 and Surry PRA.

2) An analysis of the Asco NPP operational experience throughout the study of all scheduled and non-scheduled shutdown reports occurred in Asco NPP as well as operative incidents reports and/or LERs. In the same way, initiators studied in FSAR are analyzed.

3) A qualitative analysis of Asco NPP specific initiators, based on design and procedures at LP&SD that may lead to a specific initiator. In that sense it was proceeded to:

   - Identify the limits of the systems that contain the coolant inventory, taking into account the possible changes of configuration and procedures from the entry to the exit of a plant operational state.

   - Identify the combinations of possible human errors and failures of required equipment in the operational states, taking into account particularly the manoeuvres, surveillance requirements, maintenance jobs, change of configuration and available procedures and usual practices.

For Santa Maria de Garonna NPP: The initiating events defined in the At Power PSA has been reviewed to define their applicability. Qualitative and quantitative analysis have been performed by experts to define other applicable initiating events.

12. How has this process differed from the one used for the full power PSA?

In full power PSAs, most of the identified IEs are generic and some jobs need to be developed in searching for those plant specific ones. In LP&SD PSA, even though some generic IEs are available, the main effort consists in identifying plant specific situations that can lead to an IE.

13. How do the set initiating events identified reflect the greater level of the activity on the plant (particularly during shutdown)?

As indicated in answer to Q11 for Asco plant, IE identified through analysis of the specific system limits and configuration changes and through the identification of potential human errors are representatives of the level of the plant activity. On the other hand, it can be seen in the Q14 list that some IEs correlate with the level of activity. Examples are the Special Events in Garonna NPP: fuel misloading, and control rod withdrawal among others.

14. Have you identified initiating events that are unique to LPSD conditions? If so, what are they?

A number of them have been identified. Plant by plant:

Asco NPP:

- Rupture or leakage in RHR lines.

- Rupture or leakage in RHR pump seals.

- Inadvertent opening of RHR safety valves.
- Flow transfer to RWST (Refuelling Water Storage Tank).
- Transfer to RWST from the refuelling cavity.
- Leakage on the temporary seal of the retracted thimbles (nuclear instrumentation seal).
- Refuelling cavity seal leakage.
- Nozzle-Dams rupture or leakage.
- Maintenance induced rupture or leakage in CVCS.
- Maintenance induced rupture or leakage in RHR.
- Inadvertent closure of RHR suction.
- Failures of the RHR train in service.
- RCS excessive drainage (during mid-loop operation).
- Gas entrance and/or vacuum generation.
- Total opening of the RHR discharge valves.
- Uncontrolled boron dilution through Make-up Water System.
- Uncontrolled boron dilution via thermal regeneration.
- Uncontrolled control rods extraction (from subcritical condition).
- Start up of an inactive RCP.
- Reactor core miss loading events.
- Loss of a Cooling Water System loop (RHR in service).
- Loss of a Service Water System loop (RHR in service).
- Loss of a 6,9 kV emergency bus (RHR in service).

Santa Maria de Garonna NPP:

a) Loss-of-coolant accident

LOCAs specific of LP&SD PRA

- Lines connected to reactor vessel
- Reactor head cooling system

- Main steam
- High pressure core injection
- Feedwater
- Reactor water clean-up
- Isolation condenser
- Control rods drive
- Standby liquid control
- Shutdown cooling
- RPV instrumentation
- Sample system
- Core spray
- Low pressure core injection
- Reactor vessel
- Fuel pool and lines connected to fuel pool
- Shutdown cooling
- Fuel pool cooling & clean-up
- Condensate distribution
- Low pressure core injection
- Refuelling cavity

b) Transients

Transients specific of LP&SD PSA

- Loss of support system events
- Loss of 4,16 kV bus bar
- Loss of 400 V bus bar
- Loss of 120 V a.c. control bus bar
- Vessel instrumentation system line break
- Loss of residual heat removal capability
- Loss of shutdown cooling system
- Loss of shutdown cooling system due to failure of Reactor building component cooling water system

c) Special events

- Control rod withdrawal
- Fuel bundle misloading
- Reactor vessel/cavity overfilling/overpressurisation
- Core instability

15. How have you justified that the process used to identify the set of initiating events addressed is adequate and complete

See answer to Q6.

16. How does the process used to identify the set of initiating events relate to the objectives of the PSA and to the actual or (foreseeable) future use of the PSA?

Completeness is one requirement for an LP&SD PSA to be used in applications. In this sense a sound search of all of the feasible Initiating Events is mandatory.

SWEDEN

11. What process used to identify initiating events in the LPSD PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q11	F1/2	Initiating events have been chosen from the Finnish TVO study, international experience and with consultation of staff in connection to the outage, shutdown and start up.
	R1	Guidelines and screening.
	R2-4	R2, Initiating events from full power PSA that could occur during LPSD + standard IE during an outage, based on EdF judgement, earlier experience from shutdown PSAs (own, French and international).
	B2	The findings from PSA Level 1 together with standard IE during an outage.
	O1	The following methods (sources of information) was covered: Full power PSA model, all initiating events covered for full power should be covered for the low power PSA. If some initiating events are excluded it should be motivated how come. Operating personnel may cause an initiating event (human failure). Critical actions must therefore be identified, categorised and quantified. Those actions that have been categorised as critical are those that may have an impact on: - Systems for reactor shut down - Systems for reactor vessel injection - Systems for residual heat removal An action may cause a reactor scram (SS) or just a degradation of one or more system used for the above systems. Survey of licensee event reports (LERs) for the last 10 yeas of operation in order to determine if any other than the above mentioned initiating events should be covered in the study. Initiating events depending on system configuration, i.e. if a stand-by system is taken into operation during the transition from one phase to another new initiating events is created: No start of required system in proceeding phase. Spurious stop of required system in next phase. Answer only valid for low power PSA, not outage period.
	O2	IEs from power operation was modified
	O3	IEs from power operation were modified
SKI REMARK	For low power the same initiating events + in some cases a few man made are used. For shutdown phase the full power statistics and interview with the staff are normally used. Some NPPs also uses HRA analysis to find the initiating events. Guidelines and different screening processes are used.	

12. How has this process differed from the one used for the full power PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q12	F1/2	Consultation with staff and observation the outage period.

	R1	No difference between low power and full power operations.
	R2-4	R2, IEs unique for an outage have been identified, EdF experience has been used, less generic data available
	B2	IEs unique for an outage have been identified.
	O1	It differs mainly in the way that the full power PSA doesn't take into account that an initiating event may be caused by operating personnel, which may be realistic assumption. Answer only valid for low power PSA, not outage period.
	O2	It has not
	O3	It has not
SKI REMARK	See Q11.	

13. How do the set initiating events identified reflect the greater level of the activity on the plant (particularly during shutdown)?

CSNI Question number	Answer from Swedish plant	Answer
Q13	F1/2	A rather high frequency of loss of RHR (7,8E-2)
	R1	Increased number of HRAs
	R2-4	R2, Shutdown specific IEs due to human error
	B2	This is a weak spot in the study.
	O1	By adding new initiating events if found or increasing initiating event frequencies if necessary. Answer only valid for low power PSA, not outage period.
	O2	During shutdown they are more to be regarded as shutdown (overhaul) operations than initiating events
	O3	During shutdown they are more to be regarded as shutdown (overhaul) operations than initiating events
SKI REMARK	Since most the LP and SD analyses have identified many manmade initiating events, they reflect more on these kinds of activities.	

14.Have you identified initiating events that are unique to LPSD conditions? If so, what are they?

CSNI Question number	Answer from Swedish plant	Answer
Q14	F1/2	Yes: Different kind of LOCA's and loss of RHR.
	R1	There are 10 unique conditions during low power operation and 12 conditions analysed during outage according to the guideline.
	R2-4	R2, Yes, see Q1
	B2	See the answer above.
	O1	Yes we have, mostly such that represent human failure and such related to change in system configuration, e.g. taking system in to (or out of) service and manoeuvring control rods are typical actions may cause an initiating event. Answer only valid for low power PSA, not outage period.
	O2	Yes,

	O3	Yes, loss of residual heat removal, 321.
SKI REMARK	Unique initiating events to LPSD are LOCAs during service on RCPB and auxiliary systems + loss of residual heat removal system.	

15.How have you justified that the process used to identify the set of initiating events addressed is adequate and complete

CSNI Question number	Answer from Swedish plant	Answer
Q15	F1/2	Both internal and external review of the LPSD PSA has been done. The review has pointed out weakness in the LPSD analysis in accordance to HRA part and identification of possible man made initiating events.
	R1	Not a big issue. In the future new initiating events could be addressed**.**
	R2-4	R2, The initiating event selection is based on best available knowledge for internal events. In the future, area events and further IEs might be addressed
	B2	No
	O1	This has been done in an iterative process between the personnel that conducts the PSA and personnel with great operating experience. Answer only valid for low power PSA, not outage period.
	O2	No comment
	O3	A reviewing process was carried out to ensure the accuracy of the identified events. The completeness of the set of identified events can never be verified.
SKI REMARK	SKI has yet only reviewed one domestic LPSD study, but this process and the results from the joint Nordic project will point out weaknesses in the identifying process.	

16. How does the process used to identify the set of initiating events relate to the objectives of the PSA and to the actual or (foreseeable) future use of the PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q16	F1/2	The objectives of the LPSD PSA are met considering the answer on question Q9.
	R1	It will relate to our goals for the present and future PSAs.
	R2-4	R2, It will relate to our goals for the present and future PSAs.
	B2	Currently there is a mismatch between PSA Level 1 and LPSD PSA
	O1	The initiating events are identified with purposes for the PSA in mind, see answer to question Q2. Answer only valid for low power PSA, not outage period.
	O2	The objective has been to make the IE set adequate for the purposes of the PSA
	O3	The objective has been to make the IE set adequate for the purposes of the PSA
SKI REMARK		See Q9 and Q14

UNITED KINGDOM

11. What process used to identify initiating events in the LPSD PSA?

Operational experience feedback review, including reviews of LERs, SERs SOERs etc. and reference to the EPRI report TR-113051

12. How has this process differed from the one used for the full power PSA?

It followed the same process as the full power PSA. However, fault initiating frequency apportionment was used to determine frequencies, rather than just being factored by the time at risk, e.g. some shutdown states were more likely to suffer some faults and the respective Initiating Fault Frequencies (IFFs) were therefore factored accordingly (Enhancement of IFFs taking account of e.g. activities, mode changes, greater maintenance etc.).

13. How do the set initiating events identified reflect the greater level of the activity on the plant (particularly during shutdown)?

The set initiating events reflect more the different operating regime, e.g. on RHRS, rather than the outage activities themselves.

14. Have you identified initiating events that are unique to LPSD conditions? If so, what are they?

Unique faults are cold overpressure faults, drain down events, core misload during low power physics tests, and boron dilution.

15. How have you justified that the process used to identify the set of initiating events addressed is adequate and complete?

By review and analysis, see Fault Schedule e.g. comparison with other fault schedules, review of Sizewell's unique design features, review of operating experience feedback (OEF), FMEA etc. (done for both power and shutdown)

16. How does the process used to identify the set of initiating events relate to the objectives of the PSA and to the actual or (foreseeable) future use of the PSA?

It is intricately linked and is one of the main influences. For Sizewell B, using the PSA for licensing effectively demanded that the LPSA be 'complete'. Using it as a 'Living' PSA also meant that changes to operating practices had to be taken into account, if it could affect initiating events.

UNITED STATES

BWR

11. What process used to identify initiating events in the LPSD PSA?

Using the definition of an initiating event, detailed searches were conducted to identify events that could result in the occurrence of an initiating event. The searches included:

- examining previously documented research work to identify initiating events,

- examining plant procedures to identify functions and/or equipment that if lost would result in an initiating event.

12. How has this process differed from the one used for the full power PSA?

Essentially, the process is the same as the one used for full power, i.e., a search is made for all possible ways to cause an initiating event occur.

13. How do the set initiating events identified reflect the greater level of the activity on the plant (particularly during shutdown)?

The list of initiating events includes loss of systems that would normally, for full power, not result in an initiating event, and includes initiating events resulting from human action that would/could not occur during power operation.

14. Have you identified initiating events that are unique to LPSD conditions? If so, what are they?

- Diversion to Suppression Pool via RHR
- LOCA in connected system (RHR)
- Test/Maintenance induced LOCA
- Isolation of operating shut down cooling (SDC) loop
- Isolation of reactor water cleanup (RWCU) as decay heat removal (DHR)
- Isolation of ADHRS only
- Isolation of SDC common suction line
- Isolation of common suction line for ADHRS
- Loss of operating RHR shutdown system
- Loss of RWCU as DHR
- Loss of ADHRS only
- Loss of SDC common suction line
- Loss of common suction line for ADHRS
- Refuelling accident (rod or fuel misposition)

15. How have you justified that the process used to identify the set of initiating events addressed is adequate and complete

The process was subjected to a high level review and the same people reviewed the subsequent list of initiating events.

16. How does the process used to identify the set of initiating events relate to the objectives of the PSA and to the actual or (foreseeable) future use of the PSA?

One objective of the analysis was to assess the risks of severe accidents initiated during plant operational states other than full power operations. To accomplish this objective, the complete set of initiating events for the state (POS 5) being evaluated must be identified. The actual use of the PRA involved the identification of the important initiating events and the results, both core damage frequency and various risk (consequence) measures, from the analysis.

As to the actual or (foreseeable) future use of the PRA, see answer to Q2b.

PWR

11. What process was used to identify initiating events in the LPSD PSA?

The approach followed in the Surry LP&S study includes review of existing shutdown PRA studies, review of procedures used during shutdown, review of initiators for power operations, review of NRC generic letters and information notices, review of studies of shutdown issues and scenarios, and review of US operating experience. The objective was to identify and understand all known initiating events and accident scenarios.

For plant operational states similar to that of full power operation, i.e., those POSs in which the normal mode of heat removal is through the steam generators, the initiating events of the full power PRA are used. The quantification of initiating events was done by collecting operational experience, i.e., LER searches, of the specific plant conditions for transients, and using subjective frequency reduction factors for LOCAs to account for lower operating pressures.

For plant operational states that use RHR as the normal means of decay heat removal, a Licensee Event Report (LER) search is the main method of identifying initiating events. The search covered a period of 10 calendar years. The identified events were then grouped based on their impacts on the plants, e.g. spurious safety feature actuation and loss of a bus. The Surry specific design was taken into consideration in screening of the events, i.e., some events that occurred at other plants were eliminated based on the differences in plant design. Events were eliminated if either the event could not occur at Surry, or it would not have caused an initiating event if it occurred.

12. How has this process differed from the one used for the full power PSA?

The general approach is the same as that of full power PSA. Due to the shutdown specific plant conditions, the initiating events are not the same as those of full power PSA.

13. How do the set initiating events identified reflect the greater level of the activity on the plant (particularly during shutdown)?

The activities during shutdown may be the causes of initiating events. The higher level of activity during shutdown may increase the likelihood of the initiating events.

14. Have you identified initiating events that are unique to LPSD conditions? If so, what are they?

Most of the initiating events at shutdown are unique to the shutdown conditions. Loss-of-RHR initiating events are defined as interruptions of decay heat removal. Loss of inventory due to pipe rupture or flow diversion is also dependent on the shutdown condition. The reactivity accidents identified are also based

on the shutdown conditions. Cold over pressurization and failure of weaknesses in the RCS pressure boundary are other examples of accidents unique to shutdown.

15. How have you justified that the process used to identify the set of initiating events addressed is adequate and complete

The process involved extensive reviews of sources of information for identification of initiating events, and included technical reviews.

16. How does the process used to identify the set of initiating events relate to the objectives of the PSA and to the actual or (foreseeable) future use of the PSA?

1) NUREG/CR-6144-The objective of the PSA is to assess the risk during shutdown. The process used is to identify the initiating events during shutdown.

2) Screening analysis of NUREG/CR-6144-Same as 1)

3) NUREG/CR-6616- The objective of NUREG/CR-6616 is to perform sensitivity calculations to compare the risks due to performance of preventive maintenance during shutdown vs. during full power. It was judged that the initiating events identified in 1) could be further grouped into only 5 initiating event groups.

4) For possible uses see Q2b.

B.2 Screening and Grouping of Initiating Events

PRELUDE AND QUESTIONS

The methods used for screening and grouping initiating events in full-power PSAs are also used in LPSD PSAs. And many initiating events analyzed at full power are also analyzed in LPSD PSAs. However, the adequacy of these methods and tools for LPSD conditions is being disputed.

Lack of understanding of the mechanisms (combinations of human and/or hardware failures) leading to initiating events during LPSD has been identified as a concern for LPSD risk analysis. For example, reactor coolant system (RCS) drain-down initiating events are included in LPSD risk analyses. However, many believe that we do not have an adequate understanding of the causes of these events, and thus, have no way to systematically estimate and, if necessary, reduce their frequency. Also, some are questioning if the activities taking place during LPSD operations (e.g., multiple and/or concurrent maintenance activities) can introduce new mechanisms for initiating events which are not well understood or modelled.

The criteria used in full power PSAs for screening and grouping of initiating events (in order to reduce them to a manageable set) are used in LPSD PSAs. However, some believe that the applicability of these criteria at LPSD conditions must be examined. The concern is that important LPSD initiating events could be eliminated or inappropriately grouped because of an incomplete understanding of an event or its consequence. Please respond to the following questions.

17. In each of the POSs modelled in the PSA, have you applied a screening process to eliminate from the analysis those initiating events that are assessed not to be significant to the risk? If so

a. how has this screening process been carried out?

b. How has this screening process differed from the one carried out for full power PSAs?

18. Have you grouped the initiating events identified for each of the POSs? If so

a. How has this grouping process been carried out?

b. How does it differ from "initiating event grouping" performed for a full power PSA?

19. What areas of your analysis are, in your opinion, most in need of improvement?

20. What are the areas that would most benefit from further research?

RESPONSES

BELGIUM

17. In each of the POSs modelled in the PSA, have you applied a screening process to eliminate from the analysis those initiating events that are assessed not to be significant to the risk? If so

No

18. Have you grouped the initiating events identified for each of the POSs? If so

No

19. What areas of your analysis are, in your opinion, most in need of improvement?

The areas remain to be determined. Points to examine are, for instance, to add other POSs (e.g. intermediate shutdown on SG) and to have better POS delimitation using criteria based on thermal-hydraulic analyses.

Germany

17. In each of the POSs modelled in the PSA, have you applied a screening process to eliminate from the analysis those initiating events that are assessed not to be significant to the risk? If so

a. how has this screening process been carried out?

A qualitative screening was performed with the consideration of:

- which IE are analysed in the PSA for power operation and have minor requirements on success criteria during LP&SD (e.g. transients during shutdown),
- which IE have a short grace time and hence a higher relevance compared to others (i.e. because of lower possibility for recovery actions) and in
- which POS occur the highest demands on the minimal success criteria compared to other POS for the same IE.

b. How has this screening process differed from the one carried out for full power PSAs?

The screening parameters are different.

18. Have you grouped the initiating events identified for each of the POSs? If so

a. How has this grouping process been carried out?

- The classical grouping has been used:
- Transients
- Loss of coolant accidents
- Deboration accidents
- Reactivity accidents

b. How does it differ from an "initiating event grouping" for full power PS As?

The grouping is identical.

19. What areas of your analysis are, in your opinion, most in need of improvement?

Since the PSA was focussed on the main contributors, i. e. on IEs in POSs during shutdown and the first mid-loop operation, an analysis of IEs in POS after refuelling would improve the PSA.

20. What are the areas that would most benefit from further research?

Concerning IEs further research on possibilities for human failures during maintenance which may lead to an IE might be most beneficial.

HUNGARY

17. In each of the POSs modelled in the PSA, have you applied a screening process to eliminate from the analysis those initiating events that are assessed not to be significant to the risk? If so

a. how has this screening process been carried out?

There could be two causes for elimination of an initiating event from the further analysis: (1) an initiating event identified in other PSA studies but found inapplicable to the Paks NPP and (2) an initiating event with very low likelihood (generally with frequency less than 1.0E-7/year). The first cause was identified during the expert evaluation of the applicability of the initiating events found in other PSA studies, while the second was identified during the quantification of the initiating event frequencies.

b. How has this screening process differed from the one carried out for full power PSAs?

There was not any difference.

18. Have you grouped the initiating events identified for each of the POSs? If so

a. How has this grouping process been carried out?

Initiating events were grouped by how they lead to core damage (direct, indirect causes) and what the characteristics of the physical processes were following their occurrence. Excess heat generation in the core, or loss of core cooling in consequence of an initiating event can lead to core damage. The latter can be a degradation of the heat removal or loss of coolant. Initiating events were grouped into three main groups based on their physical characteristics: initiating events causing loss of primary coolant (A-E), transient initiating events (F-J, M-N), special, common cause initiators (K-L). Grouping of initiating events was finalized partly by the use of the results of thermohydraulic calculations, and partly by a review of the scope of available mitigating systems POS by POS. Events leading to nearly the same consequences, and causing or requiring the operation of the same safety systems were grouped into the same group. Groups can change from POS to POS the most typical example of which is the grouping of initiating events with loss of primary coolant.

b. How does it differ from an "initiating event grouping" for full power PS As?

There was not any difference. *LOCA grouping?

19. What areas of your analysis are, in your opinion, most in need of improvement?

Determination of probability/frequency of specific erroneous human interactions (by plant staff or maintenance personnel) that can result in an initiating event.

20. What are the areas that would most benefit from further research?

The potential for human induced initiating events (see Q19).

JAPAN

17. In each of the POSs modelled in the PSA, have you applied a screening process to eliminate from the analysis those initiating events that are assessed not to be significant to the risk?

a. how has this screening process been carried out?

NUPEC screens the initiating events identified on the following viewpoints:

- The event should be screened out if the frequency due to it is estimated to be negligibly small (e.g. $< 10^{-6}$/ry)
- The event should be screened out if it is not relevant to core damage
- The event, where the plant response is similar to other events, can be grouped and only the representative event is considered

b. How has this screening process differed from the one carried out for full power PSAs?

The screening process shown in A20 is the same as full power PSAs.

18. Have you grouped the initiating events identified for each of the POSs?

a. How has this grouping process been carried out?

NUPEC groups the initiating events on the viewpoint of the similarity on their plant response based on the plant configurations.

b. How does it differ from an "initiating event grouping" for full power PS As?

The procedure for "initiating event grouping" in LPSD PSA is the same as that of Full Power PSA.

19. What areas of your analysis are, in your opinion, most in need of improvement?

1) Inclusion of RIA due to spurious opening of isolation valves from water by human errors and so on

2) Inclusion of the effect of external events, such as fire, flood and seismic, to identification of initiating events

3) Addressing LOCA scenario, such as possibility of pipe break LOCA and their sizes

4) Addressing the credibility of gravity injection and / or re-flux cooling during mid-loop operation based on the thermal hydraulic analyses.

20. What are the areas that would most benefit from further research?

See response to Q19 (A22)

KOREA

17. In each of the POSs modelled in the PSA, have you applied a screening process to eliminate from the analysis those initiating events that are assessed not to be significant to the risk? If so

a. how has this screening process been carried out?

At first, the grouping approach was performed using initiating events identified in the pre-steps. This grouping approach gives benefit to reduce the number of event trees and accident sequences which have to be developed. This grouping approach is an iterative process with the development of the event trees and the plant models. In this process, clearly defined initiating events which can not occur during shutdown modes were screened. The anticipated transient without scram (ATWS), containment integrity, heavy loads and fuel handling, mode change events, etc. were screened out in this stage.

b. How has this screening process differed from the one carried out for full power PSAs?

Some remarkable factors, which are not considered in full power PSA, such as pressure and operation mode characteristics, are important for this screening process.

18. Have you grouped the initiating events identified for each of the POSs? If so

a. How has this grouping process been carried out?

After making a list, the iterative process of review and evaluation was performed to reduce the number of initiators which would be into the final set of initiating events for event tree analysis. First, the events were reviewed to identify events which could be excluded because they were outside of the scope of analysis or were precluded because of improved system design features and to determine if any events

needed to be added because of unique design features. Events were considered to be outside the scope of analysis if they would be handled by full power operation mode.

b. How does it differ from an "initiating event grouping" for full power PS As?

Even if the event can occur during shutdown modes, the event must be defined its applicable POSs. Since the RCS conditions such as temperature and pressure and plant configurations are different in each POS, identifying the applicable POSs for each initiating event is very important. This is the characteristics of shutdown modes. For some cases, the identified initiating events are not PSA scope. Also, for some cases, identified events are not an initiating event. These were screened out from the final initiating events set.

19. What areas of your analysis are, in your opinion, most in need of improvement?

None.

20. What are the areas that would most benefit from further research?

None.

MEXICO

17 In each of the POSs modelled in the PSA, have you applied a screening process to eliminate from the analysis those initiating events that are assessed not to be significant to the risk? If so

a. how has this screening process been carried out?

No screening process based on their significance to the risk (frequency) has been implemented. Applicability at the plant characteristics drives the screening process.

b. How has this screening process differed from the one carried out for full power PSAs?

See the above answer.

18. Have you grouped the initiating events identified for each of the POSs? If so

a. How has this grouping process been carried out?

As for the full power case, the criteria used to group initiating events are based on plant response similarities.

b. How does it differ from an "initiating event grouping" for full power PS As?

The major difference is that some groups can not be retained, because the plant response is different, when the analysis is for coming down from power or in going back up to power, for instance changes in system availability due to technical specification requirements.

19. What areas of your analysis are, in your opinion, most in need of improvement?

Exhaustive search of initiating events (internal and external operational experience, as well as conduct FMEA) are most in need of improvement.

20. What are the areas that would most benefit from further research?

Selection of initiating events, reducing the lack of understanding of combinations of human and/or hardware failures capable to lead to an initiating event would benefit from further research.

THE NETHERLANDS

17. In each of the POSs modelled in the PSA, have you applied a screening process to eliminate from the analysis those initiating events that are assessed not to be significant to the risk? If so

a. how has this screening process been carried out?

b. How has this screening process differed from the one carried out for full power PSAs?

18. Have you grouped the initiating events identified for each of the POSs? If so

a. How has this grouping process been carried out?

b. How does it differ from an "initiating event grouping" for full power PS As?

19. What areas of your analysis are, in your opinion, most in need of improvement?

20. What are the areas that would most benefit from further research?

The grouping of the initiating events was based on mitigation requirements and equipment availability conditions such as the following:

- Mitigating functions and systems
- Timing for systems and operator actions
- Unavailability of mitigating equipment due to the initiator
- Misleading information of indicators
- Success criteria

The grouping depends on the analysis of thermal hydraulic calculations, the evaluation of system success criteria, and the timing considerations that are developed in the accident delineation task. The grouping is performed specifically for NPP Borssele.

The next table shows the various initiating event categories.

Transient initiators

- TRL Loss of Main Feedwater supply
- TRL Loss of Main Heat Sink (Turbine bypass and condenser)
- TRL Loss of MFW an MHS

- T T/H-Flux perturbations requiring scram, with MFW and MHS available
- TYD RCP trip, T/H-Flux perturbations requiring reactor runback, with MFW and MHS available
- TOP Scram or Manual Shutdown as Initiating event
- LBS SGTR
- LBM Multiple SGTR
- TR1 MSLB or MFLB inside containment
- TR2 MSLB or MFLB in Ring Room
- TR3 MSLB or MFLB outside containment
- TTJ Loss of RHR
- TTG Loss of Fuel Pool Cooling

Support system initiators

- CAD Loss of Offsite Power
- CTF Loss of system TF (Closed cooling water)
- CVF Loss of system VF (Main cooling water)
- CBA Loss of 6 kV ac bus BA
- CBB Loss of 6 kV ac bus BB
- CBU Loss of 6 kV ac emergency bus BU
- CBV Loss of 5 kV ac emergency bus BV
- CRL Loss of Feedwater tank inventory

LOCA initiators

- LL Large LOCA
- LM Medium LOCA
- LMS Large Small LOCA
- LS Small LOCA
- LV Very Small LOCA

- ISL Interfacing Systems LOCA

Special Events modelled in event tree but not treated as IE

- TZA ATWS wit MFW/ Reactivity addition
- TZR ATWS without MFW
- CUV Station blackout without LOCA
- LUV Station blackout with LOCA

Area and External Events

- FDC Fire/Flood induced loss of 24/220 DC batteries/buses
- FSA Turbine missile strikes building 45
- FYA Fire/flood fails primary instr. Room
- E-P External Events which fail main systems
  - Airplane crash
  - Dike failure
  - External flood
- EAD Cargo ship LPG flash fire
- ECD External event leading directly to core damage
  - Airplane crash
  - Tornado overpressure > .3 bar
  - External flood > 6.7 meter
  - Cargo ship LPG explosion, overpress. > .3 bar
- EHI External events disabling operators
  - Toxic gas clouds from industry/ rail and shipping transport
- EM1 Turbine missile impact containment from adjacent coal fired plant
- EPO External events which fail main systems
  - Airplane crash
  - Tornado pressure .1 bar< overpressure < .3 bar

- External flood 5.7 m < depth < 6.7 m
- Cargo ship LPG explosion overpress. .1 - .3 bar

- EVF External flood 4.1m < depth < 5.7m

SPAIN

17. In each of the POSs modelled in the PSA, have you applied a screening process to eliminate from the analysis those initiating events that are assessed not to be significant to the risk? If so

a. how has this screening process been carried out?

A feasibility matrix has been developed where IE are crossed versus POS to determine if a given IE is possible to happen or not at each POS. No other reduction or screening was applied.

b. How has this screening process differed from the one carried out for full power PSAs?

In the full power PSA, only the IEs that are not applicable to a specific plant are removed from the analysis. There is no other screening process.

18. Have you grouped the initiating events identified for each of the POSs? If so

a. How has this grouping process been carried out?

At **Asco NPP**, after defining the IEs, the resulting I.E.s are grouped into 14 groups that can be summarized in LOCAs, loss of RHR and transients. For LOCAs and loss of RHR the groups are generated according to the mitigation systems required. For the transients case, the dependency between initiating event and available systems is taken into account. At **Garonna NPP,** the large number of IE identified is collapsed in 27 groups of initiating events.

b. How does it differ from an "initiating event grouping" for full power PS As?

The applied method is similar to that of full power PSA: to group IEs according to the required plant response.

19. What areas of your analysis are, in your opinion, most in need of improvement?

See answer to question Q68.

20. What are the areas that would most benefit from further research?

See answer to question Q69.

SWEDEN

17. In each of the POSs modelled in the PSA, have you applied a screening process to eliminate from the analysis those initiating events that are assessed not to be significant to the risk? If so

a. how has this screening process been carried out?

b. How has this screening process differed from the one carried out for full power PSAs?

CSNI Question number	Answer from Swedish plant	Answer
Q17		See answer Q11
	F1/2	Initiating events have been chosen from the Finnish TVO study, international experience and with consultation of staff in connection to the outage, shutdown and start up.
	R1	By an engineering team.
	R2-4	R2, In 2001 update, EdF engineering judgement+results from 1992 study were used to screen out certain initiating events for some POS.
	B2	Through primitive engineering judgement
	O1	When an initiating event that has been included in the full power PSA is excluded from the low power PSA this is verified by reviewing the combination of initiating event frequency, system barriers and the contribution of that particular initiating event to the core damage frequency in the full power PSA. All initiating events that are not relevant are also excluded, for example the initiating event "loss of main condenser" is not considered as soon as the main condenser has been taken out of service. Answer only valid for low power PSA, not outage period.
	O2	Common sense and engineering judgement
	O3	Common sense and engineering judgement
SKI REMARK	The plant specific studies have put a lot of effort on discussing the actual process and the connected risks with the service personnel at the plant.	
Q17 b	F1/2	The visual observations of the outage, shutdown, start up and the following consultations of staff.
	R1	No big difference but more of HRA.
	R2-4	R2, Similar to full power PSA, more HRA
	B2	Due to the mismatch mentioned above it is not useful to elaborate on this
	O1	See answer to question Q11 and Q12. Answer only valid for low power PSA, not outage period
	O2	It has not been so dependent on international guides and standards
	O3	It has not been as dependent on international guides and standards as the full power PSA.
SKI REMARK	See Q17	

18. Have you grouped the initiating events identified for each of the POSs? If so

a. How has this grouping process been carried out?

b. How does it differ from an "initiating event grouping" for full power PS As?

CSNI Question number	Answer from Swedish plant	Answer
Q18	F1/2	They are grouped in the three categories a) over reactor water level and b) under reactor water level c) loss of RHR.
	R1	With HRA.

	R2-4	R2, No answer
	B2	No answer
	O1	No particular grouping has been made especially for the low power PSA, the same grouping as in full power PSA has been used (CCI, Transients, Pipe break, …). Answer only valid for low power PSA, not outage period.
	O2	In a similar way as for the full-power PSA
	O3	The grouping process was based on safety-systems requirements.
SKI REMARK	For LP the initiating events are grouped in the same way as for full power operation PSAs. For SD the initiating events in several cases have been grouped in LOCAs over and under the core + loss of RHR.	
Q18 b	F1/2	No splitting in these categories in full Power PSA. More detailed splitting for power operation PSA.
	R1	No differences.
	R2-4	R2, No answer
	B2	No answer
	O1	It doesn't differ at all. Those initiating event that are unique for the low power PSA can be included in the same "grouping" as those used for full power PSA. Answer only valid for low power PSA, not outage period.
	O2	Not much
	O3	Since the demands on the safety systems are different during LPSD than at full power, there are differences, though minor, in how initiating events are grouped. One example is the treatment of different types of LOCA's, which during LPSD are treated only as a group and not individually, or according to size.
SKI REMARK	See Q18.	

19. What areas of your analysis are, in your opinion, most in need of improvement?

CSNI Question number	Answer from Swedish plant	Answer
Q19	F1/2	Identification of "man made" initiating events. More detailed identification of IE from a general point of view (e.g. CCI) and also modelling of considered system functions with full dependencies (e.g. I&C electrical and area dependencies)
	R1	No immediate improvements needed.
	R2-4	R2, No answer
	B2	Can't tell
	O1	See answer to question Q9. Answer only valid for low power PSA, not outage period.
	O2	LOCA frequencies
	O3	LOCA frequencies
SKI REMARK	See Q9	

20. What are the areas that would most benefit from further research?

CSNI Question number	Answer from Swedish plant	Answer
Q20	F1/2	Identification of "man made" initiating events (research?)
	R1	No specific area identified today.
	R2-4	R2, No specific area identified today.
	B2	What is most needed is an attempt to apply the methodology used in Finland and Sweden.
	O1	See answer to question Q9. Answer only valid for low power PSA, not outage period.
	O2	LOCA frequencies
	O3	LOCA frequencies
SKI REMARK	See Q9	

UNITED KINGDOM

17. In each of the POSs modelled in the PSA, have you applied a screening process to eliminate from the analysis those initiating events that are assessed not to be significant to the risk? If so

a. how has this screening process been carried out?

b. How has this screening process differed from the one carried out for full power PSAs?

a) The screening process looked at the classes of faults and determined whether any of the 'states' were bounding, i.e.

- Can the fault occur?
- Is it of concern?
- Does it need individual consideration?
- Can it be bounded by a fault in a different POS?

b) The main difference is that more POSs need to be considered for LPSD. The fault analyses for each POS were reviewed to determine whether the analysis was 'explicit' or 'bounding' and whether the 'bounding' was still valid. Additional supporting arguments and analyses were therefore identified.

18. Have you grouped the initiating events identified for each of the POSs? If so

a. How has this grouping process been carried out?

The grouping is based on having similar safeguards requirements.

b. How does it differ from an "initiating event grouping" for full power PS As?

The process used was effectively the same.

19. What areas of your analysis are, in your opinion, most in need of improvement?

None.

20. What are the areas that would most benefit from further research?

None

UNITED STATES

BWR

17. In each of the POSs modelled in the PSA, have you applied a screening process to eliminate from the analysis those initiating events that are assessed not to be significant to the risk? If so

a. how has this screening process been carried out?

The screening process is dependent upon the initiating event. If the frequency can be shown to be below some specified criterion, then the event can be removed from consideration. If the event is not below the specified criterion or if other considerations are necessary (e.g., containment bypass), then other supporting arguments are required before the event can be eliminated (i.e., screened).

Some initiating events were screened from analysis due to the complex nature of the event and the lack of adequate methods to identify the events (e.g., some types of human induced initiators).

b. How has this screening process differed from the one carried out for full power PSAs?

The process is essentially the same.

18. Have you grouped the initiating events identified for each of the POSs? If so

a. How has this grouping process been carried out?

The process grouped or combined events that have the same or approximately the same effects on responses by various systems. Final responses were based on the most limiting set or responses.

b. How does it differ from an "initiating event grouping" for full power PS As?

The process is essentially the same.

19. What areas of your analysis are, in your opinion, most in need of improvement?

Identification of events involving combinations of human errors, or combinations of human errors and other failure mechanisms need improvement.

20. What are the areas that would most benefit from further research?

See response to Q19.

PWR

17. In each of the POSs modelled in the PSA, have you applied a screening process to eliminate from the analysis those initiating events that are assessed not to be significant to the risk? If so

a. how has this screening process been carried out?

Some potential initiating events were screened based on the plant specific design. For example, spurious closure of the valves is not considered an initiating event if there is no auto-closure interlock. Some initiating events were screened based on their likelihood or ease of mitigation. For example, it may be shown that charging pumps can operate an hour without cooling before exceeding bearing temperature limit, and cross tie to the other unit's charging system is possible. Therefore, loss of charging pump cooling system was screened.

b. How has this screening process differed from the one carried out for full power PSAs?

In some full power PSAs, a quantitative screening criterion is used on initiating event frequency. In this LPSD study, no quantitative criterion was used.

18. Have you grouped the initiating events identified for each of the POSs? If so

a. How has this grouping process been carried out?

The general rule for grouping initiating events is that the initiating events in the same group should have approximately the same impacts on the plant such that: (1) the same event tree can be used to model the mitigation functions that are challenged and (2) the frequencies of the initiating events in the same group can be summed to calculate the frequency of the initiating event group.

In NUREG/CR-6616 and 5718, all loss of RHR initiating events were grouped into a single group, in order to reduce the number of event trees that have to be developed.

b. How does it differ from an "initiating event grouping" for full power PS As?

There is no difference with the exception that NUREG/CR-6616 grouped initiating events into only 5 groups.

19. What areas of your analysis are, in your opinion, most in need of improvement?

Frequency of LOCAs due to pipe ruptures and flow diversions in all LPSD conditions should be further evaluated. The reduction factors for pipe rupture appeared somewhat arbitrary and should be further evaluated. Finally, methods for identifying flow diversion paths (eg, the Wolf Creek event) should be developed.

20. What are the areas that would most benefit from further research?

See response to Q19.

APPENDIX C - RISK METRICS

PRELUDE AND QUESTIONS

The definition of "core damage" for full power or LPSD PSAs is not consistent or uniform. Different analysts use different core performance parameters (temperature, pressure, coolant volume) to define "core damage." Varying definitions of "core damage" limits the usefulness of PSA result for supporting decision-making and tradeoffs.

Core damage frequency (CDF) is estimated typically on a per year basis (average CDF). However, the use of PSA in many applications (e.g., configuration risk management) has led to the use of CDF estimations on a per outage, month, day, and hour basis. The different types of CDFs provide different risk perspectives. A per hour (or per day) CDF provides a perspective of the CDF variability within a POS or from one POS (or configuration) to the next. Studies have shown that the hourly LPSD CDF associated with some POSs (e.g., hot shutdown) can be several magnitudes higher than the average (per year) total CDF. A per outage CDF provides an overall perspective of the risk during that outage and is used for comparing the shutdown CDF with the total plant CDF. Furthermore, other risk metrics (e.g., time to boil) have been developed and are used to evaluate shutdown risk. The estimation or use of the different LPSD CDF metrics is not consistent or uniform. For example, what are the data needs for estimating the different types of CDF metrics? Which metrics, if any, can be used to compare risks, or to produce a total plant risk? Which metric provides a basis for decision-making for future plant performance?

Establishing a radioactive release risk metric specifically for LPSD PSAs is also an issue discussed among PSA users. Currently, large early release frequency (LERF) and late release measures developed for evaluating full-power conditions are also used in LPSD conditions. Although LERF is considered by most PRA analysts to be the most important (for Level 2 analysis) risk measure for accidents at full power, many believe that the accident phenomenology and source terms involved in LPSD are different from those at full power and that LERF may over (or under) estimate the risk associated with LPSD conditions. However, no other measure has been generally agreed upon as being more appropriate. Please respond to the following questions.

21. What risk metrics were used for the Level 1 LPSD PSA? [This would typically include core damage frequency but may also include others such as LERF and core boiling frequency.]

22. Were risk metrics other than CDF used? If yes, what insights have you gained regarding their usefulness in the decision-making process?

23. Where CDF has been used as a risk metric, how has "core damage" been defined?

24. How does the definition of CDF for LPSD PSAs compare with the definition of "core damage" used in the full power PSA?

25. What is the basis for using the same or different core damage definitions in the full power and LPSD PSAs?

26. What CDF definitions have you or are you using in the different applications of your LPSD PSA? Why have they been chosen?

RESPONSES

BELGIUM

21. What risk metrics were used for the Level 1 LPSD PSA? [This would typically include core damage frequency but may also include others such as LERF and core boiling frequency.

CDF only was used.

22. Were risk metrics other than CDF used? If yes, what insights have you gained regarding their usefulness in the decision-making process?

No

23. Where CDF has been used as a risk metric, how has "core damage" been defined?

Core damage is defined as "the average clad temperature exceeds 1000°C for a minimum of 20% of the total cladding mass in the core".

24. How does the definition of CDF for LPSD PSAs compare with the definition of "core damage" used in the full power PSA?

The definition of CDF for the LPSD is core uncovery.

25. What is the basis for using the same or different core damage definitions in the full power and LPSD PSAs?

The definition of CDF for the LPSD is less specific, since no thermal-hydraulic analysis is available for the shutdown modes.

26. What CDF definitions have you or are you using in the different applications of your LPSD PSA? Why have they been chosen?

None.

GERMANY

21. What risk metrics were used for the Level 1 LPSD PSA? [This would typically include core damage frequency but may also include others such as LERF and core boiling frequency.

Hazard state frequencies due to uncovered core.

Hazard state frequencies due to accumulation of low borated coolant in the primary circuit because of internal boron dilution or faulty injection of unborated coolant.

Hazard state frequencies due to uncovered fuel in the fuel storage pond.

22. Were risk metrics other than CDF used? If yes, what insights have you gained regarding their usefulness in the decision-making process?

No other metrics as mentioned in Q21 have been used.

23. Where CDF has been used as a risk metric, how has "core damage" been defined?

The hazard state frequency due to uncovered core is defined as the unavailability of the system functions for core cooling without consideration of repair and accident management measures.

The hazard state frequency due to accumulation of low borated coolant is defined as the unavailability of the system functions to prevent the injection of unborated coolant or to prevent the accumulation of low borated coolant as a result of the loss of the decay heat removal system.

The hazard state frequency due to uncovered fuel in the fuel storage pond is defined as the unavailability of the system functions for fuel storage pond cooling without consideration of repair and accident management measures.

All hazard states have been defined as per year.

24. How does the definition of CDF for LPSD PSAs compare with the definition of "core damage" used in the full power PSA?

Concerning the hazard states, the definition is identical.

25. What is the basis for using the same or different core damage definitions in the full power and LPSD PSAs?

The same physical behaviour of the primary circuit / core if the cooling is lost.

26. What CDF definitions have you or are you using in the different applications of your LPSD PSA? Why have they been chosen?

No other definitions have been used.

HUNGARY

21. What risk metrics were used for the Level 1 LPSD PSA? [This would typically include core damage frequency but may also include others such as LERF and core boiling frequency.

For the different POSs both the CDF and the core damage probability (CDP) were calculated. The first is useful for drawing up the risk profile of a refuelling shutdown, while the latter makes it possible to sum the core damage risk originating from different plant operational states. The summation of CDPs over the POSs gives an annual probability of core damage originating from LPSD states. In addition to CDP and CDF boiling probability/frequency was also quantified.

22. Were risk metrics other than CDF used? If yes, what insights have you gained regarding their usefulness in the decision-making process?

In addition to the core damage probability/frequency, boiling probability/frequency was also quantified within the LPSD PSA due to the potential onsite impact. Up to now it has not been used in the decision-making process, but upon completing the level 2 (and level 3) studies it may become more useful.

23. Where CDF has been used as a risk metric, how has "core damage" been defined?

The success criteria to avoid core damage were considered to be the same as in the PSA for nominal power operational mode, namely:

a) Emergency core cooling criteria are met:

   - the maximum fuel rod cladding temperature does not exceed 1200 ^{o}C

   - the total oxidation of the cladding does not exceed 17 % of the total cladding thickness before oxidation

   - the total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1 % of the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react

   - changes in the core geometry are such that the core remains amenable to cooling

b) The radially averaged fuel enthalpy shall not exceed 963 J/gUO2 (230 cal/g) at any axial location in any fuel rod. The reference temperature for enthalpy is 298 K.

24. How does the definition of CDF for LPSD PSAs compare with the definition of "core damage" used in the full power PSA?

As concerns the definition of "core damage" they are the same.

25. What is the basis for using the same or different core damage definitions in the full power and LPSD PSAs?

The same definitions are used in order to be able to compare and sum up the resulting risks.

26. What CDF definitions have you or are you using in the different applications of your LPSD PSA? Why have they been chosen?

In our view the definition of "core damage" should not change throughout the different applications.

JAPAN

21. What risk metrics were used for the Level 1 LPSD PSA? [This would typically include core damage frequency but may also include others such as LERF and core boiling frequency.]

NUPEC uses CDF as the risk metric. Since Level-2 LPSD PSA is under-going at NUPEC, LERF will be another risk metric in the near future.

22. Were risk metrics other than CDF used? If yes, what insights have you gained regarding their usefulness in the decision-making process?

Currently, NUPEC uses only CDF as risk metrics. In the near future, LERF, which can provide the information on the effect on public. Both CDF and LERF are useful for decision-making in terms of the experiences on the application of full power PSA and LPSD PSA.

23. Where CDF has been used as a risk metric, how has "core damage" been defined?

NUPEC basically defines core damage (CD) by fuel cladding temperature (1200 deg-C). However, in specific POSs such as mid-loop operation, NUPEC defines CD as the uncovering the TAF (i.e., the top of active fuel). On the other hand, for reactivity initiation accident, CD is defined when recriticality is attained due to RCS boron dilution.

24. How does the definition of CDF for LPSD PSAs compare with the definition of "core damage" used in the full power PSA?

In full power PSA, the definition of core damage is the temperature of the fuel-clad surface increasing to 1200deg-C in any events. The definition of core damage for LPSD PSA is shown in A26.

25. What is the basis for using the same or different core damage definitions in the full power and LPSD PSAs?

The cladding temperature of 1200 deg-C is defined in Japanese safety evaluation guideline for LWRs as the criterion of peak cladding temperature during events. The criterion of uncovering top of active fuel is based on that fuel -cladding temperature can reach to 1200 deg-C in some time. This criterion may be a little bit conservative.

26. What CDF definitions have you or are you using in the different applications of your LPSD PSA? Why have they been chosen?

Even in the application of LPSD PSA, NUPEC uses the common criteria for core damage. LERF can be applied to RIR area as well as PSA area in the near future.

Two additional questions / responses were included:

What areas of your analysis are, in your opinion, most in need of improvement?

The criterion of uncovering top of active fuel is used due to the limitation of analytical codes. Therefore, the improvement / development of analytical codes for thermal-hydraulic behaviour during LPSD, especially mid-loop operation of PWRs, is important in order to set the better success criteria.

What are the areas that would most benefit from further research?

See above.

KOREA

21. What risk metrics were used for the Level 1 LPSD PSA? [This would typically include core damage frequency but may also include others such as LERF and core boiling frequency.

The CDF was used.

22. Were risk metrics other than CDF used? If yes, what insights have you gained regarding their usefulness in the decision-making process?

No.

23. Where CDF has been used as a risk metric, how has "core damage" been defined?

A stable endpoint of the event trees is the establishment of a decay heat removal path or a feed and bleed path before core damage. The core damage was assumed if the collapsed water level fell into the core mid-plane. The terms "core damage" and "core uncovery" are used interchangeably in this report.

24. How does the definition of CDF for LPSD PSAs compare with the definition of "core damage" used in the full power PSA?

The definition of core damage for LPSD PSA is more or less different than that of the full power PSA. In the full power PSA, core damage was assumed if the collapsed water level fell into the fuel assembly top-plane.

25. What is the basis for using the same or different core damage definitions in the full power and LPSD PSAs?

The difference in the LPSD core damage definition comes from the decrease of core decay heat level. Some realistic assumptions were used.

26. What CDF definitions have you or are you using in the different applications of your LPSD PSA? Why have they been chosen?

The occurrence rate per year is normally used. In some case, the occurrence rate per hour is used to review the impact of overhaul (maintenance) periods for each POS.

MEXICO

21. What risk metrics were used for the Level 1 LPSD PSA? [This would typically include core damage frequency but may also include others such as LERF and core boiling frequency.

CDF was used.

22 Were risk metrics other than CDF used? If yes, what insights have you gained regarding their usefulness in the decision-making process?

No other risk metrics are used

23. Where CDF has been used as a risk metric, how has "core damage" been defined?

In the POS identified, the control rods were fully or partial inserted. Loss of reactor coolant due to transients or LOCA without reposition leads to uncovery of the reactor core from different levels of power. Therefore, the main difference respect to full power is the level of power, which means difference in the time to reach the same condition expressed to core damage. Based on it, the definition is quite similar as that expressed for full power.

24. How does the definition of CDF for LPSD PSAs compare with the definition of "core damage" used in the full power PSA?

The definition of CDF for LPSD PSA is quite similar to the one used for full power PSA

25. What is the basis for using the same or different core damage definitions in the full power and LPSD PSAs?

See answer to question 23

26. What CDF definitions have you or are you using in the different applications of your LPSD PSA? Why have they been chosen?

See answer to question 23

THE NETHERLANDS

21. What risk metrics were used for the Level 1 LPSD PSA? [This would typically include core damage frequency but may also include others such as LERF and core boiling frequency.]

22. Were risk metrics other than CDF used? If yes, what insights have you gained regarding their usefulness in the decision-making process?

23. Where CDF has been used as a risk metric, how has "core damage" been defined?

24. How does the definition of CDF for LPSD PSAs compare with the definition of "core damage" used in the full power PSA?

25. What is the basis for using the same or different core damage definitions in the full power and LPSD PSAs?

26. What CDF definitions have you or are you using in the different applications of your LPSD PSA? Why have they been chosen?

The Borssele PSA is a full scope level-3 PSA where for level-1 CDF and ΔCDF due to planned and unplanned outages and/or events (outcomes risk monitor) are calculated. Also for each POS both the frequencies [year $^{-}$1] and contribution [%] to the Total Core Damage Frequency (TCDF) of the various accident sequences from the various defined IEs are calculated. Also the importance measures (Risk

Reduction, Risk Achievement and Fussel-Vesely) are calculated and used. There are no differences in risk measures as calculated in the LPSD POSs and the full power POS.

In case of fuel (un)loading and when all the fuel is in the fuel pool the definition of core damage is extended to widespread fuel damage inside fuel pool.

A reason for having a unified definition of risk metrics in the different POSs lies in the fact that risk informed decision-making has to be done on a risk picture that is integrated and as complete as possible and thereby making issues comparable. Issues like, assessing the risk impact of shifting maintenance activities from the refuelling outage to power operation, in order to shorten the refuelling outage and thereby improving the overall plant availability, is far easier if all the definitions and results are similar for all the POSs.

For the identification of core damage, the following conditions are accepted as successful prevention of core damage. For the core damage analysis, exceedance or violation of the conditions defined below defines the onset of core damage.

- The calculated peak fuel rod clad temperature is below 1500 K
- The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of zircaloy in the reactor.
- The clad temperature transient terminates at a time when the core geometry is still amenable to cooling, and the localized cladding oxidation limit of 17% is not exceeded during or after quenching.
- The core remains amenable to cooling during and after a break in the primary.
- The core temperature reduces and long term decay heat removal is established for an extended period of time.

As described above, violation of the conditions define the start of core damage. However, these acceptance criteria are not meant to be rigidly enforced at the expense of a realistic plant risk analysis. If the cause of violation of these conditions is arrested and the core damage that results is limited to a small localized area; then those scenarios are accepted and are not included as contributors to core melt in the PSA. However, if the cause of the conditions persists and leads to widespread core damage, than those scenarios are classified as core melt and included in the PSA. The localized core damage scenarios are judged to be insignificant in terms of risk when compared with widespread, core melt scenarios.

In performing the core degradation analysis for the containment evaluation, some sequences are identified in which core damage occurs, but total core damage is arrested prior to vessel melt through, thus limiting the release of fission products. In these cases, the frequencies of fission product release of various magnitudes are realistically assessed, and not based simply on the frequency of the onset of core damage.

SPAIN

21. What risk metrics were used for the Level 1 LPSD PSA? [This would typically include core damage frequency but may also include others such as LERF and core boiling frequency.

CDF was used.

22. Were risk metrics other than CDF used? If yes, what insights have you gained regarding their usefulness in the decision-making process?

Not applicable.

23. Where CDF has been used as a risk metric, how has "core damage" been defined?

In principle, Core Damage has been defined as in full power PSA related to cladding temperature. However, in some cases, depending on available calculations, a surrogate definition has been "core uncovery"

24. How does the definition of CDF for LPSD PSAs compare with the definition of "core damage" used in the full power PSA?

Although the general definition is the same, the core uncovery criteria is more conservative.

25. What is the basis for using the same or different core damage definitions in the full power and LPSD PSAs?

Both studies become more comparable if the same metrics are used. Use of, for instance, time to boil could be useful in providing risk insights, but it is not enough to remove the CDF determination.

26. What CDF definitions have you or are you using in the different applications of your LPSD PSA? Why have they been chosen?

No LP&SD PSA quantitative applications have yet been submitted. The use of CDF is expected. The use of LERF will depend of the extension of the studies up to level 2 and the definition of a proper metric for the new scope.

SWEDEN

21. What risk metrics were used for the Level 1 LPSD PSA? [This would typically include core damage frequency but may also include others such as LERF and core boiling frequency.]

CSNI Question number	Answer from Swedish plant	Answer
Q21	F1/2	CDF/year
	R1	In a similar way as the power operation only CDF.
	R2-4	R2, CDF
	B2	No answer
	O1	Basically the same consequences (risk metrics) as for the full power PSA have been used and they are: Core Damage Frequency (CDF) due to failure of shutting down reactor (HS1), insufficient injection (HS2) and insufficient decay heat removal (HS3). Frequency for cold over pressurisation of the RPV (ÖT1). Only economical consequences (not a core damage consequence). Frequency for fast over pressurisation of the RPV (ÖT2). Defined as a core damage consequence. Consequences HS1 and ÖT2 above are not really core damage

		consequences, they should be further developed to either HS2 or HS3. Answer only valid for low power PSA, not outage period.
	O2	CDF, RHR loss, Warm and cold over pressurisation, uncover of core with no tank lid and fuel pool overheating
	O3	CDF, RHR loss and warm and cold over pressurisation.
SKI REMARK	All studies consider CDF	

22. Were risk metrics other than CDF used? If yes, what insights have you gained regarding their usefulness in the decision-making process?

CSNI Question number	Answer from Swedish plant	Answer
Q22	F1/2	
	R1	No.
	R2-4	R2, No
	B2	No answer
	O1	See consequence type 2 and 3 for Q21. The low power PSA is not finalised yet so no particular insights have been made so far. Answer only valid for low power PSA, not outage period.
	O2	Support for strategy with no fuel in tank during main coolant pump removal, service or installation
	O3	None.

23. Where CDF has been used as a risk metric, how has "core damage" been defined?

CSNI Question number	Answer from Swedish plant	Answer
Q23	F1/2	Fuel temp. exceeding 1204 °C
	R1	Fuel temperature 1204°C
	R2-4	R2, No answer
	B2	No answer
	O1	Fuel cladding temperature >1204 °C. Answer only valid for low power PSA, not outage period.
	O2	Over 1204 C at any point in core
	O3	Above 1204 C at any point in core
SKI REMARK	For Swedish reactors a fuel temperature over 1204° C is defined as core damage.	

24. How does the definition of CDF for LPSD PSAs compare with the definition of "core damage" used in the full power PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q24	F1/2	Same def.

	R1	Same as power operations
	R2-4	R2, No answer
	B2	No answer
	O1	The same definition is used. Answer only valid for low power PSA, not outage period.
	O2	Identical
	O3	Identical
SKI REMARK	Same definition as for full power operation	

25. What is the basis for using the same or different core damage definitions in the full power and LPSD PSAs?

CSNI Question number	Answer from Swedish plant	Answer
Q25	F1/2	Uniformity in the definition of the safety limit for the fuel barrier
	R1	We don't see any significant differences during lower power operations.
	R2-4	R2, No answer
	B2	No answer
	O1	Used definition is general in that way that it is independent of operating mode (full power or low power). Answer only valid for low power PSA, not outage period.
	O2	No reasons for different definitions
	O3	No reasons for different definitions
SKI REMARK	This is the only definition used in Sweden.	

26. What CDF definitions have you or are you using in the different applications of your LPSD PSA? Why have they been chosen?

CSNI Question number	Answer from Swedish plant	Answer
Q26	F1/2	See Q23, Q25 - Fuel temp. exceeding 1204 °C - Uniformity in the definition of the safety limit for the fuel barrier
	R1	The same as power operations.
	R2-4	R2, No answer
	B2	No answer
	O1	See answer to Q21. The choice of these was a prerequisite for the analysis. Answer only valid for low power PSA, not outage period.
	O2	See Q23
	O3	See Q23
SKI REMARK	See Q23	

UNITED KINGDOM

21. What risk metrics were used for the Level 1 LPSD PSA? [This would typically include core damage frequency but may also include others such as LERF and core boiling frequency.

Core damage is used. LERF is not used as such but the frequency of large/ small uncontrolled release (UCR) is calculated. Core boiling is not used,

22. Were risk metrics other than CDF used? If yes, what insights have you gained regarding their usefulness in the decision-making process?

Level 3 developed but insights mainly dominated by core damage concerns

23. Where CDF has been used as a risk metric, how has "core damage" been defined?

Fuel melt (i.e. exceeding fuel rod structural integrity limit)

24. How does the definition of CDF for LPSD PSAs compare with the definition of "core damage" used in the full power PSA?

Same

25. What is the basis for using the same or different core damage definitions in the full power and LPSD PSAs?

Although same core damage definition used, non core melt considerations also taken into account in determining risk, e.g. gap release and PCI concerns have also been evaluated.

26. What CDF definitions have you or are you using in the different applications of your LPSD PSA? Why have they been chosen?

The safety case is based on an annual average maintenance model. However this does not prevent use of hourly risk/ outage risk being calculated for risk comparison purposes.

UNITED STATES

BWR

21. What risk metrics were used for the Level 1 LPSD PSA? [This would typically include core damage frequency but may also include others such as LERF and core boiling frequency.

The Level 1 risk metric was CDF (per calendar year).

22. Were risk metrics other than CDF used? If yes, what insights have you gained regarding their usefulness in the decision-making process?

Level 3 risk metrics included:

- Early fatality risk
- Total latent cancer risk

- Population dose within 50 miles
- Population dose within 1000 miles
- Individual early fatality risk: 0 to 1 mile
- Individual latent cancer risk: 0 to 10 miles

Use of Level 3 risk metrics provides a more complete representation of the risk from various initiating events. Level 3 metrics may indicate that while the LPSD CDF from initiating events is low compared to full power CDF, the risk from the LPSD initiating events may be comparable to the risk from full power initiators.

While Level 3 results can be used in decision-making activities, enhancements to the inputs needed for a Level 3 analysis and improvements in Level 3 modelling could provide additional stimulus for such use. Examples of such enhancements or improvements include: 1) a more complete understanding of the shutdown-specific issues affecting the development of the source term used in the consequences calculations and 2) a better understanding of the uncertainties associated with the parameters that significantly affect the overall uncertainty in risk results.

Metrics for Level 2 and 3 are useful in risk-informed regulatory activities such as the inspection program Significance Determination Process, as well as for support of the proposed shutdown rule.

23. Where CDF has been used as a risk metric, how has "core damage" been defined?

Core damage was defined using various definitions that depended on the type of calculation performed (i.e., whether it was a screening calculation or a detailed calculation). Core damage was defined as:

- Time to top of active fuel,
- Time to one-third core height,
- Time to core heatup, or
- Time to first gap release.

24. How does the definition of CDF for LPSD PSAs compare with the definition of "core damage" used in the full power PSA?

All definitions identified above are definitions that have been used in full power PRAs.

25. What is the basis for using the same or different core damage definitions in the full power and LPSD PSAs?

See Q23. The basis should be consistency among the analyses. However, this will require that a standard definition of core damage be adopted and used.

26. What CDF definitions have you or are you using in the different applications of your LPSD PSA? Why have they been chosen?

See responses to Q25 and Q26.

PWR

21. What risk metrics were used for the Level 1 LPSD PSA? [This would typically include core damage frequency but may also include others such as LERF and core boiling frequency.

1) NUREG/CR-6144- The internal event PSA of mid-loop operation at Surry is a level 3 PSA. Its risk measures include core damage frequency, population dose, and early and late fatality risks to individual and population. The internal fires, internal floods, and seismic analysis are level 1 PSAs, and use core damage as the risk measure.

2) Screening analysis of NUREG/CR-6144- The screening analysis of all POSs is a level 1 PSA using core uncovery frequency as the risk measure.

3) NUREG/CR-6616 and 5718- The risk measures of NUREG/CR-6616 include core damage frequency, population dose, and early and late fatality risks to individual and population. The risk measure of NUREG/CR-5718 is core damage frequency.

22. Were risk metrics other than CDF used? If yes, what insights have you gained regarding their usefulness in the decision-making process?

Yes, see answer to Q21.

Level 2 and Level 3 information can show important risk insights. For example, some low core damage frequency event may be shown to have increased relative importance when other considerations such as containment integrity and dose are factored into the study. Metrics for Level 2 and 3 are useful in risk-informed regulatory activities such as the inspection program Significance Determination Process as well as for support of the proposed shutdown rule.

23. Where CDF has been used as a risk metric, how has "core damage" been defined?

1) NUREG/CR-6144- Core damage is defined as initiation of cladding failure. Based on a MELCOR calculation cladding failure is likely to occur when the water level in the core region is reduced to about 2.5 feet above the bottom of the core. This criterion was used in a simpler thermal hydraulic code to determine time to core damage in different scenarios.

2) Screening analysis of NUREG/CR-6144- Core uncovery is assumed to lead to core damage.

3) NUREG/CR-6616and 5718- Same as 1.

24. How does the definition of CDF for LPSD PSAs compare with the definition of "core damage" used in the full power PSA?

In NUREG-1150 for Surry, core damage is defined as the uncovery of the top of active fuel without imminent recovery.

25. What is the basis for using the same or different core damage definitions in the full power and LPSD PSAs?

1) NUREG/CR-6144- The definition of core damage/cladding failure as water level reaching2.5 feet above the bottom of the core is considered more appropriate for shutdown conditions, due to lower decay heat level during shutdown. RCS coolant temperature, pressure, and level for

different scenarios were calculated using a simple model based on heat and mass balance. It has been verified in a MELCOR calculation that cladding failure occurs at the water level.

2) Screening analysis of NUREG/CR-6144- The same definition as in full power is used due to the screening nature of the study. It is conservative.

3) NUREG/CR-6616 and 5718- Same as 1.

26. What CDF definitions have you or are you using in the different applications of your LPSD PSA? Why have they been chosen?

See answers to Q25.

APPENDIX D - SEQUENCE ANALYSIS

D.1 Accident Sequence Modelling

PRELUDE AND QUESTIONS

Accident sequences depict plant responses to initiating events. The processes used to develop accident sequences in LPSD PSAs are similar to those used for full power PSAs. However, some experts have expressed the concern that adopting underlying assumptions and success criteria used in full power PSAs may result into overlooking or under-estimating the significance of some LPSD accident sequences. For example, a French LPSD study identified fast-acting reactivity insertions (i.e., the rapid insertions of large slugs of un-borated water into the reactor core) as a potentially significant risk contributor in LPSD PSAs. Also, a United States (Surry Nuclear Power Plant) study identified cold over-pressurization (i.e., in the event of a loss of decay heat removal in a pressurized water reactor, the potential exists for a small rise in RCS temperature which can result in a significant rise in pressure that potentially compromises the system's pressure boundary) as important. However, the modelling of such events has been limited. Please respond to the following questions.

27. How have accident sequences been modelled in the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

b) Why have these differences been necessary?

28. How success criteria have been defined in the LPSD PSA?

a) How does this different from what was done in the full power PSA?

29. Why have these differences been necessary?

RESPONSES

BELGIUM

27. How have accident sequences been modelled in the LPSD PSA?

By means of event tree analyses

a) How does this differ from what was done in the full power PSA?

Identical.

28. How success criteria have been defined in the LPSD PSA?

The success criteria in the LPSD have been identified on the basis of the availability of mitigating systems for core uncovery.

a) How does this different from what was done in the full power PSA?

No thermal hydraulic analyses have been carried out for the LPSD.

29. Why have these differences been necessary?

The lack of thermal hydraulic analysis for the LPSD.

GERMANY

27. How have accident sequences been modelled in the LPSD PSA?

Event tree analysis for any IE in each considered POS

a) How does this differ from what was done in the full power PSA?

There is no difference.

28. How success criteria have been defined in the LPSD PSA?

The success criteria is the sufficient response of the operational and safety systems including the human actions in time to avoid hazard states.

a) How does this different from what was done in the full power PSA?

There is no difference.

HUNGARY

27. How have accident sequences been modelled in the LPSD PSA?

Development of accident sequences for the PSA of off-power plant operational states was carried out in the following steps:

1) The scope of endstates for which analyses should be performed was determined. Two major categories of undesirable endstates were distinguished, namely core damage and - mainly during plant operational states with open reactor - boiling.

2) In the next step safety functions were identified, that must be ensured in order to prevent the development of the above undesirable endstates.

3) The scope of systems was determined, that can be able to fulfil the required safety functions. Also their minimal configuration and operational conditions (success criteria) were determined that are necessary and sufficient to fulfil those safety functions. Results of supporting deterministic analyses were taken into account during determination of the consequences of special, shutdown-specific initiating events.

4) Availability of the systems identified above was examined in each of the POSs. Outage work schedules, procedures and practices were studied in detail for that purpose.

5) In the last step each initiating event was examined within each POS with respect to loss of what safety functions can lead to one of the undesirable endstates. Finally the event tree was built up for each given initiating event taking into account (available) systems capable of fulfilling the required safety functions in the given POS.

a) How does this differ from what was done in the full power PSA?

In principle, the approach of developing accident sequences does not differ from the one followed in the full power PSA. However, in addition to the standard steps of a PSA procedure, development of accident sequences required an evaluation of availability of mitigating systems as defined by the actual systems configuration and the maintenance characteristics for a given plant operational state. In this way state dependent accident sequences were really developed. Also, emphasis was laid on analyzing outage specific scenarios, such as inadvertent boron dilution, drop of a heavy load, pressurized thermal shock and loss of natural circulation.

b) Why have these differences been necessary?

Simply stating, the full power PSA is related to one plant operational state, while the LPSD PSA is related to many of them. In principle, the LPSD PSA consists of as many separate PSA models as the number of the POSs defined.

28. How success criteria have been defined in the LPSD PSA?

The most important task of the accident sequence development process is the definition of success criteria. The analyses were aimed at evaluating long term heat removal in accordance with some competent shutdown PSA studies. By “long term” the time interval is meant during which mitigation of the initiating event and its consequences is ensured in such a way that afterwards core damage does not occur afterwards even if there is a failure of the mitigating systems, or the likelihood of core damage is negligible, e.g. in the worst case the core is fully unloaded. Such a time interval was identified case by case and requirements for the operation of mitigating systems were defined on the basis of that. It should be noted that - applying a conservative approach - during modelling of these long term processes neither allowed outage time nor repair of failed active components of redundant trains were considered.

System success criteria for the systems in different accidental situations starting in different plant operational states of the off-power operation were partly defined by using the results of thermal-hydraulic calculations performed earlier for the purposes of the PSA study for the full power operational mode. In addition further specific analyses were carried out in support of the LPSD PSA. The following factors were considered during definition of system success criteria:

- important plant parameters characterizing the various plant operational states
- availability of mitigating systems for the different initiating events in different plant operational states
- time available for human intervention (in most cases it is the function of the decay heat characterizing the plant operational states).

a) How does this different from what was done in the full power PSA?

In principle, the approach of defining success criteria does not differ from the one followed in the full power PSA. However, in addition to the standard steps of a PSA procedure, definition of success criteria required an evaluation of availability of mitigating systems as defined by the actual systems configuration and the maintenance status characteristic for a given plant operational state. In this way state dependent success criteria were really defined. Also, emphasis was laid on analyzing outage specific scenarios, such as inadvertent boron dilution, drop of a heavy load, pressurized thermal shock and loss of natural circulation.

29. Why have these differences been necessary?

See Q27/b.

JAPAN

27. How have accident sequences been modelled in the LPSD PSA?

Event Trees are developed for each POS of each initiating event, in which headings are lined up considering the mitigation systems and operator actions that can be credible to prevent accident progression.

Major steps for the above process are described in more detail as follows.

1) Expect the plant and system responses after the initiating event occurred.

2) Find the cause or factor of core damage out.

3) Estimate the time period to core damage using the thermal analysis code or hand calculation.

4) Find the counter-measures for core damage out considering the time interval to core damage and capability of counter-measure.

5) Express the above accident sequence by the event tree (ET) method.

6) Make the ETs for all initiating events.

a) How does this differ from what was done in the full power PSA?

In the LPSD PSA of PWR, we consider some unique mitigation measures that are not considered in the full power PSA, as follows.

1) In the case of loss of coolant inventory and/or decay heat removal function during mid-loop operation, we take advantage of the enhanced reflux cooling combined with manual opening of SG secondary side PORVs, or the coolant injection from Refuelling Water Storage Tank(RWST) driven by the static head difference between RWST and RCS main coolant pipes, depending on the POS.

2) Credit is taken for manually starting the standby RHR system or recovery of failed RHR system

In the case of BWR, these approaches are basically the same as full power PSA.

b) Why have these differences been necessary?

Unique mitigation measures noted above for PWR are necessary because there exists essentially no other decay heat removal measures, when decay heat removal function is lost in shutdown operation.

28. How success criteria have been defined in the LPSD PSA?

In the LPSD PSA of PWR, we define different success criteria for each event heading of each POS, in terms of the required capacity of the relevant mitigation systems to avoid core damage. Occurrence of core damage is judged by the PCT (Peak Cladding Temperature) exceeding 1200 degree C or by the uncovery of the core top, depending on whether PCT calculation by RELAP5 mod.2 code is available in the POS.

In the case of BWR, "core uncover" was applied for the loss of decay heat removal scenario. Whereas less than "1200 degree C "on fuel cladding temperature was applied for the loss of coolant scenario, which is the same success criteria as the full power PSA.

a) How does this different from what was done in the full power PSA?

The only difference from full power PSA is the criteria for core damage based on the uncovery of core top in case PCT calculation is not available. This is because the fuel coolability after the core uncovers might have large uncertainty, and the time period between core uncovery and exceeding1200 degree C would not play an important role in CDF evaluation.

29. Why have these differences been necessary?

The above difference is due to the fact that PCT calculation by RELAP5 mod.2 code becomes invalid in some period of mid-loop operation of PWR. For the BWR, we would like to avoid the complex evaluation for success criteria.

Two additional questions / responses were included:

What areas of your analysis are, in your opinion, most in need of improvement?

For the PWR, as mentioned in Q32, we take credit of injection from RWST by gravity force as well as reflux cooling. Although we analyzed effectiveness of these cooling mechanism using RELAP5 mod2 for 4-loop PWR, further verification by some experiments seems to be needed. For LOCA events in BWR, the occurrence of the core damage is judged by PCT exceeding 1,200-degree C. This definition differs to that for other initiating events. We estimate PCT conservatively through hand calculation. Therefore, the fuel temperature transient during the loss of coolant sequence related to the system or equipment maintenance would be in need of improvement.

What are the areas that would most benefit from further research?

Further refinement of above-mentioned cooling mechanism would lead to improving the accuracy of accident sequences and success criteria during POSs including mid-loop operation of PWR or LOCA sequence of BWR resulting in increasing the allowed time for operator actions.

KOREA

27. How have accident sequences been modelled in the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

Unlike the full power PSA, for example, the following assumptions are used in fire risk analysis during shutdown mode:

- The initiating events and associated core damage scenarios are limited to the Loss of Decay Heat Removal, Loss of Coolant Accident, and Loss of Offsite Power scenarios from Mode 4 through 6. These scenarios were determined to represent the dominant challenges to the decay heat removal and RCS inventory control safety functions.

28. How success criteria have been defined in the LPSD PSA?

System configuration, including maintenance practices, during shutdown operation was reflected in the determination of success criteria.

29. Why have these differences been necessary?

Because the system configuration may be different according to each POS, the success criteria should be varied for each POS.

MEXICO

27. How have accident sequences been modelled in the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

There are no significant differences.

b) Why have these differences been necessary?

See the above answer.

28. How success criteria have been defined in the LPSD PSA?

a) How does this different from what was done in the full power PSA?

There are no significance differences if the POS requirements were specified for just one plant condition. The plant condition representative of the POS is selected to be the most demanding.

29. Why have these differences been necessary?

The differences have been necessary for the different plant conditions (level of power, pressure, water level, etc) covered by the POS.

THE NETHERLANDS

27. How have accident sequences been modelled in the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

b) Why have these differences been necessary?

28. How success criteria have been defined in the LPSD PSA?

a) How does this different from what was done in the full power PSA?

29. Why have these differences been necessary?

The small-event-tree/large-fault-tree approach is used to define the accident sequences. In addition to the functional success criteria approach, NPP Borssele transient responses and prior PRAs of similar reactor types have been reviewed to identify any event tree headings necessary to properly model all reactor safety functions. The success criteria are similarly developed from plant specific NPP Borssele analyses and prior PRAs. In addition the important considerations from a level-2 perspective have been reviewed to identify any systems/functional headings necessary to adequately characterize the accident progression associated with the core damage sequences.

For each of the initiators, a unique set of success criteria is developed. After the success criteria are defined, the event trees are constructed. Operator actions are included explicitly in the fault trees and equations that support the event trees. The ground rules that form the basis for the event tree analysis and success criteria which are relevant for LPSD scenarios are listed below:

- All successful sequences are carried out to the point where stable conditions exist with successful long term core cooling. In general, sequences are terminated at 24 hours. Sequences with the primary system pressure boundary breached are modelled through cold shutdown. Plant conditions. Sequences with the primary system intact are modelled to the point of hot plant shutdown. For sequences ending with the plant in hot shutdown conditions, long term feedwater makeup and steam removal is modelled explicitly in top logic fault trees.

- Primary system inventory makeup is not explicitly required if primary integrity is maintained from the onset of the event. It is assumed that normal pressuriser water level is sufficient to accommodate primary inventory shrinkage from full power to hot shutdown, or if any inventory makeup is required in hot shutdown, the probability of failing to provide it is negligible.

- Boration of the reactor is not required if hot shutdown temperatures and primary integrity are maintained. Boration is required for steam or feedwater line breaks with failure to isolate the affected steam generator, due to the resultant cooldown. Boration is also required for ATWS and reactivity addition scenarios. Chemical boration is estimated to be ineffective in mitigating a large, fast reactivity accident.

- During non-power operations the P-SRV demand rate only applies to events initiating from the hot steaming plant operational state. The P-SRV demand rate in hot standby is estimated to be well below the 1 E-03 demand used for power. Further the transient initiating event frequencies are two orders of magnitude lower than that of power (due to the shorter POS duration) causing transient induced LOCAs at hot steaming to be negligible.

- During power and non-power hot steaming plant conditions , the minimum very small break LOCA (LV) size is based on the capacity of 1 Volume Control (TA) pump. In other non-power POSs, the very small LOCAs require injection plus a PORV opening (if the RCS is closed) for success.

- During power and non-power hot steaming plant conditions , the maximum very small break LOCA (LV) size is based on the capacity of 2 Volume Control (TA) pumps. In other non-power POSs, the very small LOCAs require injection plus break flow for sufficient core cooling. During power and non-power hot steaming conditions the maximum small LOCA (LS) or minimum large small LOCA (LMS) is based on the minimum break size for the break to remove decay heat and to cool down the primary at 100 K/hour. The maximum Large Small LOCA/ minimum Medium LOCA break size is based on the minimum break size to depressurize the primary such that the high pressure injection system (TJH) shifts from 1 of 4 success to 2 of 4 and that the RHR will operate without operator action. The maximum Medium LOCA/ minimum Large LOCA size is based on the minimum break size to depressurize the primary such that the high pressure injection system is not required to prevent core damage and that RHR will operate without operator action.

- Unless specifically noted, the success criteria for the non-power hot steaming POS are identical to that of power. Plant response only differs in the electric plant, which remains connected to offsite power and does not require transfer once shutdown.

- Successful spent fuel pool cooling can be achieved through operation of at least one TG pump (Fuel pool cooling system) or through pool refill through an alternate source such as low pressure ECCS or fire fighting system.

- In midloop, loss of RHR cooling must be recovered by closed-cycle cooling such as recovery of RHR, low pressure sump recirculation mode of the low pressure ECCS, or cooling with the reserve cooling system (TE), or the Volume Control System operating in the recirculation mode. Although the bunkered primary reserve injection system (TW) automatically actuates, injects for at least 10 hours, and can be refilled it is conservatively modelled that boron crystallization may occur which blocks coolant flow. Thus one refill of TW has been credited for extending the time window, but repeated filling of the TW basin is not credited for long term success.

- In the cold shutdown POS following long term failure of RHR, secondary heat removal via a SG has been credited as a successful means of decay heat removal. Main feedwater is conservatively not modelled, as it may be out of service for test or maintenance

- In the midloop POS MAAP-4 analyses indicate the thermal-hydraulic behaviour of the reactor coolant system with the RPV head on and the vents open is very close to the case where the RPV head is removed. Thus, the times when the RPV head is on and the vents are open have been modelled as if the RPV head is removed. The case with the RPV head removed is the conservative case because the reactor coolant system inventory is depleted faster in this case.

Initiating events with a similar plant response have been grouped together for input to the task of accident sequence delineation., with the most restrictive success criteria used. In addition, POSs with the same basic plant conditions (e.g., the same mitigative system available) are also grouped for the development of event trees.

SPAIN

27. How have accident sequences been modelled in the LPSD PSA?

Accident sequences have been modelled by systemic/functional event trees.

a) How does this differ from what was done in the full power PSA?

It is the same method that in full power PSA.

b) Why have these differences been necessary?

Not applicable.

28. How success criteria have been defined in the LPSD PSA?

Success criteria are defined as required to fulfill each of the mitigation functions required. This depends on the IE, the POS, and a number of boundary conditions. Then, the success criteria are defined specifically in a case by case basis. They are based on full power calculations when possible, shutdown calculations (mainly hand made ones), and expert judgments.

a) How does this different from what was done in the full power PSA?

In the full power PSA, success criteria are more or less based on expert judgment. They are more based on FSAR Chapter 15, specific TH calculations, etc.

29. Why have these differences been necessary?

The differences have been necessary because of the lack of codes adapted to LP&SD conditions and the fact that some conservatism due to calculations does not distort the results too much.

SWEDEN

27. How have accident sequences been modelled in the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

b) Why have these differences been necessary?

CSNI Question number	Answer from Swedish plant	Answer
Q27	F1/2	
	R1	In the same way as the power operations.
	R2-4	R2, Through appropriate scenarios supported by EdF's thermo hydraulic studies for the 900 MWe units. Engineering judgement and information from plant personnel was used to identify differences that could be significant.
	B2	Through appropriate scenarios supported by simplistic calculations
	O1	When modelling accident sequences we have started with the full power functional block diagram and event trees. Those have then adjusted to

		the identified POSs (se answer to questions Q3 and Q4) and been reviewed by experienced operating personnel in order to verify that the event trees used are according to how the plant is being operated during low power condition. As stated in answer to question Q11 a special process have been used in order to identify initiating events. This process is designed to identify initiating events (accident sequences) that are unique for low power conditions, e.g. reactivity transients such as the one mentioned above. Answer only valid for low power PSA, not outage period.
	O2	With event trees
	O3	With event trees
SKI REMARK	All Swedish utilities use the Riskspectrum computer program for PSA calculations. For LP a modified model of the power operation PSA is used. For SD mainly event trees are used, in certain cases together with fault trees.	
Q27 a	F1/2	No difference according too full power operation. But there is a qualitative analysis of cold over-pressurisation.
	R1	No difference.
	R2-4	R2, For the full power PSA, Ringhals and Westinghouse deterministic analysis been used.
	B2	The approach is different in LPSD where standardised deterministic analyses have not been used.
	O1	See answer to question Q11 and Q27. Answer only valid for low power PSA, not outage period.
	O2	No way
	O3	In no way
SKI REMARK	See Q27	
Q27 b	F1/2	
	R1	Not necessary.
	R2-4	R2, EdF thermo hydraulic analyses were found to be better suitable, since they take EdF operating procedures into consideration.
	B2	Lack of knowledge, lack of resources lack of tools
	O1	By just copying accident sequences used for full power and adapt them without adjustment will not result in an adequate PSA for low power conditions. Answer only valid for low power PSA, not outage period.
	O2	Not applicable
	O3	Not applicable
SKI REMARK	There has not yet been any need for detailed fault trees during SD and the existing fault tree models are constructed for power operation.	

28. How success criteria have been defined in the LPSD PSA?

a) How does this different from what was done in the full power PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q28	F1/2	No CD (1204 °C)
	R1	The same as power operation.
	R2-4	R2, Success criteria are based on EdF thermo hydraulic analysis
	B2	In many cases requirements in the Technical Specifications have been used.
	O1	Here we think that success criteria are the same as "system requirements", if that is correct the answer is that the same success criteria as in full power PSA have been used. The reason for doing this is that it is conservative and no new system analyses (with MAAP for instance) need to be performed. Answer only valid for low power PSA, not outage period.
	O2	OK = Not any of the states under Q21 / System success criteria are usually identical to those in full power PSA.
	O3	OK = None of the states listed under Q21 / System success criteria are usually identical to those in full power PSA.
SKI REMARK	Success criteria are normally; Covered core (sealed leakage) ore re-established RHR.	
Q28 a	F1/2	No difference.
	R1	No difference.
	R2-4	R2, For the full power PSA, Ringhals and Westinghouse deterministic analysis been used.
	B2	No answer
	O1	There is no difference. Answer only valid for low power PSA, not outage period.
	O2	Not much
	O3	Not much
SKI REMARK	Different time windows. Different barrier protection.	

29. Why have these differences been necessary?

CSNI Question number	Answer from Swedish plant	Answer
Q29	F1/2	
	R1	Not necessary.
	R2-4	R2, EdF material more suitable, as Ringhals PWR is using EdF's instruction package for shutdown states now.
	B2	No answer
	O1	Question not relevant, see answer to question Q28. Answer only valid for low power PSA, not outage period.
	O2	Not applicable
	O3	Not applicable

SKI REMARK See Q28a

UNITED KINGDOM

27. How have accident sequences been modelled in the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

Sequences developed in the same manner by considering the specific implications of the fault and conducting new transient analysis if required

b) Why have these differences been necessary?

See above

28. How success criteria have been defined in the LPSD PSA?

a) How does this different from what was done in the full power PSA?

The success criteria depends on the systems available and the condition of the plant.

29. Why have these differences been necessary?

The differences are required because of the differing decay heat levels and resulting temperatures and pressures.

UNITED STATES

BWR

27. How have accident sequences been modelled in the LPSD PSA?

The process used to model accident sequences is the same as used in full power PRAs. Initiating events were identified; event trees were developed to model the functional (and ultimately the systemic) impact of various system failures on the progression of the accident and to account for the fraction of time spent in the POS; fault trees were developed to identify the various combinations of failures that lead to the top events in the event trees; and recovery actions were identified that eliminated or minimized the impact of equipment failures.

a) How does this differ from what was done in the full power PSA?

There is no difference except for the inclusion of the fraction of time spent in the POS.

b) Why have these differences been necessary?

The differences have been needed to estimate the risk measures in the POS.

28. How success criteria have been defined in the LPSD PSA?

Success criteria were defined as combinations of sufficient equipment response (or operator actions) to ensure the occurrence of function (or actions) necessary to prevent the onset of core damage.

a) How does this different from what was done in the full power PSA?

The process is the same; however, the actual success criteria may be different as a result of the specific conditions being analyzed. For example, the success criteria for a large break LOCA in POS 5 required flow from two low-pressure coolant injection pumps to prevent core damage rather than from only one pump for the same accident that occurred at full power conditions.

29. Why have these differences been necessary?

The differences have been necessary because different conditions exist during at-power and shutdown.

PWR

27. How have accident sequences been modelled in the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

The method for modelling the logic of accident sequences during shutdown is the same, i.e., event trees. The general method of developing accident scenarios is the same. Care was taken to account for the specific plant configuration in each plant operational state.

1) NUREG/CR-6144- In quantification of the accident sequences, the fraction of time the plant spent in a POS was included. The only procedure that is specifically written for mitigating accidents during shutdown is the loss of RHR procedure. Therefore, the strategies for mitigating some accidents, e.g. loss of offsite power, have to be developed using information for power operation. Interactions with the plant were necessary to confirm the strategies. In order to account for the changing decay heat level, time windows after shutdown were defined based on the success criteria of the mitigation functions.

2) Screening analysis of NUREG/CR-6144- Both refuelling outages and non-refuelling outages were considered due to differences in plant configurations. The event tree development was performed in a talk through format. Typically, a team of 4 met in the conference room. The members include engineers with background and training in the area of plant operations, PRA, and thermal hydraulics. The plant's initial conditions and the responses to the initiators are discussed. The related procedures, i.e., operating procedures and abnormal procedures, and other documents were reviewed and the accident scenarios were developed. In quantification of the accident sequences, the fraction of time the plant spent in a POS was included.

3) NUREG/CR-6616 and 5718- The approach is similar to that of 1) The plant information collected in 1) was used without additional input from the plant. The same time windows as those defined in 1) were used, and simple thermal hydraulic calculations were performed to determine the success criteria and accident timing. The initiating events defined in 1) were grouped into 5 initiating event groups to simply the modelling effort. In, NUREG/CR-6166, the conditional CDF, given the plant is in the time window, was calculated. Sensitivity calculations were performed to determine the increase in risks due to maintenance of different equipment. In NUREG/5718 (draft), the CDF, including the fraction of time the plant is in each time window, was calculated using the average maintenance unavailabilities collected from past outages.

b) Why have these differences been necessary?

The differences were necessary to capture the uniqueness of shutdown conditions.

28. How have success criteria been defined in the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

1) NUREG/CR-6144- An important feature of the study is the use of 4 time windows after shutdown to account for the effects of different levels of decay heat on success criteria and timing of accidents. Due to the unique plant configuration while the plant is at mid-loop, the unique methods of heat removal considered in the PSA, reflux cooling and gravity injection, have system requirements that are different from those considered in a full power PSA. The timing and success criteria of these methods were determined by supporting thermal hydraulic calculations. Other methods of heat removal, i.e., RHR system, feed-and- spill, and spray recirculation, are similar to those of full power operation. In the case of feed-and-spill, the Surry plant has a technical report that determines the timing and success criteria for using different pumps including LHSI pumps based on the requirement to maintain sub-cooling in the RCS. The success criteria in some cases turn out to be more stringent than that of feed-and-bleed in the full power PSA. It was modified to that of full power PSA. Due to lower decay heat in later windows, recirculation and spray recirculation are not needed.

2) Screening analysis of NUREG/CR-6144- For those POSs whose plant configuration is similar to that of power operation, i.e., hot standby and start-up, the full power model was used with modifications to account for obvious differences, e.g., no reactor trip is needed if the reactor is already tripped. For those POSs in which the RHR system is initially running to remove decay heat, a generic event tree was developed based on abnormal procedure for loss of RHR. It includes all possible methods of mitigating the initiating event. The POS specific conditions of the plant were used to determine if a method of mitigation is possible. If a method is considered possible, the success criterion was determined either by judgment or based on simple thermal hydraulic considerations. For loss of offsite power and station blackout, the loss of RHR event tree was modified to account for the specific impacts of the initiating events.

3) NUREG/CR-6616 and 5718- Timing and success criteria of the loss of RHR event trees were determined using the simple thermal hydraulic program developed in 1). The success criteria for LOCAs were determined by judgment. For loss of offsite power and station blackout, the loss of RHR event tree was modified to account for the specific impacts of the initiating events.

29. Why have these differences been necessary?

The plant configuration during shutdown could be quite different from that of power operation, i.e., mid-loop operation, and the method of mitigation is very different. For those POSs other than mid-loop, the RCS condition is different from that of power operation, and accident timing and success criteria have to be determined for the specific plant conditions.

D.2 Success Criteria

PRELUDE AND QUESTIONS

There is a lack of thermal-hydraulic codes specifically designed for LPSD conditions. Analysts use either the success criteria developed for full-power PSAs, their own simplistic calculations, or calculations from thermal-hydraulic codes developed for full power conditions. However, it is not clear that the phenomena involved in an accident occurring at shutdown can be simulated efficiently and correctly with computer codes developed for full power. The concern is that the codes developed for full power may not model shutdown accidents efficiently and correctly. Another concern is that the use of full power codes or very

simplistic (back-of-the-envelope) calculations produces either overly conservative or overly optimistic estimates and therefore misleading results. Another issue is code efficiency; for example, the US NRC had experienced long run times and code instability when full power codes were used for LPSD studies.

30. How has the success criteria used in the LPSD PSA been determined?

31. What analyses have been carried out for LPSD conditions [this would include thermal-hydraulic analysis, neutronics, and analysis of other effects such as cold overpressurisation, etc.)?

a) What codes have been used to carry out the analyses for LPSD conditions?

b) How does this differ from what was done in the full power PSA?

32. What analysis has been carried out which is unique to LPSD conditions (e.g., are they difficult to analyze, are the sequences analyzed efficiently, are the results credible)?

RESPONSES

BELGIUM

30. How has the success criteria used in the LPSD PSA been determined?

For the cold shutdown for intervention POS, the success criterion has been "no boiling conditions in the RCS".

31. What analyses have been carried out for LPSD conditions [this would include thermal-hydraulic analysis, neutronics, and analysis of other effects such as cold overpressurisation, etc.)?

a) What codes have been used to carry out the analyses for LPSD conditions?

None (see also response to Q19; potential point of improvement)

b) How does this differ from what was done in the full power PSA?

For full power conditions, RELAP 5 calculations were used.

32. What analysis has been carried out which is unique to LPSD conditions (e.g., are they difficult to analyze, are the sequences analyzed efficiently, are the results credible)?

None

GERMANY

30. How has the success criteria used in the LPSD PSA been determined?

Thermal hydraulic calculations have been used (for POSs with closed primary circuit) and engineering analysis.

31. What analyses have been carried out for LPSD conditions [this would include thermal-hydraulic analysis, neutronics, and analysis of other effects such as cold overpressurisation, etc.)?

Thermal hydraulic analysis has been carried out for the cases:

Loss of the RHR during mid-loop-operation and heat removal via a steam generator as a back up for the RHR

a) *What codes have been used to carry out the analyses for LPSD conditions?*

Faulty reactor protection signal during level lowering on mid-loop.

The recriticality effect due to the transport of a slug of unborated coolant in the pump suction line through the core has been analysed by mixing and coupled thermal hydraulic- neutron kinetic analysis. (Further development is necessary)

b) *How does this differ from what was done in the full power PSA?*

Thermal hydraulic analysis: ATHLET (adapted to LP&SD conditions).

Recriticality analysis: ATHLET-QUABOX/CUBBOX.

The ATHLET-code has been adapted to LP&SD conditions:

Extension of the model for components which are not needed so far in the FP-analysis

Extension of the model to analyse effects which are important for LP&SD (e.g. junction of suction line, modelling of steam generator tubes)

Extension of physical models which are important for LP&SD (e.g. front of unborated plugs and tracking system)

The coupling of thermal hydraulic and neutronics was developed because of the demands from LP&SD analysis.

32. What analysis has been carried out which is unique to LPSD conditions (e.g., are they difficult to analyse, are the sequences analysed efficiently, are the results credible)?

The analysis concerning the accumulation of low borated coolant in the pump suction line after the loss of the RHR-system – due to condensation in SG-tubes - is unique to LP&SD. It has been shown that this effect occurs. Special experiments have approved the insights. Measures could be derived from the analysis to prevent this scenario.

The analysis of the scenario after a faulty activation of the reactor protection signal is also unique for LP&SD. New insights of the influence of the pressure limitation system have been gained.

The coupling of thermal hydraulic and neutronics was developed to analyze the recriticality effect if a plug of unborated coolant is transported through the core. The coupled codes need still further validation for assured results.

HUNGARY

30. How has the success criteria used in the LPSD PSA been determined?

See Q28.

31. What analyses have been carried out for LPSD conditions [this would include thermal-hydraulic analysis, neutronics, and analysis of other effects such as cold overpressurization, etc.)?

The results of the supporting analyses have been used mainly for the identification of success criteria of event tree headings. Results of the following types of analyses were taken into account in the LPSD PSA study:

a) about 100 thermal hydraulic calculations performed by KFKI/AEKI (Central Research Institute for Physics/Atomic Energy Research Institute, Budapest, Hungary) and VUJE/SEI (Slovakia) up till 1994 with RELAP5/mod2 and ATHLET codes for the purpose of defining LOCA categories and success criteria for the PSA for nominal power operational mode of Paks NPP Unit 3 assuming different break sizes and available safety system configurations

b) calculations performed by KFKI/AEKI (Hungary) in 1995 with ATHLET and RELAP5/mod3.1 codes to clarify some success criteria of LOCA event trees related to shutdown states of Paks NPP Unit 2 when (1) cooling down is in progress, but hydroaccumulators are isolated from the primary circuit, and (2) the reactor is open, heat removal is ensured by the secondary side decay heat removal system, there is natural circulation in the primary circuit

c) calculations of reactivity transients performed as part of the DBA analyses by KFKI/AEKI (Hungary) within the national AGNES project between 1991-1994 carried out with the purpose of re-evaluating the safety of the Paks NPP on the level of the ´90s

d) calculations performed by AEA Technology (UK) with CFX -4 code, and by KFKI/AEKI (Hungary) in 1998 with ATHLET, coupled ATHLET-KIKO3D, SMATRA and SMABRE codes for the inherent and the dominant external dilution scenarios identified for Paks NPP within the PHARE project (PH2.08/95) aiming at the systematic identification and evaluation of boron dilution faults (calculations of inherent dilution scenarios included small break LOCA and ATWS calculations, while calculations of external dilution scenarios included three homogeneous and one inhomogeneous dilution scenarios all through the make-up water system).

a) What codes have been used to carry out the analyses for LPSD conditions?

See above.

b) How does this differ from what was done in the full power PSA?

Supporting analyses carried out to define success criteria for the full power PSA served as a basis for the LPSD PSA. However, shutdown conditions may differ significantly from those of the full power operation that required adjustment, modification or special input parameters of the calculating codes in many cases.

32. What analysis has been carried out which is unique to LPSD conditions (e.g., are they difficult to analyze, are the sequences analyzed efficiently, are the results credible)?

See items 2 and 4 under Q31.

JAPAN

30. How has the success criteria used in the LPSD PSA been justified?

For the PWR, success criteria were calculated using RELAP5 mod.2 code. Justification of the calculations, however, has not yet been completed by experiments or other relevant data.

For the BWR, a hand calculation was made for the loss of decay heat removal sequence, and the thermal hydraulic code used for the full power PSA was applied for the loss of coolant sequence.

31. What analyses have been carried out for LPSD conditions [this would include thermal-hydraulic analysis, neutronics, and analysis of other effects such as cold overpressurization, etc.]?

For the PWR, we performed shutdown state thermal hydraulic calculations using RELAP5 mod.2., including the POS under the mid-loop operation prior to fuel unloading. In this POS two types of calculation were performed. One was reflux cooling by SG secondary water when auxiliary feed water to SG was not available and the other was gravity injection from RWST water into RCS when SG was isolated by installing seal plugs (SG nozzle dam) to SG hot and cold legs. In addition to this, we calculated inadvertent boron dilution during the POS of plant start up using α-FLOW code.

For the BWR: (1) core uncover time from the loss of decay heat removal, and (2) fuel heat-up rate at the loss of coolant condition were calculated.

a) What codes have been used to carry out the analyses for LPSD conditions?

The RELAP5 code was used for the analyses of both mid-loop operation of PWR and loss of coolant sequence of BWR. In addition the α-flow code was used for boron dilution of PWR.

b) How does this differ from what was done in the full power PSA?

For the PWR, the RELAP5 mod.2 code was used in full power PSA as well. In addition, the CONTEMPT-LT code was used in full power PSA to calculate the containment sump temperature response during ECCS recirculation operation. The α-flow code is used for LPSD only.

For BWR, we used the same code as the full power PSA.

32. What analysis has been carried out which is unique to LPSD conditions (e.g., are they difficult to analyze, are the sequences analyzed efficiently, are the results credible)?

As mentioned above two unique cooling mechanisms to LPSD conditions have been analyzed for PWR. In order to analyze the very special condition of air-steam mixed low-pressure two-phase flow, low-power natural circulation and phase separation characteristics in reflux cooling, we made a special node arrangement to RELAP5 mod.2 code, in which hot leg node was separated into two nodes corresponding to air-steam phase and liquid phase, respectively. As a result, we could analyze the effect of reflux cooling successfully.

In the analysis of gravity injection of RWST water, some additional nodes and junctions were attached to simulate the gravity injection system (e.g. tank, pipings, valves, etc)

Two additional questions / responses were included:

What areas of your analysis are, in your opinion, most in need of improvement?

It may be necessary to define "how much conservative it has" more precisely in the process of further verification of the code, especially in the cooling model for the uncovered fuel.

What are the areas that would most benefit from further research?

Same as above.

KOREA

30. How has the success criteria used in the LPSD PSA been determined?

Typical examples are as follows:

- The SCS trains can be aligned to the IRWST(when available) and be used to remove decay heat in a feed and bleed operation. Also, two SIS trains are available for feed and bleed (when IRWST is available). As a final method to cool the core, the CVCS can be used to inject coolant water from the Boric Acid Storage Tank (BAST). During some parts of the outage, the steam generators are available for heat removal. No credit was given for the steam generators as a DHR path.

- Also, with a four train Safety Injection System, T/S recommendations will have two SIS trains available in most shutdown modes. For the LOCA analysis, the shutdown modes were grouped in an event tree with similar equipment available. Like as the analysis for LDHR, it is assumed that SGs and SITs are not available as a DHR path or coolant inventory. Because of no enough analysis information for LOCA during shutdown operation, same success criteria of full power PSA were used for each event tree heading.

- When the loss of offsite power is coupled with failure of one EDG and failure of the combustion gas turbine, the Station Blackout (SBO) is caused. There are some mitigation systems to cope with SBO event in APR-1400, such as Auxiliary Feedwater Turbine Driven Pumps and Passive Secondary Cooling System. However these systems cannot be available during shutdown mode operation after Mode 4 due to maintenance. In this analysis, it was assumed that there is no mitigation system during shutdown operation if SBO occurs. Hence, operator must recover only offsite power as soon as possible. If operator cannot recover offsite power before time to core damage that was calculated from the thermal-hydraulic analysis, it was assumed that the event directly leads to the core damage.

31. What analyses have been carried out for LPSD conditions [this would include thermal-hydraulic analysis, neutronics, and analysis of other effects such as cold overpressurization, etc.]?

a) What codes have been used to carry out the analyses for LPSD conditions?

Deterministic T/H codes (e.g., RELAP5) are used.

b) How does this differ from what was done in the full power PSA?

It does not differ with the case of full power PSA.

32. What analysis has been carried out which is unique to LPSD conditions (e.g., are they difficult to analyze, are the sequences analyzed efficiently, are the results credible)?

None. Conservative deterministic assumptions were adopted.

MEXICO

30. How has the success criteria used in the LPSD PSA been determined?

For the POS analyzed, bounding and hand calculation were performed.

31. What analyses have been carried out for LPSD conditions [this would include thermal-hydraulic analysis, neutronics, and analysis of other effects such as cold overpressurization, etc.)?

a) What codes have been used to carry out the analyses for LPSD conditions?

See the above answer

b) How does this differ from what was done in the full power PSA?

See answer to question 30

32. What analysis has been carried out which is unique to LPSD conditions (e.g., are they difficult to analyze, are the sequences analyzed efficiently, are the results credible)?

See answer to question 30

THE NETHERLANDS

30. How has the success criteria used in the LPSD PSA been determined?

31. What analyses have been carried out for LPSD conditions [this would include thermal-hydraulic analysis, neutronics, and analysis of other effects such as cold overpressurisation, etc.)?

a) What codes have been used to carry out the analyses for LPSD conditions?

b) How does this differ from what was done in the full power PSA?

32. What analysis has been carried out which is unique to LPSD conditions (e.g., are they difficult to analyze, are the sequences analyzed efficiently, are the results credible)?

In principle the accident sequences are modelled in the same manner for both power and non-power POSs. The event tree model is the central analytical tool used to determine the various ways to mitigate core damage following an initiating event. It enables to analyze the potential for core damage in terms of the ways that safety and non-safety systems and operators can respond to a particular initiating event. An important part of developing the event trees is to reflect the inherent functional and physical dependencies between each phase in a sequence, and at the same time, model the interaction between operators and systems as the sequence unfolds. The set of systems and operator actions satisfying the safety functions define the set of success criteria for the initiating event and associated event tree.

The success criteria for any initiating event are defined as the minimal number of systems, system trains, or operator actions required to bring the plant to a subcritical state and to adequately remove heat from the

core and containment; ultimately establishing long-term stable conditions and preventing core damage. Stable conditions **vary** and include hot shutdown, cold shutdown, or any other condition where heat removal from the core and from the containment could continue for an extended period of time, with no requirement for additional systems to operate. In cases where primary integrity is breached, then the sequence development is carried out through cold shutdown conditions to ensure any release would be minimal.

Predominantly, the differences in the sequences lie in the differences in IEs, such as PTS, boron dilution scenarios, or loss of RHR. See e.g., the differences in the respective definitions of IEs in the full power PSA and LPSD PSA (see answer to Q11 & Q12). Also the success criteria for a particular IE may differ for the various POSs.

The data for events and success criteria following a boron dilution accident have been developed based on engineering judgment due to a lack of historical, experimental, and simulator data. As an example: "For the hot steaming POS, the conditional probability of reactor pressure vessel failure once the event occurs is estimated to be one. Further recovery events have been ruled out based on the extremely short time to the power excursion and associated pressure pulses, which lead to core damage through RPV rupture. For the cold shutdown and midloop POSs, the conditional probability of RPV failure is estimated to be ½. Further, recovery events in these conditions (manual boration) are modelled based on the slower development of the power excursion (lower core flow rates), lower initial pressures and temperatures, and greater volume for expansion. "

For the determination of the success criteria mainly MAAP-4 was used as a code. In some cases MELCOR was used.

For the Low Temperature Over-Pressurization Event (LTOP) an evaluation for potential human errors of commission was performed because administrative controls play a large role in the plant's defense of LTOP.

SPAIN

30. How has the success criteria used in the LPSD PSA been determined?

See answer to Q28.

31. What analyses have been carried out for LPSD conditions [this would include thermal-hydraulic analysis, neutronics, and analysis of other effects such as cold overpressurization, etc.)?

Some Thermal Hydraulic calculations have been used, most of them generic. Also hand made calculations have been performed for draining down, heating up and vaporization.

a) What codes have been used to carry out the analyses for LPSD conditions?

Not applicable.

b) How does this differ from what was done in the full power PSA?

In the full power PSA, most of the success criteria are based on TH calculations.

32. What analysis has been carried out which is unique to LPSD conditions (e.g., are they difficult to analyze, are the sequences analyzed efficiently, are the results credible)?

As mentioned in Q31, some hand made calculations have proven to be accurate, but slightly conservative. The derived results are then credible and the calculations are simpler and cheaper.

SWEDEN

30. How has the success criteria used in the LPSD PSA been determined?

CSNI Question number	Answer from Swedish plant	Answer
Q30	F1/2	There are the same success criteria as for full power operation (mainly from SAR).
	R1	The same as used in the power operation assessments.
	R2-4	R2, EdF's thermo hydraulic studies for the 900 MWe units. Engineering judgement and information from plant personnel was used to identify differences that could be significant.
	B2	See the answer above
	O1	The same criteria as in full power PSA have been used, see answer to question Q28. Answer only valid for low power PSA, not outage period.
	O2	See Q28
	O3	See Q28
SKI REMARK	Pre defined end states	

31. What analyses have been carried out for LPSD conditions [this would include thermal-hydraulic analysis, neutronics, and analysis of other effects such as cold overpressurisation, etc.)?

a) What codes have been used to carry out the analyses for LPSD conditions?

b) How does this differ from what was done in the full power PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q31a	F1/2	MAAP 4, POLCA, GOBLIN, BISON
	R1	Similar to them in power operations.
	R2-4	R2, No ans
	B2	No well-established codes have been used
	O1	So far no separate analyses have been carried out for LPSD (remember that no shutdown PSA have been performed yet). Answer only valid for low power PSA, not outage period.
	O2	Criteria adopted from full power PSA. Analyses carried out with hand calculations or MAAP
	O3	Criteria adopted from full power PSA. Analyses carried out with hand calculations or MAAP
SKI REMARK	For LP the same analyses as for power operation. SD mainly hand calculations.	

Q31b	F1/2	No difference.
	R1	No difference.
	R2-4	No answer
	B2	No answer
	O1	It doesn't. Answer only valid for low power PSA, not outage period.
	O2	It doesn't
	O3	It doesn't
SKI REMARK	See Q31	

32. What analysis has been carried out which is unique to LPSD conditions (e.g., are they difficult to analyze, are the sequences analyzed efficiently, are the results credible)?

CSNI Question number	Answer from Swedish plant	Answer
Q32	F1/2	
	R1	Straightforward assessments with credible results.
	R2-4	R2, No answer
	B2	Some public reports have been used where generic issues concerning LPSD are handled.
	O1	None. Answer only valid for low power PSA, not outage period.
	O2	Analyses regarding volumes (hand calculations)
	O3	Analyses regarding water volumes (hand calculations)
SKI REMARK	Hand calculations for SD.	

UNITED KINGDOM

30. How has the success criteria used in the LPSD PSA been determined?

By examination and analysis of the fault transient.

31. What analyses have been carried out for LPSD conditions [this would include thermal-hydraulic analysis, neutronics, and analysis of other effects such as cold overpressurisation, etc.)?

a. What codes have been used to carry out the analyses for LPSD conditions?

A modified version of LOFT4 code has been used as well as LOFTSGDT which has been specifically created under appropriate QA procedures. Also RELAP 5 Mod 3.2 for decay heat removal at S/D.

b. How does this differ from what was done in the full power PSA?

The different approach is required because of the very different operating conditions existing, e.g. temperature differences between the primary and secondary circuits.

32. What analysis has been carried out which is unique to LPSD conditions (e.g., are they difficult to analyse, are the sequences analysed efficiently, are the results credible)?

Reverse heat transfer from primary to secondary circuits from staring RCP(s) in error. Cold overpressure faults through starting HHSI pump(s) in error. Code changes necessary (RELAP 5 Mod3.2) to evaluate credible sequences.

UNITED STATES

BWR

30. How has the success criteria used in the LPSD PSA been determined?

Many of the success criteria used in the POS 5 analysis were the same as the success criteria used in the full power analysis. The major reason for this is that typically success criteria are not defined as less than full flow from a pump (e.g., 50%) due to the difficulty of determining and modelling the combinations of failures that would lead to the reduced flow. For those cases where the success criteria were different, various thermal hydraulic calculations were made using either MELCOR or simplified project-developed codes.

31. What analyses have been carried out for LPSD conditions [this would include thermal-hydraulic analysis, neutronics, and analysis of other effects such as cold overpressurisation, etc.)?

Thermal-hydraulic calculations included:

- Determination of the number of low-pressure injection pumps needed for a large loss of coolant accident during low decay heat conditions
- Determination of when certain systems could adequately remove decay heat
- Determination of time to various conditions (e.g., core damage, overpressurisation of piping, and isolation set-points)

a) What codes have been used to carry out the analyses for LPSD conditions?

MELCOR and simplified project-developed codes were used.

b) How does this differ from what was done in the full power PSA?

The major differences lie in 1) the specific conditions that were analyzed (i.e., those unique to POS 5 versus full power), and 2) the codes used to determine the success criteria (e.g., MELCOR for selected POS 5 initiators versus General Electric calculations).

32. What analysis has been carried out which is unique to LPSD conditions (e.g., are they difficult to analyze, are the sequences analyzed efficiently, are the results credible)?

Unique conditions analyzed include:

- Determination of time to core damage from an initially cold (less than 200 degree F) non pressurized reactor vessel state

- Determination of the number of low-pressure injection pumps needed for a large loss of coolant accident during low decay heat conditions

- Determination of time to various conditions (e.g., core damage, overpressurisation of piping, and isolation set-points) from an initially cold (less than 200 degree F) non pressurized reactor vessel state

While the results from these analyses are credible, some of the analyses required long calculation times due to problems with calculational instabilities. (Note: it is unclear whether these same calculational instabilities exist in the current code version.)

PWR

30. How has the success criteria used in the LPSD PSA been determined?

See answer to Q29.

31. What analyses have been carried out for LPSD conditions [this would include thermal-hydraulic analysis, neutronics, and analysis of other effects such as cold overpressurization, etc.)?

a) What codes have been used to carry out the analyses for LPSD conditions?

1) NUREG/CR-6144- MELCOR was used to model gravity injection while the plant is at mid-loop. It was also used to determine the water level in the core when cladding failure occurs, such that a simpler computer program can used this criterion to determine time to core damage for many different scenarios. The simpler computer program models the RCS as a single node with feeding from pumps and relieving through PORVs. The success criteria for reflux cooling were determined by reviewing the analysis performed by INEL and Westinghouse. INEL used RELAP/MOD3, and Westinghouse used TREAT/NC.

2) Screening analysis of NUREG/CR-6144- Timing and success criteria were determined by back-of- the- envelop calculations, i.e. energy and mass balance calculations. For each POS when RHR is initially running, a conservative estimate of time to core damage was calculated based on the plant condition. The cold pressurization scenario was based on two challenges to the system. First, the RCS heats up subsequent to a loss of RHR. Thermal expansion of the coolant causes the pressuriser to become solid. A subsequent rapid increase in RCS pressure causes the RHR system to become over-pressurized. It was assumed that the pressure increase was too fast for the relief valves to respond. Second, when the bulk coolant temperature in the RCS reaches the saturation temperature, boiling starts in the vessel, and the relief valves are relieving liquid. When the decay heat is high, at the design pressure of the RHR system, the rate at which steam is generated in the vessel is higher than the combined relief capacity of the relief valves.

3) NUREG/CR-6616 and 5718- The simple computer program of 1) also has a node modelling the secondary side of the steam generators, and allows modelling heat removal through the stem generators. It was used to determine the timing and success criteria. It was also used in determining the time to over pressurization of the RHR system.

b) How does this differ from what was done in the full power PSA?

The basic physical principles are the same. But the plant configurations and the initial physical conditions are different during shutdown.

32. What analysis has been carried out which is unique to LPSD conditions (e.g., are they difficult to analyze, are the sequences analyzed efficiently, are the results credible)?

See also answer to Q31.

1) NUREG/Cr-6144- Reflux cooling was difficult to model. In a separate study, NUREG/CR-5819, the reactivity accident identified by the French was analyzed. It was found that the mixing of injected un-borated coolant and borated coolant in the cold leg was difficult to model.

2) Screening analysis of NUREG/CR-6144- The over pressurization scenario was analyzed based on simple hand calculations. More detailed analysis is desirable.

3) NUREG/CR-6144 and 5718- The over pressurization scenario was analyzed based on the same considerations of 2). More detailed analysis is desirable.

APPENDIX E - SYSTEM MODELING

PRELUDE AND QUESTIONS

The practice of converting system models developed for full-power PSAs to models for LPSD PSAs appears to be popular among risk analysts as they attempt to efficiently use limited resources. However, it is not clear that all different facets of shutdown operations are well incorporated when modifying full power models. One concern is that the LPSD systems analysis needs to address system interactions and dependencies that, due to system realignments that take place during outages, may be different than those at full power. Some PSA users have expressed the opinion that "standardized" guidance and criteria for modifying full-power system models for LPSD PSAs is needed to ensure that models are accurately modified.

33. How has the safety system modelling been carried out in the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

RESPONSES

Belgium

33. How has the safety system modelling been carried out in the LPSD PSA?

The safety system fault trees for power modes have been used. However unavailabilities due to preventive maintenance have been included. In addition in some particular situations, system alignments have been altered (for instance hot leg alignment of safety injection during cold shutdown for intervention).

a) How does this differ from what was done in the full power PSA?

See above

Germany

33. How has the safety system modelling been carried out in the LPSD PSA?

The safety-system modelling has been carried out by modifying the fault trees from the full power PSA.

a) How does this differ from what was done in the full power PSA?

Unavailabilities of safety systems due to planned maintenance are considered in LP&SD modelling.

Hungary

33. How has the safety system modelling been carried out in the LPSD PSA?

The systems modelled were categorized as main (or front line) and support systems based on the role they play during accident mitigation and on their functional interconnections. Main systems are those ones that fulfil the various functions represented by the nodes (headers) of the event trees. Successful operation of the main systems generally depends on the operation of other auxiliary systems, these are called support systems. Failure of a support system leads, in most cases, to unavailability of multiple systems. Dependency matrices that have a significant role during integration (merging) of the different fault trees describe the functional interconnections between systems.

During system analysis the main tasks were to (1) identify those basic events/conditions which can lead to the functional failure of a given system with respect to a given success criterion and (2) define the logical relations between basic events and system failures (top events). These logical relations were described in a structured graphical form of fault trees, which have the following characteristics:

1) Modular structure due to the small event tree/large fault tree approach.

2) Based on this approach the front line and support system fault trees were developed in parallel in separate fault tree modules, and the relevant modules were linked through so-called transfer gates that were defined in accordance with the system-related dependency matrix. The use of transfer gates and boundary conditions is very much supported by the Risk Spectrum PSA code package.

3) Complex structure due to integrated functional system analysis.

4) During fault tree development one large fault tree was produced for each system, which integrates numerous fault tree parts valid under different operational and emergency conditions. The fault tree sections for a specific set of boundary conditions can be activated within the general system fault tree by using so-called house events describing the boundary conditions.

5) Multilevel component failure modes.

6) The failure modes and the associated component boundaries represented in the fault trees were partly defined by standard physical component boundaries, partly by available component reliability data. This means that component boundaries and failure modes were determined on a case-by-case basis, especially during modelling of electrical and I&C support systems.

a) How does this differ from what was done in the full power PSA?

The basic procedure used for system modelling was the same as the one applied for full power conditions. However some important features of low power and shutdown operations had to be taken into account. These were as follows:

- In the various POSs in low power an shutdown conditions the scope of initiating events and the development of accident sequences may be different from that of full power operation due to differences in reactor state, availability of mitigating systems as well as due to the numerous operational and maintenance activities. These features required treatment of more initiating events and accident sequences as compared to the full power PSA.

- Human interactions have even greater importance than in the full power case. Special attention was paid to modelling and quantifying various types of inappropriate human actions as given in Section 8.3.2.

Because of the above characteristics the fault tree logic developed for low power and shutdown conditions is more complex than that of full power operation of the reactor.

Japan

33. How has the safety system modelling been carried out in the LPSD PSA?

We developed Event Trees for each POS of each initiating event by locating credible mitigation systems and operator actions into event headings, which were chosen based on the analyses of accident sequences performed in the process of defining success criteria. The Fault Tree method was used, referring to AESJ-SC-P001 and NUREG –0492, to estimate unavailability of each mitigation system in the Even Tree headings.

a) How does this differ from what was done in the full power PSA?

The above mentioned method is the same as that of full power PSA except for the following in the PWR. One is that all safety systems (except emergency diesel generator) are assumed to start by manual actions only. Second is that credit is taken for an automatic switchover of charging pump suction from volume control tank to RWST in case the volume control tank decreases to low water level.

Two additional questions / responses were provided in the response:

What areas of your analysis are, in your opinion, most in need of improvement?

Analysis of reflux cooling effect by the steam condensation in SG could be improved.

What are the areas that would most benefit from further research?

We do not find any specific concern at the moment.

Korea

33. How has the safety system modelling been carried out in the LPSD PSA?

The basic assumptions in the fault trees are same with the full power PSA. The fault trees are modified versions of those used in the full power PSA, reflecting plant configuration changes for each POS.

a) How does this differ from what was done in the full power PSA?

When comparing with the full power PSA, there is no difference in the system modelling.

Mexico

33. How has the safety system modelling been carried out in the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

Basically the approach used in full power is the same used for LPSD system modelling. The system fault trees for full power were modified, when necessary, to represent new top events or developing new fault trees for systems not modelled in full power.

The Netherlands

33. How has the safety system modelling been carried out in the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

For the LPSD PSA either the existing models have been converted, with a special emphasis on the HRA, or separated fault trees have been developed. Also a special study made on the human errors of commission during LPSD states.

Spain

33. How has the safety system modelling been carried out in the LPSD PSA?

Systems (safety and non safety related) have been modelled by means of large fault trees where the support systems are analyzed separately from the frontline systems and linked for quantification. Different boundary conditions are taken into account by means of "house events".

a) How does this differ from what was done in the full power PSA?

The system analysis methodology is the same as in full power PSA.

Sweden

33. How has the safety system modelling been carried out in the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q33	F1/2	LP nearly full scope. Simplified for shutdown
	R1	Similar to power operation analysis.
	R2-4	R2, Simplified modelling based on engineering judgement
	B2	Engineering judgements
	O1	The same system requirements as in full power PSA have been used. The only change is that separate event trees have been developed for low power PSA. Answer only valid for low power PSA, not outage period.
	O2	With modified fault trees from full power PSA

	O3	With modified fault trees from the full power PSA
SKI REMARK	See Q27.	
Q33a	F1/2	Full scope for full power (all dependencies)
	R1	No big changes.
	R2-4	R2, Simplified FTs
	B2	No FT-analysis was performed
	O1	It doesn't. Answer only valid for low power PSA, not outage period.
	O2	It doesn't
	O3	It doesn't
SKI REMARK	See Q27	

United Kingdom

33. How has the safety system modelling been carried out in the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

The systems availability and mode of operation, e.g. auto/manual has been specifically identified by each of the 'potential' operating states identified.

United States

BWR

33. How has the safety system modelling been carried out in the LPSD PSA?

Typically, fault trees have been developed for the various top events in the event trees. In some cases, data values have been used for systems (top events) if sufficient data are available or detailed system modelling is not desired or needed.

a) How does this differ from what was done in the full power PSA?

The same techniques have been used in full power PRAs.

PWR

33. How has the safety system modelling been carried out in the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

1) NUREG/CR-6144- The fault tree models, developed as part of NUREG-1150 study, were reviewed and modified (when necessary) to develop two fault tree models for the plant applicable to shutdown and to low power operation for each system. The system configuration during shutdown was identified by reviewing the operating procedures used during shutdown, shift supervisor's log books, and the system training manual.

2) Screening analysis of NUREG/CR-6144-The NUREG-1150 fault trees were used with modifications to include the maintenance unavailabilities estimated using plant experience for different POSs.

3) NUREG/CR-6616 and 5718- The fault trees developed in 1) were used.

APPENDIX F - DATA

PRELUDE AND QUESTIONS

The data needs (e.g., data for initiating event frequencies, component failure probabilities, component unavailabilities, and common cause failure probabilities) are generally the same for LPSD and full power PSAs. However, the probability estimates used may need to be different.

Initiating event frequencies estimated for LPSD operations may be different from those at full power operations as a result of taking into consideration the higher level of maintenance activity during shutdown conditions. This factor needs to be addressed in assigning the initiating event frequencies. Estimation of initiating event frequencies should take into account the actual plant conditions for each POS modelled. For example, for some events (e.g., drain-down events) the initiating event frequency is highly dependent on plant configuration. One question is whether the currently available initiating event frequency estimates reflect actual LPSD experience.

Failure rates estimated for *systems in standby* during full power operations may be inappropriately used in analyses for shutdown operations during which these systems are *normally* operating. Also, system interactions and dependencies at LPSD may be different than those at full power due to system realignments that take place during outages. Therefore, methods and data used to derive probability estimates for common-cause and dependent failures for full power analyses may not be appropriate for shutdown analyses.

34. How have the data used in the LPSD PSA been derived? [The reply to this question should address the data used for initiating event frequencies, components failure probabilities, the time spent in each of the POSs, etc.]

35. How component unavailabilities due to maintenance were used in the LPSD PSA?

a) How have the unavailabilities been justified?

b) How does the justification of the unavailabilties take into account operating experience?

36. How does the LPSD data compare to the corresponding data used in the full power PSA?

a) Where there are changes?

b) Why they have been necessary?

37. How have the common-cause failure probabilities used in the LPSD PSA been derived?

a) How does this compare with what was done in the full power PSA?

b) Where there are changes?

c) Why they have been necessary?

RESPONSES

Belgium

34. How have the data used in the LPSD PSA been derived? [The reply to this question should address the data used for initiating event frequencies, components failure probabilities, the time spent in each of the POSs, etc.]

The initiating event frequencies have been based on operating experience (example loss of RHR loop) and generic data (example LOCAs). In the majority of the cases this implied using the frequency of the same initiator during power operation, and weighing the frequency of this initiator for the shutdown POS based on the time spent in this particular POS.

The component failure probabilities are identical.

The time spent in each of the POS has been derived from a recent plant operating profile.

35. How component unavailabilities due to maintenance were used in the LPSD PSA?

Two types of component unavailabilities due to maintenance have been included: (1) preventive maintenance; (2) corrective maintenance.

a) How have the unavailabilities been justified?

The unavailabilities have been based on plant specific operating practices and operating experience on equipment availabilities.

b) How does the justification of the unavailabilties take into account operating experience?

See above.

36. How does the LPSD data compare to the corresponding data used in the full power PSA?

The system component failure data are identical.

a) Where there are changes?

Not relevant.

b) Why they have been necessary?

Not relevant.

37. How have the common-cause failure probabilities used in the LPSD PSA been derived?

a) How does this compare with what was done in the full power PSA?

Identical.

b) Where there are changes?

Not relevant

c) *Why they have been necessary?*

Not relevant

Germany

34. How have the data used in the LPSD PSA been derived? [The reply to this question should address the data used for initiating event frequencies, components failure probabilities, the time spent in each of the POSs, etc.]

The initiating event frequencies are mainly based on operating experience. In some cases of dependencies between IE and response of the safety systems there have been performed fault trees which were connected in the event tree.

The component failure probabilities have been mainly derived from the full power PSA. Data of components which were not used in the full power PSA had been derived via a plant specific or if not possible via a generic evaluation.

The time of the POSs was derived from the standard refuelling outage (14 days).

35. How component unavailabilities due to maintenance were used in the LPSD PSA?

Conservatively the minimal permitted equipment has been modelled according to the operational manual.

a) *How have the unavailabilities been justified?*

It is not allowed to fall below the minimal permitted equipment.

b) *How does the justification of the unavailabilties take into account operating experience?*

Insights from operation experience was used for the determination of unavailabilities of credited operating systems.

36. How does the LPSD data compare to the corresponding data used in the full power PSA?

a) *Where there are changes?*

Some new data has been determined for components which had not been considered in full power PSA.

The time to failure of tested components and the mission times of components in operation have been adapted to LP&SD conditions.

b) *Why they have been necessary?*

Some components have been tested or demanded during the shut down process.

37. How have the common-cause failure probabilities used in the LPSD PSA been derived?

The CCF probabilities have been mainly derived from full power PSA.

a) *How does this compare with what was done in the full power PSA?*

b) *Where there are changes?*

In the case of CCF of the RHR-pumps a new data set has been evaluated due to the failure mode: pump fails to run. In full power PSA only the failure mode: pump fails to start has been considered.

c) *Why they have been necessary?*

In the full power PSA the RHR-System is a stand by system. The CCF-failure mode is dominated from "fails to start". In LP&SD the RHR-System is in operation.

Hungary

34. How have the data used in the LPSD PSA been derived? [The reply to this question should address the data used for initiating event frequencies, components failure probabilities, the time spent in each of the POSs, etc.]

During the estimation of the initiating event frequencies, first the associated initiators have been studied in each plant operational state (POS) separately. Quantification has been performed for the effects identified including (1) internal causes, and others like (2) erroneous change of the operational loop, (3) erroneous draining of the operational loop, (4) maintenance activities, (5) erroneous system alignment, (6) heavy load drop, and (7) other erroneous interactions. Occurrence frequency of an initiating event was quantified due to each effect identified, then these frequencies were combined for the initiating events POS by POS.

Estimation of the failure rates, in general, was based on the database developed for the level 1 PSA studies for nominal power operational mode. This database includes generic as well as plant-, or unit-specific data. Failure rates were determined in the following steps:

- First, data from eight different, internationally accessible sources (e.g. IAEA-TECDOC-478, EIREDA, Swedish T-Book, IEEE database, etc.) were gathered, and they were combined by the approach described in the German Risk Study. It assumes that point values of the source data follow a lognormal distribution, thus the combined mean value and the error factor were defined by putting them into an appropriate coordinate system. The latter pair of data characterized the prior distribution of the generic data.

- Second, operational data were gathered and analyzed, then the plant-specific and generic data were combined by the Bayes method. During this the prior distribution of the generic data was corrected by the plant-specific information. The resulting posterior distribution was characterized again by its mean value and error factor.

In order to validate the above data for the low power and shutdown states the source databases were reviewed from the point of view of whether or not they take into account changing operational characteristics in their reliability data. Since none of the databases supported such a distinction in the data for different operational states, generic data used in the level 1 PSA studies for nominal power operational mode have been used also in the LPSD PSA study. In addition, plant-specific data as well as combined reliability data have been updated based on the operational experience gathered since the last PSA studies for nominal power.

Duration of each plant operational state identified was determined by a detailed review of shutdown schedule plans and past operational records.

35. How component unavailabilities due to maintenance were used in the LPSD PSA?

POSs were defined in such a way that a component in a given POS is either on maintenance (during the whole POS duration) or not. In the first case component unavailability is given as TRUE event due to the fact that components on maintenance are really unavailable, while in the latter case the "nominal" unavailability (originating mainly from the full power PSA) is used. Increased unavailability due to greater level of activity at the plant was taken into account through type A human errors introduced into the model as a result of the HRA.

a) How have the unavailabilities been justified?

It is considered that this kind of true unavailability does not need justification.

b) How does the justification of the unavailabilties take into account operating experience?

See above.

36. How does the LPSD data compare to the corresponding data used in the full power PSA?

For data compilation extensive use was made of the component reliability data base of the level 1 PSA for full power operation as follows:

- The data bases of the earlier full power level 1 PSA were reviewed and, as a result, component reliability data were set up for the purpose of LPSD PSA.
- A review of available data sources revealed that there was no sufficient evidence or data base support to adjust component failure rates in accordance with lower plant parameters in most low power and shutdown POSs (e.g. lower pressure and temperature of medium). Therefore failure rates used in the full power PSA were assumed valid in low power and shutdown conditions too.

Time related information (T_{mis}, T_{rep}, T_{per}, T_{test}) was determined on a case-by-case basis for each component failure mode as strictly plant specific data.

It is mentioned that the mission time (T_{mis}) of each component was defined in accordance with system success criteria determined during the development of accident sequence models. Since system success criteria are dependent on the accident sequences, mission time of a given component may vary over the accident sequences to reflect varying system requirements in terms of expected time length of failure free operation. Mission times applied in the LPSD PSA model range from 1.5 hour (e.g. for high pressure ECCS in injection mode) to 336 hours (e.g. for low pressure ECCS in recirculation mode). Mission times were developed on a case by case basis.

Time data related to test and repair (T_{rep}, T_{per}, T_{test}) were defined based on plant operational experience (T_{rep}), and on limiting conditions of operation prescribed in the Technical Specifications and operational procedures (T_{per}, T_{test}).

In general, repair times were estimated to be similar to the values used in the PSA for full power operation. Although in some cases there may be slight differences in times to repair of the same

component under different conditions, their effect on component availability was assumed to be negligible and thus average repair times were used.

The same values were used for test intervals within the whole LPSD PSA model. In most cases this approach is slightly conservative because the average unavailability calculated from the test interval is usually higher than real in low power and shutdown modes (i.e. that could be obtained by taking into account in each POS the time elapsed after the last system test during an outage). For specific outages, when a given component is near the end of its test interval, use of average unavailability may result in some underestimation of component unavailability, but its effect was found negligible on the final results. During tests the systems are ready to operate on demand, i.e. unavailability of components due to tests was generally not considered.

a) Where there are changes?

There are differences in the mission times due to the differences in system success criteria.

b) Why they have been necessary?

The accident progression after an initiating event occurring in shutdown conditions is slower than in the case of the full power PSA due to the lower level of decay heat. In addition, the generic mission time in the full power PSA was 24 hours, while in the LPSD PSA the processes are followed until the ultimate success (in the worst case until the core is fully unloaded). Due to this there are sometimes much longer mission times in the case of the LPSD PSA than in the full power PSA.

37. How have the common-cause failure probabilities used in the LPSD PSA been derived?

A parametric model was applied to model the effects of residual dependencies in which identification of common cause component groups and quantification of CCF probabilities were performed similarly to the level 1 PSA for full power operation. Basically the same CCF groups were included in the PSA for low power and shutdown operating modes as in the PSA for plant operation at full power. It is considered that the important root causes and coupling mechanisms, especially the design and production related ones, are independent of the plant operational states modelled in the LPSD PSA. The original list of CCFs (taken from the Paks PSA for full power operation) was reviewed by a group of experts (plant engineers and PSA analysts), and no modification was proposed.

a) How does this compare with what was done in the full power PSA?

See above.

b) Where there are changes?

There are basically no differences between the CCF modelling and quantification in the LPSD PSA and in the full power PSA.

c) Why they have been necessary?

See above.

Japan

34. How have the data used in the LPSD PSA been derived? [The reply to this question should address the data used for initiating event frequencies, components failure probabilities, the time spent in each of the POSs, etc.]

1) Initiating event frequencies

We calculated initiating event frequencies based on the operating experiences of domestics PWRs through March, 1993. In case there is no actual occurrence for specific initiating event, we calculated frequency for this initiating event by assuming log-normal distribution with upper bound of one occurrence and error factor of 10. In case of LOCA frequency in BWR, however, a factor of one-tenth is applied to the calculated result to the full power PSA.

2) Component failure probabilities

We basically use the same component failure probability data in the LPSD PSA as those of full power PSA. These include mechanical component failure data from US-LER, electric components failure data from US-IEEE std 500, and failure data of starting emergency diesel generators from Japanese operating experiences. We also consider the restoration of off-site power and emergency diesel generators based on Japanese operating experiences.

3) The time spent in each of the POSs

We determine the time spent in each of the POSs referring to the schedules of actual refuelling and maintenance outages of domestic PWRs ,in which various schedules are averaged into a single standard.

35. What component unavailabilities during maintenance were used in the LPSD PSA?

Component unavailabilities for major safety-related components that are placed out of service for planned maintenance in LPSD period were used, which include systems or components such as High Pressure Injection System, Charging Pump, Auxiliary Feed Water Pump, Component Cooling Water Pump, Component Cooling Seawater Pump and Diesel Generator for PWR, and RHR, ECCS , make-up water system and fire protection system for BWR.

a) How have the unavailabilities been justified?

The out-of-service conditions of above system or components have been justified by referring to a standardized maintenance outage schedule established by domestic operating experiences as well as using the fault tree (FT) method and the same component failure probabilities as the full power PSA.

b) How does the justification of the unavailabilties take into account operating experience?

The above refuelling and maintenance schedule is derived based on the accumulated experiences of Japanese plants. However, according to the objectives of LPSD PSA, we make evaluations using the typical refuelling and maintenance schedules.

36. How does the LPSD data compare to the corresponding data used in the full power PSA?

We developed data basically based on operating experiences just like the full power PSA. The only differences from the full power PSA is the consideration of planned outage of safety related system or

components due to regular maintenance. However, we used the same component failure probabilities as the full power PSA.

a) Where there are changes?

See above answer.

b) Why they have been necessary?

The difference is to reflect the different configuration of safety system or components between LPSD and full power.

37. How have the common-cause failure probabilities used in the LPSD PSA been derived?

We use beta-factor method to estimate common-cause failure probabilities. Beta-factor values for specific components are basically derived from NUREG-1150

a) How does this compare with what was done in the full power PSA?

Beta-factor values for LPSD are essentially the same as those of full power PSA.

b) Where there are changes?

There is no change.

c) Why they have been necessary?

There is no change.

Two additional questions / responses were provided

What areas of your analysis approach are, in your opinion, most in need of improvement?

Improving data base in the area of component failure probability and common cause failure probability as well as human error data base may be necessary, because data of full power PSA are essentially used in LPSD PSA without considering the specific condition of LPSD.

What are the areas that would most benefit from further research?

The accuracy of failure probability of systems and human error may be improved.

Korea

34. How have the data used in the LPSD PSA been derived? [The reply to this question should address the data used for initiating event frequencies, components failure probabilities, the time spent in each of the POSs, etc.]

The APR-1400 plant has no operating experience and shutdown experience and thus the initiating event frequency is not fit to APR-1400 exactly. Therefore, the initiating event frequency has great uncertainty. Anyway, the APR-1400 analysis used the frequencies developed in the BNL study, NUREG/CR-6144. The use of these frequencies is quite conservative because the frequencies are based on historical occurrences and do not reflect any lessons learned, procedure improvements, or training improvements.

The APR-1400 design goal of forced shutdown frequency is 0.8/year but the frequency of the old plant is about 7/y in the 1970s. Also for the new plants the shorter overhaul duration and longer fuel cycle than old plants are achieved due to improved and enhanced maintenance equipment and higher fuel enrichment. Also, with the current emphasis on shutdown risk, such as training for mid-loop operation to the operators and preparation of operating/emergency procedures, initiating event frequencies in these modes will be decreased. The design changes that could reduce the frequency of certain events were not accounted in these initiating event frequencies. For example, there are four redundant and diverse level measuring instruments for reduced inventory operation which will reduce the frequency of loss of DHR from pump cavitation in that mode.

Since the APR-1400 is under design stage and the plant-specific outage plan is now being developed, the time duration for each POS during shutdown was collected from Yonggwang Unit 3&4 overhaul experiences. The collected outage time was converted to the appropriate form. The time in unplanned forced outages equals to 5 days/year, the forced outage specification in the EPRI URD. The fractions of outage time duration for forced outages were estimated using NUREG/CR-6144. The outage schedule assumes that the nozzle dams will be used for SG maintenance and minimize their time on reduced inventory operation. Also it is assumed that the refuelling will be performed every 18 months.

The LOOP frequency used in the APR-1400 full power PSA was taken directly to the shutdown PSA. This LOOP frequency was calculated as occurrence rate per hour at first, and then the occurrence rate multiplied by outage times for each POS.

35. How component unavailabilities due to maintenance were used in the LPSD PSA?

a) How have the unavailabilities been justified?

When comparing with the full power PSA, there is no difference in the quantification of maintenance unavailability.

b) How does the justification of the unavailabilties take into account operating experience?

Not applicable.

36. How does the LPSD data compare to the corresponding data used in the full power PSA?

a) Where there are changes?

No changes were made.

b) Why they have been necessary?

37. How have the common-cause failure probabilities used in the LPSD PSA been derived?

a) How does this compare with what was done in the full power PSA?

When comparing with the full power PSA, there is no difference in the use of common-cause failure probabilities.

b) Where there are changes?

Not applicable.

c) Why they have been necessary?

Not applicable.

Mexico

34. How have the data used in the LPSD PSA been derived? [The reply to this question should address the data used for initiating event frequencies, components failure probabilities, the time spent in each of the POSs, etc.]

Generic data reported in NUREG/CR 6143 for initiating events frequency and component failure have been used due to the lack of LVNPP specific data .

35. How component unavailabilities due to maintenance were used in the LPSD PSA?

a) How have the unavailabilities been justified?

Explicit plant configuration was assessed to consider only available components.

b) How does the justification of the unavailabilties take into account operating experience?

POS for cold shutdown and refuelling involves major unavailabilities for maintenance, therefore there is not enough work done until we reach those phases of the study.

36. How does the LPSD data compare to the corresponding data used in the full power PSA?

a) Where there are changes?

See answer to question 34

b) Why they have been necessary?

See the above answer

37. How have the common-cause failure probabilities used in the LPSD PSA been derived?

a) How does this compare with what was done in the full power PSA?

The study is working with bounding calculations and dependencies and common cause failures come from the models already used for full power, if appropriate. Specific common cause failures related with LPSD conditions have not been assessed at this point.

b) Where there are changes?

See the above answer

c) Why they have been necessary?

See the above answer

The Netherlands

34. How have the data used in the LPSD PSA been derived? [The reply to this question should address the data used for initiating event frequencies, components failure probabilities, the time spent in each of the POSs, etc.]

35. How component unavailabilities due to maintenance were used in the LPSD PSA?

a) How have the unavailabilities been justified?

b) How does the justification of the unavailabilties take into account operating experience?

36. How does the LPSD data compare to the corresponding data used in the full power PSA?

a) Where there are changes?

b) Why they have been necessary?

37. How have the common-cause failure probabilities used in the LPSD PSA been derived?

a) How does this compare with what was done in the full power PSA?

b) Where there are changes?

c) Why they have been necessary?

As part of the living PSA program, plant-specific data has been collected and evaluated for several types of components. The analyses of the plant-specific data used in the Borssele LPSD PSA have provided information on the following events:

- Test and maintenance unavailabilities for basic events that contribute significantly to the total core damage frequency
- Failure rates for three types of components: diesel-generators, pumps, and several types of valves
- Common cause failure data
- Initiating event data

These data include both power and non-power operation of the plant. Plant specific data on failures has been gathered for the following components:

- MOVs
- Safety Valves
- Isolation Valves
- Pumps

- Diesel Generators
- Cooler units
- Compressors
- Ventilators
- Heat exchangers
- Transmitters

For other electrical components such as switches, relays and breakers generic EG&G data are used.

For the Borssele PSA it was decided that the unavailability data for valves and the fail-to-start data for pumps and diesel generators is best evaluated as a function of the test frequency. The remaining unavailability data is evaluated as a function of the number of demands, with the exception of the fail-to-run data, which is dependent of the mission time. For valves and fail-to-start data for pumps and DGs these data have been transformed to hourly failure rates via: $\lambda = 2Q/T$, where: λ = failure rate [1/hour], Q = failure probability on demand [dimensionless] and T = test interval [hour].

The major components that are operated often, such as valves and pumps, have been examined to see if their failure history might have a different pattern from a normal operating component or standby component. This was also done for components which are not tested or demanded on a frequent basis (e.g., the sump recirculation valves).

In order to obtain test intervals, all demands on the relevant components have been collected. For this, the following steps have been taken:

- Investigation of all test procedures on demands of the components of interest.
- Interview of maintenance personnel in order to obtain information on components that are cycled more often than in the periodic tests
- Use the PSA schematics of the PSA-report to find the components that are implicitly tested during periodic tests (check valves)

To generate plant-specific component unavailabilities, the plant specific data are combined with generic data using a Bayesian update process.

- The plant-specific data development includes both power and non-power states. For example, the low pressure RHR-pump data come from operation during the cold shutdown and midloop POSs. It is assumed that the failure rates selected remain constant through all operational states.

Test and maintenance unavailabilities during non-power operations include the unavailability due to scheduled maintenance that is normally permitted during yearly refuelling outages.

Dependent failures are modelled in two ways for NPP Borssele. Explicit modelling of dependent failures has been conducted for those failures which could be specifically identified and quantified. Any residual dependency is accounted for through parametric common cause failure modelling [α-factor model and

impact vector approach (NUREG/CR-4780 and EPRI TR-1003822)]. NUREG/CR-4550 data base is used for modelling of events not listed in the EPRI data base.

Spain

34. How have the data used in the LPSD PSA been derived? [The reply to this question should address the data used for initiating event frequencies, components failure probabilities, the time spent in each of the POSs, etc.]

In general, component failure probabilities and rates are obtained through a Bayesian analysis from generic data and plant specific experience. The same approach is used for IE frequencies where applicable. The POS duration is based on plant experience, but in some cases, only on the most recent experience as the outage programming methodologies have changed throughout the time.

35. How component unavailabilities due to maintenance were used in the LPSD PSA?

In some cases, the unavailability of a train or a component defines by itself the POS, in other cases the component is unavailable during a fixed or expected fraction of the POS or, finally, the occurrence of unavailability is random.

a) How have the unavailabilities been justified?

The unavailabilities are obtained from the outage programme, the Tech Specs and maintenance files, that means, from plant specific information.

b) How does the justification of the unavailabilties take into account operating experience?

Both the definition and the numeric values of unavailability are based on the specific operational experience.

36. How does the LPSD data compare to the corresponding data used in the full power PSA?

Most of the items in the database are common to both full power and LP&SD PSA. Even, the 24 hours as mission time has been adopted for comparison purposes.

a) Where there are changes?

Not applicable.

b) Why they have been necessary?

Not applicable.

37. How have the common-cause failure probabilities used in the LPSD PSA been derived?

The methodology used in LP&SD PSAs is the same as in full power PSAs. It was, mainly, the beta factor method and, sometimes, the Multiple Greek Letter, as required.

a) How does this compare with what was done in the full power PSA?

See previous answer.

b) Where there are changes?

Not in the methodology, but in the application to specific populations, as modelled in the fault trees.

c) Why they have been necessary?

See previous answer.

Sweden

34. How have the data used in the LPSD PSA been derived? [The reply to this question should address the data used for initiating event frequencies, components failure probabilities, the time spent in each of the POSs, etc.]

CSNI Question number	Answer from Swedish plant	Answer
Q34	F1/2	LPSD specific data for LOCA and loss of RHR. HRA for recoveries use Swain Handbook data.
	R1	"T-boken" a Nordic project has been used as input of the assessments.
	R2-4	R2, transient statistics from plant and estimates from EdF based on interviews with operators, component failure data from Ringhals 2 power state operation PSA (T-book+ other) and data from EdF, time spend in each POS based on an "average" outage
	B2	The T-book, transient statistics, engineering judgements
	O1	Initiating event frequencies have first been derived from the full power frequency. In order to calculate the frequency for low power PSA frequencies from full power PSA have been scaled down with respect to the time spent in each POSs. If it has become clear that some operator action is capable of causing the initiating event in question the HEP for that action has been estimated with a screening model. The HEP has then been added to the frequency for full power, it may also replace the "full power frequency". Statistics for low power conditions have also been used fore some initiating events. The same component failure data as in full power PSA have been used. During low power (shutdown and start-up) conditions the system configuration isn't changed compared to full power condition. Also the same mission time as in full power PSA have been used, mostly 20 h. The times spent in each of the POSs have been estimated from interviews and from time schedules from recent years. During start-up no components are considered to be taken out for maintenance. Answer only valid for low power PSA, not outage period.
	O2	Full power PSA data has been modified with respect to primary system pressure and time spent in each of the POSs. Component data was not changed
	O3	Full power PSA data has been modified with respect to primary system pressure and time spent in each of the POSs. Component data was not changed
SKI REMARK	T-book, Swain-diagrams, time windows	

35. How component unavailabilities due to maintenance were used in the LPSD PSA?

a) How have the unavailabilities been justified?

b) How does the justification of the unavailabilties take into account operating experience?

CSNI Question number	Answer from Swedish plant	Answer
Q35	F1/2	
	R1	Yes.
	R2-4	R2, Conservatively, for each POS only minimum equipment required according to the technical specifications is modelled
	B2	Engineering judgements
	O1	During shutdown the same unavailability due to maintenance as in full power PSA have been used. No maintenance is assumed to be performed during start-up conditions. Answer only valid for low power PSA, not outage period.
	O2	As usual
	O3	As usual
SKI REMARK	T-book data	
Q35 a	F1/2	Modelling/utilisation of system demand consideration the UH that is performed under shut down.
	R1	Yes.
	R2-4	R2, Conservative first approach
	B2	Engineering judgements, operation statistics
	O1	The data source used for component unavailability does not take the plant-operating mode into account, i.e. there is no difference between component data for full power and low power. No specific justification has been made. Answer only valid for low power PSA, not outage period.
	O2	See Q34
	O3	see Q34
SKI REMARK	See Q35	
Q35 b	F1/2	See Q35a
	R1	Yes.
	R2-4	R2, No answer
	B2	To a very limited extent
	O1	As stated in answer to question Q11 a survey of LERs for the last ten years of operation have been made. One of the reasons for this was to see if something could be found that indicated that the same component unavailabilities as for full power conditions couldn't be used. Also see answer to Q35a. Answer only valid for low power PSA, not outage period.
	O2	It doesn't
	O3	It doesn't
SKI REMARK	See Q35	

36. How does the LPSD data compare to the corresponding data used in the full power PSA?

a) Where there are changes?

b) Why they have been necessary?

CSNI Question number	Answer from Swedish plant	Answer
Q36	F1/2	The same methods for data collection and analysis have been used.
	R1	Where appropriate outage data was used.
	R2-4	R2, No answer
	B2	No answer
	O1	See answer to question Q35. Answer only valid for low power PSA, not outage period.
	O2	They should be proportional
	O3	They should be proportional
SKI REMARK	Proportional to time window if statistical data is used. Mostly HRA though.	
Q36 a	F1/2	No
	R1	No
	R2-4	R2, No answer
	B2	No answer
	O1	No maintenance is assumed to be performed during start-up conditions, otherwise no changes. Answer only valid for low power PSA, not outage period.
	O2	See Q34
	O3	They should be proportional
SKI REMARK	See previous questions Q34 Q35	
Q36 b	F1/2	No
	R1	N. A.
	R2-4	R2, No answer
	B2	No answer
	O1	It is a realistic assumption and if not made unrealistic cutsets would have been derived. Answer only valid for low power PSA, not outage period.
	O2	See Q34
	O3	See Q34
SKI REMARK	See previous questions Q34 Q35.	

37. How have the common-cause failure probabilities used in the LPSD PSA been derived?

a) How does this compare with what was done in the full power PSA?

b) Where there are changes?

c) Why they have been necessary?

CSNI Question number	Answer from Swedish plant	Answer
Q37	F1/2	The same as for full power operation and use of common load model and Alfa factor model.
	R1	By actual data.
	R2-4	R2, Limited CCF modelling, (due to simplified FTs)
	B2	The treatment of CCF is limited
	O1	The same system configuration and requirements as in full power PSA gives that the same CCF-parameters have been used in low power PSA. Answer only valid for low power PSA, not outage period.
	O2	They are taken from full power PSA
	O3	They are taken from the full power PSA
SKI REMARK	The CCFs are automatically taken into account in the Riskspectrum software. Mostly for LP. Very limited for SD.	
Q37 a	F1/2	Same
	R1	Similar to the full power PSA.
	R2-4	R2, CCF analysis is performed in full power PSA
	B2	CCF analysis is performed in full power PS
	O1	There is no difference. Answer only valid for low power PSA, not outage period.
	O2	They are equal
	O3	They are equal
SKI REMARK	See Q37. HRA is dominating.	
Q37 b	F1/2	
	R1	No
	R2-4	R2, No answer
	B2	No answer
	O1	There are no changes. Answer only valid for low power PSA, not outage period.
	O2	None
	O3	None
SKI REMARK	See Q37.	
Q37 c	F1/2	
	R1	N. A.
	R2-4	R2, No answer
	B2	No answer
	O1	There are no changes. Answer only valid for low power PSA, not outage period.
	O2	Not applicable
	O3	Not applicable
SKI REMARK	See Q37.	

United Kingdom

34. How have the data used in the LPSD PSA been derived? [The reply to this question should address the data used for initiating event frequencies, components failure probabilities, the time spent in each of the POSs, etc.]

IFFs obtained via operational experience feedback (OEF) review. Component failure probabilities from availability/ reliability considerations and the time in each POS factored into the IFF.

35. How component unavailabilities due to maintenance were used in the LPSD PSA?

a) How have the unavailabilities been justified?

Review of US peer plants. Unavailabilities generated on a typical average maintenance unavailability

b) How does the justification of the unavailabilties take into account operating experience?

General OEF used initially and some evidence to suggest that in some cases this has been conservative.

36. How does the LPSD data compare to the corresponding data used in the full power PSA?

a) Where there are changes?

Changes to unavailability, also may test component just prior to S/D so interval time shorter than from say claiming IST test frequency at power.

b) Why they have been necessary?

Unavailability changes required due to higher maintenance performed at shutdown. Scheduling Refuelling testing using insights from model to reduce risk.

37. How have the common-cause failure probabilities used in the LPSD PSA been driven?

a) How does this compare with what was done in the full power PSA?

b) Where there are changes?

c) Why they have been necessary?

No differences.

United States

BWR

34. How have the data used in the LPSD PSA been derived? [The reply to this question should address the data used for initiating event frequencies, components failure probabilities, the time spent in each of the POSs, etc.]

Initiating event frequencies were derived from generic and plant-specific data sources identifying the occurrence (or non occurrence) of specific events. Component failure probabilities were the same as those used in full power analyses. Component unavailability due to test or maintenance was either assumed or

based on prior plant-specific information. The fractions of time spent in each POS were identified by examining plant operational histories to obtain the appropriate information. Human response estimates were obtained by use of the "Accident Sequence Evaluation Program Human Reliability Analysis Procedure" (NUREG/CR-4772).

35. How component unavailabilities due to maintenance were used in the LPSD PSA?

a) How have the unavailabilities been justified?

The unavailabilities were either based on plant specific practices or were based on assumptions that were deemed to be conservative.

b) How does the justification of the unavailabilties take into account operating experience?

Plant-specific operational experience was used.

36. How does the LPSD data compare to the corresponding data used in the full power PSA?

a) Where there are changes?

Component failure probabilities were the same. Initiating event frequencies for similar initiators were based on events relevant to the POS; thus, the frequencies could be different (e.g., there are loss of offsite power events that can occur during various shutdown conditions that cannot happen during power operation). Maintenance unavailabilities were estimated from actual refuelling outages. Human error estimates were based on the conditions relevant to the POS and the accident sequences; thus, typically they were different.

b) Why they have been necessary?

Changes were necessary to appropriately reflect the plant in the POS analyzed. For example, equipment unavailability was not dominated by random unavailability, but rather by planned activities; thus, new values were needed

37. How have the common-cause failure probabilities used in the LPSD PSA been derived?

Common-cause failure estimates were base on a review of generic and plant-specific information. Beta factors were used to model (estimate) the various common-cause events. Beta factor estimates were obtained from information in the following reports:

- EPRI NP-3967
- NUREG/CR-2098
- NUREG/CR-2770
- NUREG/CR-2099
- NUREG-0666

a) How does this compare with what was done in the full power PSA?

Same methodology used.

b) Where there are changes?

No. The same approach was used.

c) Why they have been necessary?

While for this specific shutdown analysis no changes were made in the common-cause failure analysis, subsequent investigation identified that the appropriateness of using generic full-power common-cause models for shutdown conditions should be re-examined, especially with regards to the unique conditions experienced during shutdown. Application of full-power models to LPSD conditions must be done prudently since system configurations can be drastically altered (e.g., compressed air to all diesel generators is switched over to a single supply line during outage, normally standby systems for full-power are normally operating for LPSD conditions) during outages.

PWR

34. How have the data used in the LPSD PSA been derived? [The reply to this question should address the data used for initiating event frequencies, components failure probabilities, the time spent in each of the POSs, etc.]

Initiating event frequency

1) NUREG/CR-6144- A two-stage Bayesian approach was used for all initiating events. That is, data from other plants were used in stage 1 to develop a prior distribution for stage 2 which used data from Surry.

2) Screening analysis of NUREG/CR-6144- Initiating event frequencies were quantified using LER data collected for the US PWRs for a duration of 10 years. With the exception of loss of offsite power, all other initiating events were quantified by dividing the number of events by the amount of exposure time. Surry specific design information was used in determining if an event that occurred at a different plant was applicable to Surry. For loss of offsite power, a two-stage Bayesian analysis was performed.

3) NUREG/CR-6616 and 5718- Initiating events of earlier studies were grouped to reduce the number of event trees that have to be developed. The frequencies of the loss of RHR initiating events estimated in 1) were added to calculate the frequency of the loss of RHR initiating event group. The loss of offsite power frequency of 1) was used in calculating the frequency of station blackouts. The frequencies of LOCAs estimated in 2) were used in calculating the frequency of large and small LOCAs.

Component failure probabilities

The component failure probabilities of the full power PSA were used.

Time spent in each of the POSs

The shift supervisor's log books, outage schedules, minimum equipment list, and monthly operating reports were reviewed to collect the data needed to estimate the frequency of shutdown, duration of plant operational states, and maintenance unavailabilities.

35. How component unavailabilities due to maintenance were used in the LPSD PSA?

a) How have the unavailabilities been justified?

Maintenance unavailability was estimated as a function of the POSs of a refuelling outage by reviewing the shift supervisor's log books for three refuelling outages. The average maintenance unavailabilies were modelled as basic events in the PSA. As expected, maintenance availabilities during shutdown are much higher than those of power operation. It is assumed that the past maintenance experience is representative of the maintenance practice at the plant.

b) How does the justification of the unavailabilties take into account operating experience?

The maintenance unavailabilities were estimated using past operating experience. Due to changing plant practices at shutdown, past experience may not be applicable to today's plant.

36. How does the LPSD data compare to the corresponding data used in the full power PSA?

a) Where there are changes?

The changes are in initiating event frequencies and maintenance unavailabilities. In addition, there are data that are needed to characterize the POSs and in quantification, for example, the durations of POSs, the likelihood that containment sump would be clogged due to the material brought inside the containment, the fraction of time the pressuriser safety valves are removed, and the number of steam generators that are available for heat removal.

b) Why they have been necessary?

The changes are needed to represent the plant conditions during shutdown.

37. How have the common-cause failure probabilities used in the LPSD PSA been derived?

No changes were made to the common-cause failure probabilities of the full power PSA.

APPENDIX G - HUMAN RELIABILITY ANALYSES

PRELUDE AND QUESTIONS

HRA is an important aspect of the LPSD PSAs because there may be much higher level of human activity during outages such as maintenance activities and plant realignment. In addition, there is a much higher reliance on humans to initiate and control safety systems because most automatic functions have been disabled. In existing LPSD PSAs, human errors have been identified as very important contributors to the risk.

However, PSA users have questioned the suitability of current HRA methods and data for evaluating human error during LPSD. There are many aspects of LPSD operations that are not well addressed by existing methods. For example, the numerous dynamic changes that take place in the alignment of plant systems and component may increase the opportunity for errors. Also, the quality of emergency procedures and personnel training for both recognizing and responding to an abnormal event may not be as good as the emergency response for full power. Therefore, many underlying hypotheses used in current HRA methods may not be suitable for LPSD analysis. For example, current methods and data do not appear to handle the longer time scales that are available to plant personnel responding to some LPSD initiating events.

38. How has the HRA been carried out for the LPSD PSA?

a) *How does this differ from what was done in the full power PSA?*

b) *Why have these changes been necessary?*

39. How has the HRA analysis been supported? For example, has a task analysis been carried out? If yes, provide a brief description.

40. How have the HEPs used in the LPSD PSA been derived?

a) *How does it differ from what was done in the full power PSA?*

b) *Why have these changes been necessary?*

41. How do the HEPs used reflect the longer timescales over which the accident sequences develop in the LPSD PSA compared with the full power PSA?

42. How have dependencies between individual human errors been modelled in the LPSD PSA?

RESPONSES

Belgium

38. How has the HRA been carried out for the LPSD PSA?

The HRA for the LPSD PSA is identical to the full power PSA.

a) How does this differ from what was done in the full power PSA?

Identical.

b) Why have these changes been necessary?

Not relevant.

39. How has the HRA analysis been supported? For example, has a task analysis been carried out? If yes, provide a brief description.

The HRA analysis has been supported by a detailed analysis of the operating procedures available during shutdown conditions and an assessment of the organisation in the control room.

40. How have the HEPs used in the LPSD PSA been derived?

The HEPs have been derived from the methodology of Swain and Guttman.

a) How does it differ from what was done in the full power PSA?

There are no differences.

b) Why have these changes been necessary?

Not relevant.

41. How do the HEPs used reflect the longer timescales over which the accident sequences develop in the LPSD PSA compared with the full power PSA?

The timescale features in the performance shaping factors are used to assess the HEP.

42. How have dependencies between individual human errors been modelled in the LPSD PSA?

Dependencies have been identified based on the steps in the operating procedures.

Germany

38. How has the HRA been carried out for the LPSD PSA?

The identification and evaluation process recommended in THERP was used. According to THERP only skill and rule based (procedure guided or memorized, well known) actions were considered. Post-accident activities (emergency procedures, system recovery/repair) and pre-accident activities causing an initiator in case of error (in the course of plant state changes, inspection or test of components and other typical maintenance tasks during shut down of the plant) were investigated.

a) How does this differ from what was done in the full power PSA?

More memorized and well known actions, more actions initiating an event in case of error and more repair activities had to be investigated.

b) Why have these changes been necessary?

Not all investigated LP&SD-events were covered by procedures. In some cases initiator frequency could not be derived from operational experience or the available time window was long enough to take repair activities into account.

39. How has the HRA analysis been supported? For example, has a task analysis been carried out? If yes, provide a brief description.

The evaluation process recommended by THERP methodology was applied. Important steps were familiarization with the plant, task identification, development of task models, task analysis with identification of error likely situations (ELS) and human error prediction based on the quantification of ELS. The evaluation of repair activities required the development of new methods.

40. How have the HEPs used in the LPSD PSA been derived?

Data base of ASEP (for screening) and THERP (for actions influencing the PSA results) were applied.

a) How does it differ from what was done in the full power PSA?

There is no difference.

41. How do the HEPs used reflect the longer timescales over which the accident sequences develop in the LPSD PSA compared with the full power PSA?

The longer timescales are reflected in the range of the THERP model.

42. How have dependencies between individual human errors been modelled in the LPSD PSA?

Dependencies of individual human errors have been modelled in a unique HEP- component "failure of diagnosis and recovery".

Hungary

38. How has the HRA been carried out for the LPSD PSA?

A classification of the human interactions was made as follows:

Type A:

Human actions before the initiating event (pre-initiator actions) are concerned with those actions by plant personnel, which are associated with maintenance, testing and alignment schedules during the planned outage period including shutdown (cooling down), refuelling phase and the startup period that degrade system availability. These possible human errors cause latent failures degrading the availability of the safety related systems.

Type B:

Human actions as initiating events (actions as initiators) consist of those actions that contribute to initiating events or plant transients. These human actions start incidents or accidents.

Type C:

Human actions in accident sequences (post-initiator actions) are performed in the response to an accident, when the plant personnel can operate standby equipment or can recover equipment that would mitigate the accident. These types of human errors cover the mitigation of the accident caused by an initiating event.

Type A - Human Actions Before the Initiating Event

The modelling and quantification approach consisted of two main tasks as follows:

1) determination of preliminary HEP values by using screening values

2) use of some feedback from plant experience to enhance the preliminary HEP values

As a start, HRA analysts constructed a model of the maintenance and test process during plant outages. Discussions were held with plant personnel on this model to ensure that the model reflects the maintenance process at Paks NPP. It includes - as a first phase - the system of work planning, work orders and work permits. The next phase is maintenance error generation, which leaves the equipment in a degraded state. As a result of inspections the number of errors left is reduced. After the maintenance operation the operational staff carried out functional tests to check out the equipment. As a result of these functional tests the numbers of latent errors present are reduced considerably. Following the functional tests, other tests are organized during the start up phase, which reduce the number of latent errors further. Each of these tests and checks can result in recovery actions, which reduce the failure probability further.

A composite preliminary value of the HEP has been produced for each system train modelled in the PSA. Data sheets filled in for the various systems were used to identify from POS to POS what component failure modes can be caused by human errors. The composite number has been calculated by assigning the associated preliminary HEPs to these component level human errors. Dependence between the maintenance tasks within a system train and potential recoveries from some of the errors (e.g. mis-positioning of a valve after test) was addressed during the quantification process.

During the system analysis process four basic types of maintenance or test related tasks were identified that are susceptible to a human error. The evidence from a number of sources was that the HEP range for latent errors under these conditions was from 1E-2 to 1E-3. The effect of checking and testing was to reduce the error rate to the lower value of human error probability (HEP). From discussions with Paks personnel, it was concluded that range at Paks could be expected to be similar to the above values. However, it was felt that it was prudent to consider more conservative values than the above. The increase in the base numbers was selected to be a factor of 3. This value was chosen by human reliability experts.

The four types were defined based on the experience of looking at the maintenance and test operations during the low power and shutdown conditions. They are the following:

1) Maintenance Task with Incompletely Designed Test

This is the case when an equipment is maintained, but the post-maintenance (functional) test cannot reveal a maintenance caused error due to incompleteness of the test.

2) Maintenance Task on Neighbouring Equipment

This is the case when there is no maintenance of the given component, but another component is maintained, which can lead to the unavailability of the component in question.

3) Test Task with the Potential for Positioning Error

This is the case when equipment (typically a valve) can be left in a wrong position after test.

4) Alignment Task with the Potential for Positioning Error

This is the case when a component (typically a valve) can be in a wrong position due to alignment or re-alignment. The following causes can lead to this type of error:

- The component is being used in system line-up.
- There is no re-configuration after a system alignment because
  - *i.* there is no procedure for re-alignment
  - *ii.* there are only generally procedures for re-alignment.

Some general assumptions have been made to produce the preliminary train level HEP values for type A actions. These are as follows:

- The main focus was on the maintenance and test of mechanical equipment. This required conservative preliminary HEP values that would cover the associated I&C and electrical faults. However, it was noted that a number of tests are performed for the I&C and electrical part that can reveal maintenance errors. This was especially true for errors that can cause shared equipment unavailability (e.g. loss of a bus-bar). The higher the level of dependence is the better the chances are for detecting an I&C or electrical fault during subsequent tests of multiple components and interlocks. Thus, even though there may be a larger potential for making an I&C or electrical maintenance error, the potentials for detecting such errors were seen to be better than for the mechanical part. Nevertheless, in the longer term there is a need for further investigation to justify this statement or otherwise. Also, I&C and electrical maintenance will have to be addressed both in the event analysis and in the construction of a general decision tree framework.

- If some of the human errors could be detected by subsequent test or checks during the outage, a recovery factor was used. The effect of recovery was taken into account at component level, if appropriate. Credit was given only for recoveries from valve positioning errors. These recoveries require simple operator actions. It should be noted, that no recoveries were considered for maintenance caused component unavailabilities. The same recovery factor was used for subsequent independent tests or checks that could lead to recoveries.

Parallel to the estimation of HEPs, 126 reports about incidents with the involvement of plant personnel were analyzed. The reports were related to low power and shutdown modes. Results of the analysis were used to correct the estimation of the probability values of the four main error types.

<u>Type B - Human Actions as an Initiating Event</u>

See Q34.

Type C - Human Actions in Accident Sequences

For the analysis of post-accident (operator) errors a so-called decision tree approach was used. This was based on a generalization of the decision tree method developed and applied in the PSA for nominal power operational mode. The need for the generalization arose from the fact that the decision tree model for the nominal power case relies mostly on simulator data. Unfortunately, these data are not directly applicable to the low power and shutdown states, mainly because the simulator is not used for training the operators on treatment of accident sequences starting from low power modes. In addition, at low power and shutdown, the conditions of post-accident actions can be very much different from the full power accidents, which required extensions to the existing decision tree model. Some examples of such differences are as follows:

- In shutdown modes some accidents develop very slowly. As a result, there is a long time available to response to the situation. This time may exceed even 10 hours. Such extra long response times have not been addressed in the full power PSA.

- In general, the emergency operating procedures are not very well developed for low power modes, or they are not even available.

- In most of the shutdown modes there are several parallel and sometimes concurrent activities (normal shutdown work in main control room, various types of maintenance, system alignments and re-alignments, functional tests, etc.) going on, which may well affect the ability of operators and other plant staff to response to an accident.

- Due to excessive maintenance, safety related systems are taken out of service. As a result, there can be fewer indications of an accident in the main control room, and/or local actions may be needed (especially recovery of power supply to equipment) as part of an emergency action.

The generalized decision tree development process takes into account the specifics of plant operation in low power and shutdown modes and is capable of integrating inputs from varied sources, e.g. field data on human errors, data from event reports, simulator data, expert opinion, etc. The basic steps in the process are as follows:

1) Draw up a list of the potential performance influences.

2) Sort the list into scenario dependent and global performance influences. The HEPs can be affected by either scenario effects (dependent on the influence of the scenario) or global effects (these influences affect all the HEPs within a given category).

3) Rank influences in order of importance and select the most important ones.

4) Select the number of branches per performance influence and draw logic tree.

5) Estimate the Weighing Factors for each performance influence. If the branches are more than binary, it will be necessary to split the weighing factor to cover the various branches.

6) Determine the anchor values for HEP.

7) Calculate HEP distribution.

8) Check HEP distribution.

9) Modify weighing factors and HEPs.

The decision tree developed for modelling and quantifying post-accident errors in the LPSD PSA makes use of the following:

- the model applied to the full power case
- an understanding of crew operation based on analysis of plant operating procedures and walkdowns performed during an outage of Unit 2 of Paks
- interviews with plant operators
- an analysis of 156 safety related events occurred during outages of the four units of Paks.

The process consists of four important steps of the HRA model development. These are: (1) identification of most important scenario dependent performance influences, (2) choice of anchor values for calculating human error probabilities, (3) construction of the decision tree, and (4) checking of HEP estimates.

During responses to accidents starting from a low power or shutdown state the most important performance influences have been found to be:

- Time available to take action
- Crew knowledge (level of training) of the situation
- Workload (distraction)
- Quality of man-machine interface
- Availability and quality of emergency operating procedures.

The definition of the so-called anchor HEP value is an important element of calculating HEP distribution in a decision tree. In the shutdown HRA model 1.0 is used as upper bound, while the lower bound is 1.0E-04, i.e. the reference figure if all performance influences are favourable. The lower value is based on interviews and expert opinion. The estimation is supported mainly by the fact that the accident progression is very slow, and the situation can be thoroughly analyzed within the extra long time.

Considering the assumptions and reasoning outlined above, a decision tree has been drawn and the associated HEP values have been calculated to each pathway in the tree. The decision tree is composed of five headings as defined by the performance influences, and the number of pathways is 144 in the tree. The calibration of HEPs has been done by assuming that multiple factors have multiple effects on human error likelihood. These multiple effects have been quantified by making use of pathway dependent weighing factor values defined for each performance influence. Quantification rules have been developed and used to calculate HEPs by taking into account both the weighing values of performance influences and the anchor HEP figures.

A number of situations were identified that could be evaluated by both the full power and the newly developed low power and shutdown decision trees. Then the calculations were performed by the two models and the results were compared. Where discrepancies were found changes were made in the

estimates for the shutdown decision tree. In addition, by expert opinion, direct estimates of HEPs were made for some actions modelled in the shutdown PSA. The comparisons with direct estimates also revealed differences and subsequently modifications were introduced into the HRA model. The modified final decision tree has been applied to quantify all post-accident errors modelled in the study.

a) How does this differ from what was done in the full power PSA?

See above.

b) Why have these changes been necessary?

See above.

39. How has the HRA analysis been supported? For example, has a task analysis been carried out? If yes, provide a brief description.

See above.

40. How have the HEPs used in the LPSD PSA been derived?

See above.

a) How does it differ from what was done in the full power PSA?

See above.

b) Why have these changes been necessary?

See above.

41. How do the HEPs used reflect the longer timescales over which the accident sequences develop in the LPSD PSA compared with the full power PSA?

See above.

42. How have dependencies between individual human errors been modelled in the LPSD PSA?

Review of pre-initiator actions had shown that generally no within-person or within-crew dependence had to be assumed between the train level HEPs because of separation between maintenance tasks of different trains. The separation is assured by:

- administrative separation of actions
- substantial time separation between tasks
- location based separation
- requirement to document check-up of task in writing.

Within a system train dependence was taken into account with the following assumption: If a maintenance task affects multiple components, it is thought that if an error is made, the error will be systematically made for all the components affected by the task. Typical examples of such completely

dependent errors are maintenance task with incompletely designed test (MT_IDT) or maintenance task on neighbouring equipment (MT_NE).

As concerns post-initiator actions they were defined in such a way that the resulting human actions could be considered either fully dependent or fully independent. In the first case the error was modelled with the same basic event, while in the latter with different basic events.

Japan

38. How has the HRA been carried out for the LPSD PSA?

The HRA is composed of three phases. In the first phase, information of plant operation is gathered. Through the review of event trees and fault trees, a PSA analyst can be familiar with operational information and conditions of tasks. In the second phase, the HRA event trees are qualitatively established by the analyses of the tasks. The task analyses are performed through breaking down the tasks to the subtasks such as contact, instruction, operation and manipulation procedure. In the third phase, the success or failure probabilities of the tasks are given by the quantitative analyses as follows. The PSA analyst estimates nominal values of human error probabilities, identifies factors and interactions affecting human performance, quantifies the effects of the factors and interactions, accounts for probabilities of recovery from errors, and calculates human-error contribution to the probability of system failure. Human error probabilities (HEP) are estimated based on the THERP method. Human error rates are also based on the "Hand Book" (NUREG/CR-1278).

a) How does this differ from what was done in the full power PSA?

It slightly differs from what was done in the full power PSA. For example, the human error probability (HEP) for diagnosis error without the time frame of the THERP (i.e., NUREG/CR-1278, Table 20-3) is derived assuming that the HEP after 1,500 minutes or more has a constant value of that within 1,500 minutes.

b) Why have these changes been necessary?

Not applicable.

39. How has the HRA analysis been supported? For example, has a task analysis been carried out? If yes, provide a brief description.

As mentioned earlier, the task analyses are performed through breaking down the tasks to the subtasks such as contact, instruction, operational and manipulation procedures. These individual units of behaviour will reflect the level of detail as well as probabilities of human errors evaluated for the each unit of behaviour.

For an example, HRA Event trees reflect the results of analyzing an operator's task. Operator's task is divided into two parts:

1) Pre-accident

- Miscalibration of instrumentations 6.4×10^{-4}/d
- Failure to restore the equipments after maintenance and/or test. 6.5×10^{-5}/d

2) Post-accident

- Plant diagnostic 10^{-1}/d～10^{-4}/d
- Manual initiation of ECCS Less than 10^{-7}/d

The treatment of dependency between operating personnel are follows:

- Operator---assistant shift supervisor High Dependence
- Operator---shift supervisor Moderate Dependence

40. How have the HEPs used in the LPSD PSA been derived?

The HEPs derivation has employed the Technique for Human Error Rate Prediction (THERP).

a) How does it differ from what was done in the full power PSA?

The techniques used for estimation of HEP during LPSD are the same as for the full power operation.

b) Why have these changes been necessary?

Not applicable.

41. How do the HEPs used reflect the longer timescales over which the accident sequences develop in the LPSD PSA compared with the full power PSA?

The plant configuration varies depending on the conditions of the LPSD, while operating equipments and mitigation systems to stand by are uniform in the full power operation. In the LPSD PSA, the tasks are analysed based on the information of each plant configuration determined by Plant Operational State (POS). The analytical results are incorporated into the HEPs calculations.

For an example, NUPEC uses the THERP method in NUREG/CR-1278 to estimate the cognitive error and miss-operation of the plant operators on LPSD in BWR plant. There are relatively large allowed time compared to the full power operation for operators to cope with the accidents during LPSD, but unfortunately there is no data for cognitive errors over twenty-four hours in the time reliability correlation curve in NUREG/CR-1278. If allowed time is over twenty-four hours, the probability of plant cognitive error would be the value of 1500 min. in the time reliability correlation curve in NUREG/CR-1278.

42. How have dependencies between individual human errors been modelled in the LPSD PSA?

In the THERP method, the dependencies between individual human errors are evaluated as conditional probabilities of success or failure given at branches of subtasks in HRA event trees. The level of dependence could be defined as complete independence, a low level of dependence, a moderate level of dependence, a high level of dependence or complete dependence.

But the THERP method cannot also suggest the treatment of the dependency between the preceding plant diagnostic error and the following diagnostic error in case plural operator actions are available for mitigation of accidents.

For PWRs, the mitigation measures by operators have complete dependency for diagnosis since the allowable time is too short. Therefore, no diversity of mitigation measure is credible once operators' diagnosis failed.

Two additional questions / responses were provided:

What areas of your analysis approach are, in your opinion, most in need of improvement?

Commission error

The THERP technique cannot handle the commission error correctly. In case of LPSD, commission error should be correctly evaluated to understand the features of LPSD PSA more realistically.

Error factor

For the uncertainty distribution of the each event during plant shutdown, the error factor is assumed as same value as that of used in the analysis of full power operation. For LOCA due to the human error at the time of maintenances, the initiating event frequency and error factor should be evaluated properly.

What are the areas that would most benefit from further research?

New methods should be sought for estimation of plant diagnostic error.

Korea

38. How has the HRA been carried out for the LPSD PSA?

Previous shutdown PSAs have identified Human Reliability Analysis (HRA) as an important part of the overall risk analysis because most of the systems must be manually actuated. For each fault tree or branch point, the HRA was performed based on the THERP method and HRA handbook data. These data and methods have been implemented for specific event-based scenarios, using simple, conservative and defensible assumptions regarding human performance and error in component tasks, to produce a simplified best-estimate analysis of anticipated HEPs for shutdown operations.

a) How does this differ from what was done in the full power PSA?

When comparing with the full power PSA, there is no difference in HRA modelling.

b) Why have these changes been necessary?

N/A

39. How has the HRA analysis been supported? For example, has a task analysis been carried out? If yes, provide a brief description.

No special supporting analysis was done.

40. How have the HEPs used in the LPSD PSA been derived?

a) How does it differ from what was done in the full power PSA?

When comparing with the full power PSA, there is no difference in the derivation of HEPs.

b) Why have these changes been necessary?

Not applicable.

41. How do the HEPs used reflect the longer timescales over which the accident sequences develop in the LPSD PSA compared with the full power PSA?

When comparing with the full power PSA, there is no difference in the derivation of longer-timescale HEPs.

42. How have dependencies between individual human errors been modelled in the LPSD PSA?

When comparing with the full power PSA, there is no difference in HRA dependency modelling.

Mexico

38. How has the HRA been carried out for the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

The study is working with bounding calculations, and human action comes from the models already used for full power, if appropriate. The specific human action related with LPSD conditions have not been assessed at this point.

b) Why have these changes been necessary?

See the above answer

39. How has the HRA analysis been supported? For example, has a task analysis been carried out? If yes, provide a brief description.

See answer to question 38. HRA in full power models consider most of the operator activities (under more stress environment due to the less time available to perform the activity) that are required for the POS analysed. Therefore these actions are considered conservative for the POS analyzed.

40. How have the HEPs used in the LPSD PSA been derived?

a) How does it differ from what was done in the full power PSA?

See answer to question 39

b) Why have these changes been necessary?

See answer to question 39

41. How do the HEPs used reflect the longer timescales over which the accident sequences develop in the LPSD PSA compared with the full power PSA?

See answer to question 39

42. How have dependencies between individual human errors been modelled in the LPSD PSA?

Not applicable.

The Netherlands

38. How has the HRA been carried out for the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

b) Why have these changes been necessary?

39. How has the HRA analysis been supported? For example, has a task analysis been carried out? If yes, provide a brief description.

40. How have the HEPs used in the LPSD PSA been derived?

a) How does it differ from what was done in the full power PSA?

b) Why have these changes been necessary?

41. How do the HEPs used reflect the longer timescales over which the accident sequences develop in the LPSD PSA compared with the full power PSA?

(The following is an excerpt from NEA/CSNI/R(98)1:"Critical Operator Actions-Human Reliability Modelling")

General Characterization of HRA.

Characterization of the Methodology used for the Analysis of the Type A, B and C Errors.

* Framework: SHARP (EPRI)

* Pre-Accident Human Interactions (Type A): THERP

* Operator Actions causing an initiating event (type B)

  - Included in the Borssele-specific initiating event frequencies; e.g. assessment of scram lists.

* Post-Accident Human Interactions (Type C):

C1 Manual backup to automatic actions

C2 Emergency procedure based actions

C3 Recovery actions

  - EPRI-NP 6560

  - P1 Failure to recognize situation in time (Borssele specific)

  - P2 Failure to process information in time (ORE/HCR model)

- P3 Manipulation error (THERP)
- Human Error Probability = P1+P2+P3

* Dependencies:

- Time window dependencies (P2/P3)
- Cognitive error (P1)

* Where in the model?

- Hand-start after failure of auto-start in FT.
- Procedural actions in ET.
- Maintenance & testing Errors in FT.
- Recovery in the cut-sets.
- Bleed and Feed (Accident Management) in ET.

Characterization of the Methodology used for the Analysis of the Errors of Commission

Two separate studies (one for power states and one for non-power states) of the so called errors of commission have been performed by G. Parry (NUS) + A. Mosleh (Univ. of Maryland). The so called 'hitline' method was used as a basis for these studies [Macwan,A., *Methodology for Analysis of Operator Errors of Commission in NPPs with Application to PRA*, Ph.D. Thesis, University of Maryland, November 1992.]. For the study on the non-Power States extensive use has been made of the research programme on "The influence of Low-Power & Shutdown conditions on Human Reliability"; an assessment through analysis of operational experience [Barriere, M., et al., *An Analysis of Operational Experience During Low Power and Shutdown and a Plan for Addressing Human Reliability Assessment Issues*, NUREG/CR-6093, June 1994].

ERRORS OF OMISSION DURING SHUT_DOWN STATES.

There is general agreement that, for a variety of reasons, the low power and shutdown phases of operation present more opportunities for making significant errors. The reasons include the following:

- There is an increased level of activity in the plant, much of it by contractor personnel who are unfamiliar with the plant.
- More systems are under manual control than during full power operations.
- There might be relatively few explicit procedures for dealing with abnormal system behaviour.
- Technical Specifications may be inadequate or non-existent.
- Training in low power and shutdown activities might be not as comprehensive as it should be, and operators are less knowledgeable about plant response to upset conditions.

- Systems may be in reduced redundancy and unfamiliar configurations, thus leaving them more vulnerable to failures and/or operator errors.

- Instruments being relied upon may be different and may have different characteristics from those used at full power.

For the Non-Power POSs the significant type C human interactions following various relevant initiating events are discussed. The human errors of commission during non-power operation states are discussed in later.

Loss of Component Cooling Water Systems TF/VF

During non-power operations, the TF (component cooling water) and VF (service water) systems continue to be key systems for successful decay heat removal. Failure of either system is important, because it fails the closed-cycle primary cooling and spent fuel pool cooling. From cold shutdown (early) through to the end of the refuelling, there is plenty of time available (minimum 4 hours, typically ≥ 15 hours) for connecting alternate sources if the primary sources fail. This is especially true for VF where hose connections are made up ahead of time and a procedure exists which describes cooling from alternate sources. The TF system is somewhat different, since there aren't any extra hose connections or procedures for alternate cooling. However, problems with leaks in one header can be recovered by transferring loads to the other header. Due to the long time intervals involved, the human error probabilities for long-term TF and VF recovery (of recoverable faults only) is 1.0 E-2 and 1.1 E-4, respectively.

Loss of Residual Heat Removal Cooling.

These initiating events occur only during non-power operations during the cold-shutdown, midloop, and core unload/load states. In all of these states the bunkered injection system (TW) automatically actuates when temperatures reach 80 ^{o}C. This provides success for the short term, but in the long term (>15 hours) the closed-cycle TJR cooling must be restored, either from the loops or from the sump. Recovery for this event is handled on a cutset by cutset basis. In the core load/unload cooling can be lost for three days without requiring make-up. Due to the long time window, the human error for failure to provide make-up to the reactor-basin/fuelpool is 1.1E-4.

Loss of Spent Fuel Pool Cooling

Due to the design and capacity of the fuel pool, there is an extremely long time window for operator actions to keep the pool full. There is at least 18 hours to the start of boiling, and using the Core Unload MAAP Run it appears that cooling could be lost for three days without makeup needed. Due to the long time window, the human error probability has a correspondingly low value of 1.1E-4

Loss of Offsite Power / Station Blackout

Also during non-power operations the Station Blackout and Loss of Offsite Power sequences lead to an operator action involving the electric plant and diesel generators. This is the operator action to manually load the diesel generators following failure of diesel generator load sequencer (Type C3). This event is assigned a screening value of 0,1, but does not appear in the dominant sequences and thus is not evaluated further.

Loss of Coolant Accidents

One human recovery action deals with mitigating failures of the seal water system, which on its turn fails the residual heat removal (RHR) system and occurs during any power and non-power plant operational state. Since the RHR system is manually actuated in non-power POSs, there is no YZ-38 signal to block the TN-signal (low seal water pressure) and the operator must jumper out the signal to continue decay heat removal (Type C3). For non-power states the time window extends to 5 hours, reducing the human error probability to 1.1E-4. Since this 'long time' event is coupled in all cases with the short time recovery event (power cases; within 30 minutes), a conditional probability of 1.6E-3 is used.

Examples of some of the type C3 events used primarily in the low-power and shut-down PSA

Event Description	P1	P2	P3	HEP
Type C3				
* Operator fails to restore RHR after spurious operation	1E-4	8.9E-3	n/a	9E-3
* Operator fails to isolate containment during Fuelpool Early POS	Screening value	Screening value	Screening Value	1E-1
* Operator fails to align TA for RHR	" "	" "	" "	8E-1
* Long term recovery failure of electric bus BU	data derived empirically from failure data, fraction of recoverable faults.			1,9 E-1
* Long term recovery failure of bus BV	" "			2,7 E-2
* Long term recovery failure of closed component cooling system.	" "			1,1 E-3
* Long term recovery failure of RHR	" "			3,5 E-3
* Long term recovery failure of service water system	1E-5	1E-4	n/a	1.1E-4
* Operator fails to recover from overdraining	1E-4	1.7E-3	n/a	1.8E-3
* Operator fails to override low seal water pressure within 5 hours	1E-5	1E-4	n/a	1.6E-3
* Operator fails to restore fuel pool cooling by filling the pool	1.1E-5	1E-4	n/a	1.1E-4

ERRORS OF COMMISSION DURING SHUT-DOWN STATES.

General Issues and Results.

Accident Response Procedures: There are few procedures available for mitigating the loss of a critical safety function during non-power. There are several hours available to response, so time stress is minimal (except perhaps in transition between POSs)

Maintenance is well controlled and the procedures for taking a component out of service and opening the pressure boundary are very clear.

The Design of Reactor Pressure Vessel (RPV) with no penetrations in the bottom head means that the RPV cannot be inadvertently drained. Further, the automatically actuated independent bunkered primary make-up/suppletion system (TW) increases recovery time by 10 hours.

The Refuelling Strategy is to completely unload the core (to minimize the time in midloop) and to primarily allow the majority of maintenance to occur during fuel pool operations, significantly reduces the risk of non-power operations.

Initiating Events. Maintenance errors which produce initiating events have occurred historically and are included explicitly in the support system initiating event models. Similarly, postulated operational errors during transitions (overdraining of the RCS or error in switching of the electric power plant) are also explicitly modelled.

Interfacing Systems LOCAs during start-up are historically a potential problem, e.g., the Biblis event. Recent hardware changes and the current procedure makes this scenario highly unlikely at NPP Borssele.

Reactivity Addition Events require large amounts of pure water to be injected in order to produce widespread core damage. Human errors of commission and omission which can produce such situations are explicitly modelled.

Post-Initiator Response. Nearly all of the actions following an initiating event are concerned with recovery from the initial problem (component/system loss), or recovery from the combination of random failures in the ECCS/RHR-system that follow the initiating event. Since the RHR system must ultimately be restored, the only important errors of commission are those which start as a recovery action but somehow lead to a long term failure of RHR.

Timing. The increased times available for recovery during the non-power POSs (5 hrs to > 3 days) produce many opportunities for alternative or innovative actions that lead to successful mitigation of potential accidents.

Identification of Error Expressions.

From the point of view of the event tree modelling, it was useful to group the error expressions of interest into:

- those associated with the initiating events, and
- those associated with the response phase.

A further subdivision was made by noting that errors may be latent, i.e., they may not be revealed, or have an impact, at the time they are made, or they may be active, i.e., they are the trigger events that result in the transition to the unwanted condition.

Error Expressions Resulting in Initiating Events.

In general, the initiating events can be characterized as occurring from one of the following groups of causes.

- Mechanical/structural failure
- Maintenance error causing functional failure (including loss of integrity and loss of supporting equipment)

  i. Maintenance directly on the operating system providing core cooling

  ii. Maintenance directly on a support system of the core cooling operating system

  iii. Maintenance induced spatial interactions affecting operating or support systems (Unrelated maintenance which fails a needed component), e.g., dropped items cutting wires or breaking pipes, other effects of maintenance on nearby components.

  iv. Wrong system/component identified and taken out of service for test, maintenance or calibration.

To identify potential maintenance error opportunities for each initiating event resulting from system failure it is necessary to identify for which POSs maintenance is allowed on the system itself, on a neighbouring system, or on any of its support systems.

- Operational failure, which is subdivided into two groups

  i. Incorrect system operation.

  The type of activities that are susceptible are those that essentially are manual control of systems. These include the following:

  * Important operator action during transitions like: shift electrical power from the turbine generator to offsite power, drain to midloop, reactor startup and shutdown (de-borate and borate)

  * Important actions during steady state plant operations like: Control of RCS and fuel pool level, temperature control of RHR- HX

  ii. Incorrect response to a partial system failure.

  For any system for which it is necessary for an operator to take manual control to align a redundant train should the operating train(s) fail, there is an opportunity for an error that causes complete loss of the function. Thus all failures which require operator intervention can lead to the identification of an error expression, which is failure of the remaining redundancies. The following steps are needed to fulfil the identification process for initiating events.

  Step 1 Construct a matrix which identifies, for each POS, which systems are in operation, and which are in standby. In addition the matrix should identify which of the systems s under manual control (either changing state or manual start of redundant trains).

  Step 2 Using the Redundancy Plan, identify which of the operating systems may be subjected to maintenance as a function of POS, and for each of these systems,

review the system configuration and operating practices to identify if there are potential system failure modes, using the following screening guidelines:

* For closed loop systems a major concern is loss of system integrity. This could be caused by maintenance on a segment of the system that supports non-safety related, or non-essential loads, as well as by maintenance on a pump train.

* For an open system individual load headers are probably too small to cause a significant flow diversion, particularly as the source of coolant is infinite. The search should be focused on the main headers or pump trains.

* Problems might arise with changing headers for maintenance, when it might be possible to dead head pumps.

  – Easily detectable errors that can be readily recovered need not to be considered.

Step 3 For the operating systems, identify if there are any systems not restricted by the redundancy plan which are co-located. Maintenance errors resulting from errors while working on co-located systems should only be considered for energetic impacts, e.g., floods, fires and dropped heavy loads. Consider mainly those impacts causing damage of all trains or common headers.

Step 4 For each POS following that in which maintenance can be performed, the potential may exist for latent errors. Identify if a potential failure mode of the redundancies that are in standby exists.

Step 5 For each operating system that is under manual control, identify if there is any way in which an operator, while aligning a system to correct for a partial failure, could create a failure mode (e.g., flow diversion, dead heading or cavitating pumps, etc.) The principal concern is non recoverable failures.

Step 6 Review the start up and shutdown procedures with a HAZOP mentality, to identify potential failure modes/initiating events.

Error Expressions Associated with Response.

Also here identification of failure mechanisms of systems that could be the result of errors of manipulation during alignment of alternate systems which lead to failure of remaining redundancies or failure of other alternatives. Again, flow diversion, run out of pumps, dead heading of pumps, inadvertent isolation are searched for.

Step 1 Identify Failure Modes of Systems That Could Result from Operator Intervention.

Only postulate such failures when operator action is necessary (e.g, manual system, failure within a preferred redundant system with no automatic backup, etc.)

For automatic systems identify "operator isolates or prematurely secures system" as a potential error expression.

In addition, there are latent errors that can lead to failures of the alternate systems.

Two major sub-groups can be recognized:

(A) Control room related operational activities, and

- response related to loss of safety function, or trip
  - i. written response procedure available
  - ii. no written response procedure
- response related to equipment failure during particular evolutions that requires operator actions to maintain the safety function
  - i. written procedure available
  - ii. no written procedure available
- deliberate changes of plant status due to transitions between POSs.
  - i. Interpretation of Plant Status
  - ii. Sequencing of steps
  - iii. Actuation of Equipment
  - iv. Termination of a Function

(B) Maintenance activities

42. How have dependencies between individual human errors been modelled in the LPSD PSA?

General. In the initial sequence quantification it is necessary to identify the potential multiple operator actions contained in various sequences. In order to ensure that the human dependencies are not lost due to the quantification process, all actions are initially given high screening failure probabiliti11es. All human interaction events modelled directly in the fault trees are entered with failure probabilities of 1,0. This is to ensure that these events do not cause associated branches of the fault tree to be truncated upon initial formation of the fault tree equation. The fault tree equations are then examined to determine the cutsets with multiple operator actions (if any). Cutsets with multiple operator actions are next evaluated for dependencies, and the dependencies addressed.

The fault tree equations are incorporated into accident sequence equations, and the core damage frequency calculated with all operator events set to the screening values and the HI events determined. Each combination of operator actions contributing greater than 1% of the resulting core damage frequency was then evaluated to identify and address dependencies. Finally, any HIs which are originally assigned a screening value, but then appear in significant sequences, are analyzed in greater detail to enable a more accurate quantification to be performed.

Early in the initial sequence quantification it is necessary to identify the potential multiple operator actions in the various functional equation and cut sets. In order to ensure that such dependencies are not lost due to the quantification process, Type C1 and C2 actions are initially given a value of 1,0. Then after reviewing cut sets to identify the combinations of multiple operator actions at the system equation,

functional equation, and sequence equation levels, a lower screening value of 0,5 or 0,1 is used. Each combination of multiple operator actions contributing greater than 1% to the resulting core damage frequency is the further evaluated with respect to dependencies. Human interactions that are determined not to be important, and have no dependencies, are assigned values of 0,5 or 0,1. For those human interactions that are significant, probability values P1, P2 and P3 are evaluated.

Type C HIs. Two types of dependencies are addressed for the Type C human interactions. The first is related to the effect on the time available for performing an human interaction, and the second is the question of cognitive dependency between sequential, or parallel, human interactions.

Time Dependence. At the single human interaction level, a dependence occurs between the time allowable for recognition and decision making, and the time needed to perform the action. The time window is evaluated on the basis for the completion of the human interaction, thus the time available for the operators to recognize the appropriate action to take is dependent on the time it takes to perform the action. The time window T_w is determined by subtracting from it the mean time to complete the action, and using this modified time window in the ORE/HCR correlation.

When sequential actions are performed, the time of completion of one action has both a direct impact on the time available for the next, and an indirect impact. The indirect impact occurs in that the sooner the first action is performed, the quicker the condition of the plant changes. To handle this effect, conservative assumptions are made with respect to accident sequence definition. The timing of the first action is taken initially to minimize the time window for subsequent actions. The significance of this assumption can be tested by performing sensitivity analyses.

There are cases where several options are available to the crew to successfully performing a function in a given time window T_w, and those options are usually prioritized by procedures; i.e., if first option does not work, crew initiates second option and so on. For example, in an ATWS event, to make the reactor subcritical, the first option is to try to manually trip the reactor from the control room and then initiate emergency boration if the reactor remains critical. The time spent on the first option has an impact on the time available for the next action and so on.

Cognitive Dependence. When two or more human interactions are performed either sequentially, or in parallel, and are part of the same general procedure, they are cognitively correlated. In this case the errors represented by probabilities P1 and P2 are used only once to model entry into the procedure. When moving out to a second procedure, the possibility of cognitive and time-related errors can be used again if it involves more than a simple reading of instructions.

Spain

38. How has the HRA been carried out for the LPSD PSA?

Two models were used, one for actions where allowable time was shorter than one hour and the other for actions where more than one hour was available. The first type uses the same HRA model as the full power PSA does (Time Reliability Curves modified to account for performance shaping factors considered by means of success likelihood indexes). For the second type of action, a multi-attribute model was developed.

a) How does this differ from what was done in the full power PSA?

The second type of actions was not considered in the full power PSA.

b) Why have these changes been necessary?

Because of the fact that many actions in LP&SD conditions have a longer time available but are worse procedurised than in full power conditions.

Sweden

38. How has the HRA been carried out for the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

b) Why have these changes been necessary?

CSNI Question number	Answer from Swedish plant	Answer
Q38	F1/2	The same as for full power operation. Swains Handbook has been used.
	R1	The IAEA guidelines for conducting human reliability analysis.
	R2-4	R2, HRA is done using Swain methodology; grace times, performance shaping factors, dependencies etc. are based on EdF experience in simulators, using the EdF shutdown instruction package
	B2	Engineering judgements
	O1	A screening model for HEP has been developed and used for those operator actions that are unique for the low power PSA. For those operator actions that already existed in the full power PSA (recovery actions) HEP remained unchanged in low power PSA. Answer only valid for low power PSA, not outage period.
	O2	
	O3	
SKI REMARK	Swain and other HRA-references.	
Q38 a	F1/2	No difference
	R1	Greater numbers of HRA.
	R2-4	R2, Full power PSA uses very little simulator experience
	B2	Only limited HRA in full power PSA
	O1	In the full power PSA no detailed analysis of any operator action have been made yet. So far only a very rough screening has been used in order to sort out those actions that need to be further analysed. The screening model developed for the low power PSA is more justified since all actions are quantified with a model that takes some key parameters into account. Answer only valid for low power PSA, not outage period.
	O2	
	O3	
SKI REMARK	More HRA	
Q38 b	F1/2	
	R1	Manual interventions during LPSD are more frequent than operation.
	R2-4	R2, EdF HRA methodology is based on the EdF shutdown instruction package and simulator experience, gives better data for the HRA

	B2	No answer
	O1	There where many more operator actions to be evaluated in the low power PSA so it was necessary to develop a screening model that was easy to apply and also in some extent separated each action. Answer only valid for low power PSA, not outage period.
	O2	
	O3	
SKI REMARK	Operator actions dominate especially during SD.	

39. How has the HRA analysis been supported? For example, has a task analysis been carried out? If yes, provide a brief description.

CSNI Question number	Answer from Swedish plant	Answer
Q39	F1/2	By international references, plant specific data and discussions with staff.
	R1	Yes, a screening technique is used.
	R2-4	R2, Task analysis has been used by EdF to estimate IE frequencies for some IEs, and task analysis and simulator data are a basis of the times, performance shaping factors etc. for the EdF HRA approach, which has been applied to R2 with minor modifications.
	B2	No answer
	O1	For low power PSA no task analysis has been made. All different actions have been reviewed together with operating personnel and actions with critical consequences if performed erroneously where identified. Answer only valid for low power PSA, not outage period.
	O2	
	O3	
SKI REMARK	Task analyses for IE-analyses during SD.	

40. How have the HEPs used in the LPSD PSA been derived?

a) How does it differ from what was done in the full power PSA?

b) Why have these changes been necessary?

CSNI Question number	Answer from Swedish plant	Answer
Q40	F1/2	HEPs?
	R1	Developed by consultant.
	R2-4	R2, See Q39
	B2	No answer
	O1	HEP for human errors as initiating events have been derived by the use of a screening model that for each action takes into account: Complexity, Experience and Feedback

		For recovery actions another screening model has been developed that uses the time and diagnosis curve in THERP with some changes. Answer only valid for low power PSA, not outage period.
	O2	
	O3	
SKI REMARK	Fore some analyses screening models have been used.	
Q40 a	F1/2	
	R1	No difference, but larger numbers.
	R2-4	R2, EdF experience was used, see Q38
	B2	No answer
	O1	So far only a very rough screening has made in full power PSA in order to identify those operator actions that needs to be analysed in more detail. When those actions have been identified most likely a more detailed model will be used for deriving HEPs. Answer only valid for low power PSA, not outage period.
	O2	
	O3	
SKI REMARK	Much more HRA during SD.	
Q40 b	F1/2	
	R1	Changes being made because of the nature of LPSD.
	R2-4	
	B2	See Q38
	O1	There where many more operator actions to be evaluated in the low power PSA so it was necessary to develop a screening model that was easy to apply. Answer only valid for low power PSA, not outage period.
	O2	
	O3	
SKI REMARK	See Q40a.	

41. How do the HEPs used reflect the longer timescales over which the accident sequences develop in the LPSD PSA compared with the full power PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q41	F1/2	
	R1	The time scales have adjusted to the ascension in power, outage and increasing power after outage.
	R2-4	R2, EdF's calculations support appropriate grace times
	B2	No answer
	O1	The screening model used for recovery actions is based on THERPs time diagnosis curve. Answer only valid for low power PSA, not outage period.
	O2	

	O3	
SKI REMARK	Longer time for human interaction.	

42. How have dependencies between individual human errors been modelled in the LPSD PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q42	F1/2	The HRA method considers this by the time window that is used.
	R1	Yes.
	R2-4	R2, No answer
	B2	No
	O1	So far no specific dependencies have been considered. The screening models used are considered to be conservative enough in order to sort out those actions that need to be further analysed. Answer only valid for low power PSA, not outage period.
	O2	
	O3	
SKI REMARK	Very limited so far. But if the operators do not act on the first signal it is very likely that he will not act on the second or third signal either. This is taken into account in the HRA sequences performed in some studies (R1 and O2).	

United Kingdom

38. How has the HRA been carried out for the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

b) Why have these changes been necessary?

No differences in techniques used

39. How has the HRA analysis been supported? For example, has a task analysis been carried out? If yes, provide a brief description.

Task analysis (hierarchical and timeline) has been carried out

40. How have the HEPs used in the LPSD PSA been derived?

a) How does it differ from what was done in the full power PSA?

b) Why have these changes been necessary?

The HEPs have been derived in the same manner as the LPSA and have used a combination of HEART and THERP

41. How do the HEPs used reflect the longer timescales over which the accident sequences develop in the LPSD PSA compared with the full power PSA?

Each HEP assessed on its merits and improved reliability claimed with extended timescales

42. How have dependencies between individual human errors been modelled in the LPSD PSA?

Dependencies assessed using a power factoring method based on a five-level dependency scale (zero to complete dependence)

United States

BWR

38. How has the HRA been carried out for the LPSD PSA?

The general methodology used for conducting the HRA and determining the Human Error Probabilities (HEPs) for the identified human actions was the Accident Sequence Evaluation Program Human Reliability Analysis Procedure as documented in NUREG/CR-4772. While the analysts performing the LPSD PRA believed that errors of commission might be more likely to occur during shutdown conditions, no recognized approach for identifying and quantifying such errors was available at that time; thus, errors of commission were defined to be out of scope for the POS 5 analysis. However, as an outgrowth of the LPSD work, the NRC began work that ultimately resulted in the development of ATHEANA, which does allow for the identification and quantification of errors of commission, along with a more complete understanding of the factors that can affect human performance

a) How does this differ from what was done in the full power PSA?

It does not differ. Both the LPSD and full power analyses used the same technique/methodology.

b) Why have these changes been necessary?

Not applicable.

39. How has the HRA analysis been supported? For example, has a task analysis been carried out? If yes, provide a brief description.

While no formal task analysis as defined in THERP was performed, the analysts involved in the HRA did have a general understanding of the various actions necessary for successful completion of each human action modelled in the PRA. Furthermore, they had sufficient information to identify the relevant cues, had enough information to know how much time the operators would have to identify and complete the necessary actions, and knew whether the actions would involve activities outside of the control room.

40. How have the HEPs used in the LPSD PSA been derived?

The methodology used to determine the HEPs for the identified human actions is described in NUREG/CR-4772. This method was selected for several reasons:

1) The HEPs obtained using the procedures are considered to be slightly conservative relative to those that would be obtained from other methodologies such as THERP. Conservative HEP estimates were considered desirable because existing HRA methodologies have not explicitly considered the impact of potentially unique performance shaping factors which might be operative during LPSD conditions.

2) The method was used in the full power. By using the same methodology, comparisons between related full power and LPSD HEPs might be possible. Such comparisons might provide insights regarding differences in operator behaviour during the different modes of operation, and in how

the behaviour should be quantified. In addition, the HEPs for the pre-accident human actions used in the present study were taken directly from the full power analysis. Thus, at least some degree of methodological internal consistency is maintained across the HEPs for the pre- and post-accident tasks.

3) The method is straightforward to use and has been shown to produce internally consistent HEPs that appear to reflect at least the relative potential of human failures in the nuclear power plant environment. Internal consistency was seen to be an important factor for the LPSD domain, where a complete PRA has not previously been performed.

4) The procedure allows for straightforward adjustments in HEPs as a function of the results from interviews with plant personnel. In situations such as LPSD where procedures may not be all encompassing, the results of interviews with operators and other plant personnel become a critical aspect of the HRA.

a) How does it differ from what was done in the full power PSA?

Typically, the only major differences involved:

1) adjusting (generally downward) the recommended level of stress associated with an action, and

2) giving credit for operators correctly diagnosing and carrying out a non-proceduralised action if, on the basis of the site interviews, it was judged that the operators had a clear understanding of the event in question and of the requirements for responding to the event.

b) Why have these changes been necessary?

This "reduction in stress" was done for those actions where the time available to perform the action was long when compared to similar actions at power. (Note: while the POS 5 analysis team chose this approach for dealing with long recovery times, the individuals involved in the analysis recognized the need for additional research and/or confirmatory studies to either improve the techniques for estimating HEP for human actions with long recovery times or to demonstrate that HEPs derived by this approach were sufficiently robust (i.e., were a adequate representation of reality).

41. How do the HEPs used reflect the longer timescales over which the accident sequences develop in the LPSD PSA compared with the full power PSA?

See answer to Q40b.

42. How have dependencies between individual human errors been modelled in the LPSD PSA?

Diagnosis and performance of operator actions is dependent on the initiator and on what has occurred or failed to occur previously in the accident sequence. Therefore, in order to accomplish a reasonably valid HRA analysis, a sequence by sequence analysis of the human actions contained in the event and fault trees was performed to attempt to account for the dependencies among the different operator actions.

PWR

38. How has the HRA been carried out for the LPSD PSA?

a) How does this differ from what was done in the full power PSA?

1) NUREG/CR-6144- The event tree model imbeds the human actions within a series of functional top events that include hardware failures in safety systems and their support systems. The imbedded human actions are modelled in two parts: a global diagnosis event affecting all actions and the top event-related action events. Cognitive failures are the focus of the diagnosis event. For subsequent events, most cases tend to be those in which the previous event tree's functional top event failures were caused by equipment failure or were the results of the physics or process rather than the cognitive failure of the operator. In most other cases, because global diagnosis has been successful, the operators are on the right track, but the difficulties of the physical operation (e.g., residual heat removal pump venting) are causing long time delays. There are substantial cues to encourage the operators to proceed to alternative cooling options.

   Qualitative analysis of both diagnosis events and subsequent events follows a structured approach, detailing information about preceding events (summary of the event sequence to this point identifying dependencies and abnormal plant responses that could complicate the situation), indications of plant condition (what the crew sees, how it could trigger desired and undesired actions), procedural guidance (including its applicability to the situation), relevant training and experience (positive and negative), concurrent/competing actions, indications of successful action (how will the operators know the plant is responding properly), failure impact (how will the plant respond if the operators are unsuccessful), and time constraints (based on thermal-hydraulic analyses and time required for operator action).

   The qualitative evaluation of the actions and the important parameters that affect operator's performance were used to derive the human error probabilities (HEPs) by adapting the success likelihood index methodology.

2) Screening analysis of NUREG/CR-6144- A simplified screening methodology was developed. It adopts the more conservative post-accident screening HRA methodology concept of the ASEP HRA Procedure to all human actions to be quantified in this PSA. Each human action has an associated HEP composed of two parts, HEP_d and HEP_a, representing errors associated with diagnosis and action. The time available for the action is estimated based on the condition the plant is in, and is divided into two parts, time to diagnose and time to carry out the action. A table is provided for determining HEP_a and the time needed to carry out the action, depending on if the action is simple or complex and whether or not it can be accomplished inside the containment. A separate table is provided for determining HEP_d based on the time available for diagnosis.

3) NUREG/CR-6616 and 5718- The same approach as that of 1) was used.

b) Why have these changes been necessary?

The approach of 1) and 3) is essentially the same as that of a full power PSA. In some cases more operational experience and judgment are required to address long time frames and effects of limited training and experience. The approach of 2) was necessary so that a large number of HEPs could be quantified quickly.

39. How has the HRA analysis been supported? For example, has a task analysis been carried out? If yes, provide a brief description.

Based on a review of current procedures and discussions with plant personnel, a task analysis was performed that essentially mapped the procedure against plant conditions and calculated event timing. It is documented in the analysis through the human action sequences in the model and the detailed

descriptions of the qualitative factors associated with each action. All post-initiating event operator actions were analysed and quantified. Uncertainty was addressed.

40. How have the HEPs used in the LPSD PSA been derived?

a) How does it differ from what was done in the full power PSA?

Consistent use of a HEP to represent failure to diagnose is the most important difference in the PSA for shutdown POSs. That is, for each initiating event and each shutdown POS, the probability that the operator fails to diagnose is modelled. Given failure to diagnose, it is assumed that core damage would result.

b) Why have these changes been necessary?

The modelling of failure to diagnose is necessary, because every initiating event during shutdown requires operation actions to mitigate. In a full power PSA, mitigation of accidents is initially done automatically, and operator actions are needed only if automatic systems fail or to establish long term cooling after safety systems responded successfully. The approach addresses some of the issues of context and cognition currently considered by second generation methods.

41. How do the HEPs used reflect the longer timescales over which the accident sequences develop in the LPSD PSA compared with the full power PSA?

The time available for carrying out the action is a performance shaping factor used in the quantification. The modified SLIM framework allows the judgement of the analyst to be used to adjust the HEPs.

42. How have dependencies between individual human errors been modelled in the LPSD PSA?

1) NUREG/CR-6144- The dependency of all operator actions on the ability to diagnose the accident was explicitly modelled. The impact of preceding human errors is one of the performance shaping factors used in the quantification of HEPs.

2) Screening analysis of NUREG/CR-6144- The dependency of all operator actions on the ability to diagnose the accident was explicitly modelled. Multiple human errors in the same accident sequence were assumed to be independent.

3) NUREG/CR-6616 AND 5718- Same as 1).

APPENDIX H - QUANTIFICATION/SENSITIVITY STUDIES/UNCERTAINTIES

PRELUDE AND QUESTIONS

Accident sequences produced for LPSD PSA are quantified using the methods and tools available for full power PSAs. However, PSA users believe that the applicability of both methods (e.g., fault tree linking method) and tools (e.g., computer codes) should be examined. For example, a LPSD PSA may involve an extremely large number of sequences due to the large number of systems, which could be called upon for coolant injection. A LPSD PSA may also involve much smaller frequency estimates because the frequency and the fraction of time the plant is in a given POS are taken into account. Therefore, the derivation of accident frequencies, the capability of the code to adequately handle the number of sequences produced, and the truncation limits used ought to be examined.

Also, performing sensitivity studies may not be as easy or as meaningful as for LPSD PSAs. For example, different types of systems may be unavailable for the different types of outages. Therefore, insights derived for the importance of systems associated with a specific outage may not be applicable to all outages. Similarly, results of uncertainty studies performed for a LPSD PSA may not be as useful since they may represent uncertainties associated with a particular plant outage and not the uncertainties with an "average" outage.

43. How has the LPSD PSA been quantified?

a) How does this quantification differ from what was done for the full power PSA?

44. What sensitivity studies have been carried out?

a) How was the range of sensitivity studies decided on?

45. Has an uncertainty analysis been carried out?

a) How were different types of uncertainties (e.g., epistemic, aleatory) addressed?

46. Have you been able to generalize the results and use the insights produced as a result of sensitivity and uncertainty analyses?

RESPONSES

Belgium

43. How has the LPSD PSA been quantified?

The LPSD PSA has been quantified using the Risk Spectrum PSA software.

a) How does this quantification differ from what was done for the full power PSA?

There is no difference.

44. What sensitivity studies have been carried out?

Sensitivity analyses have been performed on some initiating event frequencies, component failure data and CCF modelling for operating equipment.

a) How was the range of sensitivity studies decided on?

The range of sensitivity studies has been agreed upon with the Regulator.

45. Has an uncertainty analysis been carried out?

No

a) How were different types of uncertainties (e.g., epistemic, aleatory) addressed?

Not relevant.

46. Have you been able to generalize the results and use the insights produced as a result of sensitivity and uncertainty analyses?

Not clear what is meant with "to generalize the results". No specific insights were produced as a result of sensitivity and uncertainty analyses.

Germany

43. How has the LPSD PSA been quantified?

The LP&SD PSA has been quantified using the time dependant analysis of RiskSpectrum.

a) How does this quantification differ from what was done for the full power PSA?

The full power PSA has been quantified using mean values while in the LP&SD-PSA the maximal values of the time dependant analysis have been used because the initiating events occur at a fixed point in time during the outage which are always defined at the end of the observation period. Components with a test-interval of one year have than reached the maximal unavailability.

44. What sensitivity studies have been carried out?

Fussel-Vesely-Importance studies have been carried out to identify week points and key components which may be subject to repair measures.

a) How was the range of sensitivity studies decided on?

The importance of basic elements up to 5% was analysed.

45. Has an uncertainty analysis been carried out?

An uncertainty analysis has been carried out with the program RALLY-STREUSL.

a) How were different types of uncertainties (e.g., epistemic, aleatory) addressed?

The uncertainty analysis using RALLY-STREUSL considers only aleatory uncertainties.

Beside of this, the influence of not analysed IE and differences in the measures and schedule of the outage on the result has been analysed.

46. Have you been able to generalise the results and use the insights produced as a result of sensitivity and uncertainty analyses?

The derived insights concerning the importance of components or HF are thought to be reliable in general for any outage with respective POSs.

Hungary

43. How has the LPSD PSA been quantified?

For the different plant operational states both the core damage frequency (CDF) and the core damage probability (CDP) were calculated. The first is useful for drawing up the risk profile of a refuelling shutdown, while the latter makes it possible to sum up the core damage risk originating from different plant operational states. The summation of CDPs over the POSs gives an annual probability of core damage originating from LPSD states.

a) How does this quantification differ from what was done for the full power PSA?

For the full power PSA only the CDF was calculated. In order to be able to compare and/or to sum up the core damage risk originating from both the full power and the LPSD states, the full power CDF should be converted to an annual core damage probability (CDP) by multiplying the CDF by the duration the plant operates at full power in a year.

44. What sensitivity studies have been carried out?

Consistent with the PSA for full power operation, the importance and sensitivity studies had three main objectives as follows:

- to identify the most important initiating events, accident sequences independent and dependent component failures, and human errors (importance calculations)
- to determine how the analysis results are affected by moderate changes in (1) specific modelling assumptions and the associated model structure, and in (2) input data (sensitivity analyses).
- to identify potential safety improvements measures and evaluate expected risk reduction.

Two types of analysis were performed to meet the above objectives, qualitative analysis and quantitative analysis.

Qualitative analysis was carried out parallel to producing point estimates of risk measures. The important minimal cut sets dominating the various POSs and the overall risk in low power and shutdown states were examined with respect to (1) their level of order and (2) the sensitivity of the associated basic events to specific effects (e.g. hardware failures of similar type, human errors with similar performance influences, common cause failures, etc.). Quantitative analysis was focused on those failures and human errors that had been found important either due to the probability of minimal cut sets (quantitative indicator) or as a result of the above qualitative evaluation.

Above all quantitative analysis was aimed at calculating the following measures for initiating events, component failure modes, and model parameters:

- Fussel-Vesely importance (fractional contribution - FC)
- Risk decrease factor (RDF)
- Risk increase factor (RIF)
- Sensitivity measures (S_U, S_L, $S_{U/L}$).

In the first step of quantitative analysis these measures were calculated within the Risk Spectrum code. The results yielded insights into important risk contributors and sensitivity of core damage risk to specific effects in each POS analyzed.

In the next step the quantitative analysis was extended to the overall core damage probability which is representative for the entire outage duration. This was done by integrating the POS related results of sensitivity analysis in the probability domain. The importance and sensitivity measures produced by Risk Spectrum were combined.

In the last phase of quantitative analysis the effect of potential safety improvement measures (identified during qualitative analysis of dominant minimal cut sets and evaluation of quantified risk figures) was evaluated. Estimation of expected risk reduction was the objective of this analysis. Conceptualized measures related to technical, administrative and procedural changes were the target of the analysis through which reduction can be achieved in initiating event frequencies, unavailability of mitigating systems and probability of inappropriate human interactions. The effect of some measures was quantified by modelling the expected changes in detail rather than performing sensitivity calculations in a mechanistic way (e.g. quality and availability of emergency operating procedures would change from ‘poor’ to ‘good’ if additional procedures were developed for specific emergency conditions and thus the effect of performance influence 'Procedures' was changed accordingly in the decision tree based HRA model, which resulted in modified estimates of human error probability for some post-initiator actions). For some other potential measures the expected risk reduction could not be directly modelled since the measures are at the level of conceptualization and therefore there is a lack of specific information that would be needed for adequate quantification. The effect of such measures was estimated by assigning attributes to the basic events affected by the measures and by performing sensitivity calculations for these attributes (e.g. administrative measures that can impact on a range of pre-initiator or initiator type errors). As a final step the expected values of risk reduction calculated for each POS were integrated to evaluate the global effect of the various measures on the core damage risk in all low power and shutdown states.

a) *How was the range of sensitivity studies decided on?*

See above.

45. Has an uncertainty analysis been carried out?

Yes.

a) *How were different types of uncertainties (e.g., epistemic, aleatory) addressed?*

In the course of assessing risk from low power and shutdown operational modes of the Paks NPP emphasis was put on minimizing uncertainties in quantified risk measures as far as possible. In general uncertainties are attributable to three main sources as follows:

1) incompleteness of PSA logic models (event trees and fault trees)

2) limitations in adequacy of PSA models

3) uncertainties in input data used for performing risk computations.

The following efforts were made to minimize uncertainties due to incompleteness and limitations in adequacy of the PSA models:

- Extended use was made of the logic models, data, methods and experience of the level 1 PSA for full power operation of the Paks NPP.

- The open literature on low power and shutdown PSA was reviewed in detail to ensure that international experience and recommendations would be taken into account.

- A wide range of plant documentation was examined and used for the purpose of the analysis.

- Walkdown inspections of the plant helped to confirm some modelling assumptions and make up missing data.

- Interviews were used throughout the analysis with various staffs of plant personnel to obtain specific information needed for LPSD PSA model development and quantification.

- In addition to the results of thermal-hydraulic simulations performed earlier within the level 1 PSA for full power operation of the Paks plant, use was made of a range of deterministic and probabilistic analyses performed to address specifics of accident sequences in low power and shutdown states.

- In-house reviews (within VEIKI) and external reviews of the plant personnel were carried out at each major stage of the analysis in accordance with the quality assurance program of the LPSD PSA study.

There are several factors that affect completeness and adequacy of the PSA models. Some of them are subjective (e.g. the scope of the study defines some limits beyond which the results do not apply), others are objective and represent our current understanding of physical phenomena, capabilities of modelling accident sequences. Some of the known limitations affecting the LPSD PSA for the Paks plant are as follows:

- Plant operational states of a refuelling outage were the target of the analysis. Forced shutdown due to an initiating event or violation of Technical Specifications was not included.

- Only internal initiators were taken into account. Internal fires, flooding and external hazards were not modelled.

- The reactor core was the only source of radioactivity release that was considered, other potential sources (spent fuel pool, interim storage facility) were excluded from the analysis.

- There is a limited knowledge about the development of certain accident sequences and about their consequence.

- Due to lack of experience and inherent limitation of expert judgment/opinion some initiating events, accident sequences, the corresponding plant responses, and the associated human interactions may not have been taken into account at an appropriate level of detail during model development.

- Latent and known weaknesses of plant documentation may have influenced model adequacy.

The PSA models and results need to be updated regularly so that they represent the actual state of the plant and the effect of the above limitations can be kept at a reasonably low level. There is a living PSA programme in place for the Paks NPP which is aimed at maintaining and updating the existing PSA models and data.

With respect to input data on random events (e.g. time to failure) and systematic effects (e.g. test interval) plant specific information was used as far as possible. In addition, data from generic sources and expert opinion were also needed. When these data sources were used during parameter estimation an attempt was made to determine uncertainty bounds besides a representative point estimate (usually mean value). These uncertainty bounds were the input to quantitative uncertainty analysis which has been performed exclusively for uncertainty due to variability in input data, i.e. no attempt was made to quantify modelling uncertainties.

Quantitative uncertainty analysis was done so that parameters representing input data were treated as random variables. The same procedure was followed as in the PSA for full power operation. The objective of uncertainty analysis was to determine uncertainties in core damage risk in each POS of a refuelling outage due to data uncertainty. This objective was accomplished by using an approximate solution which consisted of the following steps:

- Those parameters that were within the scope of uncertainty analysis were described by continuous distribution functions as random variables. In accordance with the practice of the full power PSA lognormal approximation of the probability distribution was used in most cases.

- Uncertainty calculations were performed for each POS using Monte Carlo simulation within the Risk Spectrum program.

- Lognormal expressions of the POS related uncertainty measures were determined by the use of non-linear optimization for curve fitting.

- The fitted lognormal probability distribution functions were characterized by their parameters and by the plots of the corresponding probability density and cumulative distribution functions.

46. Have you been able to generalize the results and use the insights produced as a result of sensitivity and uncertainty analyses?

The results were mainly used for recommending safety enhancement measures and prioritizing them.

Japan

43. How has the LPSD PSA been quantified?

Accident sequences are defined by event trees. Success or failure probabilities at each branch in the event trees are described by means of the fault tree method. The quantification may address each individual sequence or each of several groups of sequences, called "plant–damage states" (PDSs), formed by combining sequences with certain similarities. To calculate a sequence frequency, two approaches are outlined. One approach is fault-tree linking, which determines the minimal cut sets for an accident sequence or a PDS. The minimal cut sets of an accident sequence are subsequently quantified to produce an estimated frequency for an accident sequence or a PDS. The other approach, which uses event trees with boundary conditions, quantifies system models under various conditions and multiplies system failure probabilities by initiating event frequencies to estimate an accident sequence frequency. NUPEC adopts the former approach (i.e., the fault tree linking approach) for the accident sequence quantification.

a) How does this quantification differ from what was done for the full power PSA?

Differing from what was done in the full power PSA, the plant configuration varies depending on the conditions of each POS in the LPSD. The quantification is performed at each POS.

44. What sensitivity studies have been carried out?

Sensitivity studies have been carried out for analytical assumptions, model selections, data selections etc., focusing how they make CDF deviate from original CDF with different values.

For an example, the probability of plant cognitive error, the plant configuration, etc.

a) How was the range of sensitivity studies decided on?

If probabilistic importance analysis points out some equipment failure has large importance, the equipment failure will be selected for a sensitivity study case. Big contribution factors for CDF are also added to the sensitivity study case.

45. Has an uncertainty analysis been carried out?

Yes.

a) How were different types of uncertainties (e.g., epistemic, aleatory) addressed?

The different types of uncertainties such as epistemic, aleatory etc. were not addressed.

46. Have you been able to generalize the results and use the insights produced as a result of sensitivity and uncertainty analyses?

The sensitivity study promoted some improvements such as the adoption of ultra-sonic water level instrumentation and the water make up to RCS from accumulators, which would enhance the safety during mid-loop operation. Furthermore, in the sensitivity study, NUPEC confirmed that CDF during LPSD would be adequately low even if the minimum redundancies of safety systems allowed in Tech Spec were applied.

Two additional questions / responses were provided:

What areas of your analysis approach are, in your opinion, most in need of improvement?

For LOCA due to the human error at the time of maintenances, Initiating event frequency is very small. The method of uncertainty analysis for this kind of initiating event frequency should be developed.

What are the areas that would most benefit from further research?

We do not find any specific concern at the moment.

Korea

43. How has the LPSD PSA been quantified?

a) How does this quantification differ from what was done for the full power PSA?

When comparing with the full power PSA, there is no difference in the quantification process.

44. What sensitivity studies have been carried out?

To identify the effect of design or operational items to the plant risk level (CDF), sensitivity analysis was performed. The hardware systems such as AAC, DG, CS system, CVCS system and SC system were found to be significant from the importance analysis. Therefore, in the consideration of sensitivity analysis, the hardware was excluded.

Sensitivity analysis was performed for following items;

- Recovery action for restoration of shutdown cooling train
- Fire suppression action
- Operator action for LOCA isolation
- Operator actions taken from the operational experience
- Operator actions analysed by the HRA

a) How was the range of sensitivity studies decided on?

It was based on the expert judgments.

45. Has an uncertainty analysis been carried out?

a) How were different types of uncertainties (e.g., epistemic, aleatory) addressed?

No.

46. Have you been able to generalize the results and use the insights produced as a result of sensitivity and uncertainty analyses?

It is found that the enhanced operator training for shutdown cooling recovery should be performed by plant management, and emergency procedures for shutdown operation mode be prepared. Also, it is recognized that the fire suppression is very important and thus operators and fire brigade must train to

suppress fire in early stage of event, especially before SC operation failure. From the results of sensitivity analyses, it is noted that the isolation is very important and operator must find the location of leakage and isolate loop within limited time. To perform LOCA (leakage) isolation well during shutdown operation, operators must be trained and also well-designed monitoring devices must be employed. The effect of these operator actions was found to be negligible. It means that there are several mitigation systems and thus operator actions to restore SC function are not critical to the CDF. The results show that these operator actions are very critical and thus training and education must be reinforced.

Mexico

43. How has the LPSD PSA been quantified?

a) How does this quantification differ from what was done for the full power PSA?

Hand calculations of frequency for accident sequences using top event trees unavailability or HEP as appropriate.

44. What sensitivity studies have been carried out?

None

a) How was the range of sensitivity studies decided on?

45. Has an uncertainty analysis been carried out?

None

a) How were different types of uncertainties (e.g., epistemic, aleatory) addressed?

46. Have you been able to generalize the results and use the insights produced as a result of sensitivity and uncertainty analyses?

See answer to questions 44 and 45

The Netherlands

43. How has the LPSD PSA been quantified?

a) How does this quantification differ from what was done for the full power PSA?

44. What sensitivity studies have been carried out?

a) How was the range of sensitivity studies decided on?

45. Has an uncertainty analysis been carried out?

a) How were different types of uncertainties (e.g., epistemic, aleatory) addressed?

46. Have you been able to generalize the results and use the insights produced as a result of sensitivity and uncertainty analyses?

The NPP Borssele PSA has been performed for all POSs following a linked fault tree approach as described in NUREG/CR-2300 using the NUPRA software package supplemented with the NURELMCS codes (RELCON).

The process for developing equations for a single merged fault tree or a combination of merged fault trees is identical. The appropriate fault tree for the function of interest is linked in NUPRA. Linking simply entails connecting all of the lower level fault trees that combine to form the top level function. The linked fault tree is then updated with the current basic event data and with the house event BED file, POS BED file, and spatial BED file appropriate to the conditions of interest. This updating will turn on and off gates and change data as appropriate. The fault tree is then solved and quantified with the NURELMCS code using a truncation value based on the expected order of magnitude of the function unavailability, generally 1E-9. The truncation value has been selected based on previous sequence quantifications, to ensure that all sequence cutsets above 1E-10/year are retained. The result is a boolean equation. The boolean equation can then be combined with other equations or can be used in the sequence quantification directly. Cutsets and unavailabilities are also produced for insights into the dominant failures of a specific function.

NUPRA has been used to assign functional equations to nodes in the event trees and to automatically generate quantification (merge) control (OCL) files for each of the 99 internal and external initiating event trees.

Two types of uncertainty are addressed in the Borssele PSA; parameter value uncertainty and modelling uncertainty. Parameter value uncertainties are handled by defining a probability distribution on the value of each parameter such that the n-th percentile of the distribution represents the value below which the analyst has a degree of belief of n% that the true value lies. This subjective approach to the representation of uncertainty makes the propagation of parameter value uncertainty through the sequence quantification process mathematically straightforward using a Monte Carlo sampling technique.

Modelling uncertainties are treated by defining discrete or continuous probability distributions over the different modelling hypothesis. This was done by performing sensitivity analyses on the issues identified to be the most significant to the results. These sensitivity studied involved requantification of the plant model, to evaluate the impact to the results by the variance of these modelling issues.

The first example is the sensitivity analysis on human error failure probabilities. In the sensitivity analysis, all human error probabilities are first reduced and later increased by a factor of 10. The selected case is analyzed by manipulation of the single core damage equation developed from the dominant accident sequence cutset equations by modification of the data and requantification of the plant model. The sensitivity analysis shows that the total core damage frequency is reduced by 59% if all human error probabilities are reduced by a factor of ten, and increases by a factor of 9 if all human errors are increased by a factor of ten.

The sensitivity analysis examining the simultaneous unavailability of safety systems due to maintenance during the outage was not significant.

Sensitivity studies investigating the importance of the 24V dc dependencies for operator actions from the control room showed that some fault trees had to remodelled regarding manual actuation. Especially for midloop and cold shutdown situations the sensitivity analysis demonstrated that the previous FT-modelling had some oversimplifications in the modelling of the dependency of manual actuations from

the control room on the availabilities (including maintenance) of the 24V dc buses. Due to remodelling the Midloop POS contribution to the core damage frequency increased with 18% (from 1.21E-6 to 1.43E-6).

Spain

43. How has the LPSD PSA been quantified?

The quantification methodology used in LP&SD PSA is the same as in full power PSAs: fault trees linking, delete term and rare event approximations and the same level of truncation for minimal cut sets. The quantification tools depend on the specific project and the time when they were carried out, but they are always the same as in full power PSAs.

a) How does this quantification differ from what was done for the full power PSA?

There is no difference in the process; the only difference is that the sequences are 'scenario sequences' instead of 'initiating event sequences'.

44. What sensitivity studies have been carried out?

By the time being Garonna NPP has not yet performed sensitivity studies. At Asco NPP a number of studies have been carried out: CCFs elimination, Human errors elimination, Test and Maintenance unavailability events elimination, dependency among human actions and time duration of mid-loop operations.

a) How was the range of sensitivity studies decided on?

The first groups of sensitivity studies were intended to know the weight of CCF, HRA, unavailability and dependency in the minimal cut-sets and core damage frequency. In general, parameters with larger uncertainty and that govern the dominant core damage cut-sets are analyzed. The last one assessed the plant improvement after the steam generators replacement (the new steam generators require a shorter time at mid-loop)

45. Has an uncertainty analysis been carried out?

Yes, the uncertainty of the probabilities of the events that contribute to the CDF was propagated and a final distribution was obtained, along with its main parameters.

a) How were different types of uncertainties (e.g., epistemic, aleatory) addressed?

As mentioned, only the aleatory was taken into account.

46. Have you been able to generalize the results and use the insights produced as a result of sensitivity and uncertainty analyses?

The replacement of steam generators has been proven as an important safety asset, as it means a substantial reduction in CDF, given the shorter time at mid-loop they require.

Sweden

43. How has the LPSD PSA been quantified?

a) How does this quantification differ from what was done for the full power PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q43	F1/2	As for full power operation. CDF/year is calculated.
	R1	Similar to power operations.
	R2-4	R2, Only limited FT modelling
	B2	Engineering judgements
	O1	By using the same techniques as full power PSA, i.e. fault and event trees that are built and quantified in RiskSpectrum® PSA Professional. Answer only valid for low power PSA, not outage period.
	O2	Conventional frequencies and importance values for systems etc.
	O3	Conventional frequencies and importance values for systems etc.
SKI REMARK	All new analyses have been performed with RiskSpectrum.	
Q43 a	F1/2	No difference
	R1	No differences.
	R2-4	R2, Full power PSA relies more on generic event statistic
	B2	Full power PSA relies more on generic event statistic
	O1	It doesn't. Answer only valid for low power PSA, not outage period.
	O2	Less sensitivity calculations
	O3	Less sensitivity calculations
SKI REMARK	For LP almost the same. For SD mostly event trees.	

44. What sensitivity studies have been carried out?

a) How was the range of sensitivity studies decided on?

CSNI Question number	Answer from Swedish plant	Answer
Q44	F1/2	None
	R1	No need for sensitivity study identified.
	R2-4	R2, No answer
	B2	Through engineering judgements
	O1	So far no results have been derived yet and it is therefore not possible to say what sensitivity studies that will be necessary. Answer only valid for low power PSA, not outage period.
	O2	
	O3	None.
SKI REMARK	Limited in all studies. This is an essential task for the future.	

Q44 a	F1/2	
	R1	N.A.
	R2-4	R2, No answer
	B2	Through engineering judgements
	O1	See answer to question Q44 above. Answer only valid for low power PSA, not outage period.
	O2	
	O3	Not applicable.
SKI REMARK	See Q44.	

45. Has an uncertainty analysis been carried out?

a) How were different types of uncertainties (e.g., epistemic, aleatory) addressed?

CSNI Question number	Answer from Swedish plant	Answer
Q45	F1/2	No
	R1	No
	R2-4	R2, Limited sensitivity studies, based on engineering judgement for 1996 study
	B2	Through engineering judgements
	O1	No uncertainty analyses have been carried out. Answer only valid for low power PSA, not outage period.
	O2	No
	O3	No
SKI REMARK	Only for the first R2 study.	
Q45 a	F1/2	
	R1	Mainly epistemic uncertainties.
	R2-4	R2, No such approach was used.
	B2	No such approach was used.
	O1	No uncertainty analyses have been carried out. Answer only valid for low power PSA, not outage period.
	O2	Not applicable
	O3	Not applicable
SKI REMARK		

46. Have you been able to generalize the results and use the insights produced as a result of sensitivity and uncertainty analyses?

CSNI Question number	Answer from Swedish plant	Answer
Q46	F1/2	
	R1	No.

	R2-4	R2, No
	B2	No
	O1	No results have been derived yet. Answer only valid for low power PSA, not outage period
	O2	No
	O3	No
SKI REMARK	This question will be discussed in the joint Nordic project.	

United Kingdom

43. How has the LPSD PSA been quantified?

a) How does this quantification differ from what was done for the full power PSA?

Quantified using Risk Spectrum Professional in the same manner as the full power PSA and forms part of the event tree/fault tree model that is known as Sizewell's Living PSA.

44. What sensitivity studies have been carried out?

a) How was the range of sensitivity studies decided on?

Sensitivity studies undertaken were the same as at power. A review was conducted to determine whether there were any potential vulnerabilities. The dominant contributors to faults with a core melt frequency $>10^{-7}$ per year were reviewed. The review also considered the sensitivity of operator actions in Mode 4 and operator actions in other modes. Balance of risks were discussed between mitigating LOCAs as opposed to mitigating a loss of decay heat removal fault causing a cold overpressure fault (see next response also)

45. Has an uncertainty analysis been carried out?

a) How were different types of uncertainties (e.g., epistemic, aleatory) addressed?

None undertaken. Comprehensive sensitivity studies (above) looking at importance of individual systems, groups of components etc. considered to add more value.

46. Have you been able to generalise the results and use the insights produced as a result of sensitivity and uncertainty analyses?

As a result of the analysis, recommendations were made to align the plant more for decay heat removal for LOCA rather than for containment protection in Mode 4

United States

BWR

43. How has the LPSD PSA been quantified?

In general, the process used the following steps:

1) Sequence logic for the "level" (i.e., the "sequence" logic through a specified number of transfer event trees) being analyzed was generated using the event trees as contained in the IRRAS (NUREG/CR-5813) database for the initiating event being examined. (NOTE: Due to continued development of the IRRAS Code, in part to provide this project with more efficient tools, many different "versions" of IRRAS were used during this analysis. The NUREG/CR5813 reference is used only to represent the most up-to-date version that was released for general use at the time of the analysis and in no way implies that all work was performed on the referenced version.) This approach was necessary due to the large number of possible accident sequences.

2) Once the sequence logic was generated it was checked for any non-valid sequences. This step was necessary since the use of multiple transfer trees sometimes resulted in invalid sequence logic being generated.

3) After the sequence logic was checked, any changes that needed to be made to the base case values of the basic events were identified and incorporated into "Change Sets" as a means of modifying basic event values, turning on or off particular gates in the system fault trees, or simply using the name of a particular top event to represent the failure of some system.

4) After the Change Sets were finalized, the sequences that were to be analyzed were "marked" in the appropriate IRRAS menu. The appropriate truncation information was entered, and the sequences were solved using the logic information contained in the top event (system) fault trees both for the failed top events and the successful top events (to allow deletion of cut sets which are not appropriate for the sequence being analyzed (called DELETE TERM in SETS)).

5) If any of the sequences analyzed ended in core damage, the appropriate information was stored for later use. For any sequence surviving truncation, steps 1 through 4 were repeated for the next "level" of analysis. This process was repeated until all sequences either terminated in core damage or were eliminated by truncation.

6) After all core damage sequences were obtained for the IE being examined, the sequence cut sets were examined for validity and for potential operator recovery actions. For those cut sets where recovery was appropriate, a basic event was incorporated into the cut set. This basic event represents the failure of the operator to perform the required action(s).

7) Core damage sequences surviving the point estimate calculations from (6) were examined during the Time Window Analysis. This analysis incorporated specific system unavailabilities and decay heat loads for three distinct time regimes.

8) Uncertainty analyses were performed on each core damage sequence surviving the Time Window Analysis (i.e., 7.).

a) How does this quantification differ from what was done for the full power PSA?

Other than the difference in tools used in the full power versus LPSD analyses—SETS (SAND77-2051) for sequence logic solution and TEMAC (NUREG/CR-4598) for quantification and uncertainty analysis in the full power analysis versus IRRAS for both sequence logic solution and quantification and uncertainty analyses in the LPSD analysis)—the major difference between the two analyses lies in the sheer magnitude of the piecewise accident sequence solution for each initiating event that was necessary in the LPSD analysis.

44. What sensitivity studies have been carried out?

SAND94-2949 documents various sensitivity studies performed on the POS 5 LPSD analysis results. The major sensitivities addressed in the SAND report dealt with:

- LOCA frequencies
- Operator response to specific conditions
- Equipment survivability
- Plant conditions assumptions

a) How was the range of sensitivity studies decided on?

Sequences important to core damage frequency and offsite consequences were examined to determine what modelling assumptions, data, and human actions could affect the results of the analysis.

45. Has an uncertainty analysis been carried out?

Yes

a) How were different types of uncertainties (e.g., epistemic, aleatory) addressed?

Parameter uncertainty associated with initiating event frequencies, equipment failure/unavailability probabilities, and human error probabilities was estimated using Latin Hypercube sampling to produce samples for each uncertain parameter that was then propagated through the analysis. Model uncertainty was addressed by the various sensitivity studies that were performed (see Q44).

46. Have you been able to generalize the results and use the insights produced as a result of sensitivity and uncertainty analyses?

The results of the uncertainty analysis were used to compare the core damage frequency and risk results with those from the full power analysis.

While no effort has been made with regards to generalizing the results, it would be possible to produce some generalizations; however, before any such generalizations were made, it would be necessary to understand the differences and similarities between the plant analyzed (a BWR 6 Mark III) and the other plants for which generalizations were sought. Nevertheless, the shutdown study had underscored the importance of understanding shutdown risk and has helped support activities such as those discussed in Q2 and Q56.

PWR

43. How has the LPSD PSA been quantified?

a) How does this quantification differ from what was done for the full power PSA?

1) NUREG/CR-6144- The quantification differs from that of the full power PSA because it has to account for different types of outages and the amount of time the plant spent in the POSs. A core damage sequence or cutset starts with the frequency of an outage type, e.g., refuelling vs. maintenance outages. The frequency is then multiplied by the conditional probability that an initiating event occurs in a POS, calculated as the product of the frequency of the initiating event and the duration of the POS. It is then linked to the rest of the logic model.

2) Screening analysis of NUREG/CR-6144- Same as 1).

3) NUREG/CR-6616 and 5718- The approach of NUREG/CR-5718 is the same as 1). The objective of the study in NUREG/CR-6616 is to compare the risk impacts due to performing preventative maintenance of different equipment during shutdown vs. during power operation. Therefore, the risk measures were quantified conditional on the plant being in a particular time window of a particular POS.

44. What sensitivity studies have been carried out?

1) NUREG/CR-6144- None.

2) Screening analysis of NUREG/CR-6144- None.

3) NUREG/CR-6616 and 5718 The objective of the study in NUREG/CR-6616 is to compare the risk impacts due to performing preventative maintenance of different equipment during shutdown vs. during power operation. A sensitivity calculation was performed for each equipment of interest including AFW pumps, emergency diesel generators, LPI pumps, HPI pumps, CCW pumps and service water pumps.

a) How was the range of sensitivity studies decided on?

NUREG/CR-6616 - The sensitivity calculations were selected based on the major components modeLled in the PSA.

45. Has an uncertainty analysis been carried out?

a) How were different types of uncertainties (e.g., epistemic, aleatory) addressed?

1) NUREG/CR-6144- A level-3 PSA with uncertainly analysis was performed for internal events only. Only epistemic uncertainties were considered. No uncertainty analysis was performed for internal floods, internal fires, and seismic events.

2) Screening analysis of NUREG/CR-6144- No uncertainty analysis was performed.

3) NUREG/CR-6616- A level-3 PSA with uncertainly analysis was performed for internal events only. Only epistemic uncertainties were considered.

46. Have you been able to generalize the results and use the insights produced as a result of sensitivity and uncertainty analyses?

1) NUREG/CR-6144- The results of uncertainty analysis were used in a comparison with those of full power operation.

2) Screening analysis of NUREG/CR-6144- Not applicable.

3) NUREG/CR-6616- The results of uncertainty analysis were used in a comparison with those of full power operation.

4) Before generalizations are made further work would be needed to understand the similarities and differences in the other PWRs to determine how the studies could be used for this purpose. Nevertheless, the shutdown studies have underscored the importance of understanding shutdown risk and have helped to support activities such as those discussed previously in Q2 and Q56.

APPENDIX I - PLANT DAMAGE STATES

PRELUDE AND QUESTIONS

In order to assist the analysis of the physical processes taken place during an accident, it is convenient to group the various accident sequences resulted from the Level 1 analysis into "plant damage states" (PDSs). PDSs are defined on the basis of the operability of plant systems (e.g., the availability of plant spray) and certain key physical conditions (e.g., reactor coolant system pressure). Determining the PDSs for LPSD analyses may be more complicated because the technical specifications allow more equipment to be inoperable at LPSD conditions than at full power.

47. What PDSs have been defined for use in the LPSD PSA?

a) How do these relate to the PDSs used in the full power PSA?

b) What additional PDSs need to be defined?

RESPONSES

Belgium

47. What PDSs have been defined for use in the LPSD PSA?

The LPSD PSA outcomes have not been classified into PDSs.

a) How do these relate to the PDSs used in the full power PSA?

Not relevant because no assessment of consequences beyond core damage for low power and shutdown sequences.

b) What additional PDSs need to be defined?

Not relevant.

Germany

47. What PDSs have been defined for use in the LPSD PSA?

PDS have not been considered.

Hungary

47. What PDSs have been defined for use in the LPSD PSA?

The level 2 PSA including PDS definitions are currently being performed.

a) How do these relate to the PDSs used in the full power PSA?

The level 2 PSA being performed is related to both full power and LPSD states simultaneously.

Japan

47. What PDSs have been defined for use in the LPSD PSA?

a) How do these relate to the PDSs used in the full power PSA?

In NUPEC, PDSs different from PDS used in the full power is used considering specify the accident feature of LPSD.

b) What additional PDSs need to be defined?

In NUPEC, level 2PSA study for LPSD is under going. The final definitions of PDS are not yet determined.

Two additional questions / responses were included:

What areas of your analysis approach are, in your opinion, most in need of improvement?

In NUPEC, level 2PSA study for LPSD is under going. The areas which need improvement are not yet determined.

What are the areas that would most benefit from further research?

In NUPEC, core damage sequences for LPSD would be grouped by level 2PSA.

Korea

47. What PDSs have been defined for use in the LPSD PSA?

a) How do these relate to the PDSs used in the full power PSA?

b) What additional PDSs need to be defined?

LPSD PSA scope does not cover Level 2 and 3.

Mexico

47. What PDSs have been defined for use in the LPSD PSA?

a) How do these relate to the PDSs used in the full power PSA?

b) What additional PDSs need to be defined?

LPSD PSA scope does not cover Level 2 and 3.

The Netherlands

47. What PDSs have been defined for use in the LPSD PSA?

a) How do these relate to the PDSs used in the full power PSA?

b) What additional PDSs need to be defined?

The following eight parameters have been selected for use in defining the NPP Borssele plant damage states.

- Containment bypass status (BYPASS)
  - i. No bypass
  - ii. Interfacing systems LOCA
  - iii. Steam Generator Tube Rupture
  - iv. LOCA plus Isolation failure
  - v. Containment failed by initiating event at or near time of plant trip
  - vi. Containment failed by initiating event > 72 hours after plant shutdown
- Sequence Type (SEQ-TYPE)
  - i. Transient
  - ii. Small LOCA
  - iii. Large/intermediate LOCA
- Reactor Vessel Status (RX-STATE)
  - i. Vessel open
  - ii. Vessel closed
- Fuel location/ Decay heat level (FUELSTATE)
  - i. Fuel in vessel (high decay heat)
  - ii. Fuel in vessel (low decay heat)
  - iii. Fuel in pool (high decay heat)
  - iv. Fuel in pool (low decay heat)
  - v. Fuel in vessel - > 72 hours after plant shutdown

vi. Fuel in vessel – SGTR with failure to scram

vii. Fuel in vessel – SGTR with failed feedwater

- Power recovery (POWREC)

  i. Prior to vessel breach/No station blackout

  ii. Prior to containment failure

- RCS pressure before vessel breach (RCSPRESS)

  i. High

  ii. Not low but recoverable by operator action

  iii. Intermediate

  iv. Low

- In-vessel (In-pool) injection before core slumping (INVESSINJ)

  i. TW/TA/TJH injection (bunkered primary reserve injection system/volume control system/high pressure ECCS)

  ii. TW/TA/TJL/TJH injection

  iii. TJL injection (low pressure ECCS)

  iv. None available

- Containment heat removal available or recoverable (CONTHTRM)

  i. Yes

  ii. Recoverable

  iii. No

A logic diagram has been constructed with these eight parameters as decision branches to aid in the assembly of special plant damage state characteristics from the matrix of all possible combinations allowed by the eight grouping parameters. The NPP Borssele PDS logic diagram has 111 endpoints.

Spain

47. What PDSs have been defined for use in the LPSD PSA?

a) How do these relate to the PDSs used in the full power PSA?

b) What additional PDSs need to be defined?

LPSD PSA scope does not cover Level 2 and 3.

Sweden

47. What PDSs have been defined for use in the LPSD PSA?

a) How do these relate to the PDSs used in the full power PSA?

b) What additional PDSs need to be defined?

CSNI Question number	Answer from Swedish plant	Answer
Q47	F1/2	The plant damage states for shutdown are: HSOH: CD due to leak above the core. HSBL: : CD due to leak below the core HSTON: CDS due to the loss of cooling with the reactor vessel head ON HSTOFF: CDS due to the loss of cooling with the reactor vessel head OFF
	R1	As of yet no level 2 assessments of LPSD have started.
	R2-4	R2, 1996: Similar to power level PSA, some modifications
	B2	The PDS from full power level 2 PSA were not used in the LPSD PSA
	O1	The same PDSs as for full power PSA have been used. Answer only valid for low power PSA, not outage period.
	O2	At the present we have the following PDS defined for LPSD from level 1 PSA. HS1, Unsuccessful reactor shut down. This PDS describes as following: Consequence HS1: ATWS sequences initiated by transient or LOCA. The reactor remains critical due to insufficient insertion of control rods. The power production of the reactor depends on the amount of successfully inserted control rods and start of emergency core cooling system. These sequences are of the type Sx-C or T-C. HS2, Core cooling lost This PDS involves the following sequences: Consequence HS2-D12: LOCA initiated core damage sequences, in which the containment exceeds its pressure limit due to insufficient pressure suppression function. The cause to the insufficiency may be a by-pass of the downcomers from drywell to the water volume of the wet well. In addition these sequences assume that the containment over pressurisation protection system is unavailable. It is either inoperable due to closed valves or due to too large leakage area. The containment will burst and it is also possible that the coolant injection pipes will simultaneously fail. These sequences are of the type Sx-D1-D2. Consequence HS2-X-L: LOCA initiated core damage sequences, in which the core cooling is not possible due to failed high pressure injection, and the depressurisation of the primary circuit fails. The core start to melt at high pressure, but the depressurisation is still possible after the recovery of the depressurisation function or the high-pressure injection function. Containment pressure increases before the core melt. Under certain conditions the recovery of primary circuit depressurisation system together with operating core spray system interrupts the core melt process before vessel failure. These sequences are of the type Sx-U-X. Consequence HS2-X-T: Transient initiated core damage sequences, in

which the core cooling is not possible due to failed high pressure injection, and the depressurisation of the primary circuit fails. The core melting begins at high pressure, but the depressurisation is still possible after the recovery of the depressurisation function or the high-pressure injection function. Containment pressure does not remarkably increase before the core melt. Under certain conditions the recovery of primary circuit depressurisation system together with operating core spray system terminates the core melt process before vessel failure. These sequences are of the type T-U-X.

Consequence HS2-V-L: LOCA initiated core damage sequences followed by successful depressurisation of the primary circuit. However, all core-cooling systems are initially lost, and the core melt process begins at low pressure. Containment pressure increases before the core melt. Recovery of core cooling is still possible, and it terminates the melt process under certain conditions before the vessel fails. These sequences are of the type Sx-U-V.

Consequence HS2-V-T: Transient initiated core damage sequences followed by successful depressurisation of the primary circuit. However, all core-cooling systems are initially lost, and the core melt process begins at low pressure. Containment pressure does not remarkably increase before the core melt. Recovery of core cooling is still possible, and it terminates the melt process under certain conditions before the vessel fails. These sequences are of the type T-U-V.

HS3, Residual heat removal lost. This PDS involves following sequences:

Consequence HS3-L: LOCA initiated core damage sequences, in which all containment cooling, including filtered venting, is lost. The sequences lead to core damage after the containment break days after the initiating event. The sequences involve unsuccessful filtered venting followed by containment break due to overpressure. These sequences are of the type Sx-Wx-W5.

Consequence HS3-T: Transient initiated core damage sequences, in which all containment cooling, including filtered venting, is lost. The sequences lead to core damage after the containment break days after the initiating event. The sequences involve unsuccessful filtered venting followed by containment break due to overpressure. These sequences are of the type T-Wx-W5.

OT1, Containment over pressurisation followed by successful depressurisation involves:

Consequence OT1: Transient initiated sequences without core damage. Residual heat removal does not operate; containment pressure increases and the operators carry out filtered venting before containment rupture occurs. This PDS lead to minor filtered releases of radioactive materials, although the core damage does not occur. If the filtered venting does not succeed, it will lead to HS3.

HS3 and OT1 sequences have been separated in level 1 PSA. OT1 sequences are not included in level 2 PSA, because they do not lead to core damage.

OT2, Reactor over pressurisation. This PDS involves:

Consequence OT2: Transient or LOCA initiated core damage sequences, in which reactor pressure control is unavailable. The coolant injection is not possible and the primary circuit will pressurise up to 11 MPa. That pressure will open slightly the reactor tank lid, because the lid bolts will yield. Containment pressure increases before the core melt

		like in LOCA sequences. These sequences are of the type T-M; LOCA initiated sequences were not identified.
	O3	The PDS's are the same as those used in the full power PSA.
SKI REMARK	For LP the same PDSs as for power operation have been used.	
Q47 a	F1/2	Unique to LPSD
	R1	N.A.
	R2-4	R2, 1996: Similar to power level PSA, some modifications
	B2	See the answer above
	O1	The same PDSs as for full power PSA have been used. Answer only valid for low power PSA, not outage period.
	O2	They are similar
	O3	See Q47.
SKI REMARK	See Q47.	
Q47 b	F1/2	
	R1	No level 2 analysis as started as of yet.
	R2-4	R2, This can be considered after level 2 for full power PSA has been updated
	B2	The PSA Level 2 is updated. This can form a basis for a new LPSD PSA
	O1	This is a question that hasn't been reflected upon yet; when work on a Level 2 analysis is started perhaps some specific PDS for low power condition will be defined. For the time being it more of a question to decide whether an end state is a core damage consequence or not. Answer only valid for low power PSA, not outage period.
	O2	
	O3	See Q47.
SKI REMARK	The joint Nordic project will cover this question.	

United Kingdom

47. What PDSs have been defined for use in the LPSD PSA?

a) How do these relate to the PDSs used in the full power PSA?

Only 2 additional descriptors were used, one for the RCS intact and RHR connected (N_1) and one for the RCS open (N_2)

b) What additional PDSs need to be defined?

The 2 additional descriptors resulted in an additional 4 POSs being identified, to cover for sequences with/ without containment fan cooling.

United States

BWR

47. What PDSs have been defined for use in the LPSD PSA?

A total of twelve (12) PDSs were defined for this POS-5 study. A brief description of each is provided below.

Description of PDSs in NUREG/CR-6143

PDS1-1:

The accidents in PDS1-1 are initiated by a LOCA (A or S1) while the plant is in Time Window 1. At the onset of core damage, the vessel integrity is breached (via the break), the vessel is at low pressure, and the containment is open. Although offsite power is available during the accident, both the core cooling and containment cooling functions are lost for the entire accident.

PDS1-2:

The accidents in PDS1-2 are initiated by a loss of offsite power (T1) followed by a failure of the Train B emergency diesel generator either to start or to run for sufficient time to prevent core damage. The initiating event occurs while the plant is in Time Window 1. At the time of core damage the vessel integrity has been breached and the primary system is at low pressure. The containment cannot be closed and the core and containment cooling systems cannot be restored.

PDS1-3:

This PDS is similar to PDS1-2. At the time of core damage, the vessel is at system pressure with pressure relief provided by the SRVs cycling at their setpoints. The reactor vessel head vent is also open. The containment cannot be closed and the core and containment cooling systems cannot be restored.

PDS1-4:

This PDS is similar to PDS1-2. At the time of core damage the vessel integrity has been breached, the primary system is at low pressure, and the containment equipment hatch is open. It is possible to restore offsite power after the onset of core damage. Following recovery of ac power, low pressure ECCS can be used to provide coolant to the core.

PDS1-5:

The accidents in PDS1-5 are initiated by a valve mis-alignment that diverts vessel water to the suppression pool via the residual heat removal (RHR) system (H1) while the plant is in Time Window 1. At the onset of core damage, two SRVs are open, the reactor vessel head vent is closed, and the primary system is at low pressure. The containment is open and all core and containment cooling systems are lost for the entire accident.

PDS2-1:

The accidents in PDS2-1 are initiated by a LOCA (A or S1). This PDS is the same as PDS1-1 except that the accident is initiated while the plant is in Time Window 2.

PDS2-2:

The accidents in PDS2-2 are initiated by a loss of offsite power (T1). This PDS is the same as PDS1-2 except that the initiating event occurs while the plant is in Time Window 2.

PDS2-3:

The accidents in PDS2-3 are initiated by a loss of offsite power (T1). This PDS is the same as PDS1-4 except that the initiating event occurs while the plant is in Time Window 2.

PDS2-4:

The accidents in PDS2-4 are initiated by a diversion of vessel water to the suppression pool via the RHR system because of a misalignment of valves. This PDS is the same as PDS1-5 except that the initiating event occurs while the plant is in Time Window 2.

PDS2-5:

The accidents in this PDS are initiated by a loss of all SSW (T5A). At the onset of core damage, the primary system is at system pressure and the vessel head vent is open. All core and containment cooling systems are lost for the entire accident. The containment can be either open or closed. In the case where the containment is closed prior to core damage, the containment vent system is available to relieve the pressure in the containment.

PDS2-6:

The accidents in PDS2-6 are initiated by a valve misalignment that diverts vessel water to the suppression pool via the RHR system (H1) while the plant is in Time Window 2. At the time of core damage the MSIVs are open, which establishes a direct path from the RPV to the turbine building, which is outside the containment. Because the MSIVs are open, the primary system is at low pressure. Offsite power is available. The containment can be either open or closed. All core and containment cooling systems are lost for the entire accident.

PDS3-1

The accidents in PDS2-1 are initiated by a LOCA (A or S1). This PDS is similar to PDS1-1 except that the accident is initiated while the plant is in Time Window 3 and HPCS was initially available.

a) How do these relate to the PDSs used in the full power PSA?

There is no direct relationship.

Comparison, however, the following full power plant damage states (described by type of initiator or sequence) to the Grand Gulf study NUREG/CR-4551, "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Grand Gulf, Unit 1" shows that plant damage states for the LPSD study differ due to accounting for LPSD-specific events (e.g., flow diversion events which impact RHR) and due to events that are no longer applicable (e.g., ATWS).

b) What additional PDSs need to be defined?

The PDSs needed to reflect differences within the POS (POS 5). This was reflected in the configuration of the reactor coolant system boundary and the containment status. For example, the reactor vessel head vent

may be open during the accident, the containment may be open, or the suppression pool level may be at "low level" or "drained level". The PDSs also needed to distinguish between different time windows after shutdown. The time windows are defined as:

Time window 1: starts 14 hours after shutdown and has a duration of 10 hours

Time window 2: starts 24 hours after shutdown and has a duration of 70 hours

Time window 3: starts 40 days after shutdown and has a duration of 10.4 days

PWR

47. What PDSs have been defined for use in the LPSD PSA?

1) NUREG/CR-6144- PDSs different from those in the full power study (NUREG/CR-4551) are:

Addition of:

Time of Accident Initiation

1: Window 1

2: Window 2

3: Window 3

4: Window 4

Human Error

N: No human error or non-recoverable human error

D: Diagnosis error

A: Action error

RCS Status at Onset of Core Damage

L: Low pressure

G: 5% probability that pressure is high

Changes in the following which account for human error versus hardware failure and outage practices:

ECCS Status

U: Hardware failure

R: Recoverable if human error, LOSP, or 4KV is recovered

C: Failure of recirculation

Recirculation Spray Status

R: recoverable

U: not recoverable

Removal of:

Heat Removal from the Steam Generators

X: AFW operating, SGs not depressurized

Y: AFW operating, SGs depressurized

S: AFW recoverable, SGs not depressurized

D: AFW recoverable, SGs depressurized

N: no AFW operating, no AFW recoverable

Cooling for Reactor Coolant Pump Seals

Y: operating

R: not operating, but recoverable

N: not operating, not recoverable

2) Screening analysis of NUREG/CR-6144- No PDS analysis was done.

3) NUREG/CR-6616- Same as 1).

a) How do these relate to the PDSs used in the full power PSA?

No direct relationship can be established with the PDSs of the full power PSA.

b) What additional PDSs need to be defined?

See above.

APPENDIX J - CONTAINMENT PERFORMANCE

J.1 Containment Analysis

PRELUDE AND QUESTIONS

Containment performance is examined as part of the accident progression analysis and the evaluation of radioactive releases after core damage has occurred. Conventional type approaches (used in full power PSAs) can be adapted for the analysis of containment performance associated with accidents occurred during low power because they can lead to energetic events severe enough to challenge the containment in a way similar to those at full power. These approaches, however, do not appear to be credible for shutdown analysis during which the power is very low and, therefore, containment performance issues that needed to be explored may be quite different.

Containment event tree analysis

Containment event trees (CETs) are used to model and quantify the progression of accident sequences. In a full power PSA, for each general type of accident defined by a PDS, CETs are developed by considering the important characteristics of core melting processes, the challenges of the containment building, and the response of the building to those challenges. However, CET development for LPSD PSAs presents unique issues such as addressing containment isolation failure, containment bypass, and containment status (open or closed). For example, failure of operators to close the lower personnel airlock (in BWRs) or equipment hatch leads to containment being open during an accident.

48. How have you carried out the accident sequence modelling for containment?

a) How does this relate to what was done in the full power PSA?

49. How has the analysis been quantified?

50. Where a CET approach has been used, how have the branch point probabilities been derived/ justified?

RESPONSES

Japan

48. How have you carried out the accident sequence modelling for containment?

a) How does this relate to what was done in the full power PSA?

In NUPEC, container analysis is due to be carried out about the accident sequence with the large generating frequency obtained from level 1PSA of LPSD different from the accident sequence used in the full power PSA.

49. How has the analysis been quantified?

In NUPEC, the plant modelled for the container analysis of LPSD is inquiring. MELCOR code (it was developed by SNL in the United States) is applied to analysis.

50. Where a CET approach has been used, how have the branch point probabilities been derived/ justified?

In NUPEC, CET approach of LPSD would be used. The branch probabilities of CET would be derived by considering actual plant conditions.

Two additional questions / responses were included:

What areas of your analysis approach are, in your opinion, most in need of improvement?

In NUPEC, level 2PSA study for LPSD is under going. The areas which need improvement are not yet determined.

What are the areas that would most benefit from further research?

In NUPEC, the frequency corresponding to the release categories would be obtained from level 2PSA for LPSD.

The Netherlands

48. How have you carried out the accident sequence modelling for containment?

a) How does this relate to what was done in the full power PSA?

49. How has the analysis been quantified?

50. Where a CET approach has been used, how have the branch point probabilities been derived/ justified?

The events in the CET were chosen for the following purposes.

- To represent the uncertainties in physical phenomena (e.g., quenching behaviour of the melt, DCH, containment loading)
- To assess operator recovery and mitigation actions
- To assess consequential failure of important systems caused by specific phenomena (e.g., H_2 burn or by the general severe accident environment)

The numbers of headings that are required for a CET to depict the important accident progression possibilities and to define the spectrum of possible outcomes need not to be large. Additional event details required for the quantification of CET events have been relegated to decomposition event trees (DETs) which are evaluated for all branch points to determine the likelihood of each branch occurrence.

The CETs and DETs address the following scenarios and phenomenological events

- Decay heat level

i. High (prior to core reload)

ii. Low (after core reload)

- Containment bypass

i. High temperature induced failure of steam generator tubes including the possibility of the accident management measures to isolate the failed SG or to cover the leak with water

ii. Steam generator tube rupture including the possibility of the accident management measure to cover the leak with water

iii. Interfacing system LOCA

iv. LOCA with failure of containment isolation function.

- Arrest of core melt/debris within RPV or fuel pool (automatic or by operator action)

i. Depressurization of primary system due to high temperature induced leak in piping connected to the primary system

ii. Water injection into RPV and containment heat removal after core damage but before RPV break.

- Early containment failure

i. Steam or hydrogen production due to unsuccessful debris cooling

ii. Hydrogen burn including the effect of the hydrogen recombination system

iii. In-vessel steam explosion Direct Containment Heating

iv. RPV failure at high pressure

v. External causes

- Cooling of core melt/debris within RPV cavity

i. Core melt/debris coolability

ii. Water injection into cavity

- Late containment failure

i. Function of containment depressurization, e.g., filtered venting system

ii. Hydrogen burn

iii. Basemat melt-through

Three main sources of data are used to quantify the CETs

- The deterministic source term calculations (MAAP- and MELCOR) providing information of timing (e.g., speed of core degradation, amount of hydrogen produced, etc.)
- Expert judgments as presented, e.g. in the SERG-review, IDCOR (Industry Degraded Core Program)
- Results from previous PSAs for plants like NPP Borssele, e.g., NUREG-1150 and the German Risk Study, Phase-B.

From the description of the Plant Damage States and the CETs and DETs it easily can be seen that the shut-down stated are accommodated in the study of the NPP Borssele.

Containment event trees have been developed for each plant damage state. The top events in the CET consist of phenomenological events or processes and consequential system failures or functions resulting from physical phenomena, human actions, or the accident environment which are considered to be important for the assessment of further accident phenomena or events and for the definition of the source term and the time, mode and location of release of fission products. The severe accident phenomena and containment events specified in NUREG-1335 have been evaluated for inclusion in the CET. The detailed set of events developed for NUREG-1150 or NUREG/CR-4551 have also been considered.

Specific events to be included in each CET are determined to a large extent by the characteristics of the sequences in each plant damage state with which a particular CET is associated.

Event timing is a key factor in organizing the events on the CET. The accident progression is divided into distinct time periods for both power and non-power POSs for which different phenomenological processes are important and for which different recovery and mitigation actions may be effective. The general time periods are the following.

- Prior to failure of the support plate
- Prior to RPV failure
- At or within a few hours after RPV failure
- Late- many hours after RPV failure (before containment failure).

The CET structure is identical for all PDSs. However, according to particular PDS characteristics, some branches may be suppressed and the quantification differs.

Sweden

48. How have you carried out the accident sequence modelling for containment?

a) How does this relate to what was done in the full power PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q48	F1/2	With separate containment event trees connecting PDS with RC.
	R1	No level 2 assessment of LPSD has started as of yet.
	R2-4	R2, No answer

	B2	Through engineering judgements
	O1	No CET analysis has been performed yet. Answer only valid for low power PSA, not outage period.
	O2	The CETs for LPSD have been carried out by engineering judgement
	O3	The CETs for LPSD have been carried out by engineering judgement
SKI REMARK	Accidents with open containment are addressed to level 2.	
Q48 a	F1/2	With separate containment event trees connecting PDS with RC.
	R1	N. A.
	R2-4	R2, No answer
	B2	The phenomena analysis is more limited in LPSD PSA compared with Level 2 PSA
	O1	No CET analysis has been performed yet. Answer only valid for low power PSA, not outage period.
	O2	In the full power PSA we have done calculations with MAAP. These MAAP calculations have also been used for the LPSD PSA.
	O3	In the full power PSA we have done calculations with MAAP. These MAAP calculations have also been used for the LPSD PSA.
SKI REMARK	For LP the same as for power operation. For SD simplified assumptions.	

49. How has the analysis been quantified?

CSNI Question number	Answer from Swedish plant	Answer
Q49	F1/2	Frequencies for different release categories are calculated with RSW.
	R1	N. A
	R2-4	R2, No answer
	B2	Through engineering judgements
	O1	No CET analysis has been performed yet. Answer only valid for low power PSA, not outage period.
	O2	For the LPSD PSA for level 2 we have used @risk which is a complement to Microsoft Excel.
	O3	For the LPSD PSA for level 2 we have used @risk which is a complement to Microsoft Excel.
SKI REMARK	Different analysis tools have been used.	

50. Where a CET approach has been used, how have the branch point probabilities been derived/ justified?

CSNI Question number	Answer from Swedish plant	Answer
Q50	F1/2	By Fauske & associates inc.
	R1	N. A.
	R2-4	R2, No answer
	B2	Actual CETs were not used.

O1	No CET analysis has been performed yet. Answer only valid for low power PSA, not outage period.
O2	In the LPSD PSA we have used a triangle distribution for most of the branch points.
O3	In the LPSD PSA we have used a triangle distribution for most of the branch points.

SKI REMARK More use of models.

United Kingdom

48. How have you carried out the accident sequence modelling for containment?

a) How does this relate to what was done in the full power PSA?

The Level 2 PSA for Sizewell B also included 4 Plant Damage States (PDS) defined for faults at shutdown (a total of 14 core damage PDSs were defined for Sizewell B PSA). The representative accident sequences used for the 4 PDSs took on board some key thermal-hydraulic parameters in contrast to faults at power, including different RCS water inventory, lower decay heat, reactivity margins and lower RCS temperature and pressure. Some analysis was performed based on decay heat corresponding to conditions arrived at following Station Operating Instructions.

For most plant damage conditions arising from shutdown states, the accident sequences could be appropriately allocated to the core damage PDSs defined for power operation. The 4 PDSs were defined to address specifically the faults arising during operations at cold shutdown and refuelling because the accident progression and the associated phenomena are significantly different from those for faults at power (depending on RCS intact state & RCS level and also Auxiliary Building leaks when shutdown systems connected).

The accident sequences representing the 4 shutdown PDSs can be divided into 2 broad categories: those in which the RCS pressure boundary is intact at the start of the sequence, and those in which the RCS has been fully depressurised and some part of it is open to the containment atmosphere. The key distinction between these two categories is the discharge pressure at RPV failure which has a major impact on the subsequent melt debris disposition in the containment building.

50. Where a CET approach has been used, how have the branch point probabilities been derived/ justified?

The same Containment Event Tree (CET) was used for the quantification for the original four shutdown PDSs. The conditional probabilities were largely based on severe accident analysis using the MAAP3 code, using the rationale described in [Ang ML and Buttery NE, 'An Approach to the Application of Subjective Probabilities in Level 2 PSAs', Reliability Engineering and System Safety, 58(1997)145-156]

United States

BWR

48. How have you carried out the accident sequence modelling for containment?

The responses below are based on the detailed studies performed for NURE/CR-6143 and for NUREG/CR-6144.

a) How does this relate to what was done in the full power PSA?

NUREG/CR-6143: The process—an accident progression event tree (APET)— is the same; the only differences are in the number of questions contained in the tree (fewer for LPSD), the level of detail associated with various issues (less for LPSD), and use of formal expert judgement procedures (not used for LPSD).

49. How has the analysis been quantified?

NUREG/CR-6143: The EVNTRE code (NUREG/CR-5174) was used to quantify the APET

Q50 Where a CET approach has been used, how have the branch point probabilities been derived/ justified?

NUREG/CR-6143: The probabilities in the POS 5 APET were quantified using information from the following sources:

- Level 1 Analysis: The frequencies for the PDSs were obtained from the Level 1 analysis described in Volume 2 of this report.

- HRA analysis: A Human Reliability Analysis (HRA) was performed to determine the human error probability for operator actions during the core damage process (e.g., containment closure and the recovery of core cooling and containment cooling functions). For the sake of consistency, wherever possible, the same HRA models and techniques used in the Level I analysis were also used in this study.

- MELCOR calculations: A series of MELCOR calculations was performed specifically for this study. Results from these calculations helped guide the development and quantification of the APET. Specifically, the MELCOR calculations were used to determine the timing of key events (e.g., onset of core damage, vessel failure, containment failure) and the pressure, temperature, and composition histories of the containment and auxiliary building.

- Data from the NUREG-1150 PRAs: Where appropriate, data used in the NUREG-1150 full power PRA of Grand Gulf were also used in this study (e.g., structural capacity of the containment to static loads).

For those events judged to be important to risk, and for which there existed a large amount of uncertainty concerning the value to assign to the branch probability, an uncertainty distribution was assigned to the probability. Twenty-three variables in the APET were included in the uncertainty analysis. For the remaining events, those either judged to be less important or for which the branch probability was not believed to be uncertain, a single value was used.

PWR

48. How have you carried out the accident sequence modelling for containment?

NUREG/CR-6144, Vol 6.- An accident progression event tree (APET) was developed specifically to address mid-loop operation. Special issues related to shutdown were emphasized. These included the ability to close containment and achieve pressure retaining capability.

a) How does this relate to what was done in the full power PSA?

NUREG/CR-6144: The APET developed is significantly different from the full power APET described in NUREG-1150 for Surry. The full power APET had a large number of questions dealing with the specifics of core melt progression and the associated containment loads with an emphasis on determining the likelihood of early failure. This analysis indicated that there was a relatively low probability of early failure at full power. It was therefore assumed that if the containment is isolated during mid-loop operation that the likelihood of early containment failure is also low. This assumption reduced the number of top event questions needed for the mid-loop APET to a relatively small number. The APET for mid-loop operation therefore focused on determining if the hatch was open or could be closed, and the potential for late containment failure.

49. How has the analysis been quantified?

NUREG/CR-6144 Vol 6 The analysis was quantified using an accident progression event tree (APET) which is essentially a CET approach.

50. Where a CET approach has been used, how have the branch point probabilities been derived/justified?

NUREG/CR-6144 Vol 6 The timing to key events in the accident sequences (such as time for recovery prior to vessel breach, timing of late containment failure, etc.) was quantified using MELCOR results. The MELCOR code models severe accidents in light water reactors. It models the thermal hydraulic behaviour of the reactor cooling system, the heat-up and melting of the reactor core, the release of fission products from the damaged fuel to the containment building, the migration of the fission products through compartments in the containment, and the release of fission products to the environment if the containment is failed or bypassed. An important event to quantify was closing of the containment hatch. This was done by visiting the site and interacting with operating personnel to ascertain the time required to close the hatch and achieve pressure retaining capability. These times were compared to the recovery times in the accident sequences to determine the likelihood of success.

J.2 Severe Accident Analysis / Level 2 PSA Phenomenology

PRELUDE AND QUESTIONS

The accident phenomenology is different for LPSD and full power conditions. For example, the timing of an accident may be quite different at LPSD than at full power. The probability of early containment failure caused by phenomena such as direct containment heating and hydrogen combustion is expected to be lower during LPSD accidents (assuming that the containment has the same capability to hold pressure as in full power). However, the cooling capability of debris given low decay heat has not been fully investigated.

At full power, hydrogen combustion is precluded in BWR Mark I and Mark II plants because the containment atmosphere is inerted with nitrogen. However, during LPSD operation, the containment may be de-inerted and no hydrogen control systems are available; therefore, hydrogen combustion may cause an early over-pressurization of the containment that can lead to loss of containment integrity. Although low decay heat levels are considered to have some mitigating effects on drywell melt-through in BWR Mark I containments, an assessment of the impact of decay heat on this containment failure mode has not been performed. Please respond to the following questions.

51. How has the severe accident analysis been carried out?

a) What methods and codes have been used?

52. How does this differ from what has been done for full power?

RESPONSES

Japan

51. How has the severe accident analysis been carried out?

a) What methods and codes have been used?

In NUPEC, the plant modelled in order to obtain the source term of radiation study for LPSD was studied. MELCOR code is applied to analysis.

52. How does this differ from what has been done for full power?

In NUPEC, the accident analysis will be performed using the plant conditions and analytical nodes considering specific accident sequences of LPSD different from full power operation.

Two additional questions /responses were included:

What areas of your analysis approach are, in your opinion, most in need of improvement?

In NUPEC, level 2PSA study for LPSD is under going. The areas which need improvement are not yet determined.

What are the areas that would most benefit from further research?

In NUPEC, the thermal hydraulic accident progression for each core damage sequence would be known from the level 2PSA for LPSD.

Sweden

51. How has the severe accident analysis been carried out?

a) What methods and codes have been used?

CSNI Question number	Answer from Swedish plant	Answer
Q51	F1/2	MAAP 4-analysis of 4 selected, specific scenarios related to shut down conditions.
	R1	No level 2 assessment of LPSD has started as of yet.
	R2-4	R2, No answer
	B2	See the answers above
	O1	Only a Level 1 PSA has been conducted in the most recently performed study. The Level 1 model is however prepared for Level 2 analysis by definition of PDSs in the event trees as well as Level 1 consequences. Answer only valid for low power PSA, not outage period.
	O2	We have not yet done any accident analysis for the LPSD PSA. We have by engineering judgement used the analyses done for the full power PSA.

	O3	We have not yet done any accident analysis for the LPSD PSA. We have by engineering judgement used the analyses done for the full power PSA.
SKI REMARK	Most studies have not yet carried out any severe accident analyses for SD.	
Q51 a	F1/2	The selected scenarios, over the difference.
	R1	N. A.
	R2-4	R2, No answer
	B2	No answer
	O1	See answer to question Q51 above. Answer only valid for low power PSA, not outage period.
	O2	We have used MAAP
	O3	We have used MAAP
SKI REMARK	See Q51.	

52. How does this differ from what has been done for full power?

CSNI Question number	Answer from Swedish plant	Answer
Q52	F1/2	The selected scenarios, over the difference.
	R1	N. A.
	R2-4	R2, No answer
	B2	No answer
	O1	See answer to question Q51. Answer only valid for low power PSA, not outage period.
	O2	
	O3	
SKI REMARK	See Q51.	

United Kingdom

51. How has the severe accident analysis been carried out?

a) What methods and codes have been used?

MAAP3 was the main accident analysis code

52. How does this differ from what has been done for full power?

The characteristic accident sequences defined for the PDSs are obviously different to those for faults at power and took on board in the modelling some of the key system parameters mentioned in Q48. Also the implications of some aspects of severe accident phenomena more unique to shutdown sequences were also assessed. Examples are the air ingression phenomenon which would lead to the presence of an oxidising environment. The oxidising conditions have implications on zirconium oxidation (air instead of steam) and oxidation of UO_2 and ruthenium.

United States

BWR

51. How has the severe accident analysis been carried out?

The entire accident progression analysis including containment response was described in the responses to Q48, Q49, and Q50.

a) What methods and codes have been used?

The entire accident progression analysis including containment response was described in the responses to Q48, Q49, and Q50.

52. How does this differ from what has been done for full power?

The entire accident progression analysis including containment response was described in the responses to Q48, Q49, and Q50.

PWR

51. How has the severe accident analysis been carried out?

NUREG/CR-6144, Vol 6. - The analysis was carried out using both severe accident analysis codes and expert elicitation. However, because the likelihood of early containment failure was extremely low during low power operation, there was not a need for extensive expert elicitation related to severe accident phenomena, as described in the NUREG-1150 analysis for Surry.

a) What methods and codes have been used?

NUREG/CR-6144, Vol 6.- The accident sequence modelling was carried out by running MELCOR to estimate the occurrence and impact of severe accident phenomena and to establish the accident progression. This analysis was supplemented by expert elicitation related to unique shutdown issues.

52. How does this differ from what has been done for full power?

NUREG/CR-6144, Vol 6. - Because the likelihood of early containment failure was extremely low during low power operation, there was not the emphasis on severe accident analysis in quantifying the mid-loop APET that was needed for quantifying the full power APET, as described in the NUREG-1150 analysis for Surry.

J.3 Source Term Analysis

PRELUDE AND QUESTIONS

The fractions of the radioactive material released into the environment, the timing, and other release information is used to calculate the offsite consequences, together termed as "source terms." The methods and codes used for full power source term determination are modified to incorporate the unique features of the different POSs analysed. However, there is a concern that there is a lack of understanding of the behaviour of a degraded core under LPSD conditions. For example, the types of fission products that could be released from an undercooled core exposed to an air atmosphere, as in some LPSD scenarios, have not been extensively investigated. Another concern is that the source term mechanisms needed to

achieve a release with early fatalities do not exist during outages. Therefore, current LPSD analyses may have overestimated the consequences associated with LPSD accidents.

53. How have the radiological source terms been derived?

a) What methods and codes have been used?

b) How does this differ from what has been done for full power?

RESPONSES

Japan

53. How have the radiological source terms been derived?

a) What methods and codes have been used?

In NUPEC, plant modelling in order to obtain the radiological source term for LPSD has been studied. The MELCOR code, which was developed at SNL in USA, is applied to the analysis.

b) How does this differ from what has been done for full power?

In NUPEC, source term analysis for LPSD will be performed by 2004. Based on the insights from the conventional level 2 PSA for LPSD, source terms during open periods of the containment vessel would become important comparing with full power operation.

Two additional questions / responses were included:

What areas of your analysis approach are, in your opinion, most in need of improvement?

In NUPEC, level 2PSA study for LPSD is under going. The areas which need improvement are not yet determined.

What are the areas that would most benefit from further research?

In NUPEC, the radiological source term and frequencies would be obtained from the level 2PSA for LPSD.

Sweden

53. How have the radiological source terms been derived?

a) What methods and codes have been used?

b) How does this differ from what has been done for full power?

CSNI Question number	Answer from Swedish plant	Answer
Q53	F1/2	
	R1	No level 2 assessment of LPSD has started as of yet.

	R2-4	R2, The same source terms as in Level 2 PSA were used
	B2	The same source terms as in Level 2 PSA were used
	O1	See answer to question Q51. Answer only valid for low power PSA, not outage period.
	O2	Have not yet been done for LPSD PSA
	O3	Have not yet been done for LPSD PSA
SKI REMARK	MAAP is the program used in Sweden.	
Q53 a	F1/2	Three selected scenarios based on selections from the CET. Source terms calculated by MAAP 4.xx.
	R1	N. A.
	R2-4	R2, MAAP
	B2	For Level 2 PSA MAAP 3b
	O1	See answer to question Q51. Answer only valid for low power PSA, not outage period.
	O2	We are planning to use MAAP
	O3	We are planning to use MAAP
SKI REMARK	See Q53.	
Q53 b	F1/2	Three selected scenarios based on selections from the CET. Source terms calculated by MAAP 4.xx.
	R1	N. A.
	R2-4	R2, No difference
	B2	No answer
	O1	See answer to question Q51. Answer only valid for low power PSA, not outage period.
	O2	For the full power PSA we have used MAAP 4
	O3	For the full power PSA we have used MAAP 4
SKI REMARK	See Q53	

United Kingdom

53. How have the radiological source terms been derived?

a) What methods and codes have been used?

MAAP3 was the main analysis code

b) How does this differ from what has been done for full power?

See previous response

United States

BWR

53. How have the radiological source terms been derived?

The source term consists of the following information: the amount and type of radioactive material released from the containment, the timing characteristics of the release, the energy of the release, the elevation of the release, and the time when a general emergency is declared and evacuation plans are initiated (referred to as the warning time). The amount of material released is expressed as a fraction of the radionuclide inventory present in the core at the start of the accident for the radionuclides considered. Although many different radionuclides would be released from the damaged fuel during an accident, health effects from only 60 radionuclides are considered in the consequence analysis. Furthermore, because in the source term analysis the release and transport of these radionuclides are of concern (not the health effects from the radionuclides), it is possible to form groups of radionuclides that are expected to have similar release and transport characteristics. The 60 radionuclides considered in the consequence analysis are combined into nine release classes. The definition for each release class is based on the definitions used in the NUREG-1150 study as discussed in Volume 1 of NUREG/CR-4551. Similar to NUREG-1150, the timing of the release is characterized by two release segments: the first or early release segment and the second or late release segment. For each release segment, the start time of the release segment (relative to the start of the accident), the duration of the segment, the release fractions for the nine release classes, and the energy release rate are provided.

a) What methods and codes have been used?

The source term was estimated using a modified version of the parametric code GGSOR that was developed for the NUREG-1150 Grand Gulf plant study as documented in NUREG/CR-4551 Vol. 6. The specific version of GGSOR that was used in the analysis is GGSOR-P5. MELCOR was used to provide input into GGSOR

b) How does this differ from what has been done for full power?

GGSOR was modified to reflect plant configuration during plant operating state (POS) 5 and the types of accidents that are possible during this mode of operation. Specifically, GGSOR was modified to account for: (1) loss of coolant accidents (LOCAs) in the containment, (2) interfacing system LOCAs in the auxiliary building, (3) the passage of releases through the auxiliary building, (4) the passage of releases through the reactor pressure vessel head vent, and (5) the timing characteristics of the accidents initiated while the plant is in POS 5.

PWR

53. How have the radiological source terms been derived?

NUREG/CR-6144 Vol 6 - The source term model used for full power was considered suitable for use in mid-loop operation with only minor modification. The suitability was based on a comparison with point calculations from MELCOR and the views expressed by an expert review panel.

a) What methods and codes have been used?

The MELCOR code was used to obtain representative source terms and then the SURSOR code from the NUREG-1150 full power study was used for the parametric model. The SURSOR code is a fast-running

parametric model which is used to predict a large number of source terms associated with a wide range of accident sequences. The code has a number of retention factors associated with the various compartments in the flow paths of the release.

b) How does this differ from what has been done for full power?

The SURSOR code was modified and adapted to low power conditions.

APPENDIX K - CONSEQUENCE ANALYSIS

PRELUDE AND QUESTIONS

Consequence or Level 3 analysis evaluates the offsite effects of postulated sequences. The evaluation of consequences for LPSD may differ from that at full power due to the differences of the phenomenology involved. For example, the timing of a LPSD accident (e.g., during the early or late stages of the outage) has an impact on the severity of the consequences. Another concern that applies to both at power and shutdown studies relates to the quality of data and tools available for Level 3 analysis.

Radiological release modelling depicts the transport of radioactive material to the environment. The question is whether existing codes and data for Level 3 analysis are adequate. Some PSA users believe that source terms, site and surrounding characteristics, population distributions, meteorological data, plume dispersion models, should be updated, so that results from consequence analyses are appropriate for risk-informed decision making.

54. How has the radiological release modelling been carried out?

a) What methods and codes have been used?

b) How does this differ from what has been done for full power?

55. What societal risk measures have been addressed?

56. Have or will Level 3 analysis results be used in support of decision making?

Responses

Hungary

54. How has the radiological release modelling been carried out?

The level 3 PSA may be carried out after completion of the level 2 study.

a) What methods and codes have been used?

b) How does this differ from what has been done for full power?

The level 3 PSA will likely be related to both full power and LPSD states simultaneously.

55. What societal risk measures have been addressed?

The level 3 PSA may be carried out after completion of the level 2 study.

56. Have or will Level 3 analysis results be used in support of decision making?

The level 3 PSA may be carried out after completion of the level 2 study.

Japan

54. How has the radiological release modelling been carried out?

a) What methods and codes have been used?

In NUPEC, level 3 PSA for LPSD has not discussed yet. The MACCS-2 code, which was developed at SNL in USA, will be applied to the analysis. The MACCS-2 code has been modified at NUPEC for the analysis of radiological consequence of NPPs in Japan.

b) How does this differ from what has been done for full power?

In NUPEC, source term analysis for LPSD will be performed by 2004 as mentioned in Chapter 10. The results of the consequence analysis are strongly dependent on the results of source terms in level 2 PSA. Based on the insights from the conventional level 2 PSA for LPSD, source terms during open periods of the containment vessel would become important comparing with full power operation.

55. What societal risk measures have been addressed?

In NUPEC, an expectation value for radiation doses (dose-risk) around a site has been discussed for full power operation including a large early release frequency (LERF). As for LPSD, the same risk measure would be discussed.

56. Have or will Level 3 analysis results be used in support of decision making?

In NUPEC, the results of level 3 PSA would be applied to the discussions for Safety Goal and LERF in RIR as technical bases.

Two additional questions / responses were also included:

What areas of your analysis approach are, in your opinion, most in need of improvement?

In INS/NUPEC, consequence analysis for LPSD will be performed after the completion of level 2PSA study for LPSD. The areas which need improvement are not yet determined.

What are the areas that would most benefit further research?

In INS/NUPEC, the early fatality risk and the latent cancer fatality risk would be obtained from the level 3PSA for LPSD.

The Netherlands

54. How has the radiological release modelling been carried out?

a) What methods and codes have been used?

b) How does this differ from what has been done for full power?

55. What societal risk measures have been addressed?

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 Ionizing 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 licenses 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 ionizing 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 $2.5*10^{-2}$ 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.

56. Have or will Level 3 analysis results be used in support of decision making?

Yes, especially with regard to optimizing backfitting measures or SAMGs. Also the input from level-2 and level-3 analyses will be used for decision-making regarding off-site emergency planning and preparedness.

Sweden

54. How has the radiological release modelling been carried out?

a) What methods and codes have been used?

b) How does this differ from what has been done for full power?

CSNI Question number	Answer from Swedish plant		Answer
Q54	F1/2		Not in principles. The scenarios are different.
		R1	No level 2 assessment of LPSD has started as of yet.
		R2-4	R2, No level 3 analysis have been done
		B2	In the same manner as for Level 2 PSA
		O1	See answer to question Q51. Answer only valid for low power PSA, not outage period.
		O2	We have not yet done release calculations but plan to use the MAAP code
		O3	Not applicable (see Q1)
SKI REMARK	Not yet carried out in Swedish PSA but there are source term calculations.		
Q54 a	F1/2		
		R1	N. A.
		R2-4	R2, No answer
		B2	No answer
		O1	See answer to question Q51. Answer only valid for low power PSA, not outage period.

	O2	We plan to use MAAP
	O3	Not applicable (see Q1)
SKI REMARK	See Q53.	
Q54 b	F1/2	
	R1	N. A
	R2-4	
	B2	No answer
	O1	See answer to question Q51. Answer only valid for low power PSA, not outage period.
	O2	MAAP have been done
	O3	Not applicable (see Q1)
SKI REMARK	See Q53.	

55. What societal risk measures have been addressed?

CSNI Question number	Answer from Swedish plant	Answer
Q55	F1/2	
	R1	N. A.
	R2-4	
	B2	No answer
	O1	See answer to question Q51. Answer only valid for low power PSA, not outage period.
	O2	
	O3	Not applicable (see Q1)
SKI REMARK	None	

56. Have or will Level 3 analysis results be used in support of decision making?

CSNI Question number	Answer from Swedish plant	Answer
Q56	F1/2	
	R1	N. A.
	R2-4	R2, No
	B2	No
	O1	See answer to question Q51. Answer only valid for low power PSA, not outage period.
	O2	No. Level 3 analyses are not planned for
	O3	Not applicable (see Q1)
SKI REMARK	Level 3 studies are not considered in Sweden for the moment.	

United Kingdom

54. How has the radiological release modelling been carried out?

a) What methods and codes have been used?

CONDOR was the main analysis code

b) How does this differ from what has been done for full power?

Essentially the same as at power

55. What societal risk measures have been addressed?

Individual risk of death to a member of the public and societal risk (freq. of 100 fatal cancers)

56. Have or will Level 3 analysis results be used in support of decision making?

Currently forms part of the decision making process

United States

BWR

54. How has the radiological release modelling been carried out?

a) What methods and codes have been used?

Offsite consequences were calculated with version 1.5.11.1 of the MACCS code.

A scoping study was performed to estimate onsite consequences. Doses and dose rates were estimated for a range of distances from the reactor. For comparative purposes, two different wake-effect models were used to estimate the relative concentrations downwind of the reactor. The first model was developed by Ramsdell ("Diffusion in Building Wakes for Ground-Level Releases," *Atmospheric Environment*, Vol. 24B, No. 3, 377-388, 1990). The second model actually consists of two models: a model developed by Wilson, which was used to estimate doses within 100 meters of the plant, and a model utilized by the NRC, which was used to estimate doses beyond 100 meters (Wilson in *Atmospheric Science and Power Production*, Ed. Randerson, D., DOE/TIC-27601, 299, (1984); U.S. Nuclear Regulatory Commission, *Regulatory Guide 1.145, Revision 1*, November 1982). The Ramsdell model was developed by using multiple linear regression to fit experimental results to a statistical model that included the following four variables: (1) wind speed, (2) distance, (3) building area, and (4) stability. The result was that the exponent on each of these variables was determined by the experimental results. The Wilson and NRC models are based on Gaussian plume theory. The NRC model allows for plume meander during low speed and stable atmospheric conditions for distances less than 800 m. The Wilson models uses experimental data to fit the Gaussian model to the relative concentration for distances very close to the release point. Relative concentration marks an interesting difference between the two sets of models: In the Ramsdell model, the relative concentration is somewhat proportional to the wind speed and the stability class; in the Wilson and NRC models, the relative concentration is predicted to be inversely proportional to the wind speed. Using the integrated air concentrations for each building wake-effect model, the dose and dose rate were estimated for each partitioned source term group

The major simplifying assumptions used in this analysis were that radioactive decay was neglected during the exposure time, the directional dependence of the weather was ignored, and a single radioactive release location and building area was assumed. Two different weather scenarios were used for each set of models. The first scenario assumes stable conditions (stability class F) with a wind speed of 1 m/s. The second scenario assumes unstable conditions (stability class A) with a wind speed of 5 m/s. The dose calculation considered exposure from both the immersion and inhalation pathways and is a 50-year committed dose. The dose rate only considered exposure from the immersion pathway. Doses were calculated assuming both exposure to the entire release and exposure to only the first 15 minutes of the release. A dose rate was calculated for each of the release segments defined in the source term analysis.

b) How does this differ from what has been done for full power?

For offsite consequences, no differences. (Same code used, only a different version).

The full power analysis did not produce estimates for onsite consequences.

55. What societal risk measures have been addressed?

The following risk measures were reported:

- CDF
- Early Fatality Risk
- Total Latent Cancer Risk
- Population Dose within 50 miles of the plant
- Population Dose within 1000 miles of the plant
- Individual Early Fatality Risk—0 to 1 mile
- Individual Latent Cancer Risk— 0 to 10 miles

56. Have or will Level 3 analysis results be used in support of decision making?

Results from the LPSD analysis were used in the development of NRC's draft shutdown rule; however, this rule was never adopted. Even though the shutdown rule was never adopted, industry response to NRC's shutdown related activities lead to initiatives and activities that enhanced shutdown safety and help to manage risk.

PWR

54. How has the radiological release modelling been carried out?

NUREG/CR-6144 Vol 6 – The radiological release model used for mid-loop operation was similar to the model used for full power operation in NUREG-1150. The same suite of codes was used for both studies, with modifications to reflect the conditions pertinent to mid-loop operation.

a) What methods and codes have been used?

MELCOR was used to calculate point estimate release fractions. These calculations were used to benchmark SURSOR and minor changes were made to model mid-loop operations. The PARTITION code required extensive modifications in order to generate appropriate source terms to the consequence model. This was necessary in order to account for changing radionuclide inventories for the various accident sequences during mid-loop operation because they occur a long time after shutdown. The PARTITION code is used to reduce the very large number of source terms into a smaller number of representative terms for input into the consequence analysis. The partitioning is based on early and latent health effects. The MACCS was used to calculate the off-site consequences. The MACCS code is a probabilistic consequence assessment code that takes into account the variability of weather and other site specific factors, such as population, emergency response, etc. at a specific site. The MACCS calculations included the updates to the code in response to BEIR V, which increased cancer rates approximately 3 times for the same dose.

b) How does this differ from what has been done for full power?

Although a similar suite of codes was used, significant modifications were made to all of them to account for the special conditions at low power.

55. What societal risk measures have been addressed?

NUREG/CR-6144 Vol 6 – Early fatalities and latent cancers and population dose were calculated within 50 and 1000 miles. Individual risk (early fatalities within 1 mile and latent cancer fatalities within 10 miles) was also calculated to allow comparison with the USNRC's safety goals.

56. Have or will Level 3 analysis results be used in support of decision making?

Results from the LPSD analysis were used in the development of NRC's draft shutdown rule; however, this rule was never adopted. Even though the shutdown rule was never adopted, industry response to NRC's shutdown related activities lead to initiatives and activities that enhanced shutdown safety and helped to manage risk.

APPENDIX L - RESULTS OF THE LPSD PSA / INSIGHTS

PRELUDE AND QUESTIONS

The results of a LPSD for a PSA are typically reported in terms of the core damage frequency, large (early or late) release, and fatality risks. The results are also reported in terms of total CDF and terms of CDF per each POS analyzed. Initiating events, accident sequences, and equipment and human failures, that are found to be dominant contributors to the various risk metrics are also identified and reported. On the basis of these results, insights are derived with respect to the driving factors of the risk for the plant and if there are any improvements that are potentially worth implementing. Please respond to the following questions.

57. How the main results of the LPSD PSA compare to the results of the full power PSA?

58. How have the results of the LPSD PSA been presented?

a) How does this differ from what was done for the full power PSA?

59. What confidence do you have in the results of the LPSD PSA compared to the full power PSA?

60. To what degree does the confidence in your results allow or impair risk trade-off between full power and shutdown states?

61. What insights have been obtained from the LPSD PSA?

62. What further work is required/ planned to refine the analysis that has been carried out to date?

RESPONSES

Belgium

57. How the main results of the LPSD PSA compare to the results of the full power PSA?

For some plants the results are equivalent. For other plants there is a major dominance of the LPSD results compared to the full power results.

58. How have the results of the LPSD PSA been presented?

The risk dominant sequences have been identified per POS.

a) How does this differ from what was done for the full power PSA?

Identical.

59. What confidence do you have in the results of the LPSD PSA compared to the full power PSA?

Good confidence. However the uncertainties for the LPSD PSA are higher due to the increased activities.

60. To what degree does the confidence in your results allow or impair risk trade-off b between full power and shutdown states?

Since the same methodology was applied for power states and shutdown states, reasonable confidence can be given to risk trade-off between power states and shutdown states.

61. What insights have been obtained from the LPSD PSA?

Importance of sufficient pressure relief during mid-loop operation with SG nozzle dams installed.

62. What further work is required/ planned to refine the analysis that has been carried out to date?

To be determined

Germany

57. How the main results of the LPSD PSA compare to the results of the full power PSA?

The conditional probability of hazard states of core uncovery during LP&SD was determined in total of 6.7E-6/outage and is - assuming one outage per year - within the same order of magnitude as the frequency of hazard states during power operation.

58. How have the results of the LPSD PSA been presented?

The results have been presented as frequency of hazard states per year under the assumption of one refuelling-outage per year. The contribution of the initiating events to the overall result has been presented also.

a) How does this differ from what was done for the full power PSA?

The presentation of results is similar.

59. What confidence do you have in the results of the LPSD PSA compared to the full power PSA?

The confidence in the results is a bit lower than for the full power PSA because of the contribution of not analysed initiating events. The contribution has been estimated but a detailed analysis has not been performed for these events.

60. To what degree does the confidence in your results allow or impair risk trade-off between full power and shutdown states?

The confidence is still high enough to allow a trade-off between FP und LP&SD.

61. What insights have been obtained from the LPSD PSA?

The main insights were:

- The level of the contribution of initiating events during LP&SD to the overall risk.

- The importance of procedures to cope with initiating events.
- The importance of a higher availability of redundancies of safety systems in some POSs.
- The importance of the point in time when the reactor protection system should be disconnected.

62. What further work is required/ planned to refine the analysis that has been carried out to date?

A further analysis of some initiating events would be required, but at present no plans are in discussion.

Hungary

57. How the main results of the LPSD PSA compare to the results of the full power PSA?

The core damage risk originating from LPSD states is about the same as that originating from internal initiating events occurring at full power. (It is noted again that the LPSD PSA relates to internal initiating events.)

58. How have the results of the LPSD PSA been presented?

CDF and CDP are given for each POS. The risk profile of a refuelling outage is drawn based on the CDF values. Contribution of the different POSs as well as groups of POSs (like e.g. POSs with the reactor open, the primary circuit level is low) are presented based on the CDP values, and their sub-totals. Dominant minimal cutsets are described.

a) How does this differ from what was done for the full power PSA?

Only the CDF value was given for the full power PSA. The results for the full power PSA were presented as contribution of initiating events as well as initiating event groups (like e.g. large LOCA) rather than distribution of POSs (being the full power mode only one of the POSs).

59. What confidence do you have in the results of the LPSD PSA compared to the full power PSA?

Due to the fact that the methodology followed, and the approaches used during the development of the LPSD PSA were very similar to (sometimes the same as) those used in the full power PSA it is found that the confidence in the results can be (or should be) the same.

60. To what degree does the confidence in your results allow or impair risk trade-off between full power and shutdown states?

Due to the same reason (see Q59) it allows risk trade-off between full power and shutdown states to a high degree.

61. What insights have been obtained from the LPSD PSA?

The core damage risk is dominated by those POSs when the reactor vessel is open as well as by the POSs that directly precede and follow the open reactor state. The result is mostly explained by two factors: the difference between the average lengths of the POSs and the lack of automatic plant responses in some operational states.

Primary side LOCAs are the most significant processes within those POSs that dominate risk in low power and shutdown operation. Loss of coolant events can be induced mostly by inadvertent human actions rather than hardware failures. Another important scenario is loss of natural circulation due to loss of heat removal on secondary side or termination of circulation in primary system (excluding LOCA, since it is another category as discussed above). The primary cause to loss of natural circulation is loss of secondary cooling dominated by drop of a heavy load in turbine hall, human induced loss of secondary coolant and loss of secondary decay heat removal pumps. Drop of a heavy load during certain refuelling crane operations can endanger core coolability, which is also a risk significant fault. Inadvertent dilution scenarios have a notable contribution to the core damage risk. In comparison with these potential core damage processes, scenarios with cold overpressurisation of reactor vessel are less important due to some existing defences.

62. What further work is required/ planned to refine the analysis that has been carried out to date?

- refinement in some modelling assumptions and data, especially those related to human reliability analysis
- extension to outages other than a refuelling outage.

Japan

57. How the main results of the LPSD PSA compare to the results of the full power PSA?

The total CDF is estimated through the summation of the CDF during the LPSD and the full power operation, if the initiating event frequency is defined by the reactor year basis in the full power PSA and the LPSD PSA. However, if the CDF during the LPSD is estimated as the probability during the maintenance outage, the unit of the CDF needs to be changed into the reactor year basis.

58. How have the results of the LPSD PSA been presented?

a) How does this differ from what was done for the full power PSA?

The LPSD PSA is conducted using the interval of the plant operation state (i.e., POS). In the periodic safety review in Japan, the utility makes LPSD PSA for the typical schedule of the plant maintenance outage. Therefore, the CDF is estimated as the probability for the maintenance outage.

On the other hand, the LPSD PSA by NUPEC uses the operating experience of the plant to estimate the duration of the POS. The CDF is estimated as the annual probability of the core damage (i.e., per reactor year basis).

By the way, the NUPEC and the utilities estimate the CDF during the full power operation as the annual core damage probability.

59. What confidence do you have in the results of the LPSD PSA compared to the full power PSA?

In the LPSD PSA, the frequency of the initiating event induced by the human error has very large uncertainty. This induces the large uncertainty in the LPSD CDF, compared to the CDF of the full power operation.

60. To what degree does the confidence in your results allow or impair risk trade-off b between full power and shutdown states?

We consider both the full power PSA and the LPSD PSA are useful to judge the applicability of the online maintenances. In general, the online maintenance decreases the LPSD CDF, even though that increases the CDF of the full power. Therefore, the online maintenance may decrease the total CDF including the CDF of the LPSD and the full power. In such case, the online maintenance is considered to be allowable, in viewpoint of the risk information.

61. What insights have been obtained from the LPSD PSA?

The risk level (i.e., the LPSD CDF) may change through the plant configuration or the plant outage schedule. Therefore, the PSA of the LPSD becomes a useful tool to control and manage the plant outage schedule, in viewpoint of the plant risk level.

62. What further work is required/ planned to refine the analysis that has been carried out to date?

It is desirable to clarify the objectives of the LPSD PSA. One is the LPSD PSA used to control and manage the future plant outage schedule. In this case, the methodology should be developed to set forth the criteria to define the allowable plant configuration, and so on. The other is the LPSD PSA used to evaluate the plant safety level, based on the plant operation experience. We already fully developed in this area the LPSD PSA methodology concerning the internal event through the periodic safety review (PSR). However, there are several issues, such as the POS identification where the outage is extremely prolonged from the typical maintenance schedule, the risk contribution from the external events during the outage, and so on. The concept of the time window may be utilized to define the POS for the prolonged outage schedule.

The Netherlands

57. How the main results of the LPSD PSA compare to the results of the full power PSA?

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 %

Event	contribution to TCDF (Old Plant)	contribution to TCDF (Current Modified Plant)
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 %
	6.7 E-7 1.2 %	2.3 E-7 9.8 %
EXTERNAL EVENTS DURING POWER STATES	2.8 E-7 0.5 %	1.1 E-7 2.9 %
	1.7 E-7 0.3 %	2.0 E-8 0.7 %
- Vapour Cloud Explosions (Shipping LPG)	1.1 E-7 0.2 %	6.8 E-8 1.6 %
- External Flooding	5.6 E-8 0.1 %	1.3 E-8 0.3 %
- Toxic Gas Releases		
- Long-term Loss of Offsite Power due to external hazards	5.6 E-8 0.1 %	8.6 E-9 0.2 %
- Other	5.6 E-7 0.9 %	1.66 E-9 0.0 %
INTERNAL & EXTERNAL EVENTS DURING (EARLY + LATE) HOT STEAMING POS (LOCAs dominate; added Reactivity Addition Accident during Start-up; Increased Loss of Offsite Power, due to potential problems in shifting Offsite Power	1.1 E-6 1.9 %	2.0 E-7 7.0 %
	9.0 E-7 1.6 %	1.3 E-7 4.7 %
	< 5.6 E-8 <0.1 %	5.9 E-9 0.3 %
INTERNAL & EXTERNAL EVENTS DURING (EARLY + LATE) COLD SHUTDOWN	< 2.8 E-7 <0.5 %	5.9 E-8 2.0 %
- LOCA inside containment	1.6 E-5 28.2 %	1.1 E-6 40,2 %
- IS-LOCA	7.9 E-6 14.3 %	2.4 E-7 8.5 %
- Other	6.3 E-6 11.2 %	7.3 E-7 25.9 %
	6.2 E-7 1.1 %	4.2 E-8 1.5 %
INTERNAL EVENTS DURING MIDLOOP POS	3.4 E-7 0.6 %	8.5 E-8 3.0 %
- LOCA	3.4 E-7 0.6 %	3.1 E-8 1.1 %
- Fire	3.4 E-7 0.6 %	5.7 E-9 0.2 %
- Loss of RHR		
- Loss of 6 kV ac bus BU	5.0 E-7 0.9 %	4.8 E-8 1.7 %
- Loss of component cooling	2.2 E-7 0.4 %	1.4 E-8 0.5 %
- Other	2.2 E-7 0.4 %	1.0 E-8 0.3 %
	5.6 E-8 0.1 %	2.4 E-8 0.9 %
EXTERNAL EVENTS DURING MIDLOOP POS		
- Vapour Cloud Explosions (Shipping LPG)	< 1.0 E-9	< 1.0 E-9
- External Flooding		
- Other		
	1.7 E-6 2.8 %	2.0 E-7 7.0 %
INTERNAL & EXTERNAL EVENTS DURING CORE UNLOADING/LOADING POS	- 1.0 E-6 1.8 %	2.8 E-8 1.0%
	- 5.3 E-7 0.9 %	1.7 E-7 5.9%

Event	contribution to TCDF (Old Plant)	contribution to TCDF (Current Modified Plant)
INTERNAL & EXTERNAL EVENTS FUEL POOL POS (All Fuel in In-containment Fuel-Pool) - Loss of Support Systems (Station Blackout) - Loss of Fuel-Pool Cooling TOTAL	5.6 E-5 100 %	2.83 E-6 100 %

58. How have the results of the LPSD PSA been presented?

a) How does this differ from what was done for the full power PSA?

The results of the LPSD PSA were part of the results of the total PSA. In the report the results are given of the CDFs [year $^{-1}$] for each POS, for each Initiating Event Group (LOCAs, Spatial, transients, external events and support system initiators)(per POS and per total), per initiating event (per POS and per total), per plant damage state. Also for each POS the top 100 cutsets as well as the top 100 cutsets of the TCDF.

59. What confidence do you have in the results of the LPSD PSA compared to the full power PSA?

60. To what degree does the confidence in your results allow or impair risk trade-off between full power and shutdown states?

As long as the results can be explained in a deterministic way as well, decision-making regarding risk trade-offs can be allowed by the regulatory body. Recently, due to exceeding an AOT of the pump of the reserve cooling water system, a kind of a so-called Notice of Enforcement Discretion was a.o. issued based on the fact that the risks of the LPSD POSs cold shutdown and midloop were (much) higher (the duration of the respective POSs were assumed to be 1 year). These differences could be explained in a deterministic way as well.

61. What insights have been obtained from the LPSD PSA?

<u>Summary description of the Hot Steaming POS results.</u>

The hot steaming POS is a negligible contributor, with none of the event trees contributing above 1E-9 per year. All the event trees have been screened out. This is due to the short time interval in this POS (1% - 2% of the typical year), and the fact that the plant response is typically bounded by the power POS, the quantification of this POS is usually included with the power POS. A unique sequence in this POS, the dominant transient sequence is a reactivity addition accident during start up, a prompt criticality due to rapidly injection of a slug of pure unborated water in the core. During start-up there is a failure which allows pure water to accumulate in the Volume Control Tank. Next there is a loss of offsite power, loss of 6 kV buses BA and BB, or a simultaneous loss of both reactor coolant pumps. Flow through the core stops, but the pure water continues to inject into the loop. Core damage occurs when the reactor coolant pumps are restarted. This analysis is conservative in that no credit has been taken for mixing in the loops. With a sequence frequency of 7.4 E-10/year for the short cycle type, it hardly contributes to the TCDF.

<u>Summary description of cold shutdown results.</u>

The cold shutdown is also a minor contributor, 7% of the total core damage frequency (TCDF), due to the short time interval of this POS (0.8% - 1.5% of the year), and the fact that the plant response is typically bounded by the midloop POS. If decay heat removal fails due to the loss of the low power ECCS system, than feedwater and steam systems can be used. None of the sequences in this POS contributes 1 E-07 per year or more. The LOCA and fire sequences and cutsets are generally the same as described in the midloop POS, only secondary systems are available for decay heat removal. The top sequence in this POS is a very small LOCA with a sequence frequency of 5.5 E-8 per year for the short cycle type. It contributes 1.9 % of the total core damage frequency.

Summary Description of Midloop POS results

The Midloop is the second largest contributor at ca. 42% of the total core damage frequency from all POSs. Although there is only a short time interval in this POS (2.3 % of the year for both the short and long plant cycle types), this POS is important because of the reduced inventory and the fact that there is no redundancy of the low pressure ECCS system. The accident sequence with the highest frequency in the Borssele plant is a fire sequence during midloop. The fire is a fire in a special room (containing the bus CU (Red. I) and all the cables to the battery chargers (Red. II) of dc power. The fire causes al loss of electrical buses BB, CA, CU, CB, EA, EB and EJ; along with causing a loss of off-site power.

62. What further work is required/ planned to refine the analysis that has been carried out to date?

A part of the second 10-year periodic safety review is a review of the PSA against current state-of-the-art methods and techniques. This includes the LPSD-PSA. Several suggestions for future upgrading were made. To mention are:

- Currently a mission time of 24 hours is applied. Especially, for LOSP events larger mission times are recommended as well as applying specific recovery terms based on type of cutset failures.

- The Borssele PSA often used 1E-7 as a screening criterion for initiating event frequencies. Through design modifications and safety enhancement programs, the Borssele TCDF is now about 3E-6/year. This is an order of magnitude lower than most PWRs. Initiators screened at the 1E-7/yr away sequences now are important due to the overall low TCDF. This is especially, true for external events.

- Regarding Fault Trees used for support system Initiating Events it is recommended to use the Boolean equation as input for the logic model. Currently Fault Trees are used to develop an initiating event frequency, but a basic event is used as input in the logic model. Use of the Boolean equation would preserve all dependencies between the cause of the initiating event and the mitigating systems.

- Regarding the modelling of cognitive Human Errors the HCR/ORE is used in the PSA. In HCR, time is the primary performance shaping factor. It could be worthwhile to re-evaluate the HEPs using EPRI- Cause Based Decision Tree (CBDT) methods for cognitive error probabilities.

- The advances in knowledge about late severe accident progression, such as vessel failure and containment phenomena in the last 15 years are recommended to be included in the level-2 modelling.

Spain

57. How the main results of the LPSD PSA compare to the results of the full power PSA?

The global CDF obtained is somewhat lower in the case of LP&SD PSA than in the full power case, but in the same order of magnitude.

58. How have the results of the LPSD PSA been presented?

Asco NPP has presented the CDF distribution by sequences, by scenarios and by Initiating Events, besides the global CDF. Garonna NPP has only submitted the global CDF but has been requested by CSN to better analyze and present her results.

a) How does this differ from what was done for the full power PSA?

In full power PSAs, the Spanish plants present at least CDF by sequence, by IE and globally.

59. What confidence do you have in the results of the LPSD PSA compared to the full power PSA?

CSN has got confidence in the results of the various LP&SD PSAs as they have been generated starting from the full power models and data, the methodology followed is the same, and the review and QA process are also the same.

60. To what degree does the confidence in your results allow or impair risk trade-off between full power and shutdown states?

Based on the Q59 answer, there is a sufficient confidence in the results. Even though the direct balance or trade-off is not always possible, they are enough accurate to support decision-making.

61. What insights have been obtained from the LPSD PSA?

The main insights were that, some modifications, procedures and improvements should be introduced. Additional systems or trains should be operable at given plant situations regardless the Tech Spec requirements.

62. What further work is required/ planned to refine the analysis that has been carried out to date?

Although it is not a refinement of the analysis, an increase of the analysis scope is foreseen, initially to cover internal fire and flooding assessment and Level 2 later.

Maybe some refinement regarding human induced initiating event could be recommended.

Sweden

57. How the main results of the LPSD PSA compare to the results of the full power PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q57	F1/2	Nearly of the same magnitude (1/3of the full power CDF). There are also some specific risk factors for the LPSD analyse.
	R1	No significant differences found.

	R2-4	R2, They are not comparable due to the differences in the studies
	B2	They are not comparable due to the differences in the studies
	O1	No results of the low power PSA have been derived yet. Answer only valid for low power PSA, not outage period.
	O2	They are approximately in the same region, about 1/10 of the full power results.
	O3	They are approximately in the same region, about 1/10 of the full power results.
SKI REMARK	The studies show figures that are in the same magnitude as for the power operation PSA.	

58. How have the results of the LPSD PSA been presented?

a) How does this differ from what was done for the full power PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q58	F1/2	As CDF's, PDS's and frequencies for different release categories.
	R1	As one report for S-PSA (shutdown PSA) and the LP part will be included in the power operation PSA.
	R2-4	R2, As one report for Shut down-PSA
	B2	In the same way as the results for Level 2 PSA
	O1	See answer to question Q57. Answer only valid for low power PSA, not outage period.
	O2	They have not
	O3	They have not yet been presented.
SKI REMARK	Only F1/2 study has been presented to the authority yet.	
Q58 a	F1/2	No differences
	R1	The same.
	R2-4	R2, No difference
	B2	No difference
	O1	See answer to question Q57. Answer only valid for low power PSA, not outage period
	O2	Not applicable
	O3	Not applicable
SKI REMARK	The utilities and SKI´s aim is to include the LPSD PSA presentations in the same document as for power operation.	

59. What confidence do you have in the results of the LPSD PSA compared to the full power PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q59	F1/2	Weaker than the FP PSA (based on more HRA and no model functional dependencies)
	R1	Similar to full power PSA.

	R2-4	R2, No answer
	B2	There are always parts in PSAs that you can be more confident in than others. It is very hard and sometimes not advisable to compare results, especially not the quantitative results
	O1	See answer to question Q57. Answer only valid for low power PSA, not outage period.
	O2	The same, with some minor exceptions
	O3	The same, with some minor exceptions
SKI REMARK	SD PSA is not yet at the same confidence level as the power operation PSA.	

60. To what degree does the confidence in your results allow or impair risk trade-off between full power and shutdown states?

CSNI Question number	Answer from Swedish plant	Answer
Q60	F1/2	Not suitable for trade-off's, due to incompleteness in the LPSD.
	R1	N. A.
	R2-4	R2, No answer
	B2	No answer
	O1	See answer to question Q57. Answer only valid for low power PSA, not outage period.
	O2	At present state, it would allow trade-off between full power and shutdown + start-up states
	O3	At present it would allow trade-off between full power and shutdown + start-up states
SKI REMARK	N.A.	

61. What insights have been obtained from the LPSD PSA?

CSNI Question number	Answer from Swedish plant	Answer
Q61	F1/2	Large and medium LOCA's below upper core level dominates the results. Closed lower chamber doors are vital, as well as operability of "filtrated ventilation systems" containment flooding are essential for success. Spraying of refuelling floor by independent system would reduce environmental impact.
	R1	No major issues found.
	R2-4	R2, Plant modifications have been made and the EdF shutdown instruction package has been implemented at Ringhals
	B2	All the fuel is taken out from the RPV during an outage.
	O1	See answer to question Q57. Answer only valid for low power PSA, not outage period.
	O2	We have basically not reached this stage
	O3	We have basically not reached this stage.
SKI REMARK	A lot of development work remains to analyse the results in Sweden.	

62. What further work is required/ planned to refine the analysis that has been carried out to date?

CSNI Question number	Answer from Swedish plant	Answer
Q62	F1/2	Identification of: "Man made" initiating events Area events External events (hazards) Level 2 Fault trees (system modelling and dependencies) Developing of methods (qualitative and repeatability).
	R1	No further work required.
	R2-4	R2, Further work will be considered after the current full power PSA update is finished.
	B2	A new LPSD PSA has to be performed in accordance with state of the art.
	O1	No results of the low power PSA have been derived yet. It is therefore not possible to determine what further work is required. Answer only valid for low power PSA, not outage period.
	O2	Conclusion, evaluation and some modification
	O3	Conclusion, evaluation and some modification
SKI REMARK	The joint Nordic project will set an acceptable level of what is necessary to perform for a covering study.	

United Kingdom

57. How the main results of the LPSD PSA compare to the results of the full power PSA?

The shutdown core damage frequency (including hazards) is approximately 60% of the total power and shutdown figure. (Cautionary note; based on including average maintenance in analysis. If zero maintenance used then split would be different)

58. How have the results of the LPSD PSA been presented?

a) How does this differ from what was done for the full power PSA?

The LPSD has always been a part of the Sizewell B LPSA and has always been presented in the same manner as the full power PSA.

59. What confidence do you have in the results of the LPSD PSA compared to the full power PSA?

Both are treated as ‘best estimate’ with effectively the same confidence

60. To what degree does the confidence in your results allow or impair risk trade-off between full power and shutdown states?

Since both are covered in the same manner within the same overall model, this allows risk trade-offs to be made.

61. What insights have been obtained from the LPSD PSA?

Most are associated with operation of the plant and the alignment and availability of safeguards equipment in the various shutdown states

62. What further work is required/ planned to refine the analysis that has been carried out to date?

No further work is currently planned as the model has recently been updated to align the model with improvements to operating practices at the station.

United States

BWR

57. How the main results of the LPSD PSA compare to the results of the full power PSA?

The full power analysis did not perform internal fire, internal flood, or seismic analyses. The following POS 5 information is provided for completeness:

- Internal fire events <1E-8
- Internal flood events 2.3 E-8
- Seismic events
  - i. for the LLNL (1993) hazard curves 7.1E-8
  - ii. for the EPRI hazard curves 2.5E-9

The table (see Q59) provides a comparison of selected results from the detailed POS 5 analysis versus the full power analysis for internal events.

58. How have the results of the LPSD PSA been presented?

POS 5 results were presented based on a per calendar year basis. It was chosen as a mechanism for directly comparing risk between other modes of operation, allowing one to quickly and easily identify the mode of operation that produces the most risk.

a) How does this differ from what was done for the full power PSA?

Full power results were based on a per reactor year basis rather than on a per calendar year basis.

Note that placing results from other operational states on a per-operational year basis would be difficult given the substantial changes that would/could happen during operation in a particular state for an entire year. For example, if one wanted to estimate the risk from being in POS 5 for an entire year, one must account for the potentially substantial changes in success criteria as the decay heat load is reduced during the year, thereby increasing the complexity of the model needed to represent the plant as it operates in POS 5 for an entire year.

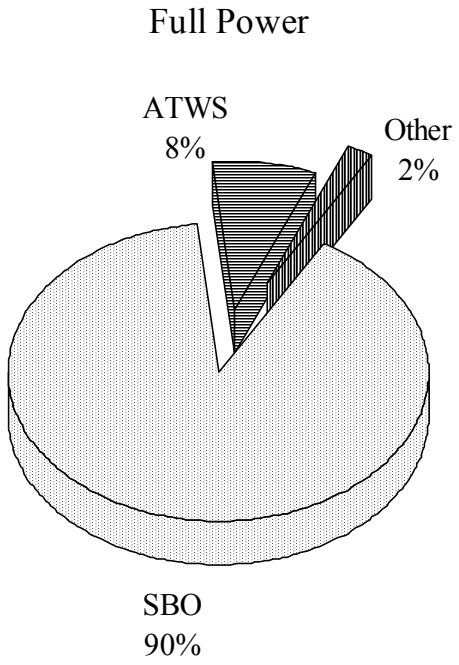
59. What confidence do you have in the results of the LPSD PSA compared to the full power PSA?

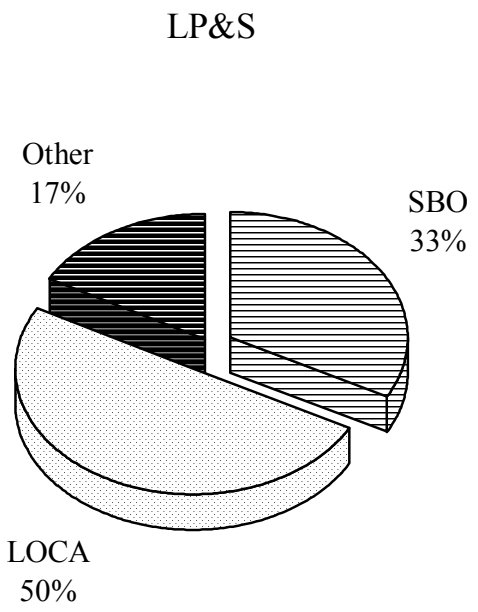
Given that this was the first NRC sponsored analysis of LPSD conditions (and specifically the detailed POS 5 refuelling analysis), the degree of confidence in the completeness of the numerical results is somewhat less than it would be for results obtained from performing a full power PRA because there are known issues (e.g., maintenance induced LOCAs) that were not considered in the scope of the POS 5 refuelling analysis. However, sensitivity studies have been performed to obtain an understanding of how certain model changes would affect the results. The results from these sensitivity analyses indicate that the overall conclusion from the POS 5 analysis (i.e., risk from LPSD conditions is comparable to full power risk) is robust.

Distributions for aggregate risk for POS 5 (internal events) and for full power (all values are per year; population doses are in person-rem)

Analysis	Descriptive Statistics			
	Percentiles			
	5th	50th	95th	Mean
	Core Damage Frequency			
POS 5	4.1E-07	1.4E-06	5.6E-06	2.1E-06
Full Power	1.8E-07	1.1E-06	1.4E-05	4.1E-06
	Early Fatality Risk			
POS 5	3.7E-11	2.8E-09	3.9E-08	1.4E-08
Full Power	2.5E-12	6.1E-10	2.6E-08	8.2E-09
	Total Latent Cancer Fatality Risk			
POS 5	4.3E-04	1.9E-03	1.2E-02	3.8E-03
Full Power	1.4E-05	2.4E-04	2.3E-03	9.5E-04
	Population Dose (within 50 miles of the site) Risk			
POS 5	1.3E-01	5.3E-01	3.1E+00	9.9E-01
Full Power	1.2E-02	1.3E-01	1.4E+00	5.2E-01
	Population Dose (within 1000 miles of the site) Risk			
POS 5	9.9E-01	4.4E+00	2.8E+01	8.7E+00
Full Power	9.0E-02	1.4E+00	1.5E+01	5.8E+00
	Individual Early Fatality Risk: 0 to 1 mile			
POS 5	4.2E-13	2.7E-11	3.0E-10	9.6E-11
Full Power	2.3E-14	5.2E-12	1.0E-10	3.3E-11
	Individual Latent Cancer Risk: 0 to 10 miles			
POS 5	2.5E-10	9.4E-10	4.9E-09	1.6E-09
Full Power	1.3E-11	9.4E-11	9.7E-10	3.4E-10

The following figures identify the dominant accident sequences (from the NUREG/CR-4550 analysis) and provide the percent contribution to the total CDF from the power and LPSD (POS 5 internal events only) analyses.


Full Power
ATWS
8%
Other
2%
SBO
90%



60. To what degree does the confidence in your results allow or impair risk trade-off between full power and shutdown states?

At the time of the analysis, early to mid 1990's, it would have been possible to perform risk trade-off between full power and shutdown states with the results of the LPSD analysis. However, before the analysis could be used to support current risk trade-offs, it would have to be updated to reflect current plant specific operational practices and conditions.

61. What insights have been obtained from the LPSD PSA?

An examination of the accident sequence analysis results from the POS 5 Phase 2 documentation reveals that three classes of accident sequences [i.e., LOCA, station blackout (SBO), and *Other*] dominate the core damage frequency results. These three classes can be generally described as follows:

LOCA - Initiated by a pipe break (in a recirculation loop) of sufficient size to cause either a large or an intermediate loss-of-coolant accident in time windows 1, 2, and 3. The time windows were defined as:

- Time window 1: starts 14 hours after shutdown and has a duration of 10 hours
- Time window 2: starts 24 hours after shutdown and has a duration of 70 hours
- Time window 3: starts 40 days after shutdown and has a duration of 10.4 days

SBO - Initiated by a loss-of-offsite power (LOSP) event in time windows 1 and 2. Onsite ac power for 1E divisions fails.

Three subclasses of SBO accident sequences were identified. They are described as follows:

1) Insufficient time for the operators to align and use the diesel-driven firewater pumps.

2) Sufficient time for operators to use the pumps, but they fail.

3) Operators successfully align and begin use of the pumps, but the SRVs close, and injection is lost as the reactor vessel pressure increases.

Within the third subclass, a distinction was made between those sequences where the pumps ran for sufficient time to allow closure of isolation valves to prevent an interfacing system LOCA.

Other - Initiated by five different events in time windows 1 and 2. The ultimate failures that resulted in core damage for the different initiating events were operator failure to: (1) close the airlock to prevent flooding of the auxiliary building, (2) close the main steam isolation valves to prevent flooding down the steam lines, or (3) steam the vessel at high pressure (i.e., recognize the need to steam the vessel at high pressure and either take steps to inject water at high pressure or to allow the vessel to pressurize and thus gain some additional time to respond to the event).

From the descriptions given above and from an examination of the sequence logic and the accident sequence cut sets in each of the classes, it can be shown that, at least for the LOCA and SBO classes, the sequences that make up these two classes are ones in which some event causes or contributes to the loss of multiple systems. In other words, the sequences that survived are not sequences resulting from totally unrelated or independent failures of multiple systems.

The following table provides the statistics for both the CDF and the fractional contribution (FC) to the CDF for each of the classes.[1] It can be seen that the LOCA class is the most important class from a mean CDF point of view. In addition, the LOCA class is also the most important class from a mean FC core damage frequency point of view (i.e., 0.51 for LOCA versus 0.33 and 0.17 for the SBO and *Other* classes, respectively).

[1] All statistics presented for LP&S calculations are based on an LHS sample size of 200. While it is recognized that these statistics would change if the sample size were increased, these changes should be small given the stratified sampling technique used by LHS. In addition, none of the observations or insights presented in this report are expected to be sensitive to changes resulting from an increased sample size.

Original POS 5 core damage frequency and fractional contribution to core damage frequency (internal events only)

Initiating Event Class	Descriptive Statistics[a]				
	Percentiles			Mean	Standard Deviation
	5th	50th	95th		
	Core Damage Frequency				
LOCA	1.1e-07	5.7e-07	4.0e-06	1.1e-06	1.7e-06
SBO	4.2e-08	2.8e-07	2.3e-06	7.4e-07	2.1e-06
OTHER	2.0e-08	1.2e-07	8.0e-07	2.4e-07	3.2e-07
TOTAL	4.1e-07	1.4e-06	5.6e-06	2.1e-06	2.7e-06
	Fractional Contribution to Core Damage Frequency				
LOCA	0.10	0.50	0.93	0.51	0.27
SBO	0.03	0.24	0.80	0.33	0.26
OTHER	0.01	0.09	0.58	0.17	0.18

[a]Based on an LHS sample size of 200.

The following two tables contain the statistics for early fatality and total latent cancer risk distributions, and the FC of each of the initiating event classes. In both tables, it can be seen that the SBO class is the most important from a mean risk point of view. In addition, the SBO is also the most important class from a mean FC to early fatality risk. For total latent cancer risk, both the LOCA and SBO classes are important contributors.

Insights from LOCAs

CDF Insights

The LOCAs that were analyzed in the LPSD project can be grouped into two categories. These are LOCAs during nonhydro conditions (i.e., at atmospheric pressure) and during hydro conditions (i.e., at approximately 1000 psi). Human errors are important for the LOCA accident sequences:

From a CDF viewpoint, concerns about the value used for the initiating event frequency of the nonhydro LOCAs in POS 5 would be immaterial if the automatic actuation of the SPMU system were functional. Automatic actuation of the SPMU would most likely eliminate all LOCAs. Elimination of the low-pressure LOCAs reduces the fractional contribution of the LOCAs to the total CDF by about a factor of 2. More important, elimination of the low-pressure LOCAs reduces the early fatality risk attributed to LOCAs by about a factor of 80. This is expected because low-pressure LOCAs occur in time windows 1 and 2, and, generally speaking, these affect early fatality risk more than time window 3 sequences.

Changes in procedures that would allow more credit to be given to the operators during the human reliability analysis for controlling injection systems, specifically the high pressure core spray (HPCS) system, would provide some reduction in the importance of the LOCA class from a CDF and total latent cancer risk point of view; however, since this change affected only accidents in time window 3 (sequences in time windows 1 and 2 were unaffected because the HPCS system is unavailable in the cut sets that survived the phase 2 analysis), no significant change would be expected in the early fatality risk measure.

Risk Insights

When considered as a group, LOCA accidents are not on average the most important contributor to early fatality risk. They are, however, an important contributor to the total latent cancer risk.

The LOCA group is not on average the most important contributor to early fatality risk because the most probable LOCA accidents occur while the plant is in time window 3 (approximately 40 days after shutdown), by which time radioactive decay has significantly reduced the inventory of short-lived radionuclides that are important in early health effects.

The LOCA group is an important contributor to total latent cancer fatality risk because it is an important contributor to the core damage frequency and because accidents from this group release a radioactive material into the environment, primarily long-lived radionuclides that are important in latent health effects.

Original POS 5 early fatality risk and fractional contribution to early fatality risk(internal event only)

Initiating Event Class	Descriptive Statistics[a]				
	Percentiles			Mean	Standard Deviation
	5th	50th	95th		
	Early Fatality Risk				
LOCA	6.4e-13	1.3e-10	3.0e-09	8.8e-10	3.6e-09
SBO	2.6e-12	1.9e-09	3.6e-08	1.2e-08	5.3e-08
OTHER	5.7e-13	1.2e-10	1.8e-09	4.2e-10	1.1e-09
TOTAL	3.7e-11	2.8e-09	3.9e-08	1.4e-08	5.4e-08
	Fractional Contribution to Early Fatality Risk				
LOCA	0.001	0.04	0.72	0.16	0.24
SBO	0.08	0.87	1.00	0.73	0.30
OTHER	0.001	0.04	0.61	0.12	0.18

[a]Based on an LHS sample size of 200.

Original POS 5 total latent cancer risk and fractional contribution to total latent cancer risk internal event only)

Initiating Event Class	Descriptive Statistics[a]				
	Percentiles			Mean	Standard Deviation
	5th	50th	95th		
	Total Latent Cancer Risk				
LOCA	6.6e-05	6.7e-04	7.0e-03	1.5e-03	2.5e-03
SBO	7.8e-05	6.0e-04	7.0e-03	2.1e-03	7.1e-03
OTHER	1.6e-05	1.3e-04	8.9e-04	2.7e-04	4.1e-04
TOTAL	4.3e-04	1.9e-03	1.2e-02	3.8e-03	7.6e-03
	Fractional Contribution to Total Latent Cancer Risk				
LOCA	0.04	0.38	0.90	0.42	0.27
SBO	0.04	0.41	0.90	0.45	0.28
OTHER	0.01	0.06	0.55	0.13	0.17

[a]Based on an LHS sample size of 200.

Insights from Station Blackouts

CDF Insights

The SBO sequences are driven by events that result in the loss of multiple systems. For example, the loss of onsite ac power prevents the use of all systems except the diesel-driven firewater pumps. The use of firewater was unsuccessful in these accidents, which were grouped into the following three classes:

1) Insufficient time for the operators to align and use the pumps.

2) Sufficient time for operators to use the pumps, but they fail.

3) Operators successfully align and begin use of the pumps, but the SRVs close, and injection is lost as the reactor vessel pressure increases.

Results from the sensitivity calculation where it was assumed that the SRVs could be either opened or kept open show that the ability to open or keep open the SRVs is relatively important—the mean fractional contribution of the SBO sequences changes from 0.33 to 0.10, indicating that approximately 23% of the total CDF comes from SBO sequences involving dependence on the SRVs. In addition, the sensitivity calculation gives an indication of the importance of the operator action associated with use of the diesel-driven firewater pumps. Failure to successfully align and use the firewater system contributes about 10% to the total mean CDF.

Risk Insights

Station blackout accidents, when considered as a group, are an important contributor to both early fatality risk and to total latent cancer fatality risk.

Insights from *Other*

CDF Insights

Accident sequences in the *Other* class were grouped into three classes:

1) Flooded containment,

2) Open main steam isolation valves (MSIV s), and

3) Loss of all standby service water (SSW).

All of the sequences in the first class involved an empty suppression pool—based on an assumption made in the LPSD analysis. If the sequences in this class are eliminated from the *Other* class, then the mean CDF for the *Other* class decreases by about a factor of 10. However, the total POS 5 mean CDF decreases by only a factor of 1.2 since the mean contribution of this class to the total CDF is only 17%. This implies that while water in the suppression pool is important to the *Other* class, it is not important to a change in the total mean CDF for POS 5.

Risk Insights

When considered as a group, the *Other* accidents are not on average the most important contributor to either early fatality risk or to the total latent cancer fatality risk. This stems primarily from the fact that the *Other* group is not on average the most important contributor to the core damage frequency and the consequences from these accidents are not large enough to compensate for the relatively low core damage frequency. This is not to say that the releases are negligible. While many different types of progressions can be found in this group, some common characteristics of these accidents include:

- Core cooling is never restored and, hence, all of the accidents progress to full core damage and vessel failure.

- The containment is either open during the entire accident or if it is closed before the core is damaged, it is either vented or fails during the accident. The principal reason the containment becomes pressurized is that containment heat is not removed; hydrogen combustion and loads that accompany failure of the reactor vessel at high pressure also contribute to containment failure.

General Insights

CDF Insights

1) The sequences generally contain some event or events that result in the failure of multiple systems

2) Requiring the availability of SRVs during POS 5 allows for additional cooling options and thus requires that something cuts across multiple system boundaries to result in core damage. This is in part what contributes to the relatively small CDF estimate obtained by the LPSD project.

3) The presence of a motor-driven high-pressure system also contributes to the relatively small CDF estimate. This is true even after taking into account this system's sometimes significant unavailability, primarily because the motor-driven pump, when available, can be used in many situations where a turbine-driven pump could not be used.

4) Automatic isolation of the low-pressure piping on high pressure during POS 5 effectively eliminates the problem of an interfacing system LOCA.

5) Operator actions play a significant role both in the progression of the accident sequences and in the estimate of the CDF associated with the sequences. Generally speaking, operator actions in POS 5 tend to be more involved and/or complex from a diagnosis viewpoint.

6) Only sequences that resulted in a loss of vessel inventory survived in time window 3. The major reason is that the decay heat load in time window 3 is small enough that only with a loss of water from a break or diversion will core damage occur within the 24-hour mission time used in the LPSD project.

Risk Insights

See above.

62. What further work is required/ planned to refine the analysis that has been carried out to date?

No additional work is planned at this time.

PWR

57. How do the main results of the LPSD PSA compare to the results of the full power PSA?

1) NUREG/CR-6144- Mid-loop PSA

Comparison of core damage frequency

Table 1 lists the key uncertainty characteristics of the core damage frequencies for mid-loop operation and power operation, and shows that the core damage for mid-loop operation has a wider spread than that of power operation. The internal event CDF of mid-loop is an order of magnitude lower than that of full power. Taking into consideration the fraction of time that that plant is at mid-loop, the conditional CDF (CCDF) is comparable to that of full power. The same is true for internal floods. The CDF and CCDF of seismic events are lower than those of full power. The internal fire CDF is comparable to that of full power. Taking into consideration the fraction of time a plant is at mid-loop, the CCDF of mid-loop is higher than that of full power. The main reason is that the routing of the cables of the equipment needed to support RHR operation or mitigate an accident during mid-loop operation is such that a single fire at a few critical locations can damage almost all the equipment needed to mitigate the accident, while during power operation much few critical locations exist.

Comparison of risks

Table 2 presents statistical measures of the distributions for seven consequence measures for accidents during mid-loop operation obtained from this study. Similar statistical measures for full power operation obtained from the NUREG-1150 study of Surry are also included in the table. It indicates that the mean risk of offsite early health effects is over two orders of magnitude lower for accidents during mid-loop operation than for full power in spite of the lack of mitigative features. This is due to the natural decay of the radionuclide inventory (because the accidents occur a long time after shutdown) particularly the short-lived isotopes of iodine and tellurium, which are primarily associated with early health effects. The statistical measures for latent cancer fatalities differ by a factor of approximately three, although the statistical measures for population dose (1000 miles) for mid-loop and full power operations are similar. This difference is largely explained by differences in the latent cancer versus dose relationship in the different versions of MACCS used in the two studies.

2) Screening analysis of NUREG/CR-6144

A comparison of the conditional CDFs of the POSs in a refuelling outage with that of the full power PSA was made. Due to the screening nature of the shutdown PSA and higher maintenance unavailabilities during shutdown, the conditional CDFs of all POSs are greater than that of the full power PSA.

3) Cold shutdown PSA

<u>NUREG/CR-6616</u>

In this study, the conditional risks during cold shutdown were calculated. Table 3 is a comparison of some of the risk measures between cold shutdown and full power, for the base case in which no maintenance is assumed. It can be seen that the conditional CDF and latent cancer fatalities at cold shutdown are of the order of magnitude as those of full power, while the early fatality risk at cold shutdown is lower than that of full power operation.

58. How have the results of the LPSD PSA been presented?

a) How does this differ from what was done for the full power PSA?

The results of LPSD PSAs were presented in the same way full power PSAs are presented. As a part of the presentation, the outages types, POSs, and time windows have to be specified.

59. . What confidence do you have in the results of the LPSD PSA compared to the full power PSA?

1) NUREG/CR-6144- The confidence in the mid-loop PSA is lower, because there are larger uncertainties in the strategies for mitigating accidents. The assumed plant conditions may no longer represent the current plant practice. The assumptions of the supporting thermal hydraulic analysis may not all be valid. The procedures for mitigating shutdown accidents are limited and do not cover all aspects of the accidents modelled.

2) Screening analysis of NUREG/CR-6144- Due to the screening nature of the study, many conservative and some inappropriate assumptions may have been made.

Table 1 Results of the Level 1 Uncertainty Analysis and Comparison with Full Power Operation (per year)

	Study		Mean	5th Percentile	50th Percentile	95th Percentile	Error Factor
Internal Events	Full Power Operation - NUREG 1150 (per year)		4.0E-05	6.8E-06	2.3E-05	1.3E-04	4.4
	Full Power Operation- IPE		7.4E-05*	—	—	—	—
	Mid-Loop Operation (per year while at mid-loop)		4.9E-06	4.8E-07	2.1E-06	1.5E-05	5.7
Internal Fires	Full Power Operation - NUREG 1150 (per year)		1.1E-05	—	—	—	—
	Full Power Operation- IPE		**	—	—	—	—
	Mid-Loop Operation (per year while at mid-loop)		2.2E-05	1.4E-06	9.1E-06	7.6E-05	7.2
Internal Flood	Full Power Operation - NUREG 1150 (per year)		***	—	—	—	—
	Full Power Operation- IPE		5.0E05**	—	—	—	—
	Mid-Loop Operation (per year while at mid-loop)		4.8E-06	2.2E-07	1.7E-06	1.8E-05	9.0
Seismic Events	Full Power Operation - NUREG 1150 (per year)	LLNL	1.2E-04	—	—	—	33
		EPRI	4.0E-05	—	—	—	4.4
	Full Power Operation- IPE		**	—	—	—	—
	Mid-Loop Operation (per year while at mid loop) mid-loop)****	LLNL	3.5E-07	1.3E-09	4.0E-08	1.4E-06	32
		EPRI	8.6E-08	2.5E-10	9.7E-09	3.7E-07	37

* point estimate
** not available
*** below truncation of 1.0E-08 per year
**** refuelling outage only (no drained maintenance)

Table 2 Comparison of Distributions of Risks for Mid-Loop and Full-Power Operation (All Values per Reactor Year; Population Doses in P-Sv per Year)

	5th Percentile		Median		Mean		95th Percentile		Sigma	
	Mid-Loop	Full-Power	Mid-Loop	Full-Power	Mid-Loop	Full-Power	Mid-Loop	Full-Power	Mid-Loop	Full-Power
Early Fatalities	1.3E-10	7.6E-10	3.6E-09	7.0E-08	4.9E-08	2.0E-06	1.6E-07	5.4E-06	1.7E-07	N.A.
Latent Fatalities within 50 mi	1.6E-04	N.A.	8.3E-04	N.A.	2.5E-03	N.A.	8.8E-03	N.A.	3.7E-03	N.A.
Latent Fatalities within 1000 mi	8.0E-04	3.1E-04	5.3E-03	2.2E-03	1.6E-02	5.2E-03	5.5E-02	1.9E-02	2.5E-02	N.A.
Population Dose within 50 mi	3.8E-03	5.9E-03	2.0E-02	2.7E-02	5.8E-02	5.8E-02	1.9E-01	2.5E-01	8.8E-02	N.A.
Population Dose within 1000 mi	1.9E-02	1.9E-02	1.2E-01	1.3E-01	3.7E-01	3.1E-01	1.3E+00	1.2E+00	5.9E-01	N.A.
Individual Early Fatalities Risk within 1 mi*	6.0E-12	1.4E-11	1.3E-10	8.7E-10	1.7E-09	1.6E-08	6.9E-09	4.9E-08	5.5E-09	N.A.
Individual Latent Fatalities Risk within 10 mi*	1.2E-10	1.6E-10	7.5E-10	4.9E-10	2.1E-09	1.7E-09	7.1E-09	8.1E-09	3.0E-09	N.A.

N.A. – Not Available

*NRC quantitative health objectives:

- • Individual early fatality risk within one mile to be less than 5×10^{-7} per reactor year.
- • Individual latent cancer fatality risk within 10 miles to be less than 2×10^{-6} per reactor year.

Table 3 Risk Comparison of Cold Shutdown vs Full Power

Operation Mode	5th	Median	Mean	95%
Core Damage Frequency (/RY)				
Power	9E-6	3E-5	4E-5	1E-4
Window 1	4E-6	1E-5	2E-5	6E-5
Window 2	1E-6	4E-5	4E-5	8E-5
Window 3	1E-5	4E-5	4E-5	8E-5
Window 4	3E-6	8E-6	1E-5	4E-5
Total Early Fatalities (/RY)				
Power	9E-9	2E-7	7E-7	4E-6
Window 1	2E-8	7E-8	1E-7	5E-7
Window 2	8E-9	3E-8	3E-8	7E-8
Window 3	4E-10	9E-10	1E-9	2E-9
Window 4	6E-12	2E-12	4E-11	1E-10
Total Latent Cancer Fatalities (/RY)				
Power	4E-4	3E-3	1E-2	3E-2
Window 1	5E-3	2E-2	4E-2	1E-1
Window 2	1E-2	3E-2	4E-2	1E-1
Window 3	1E-2	2E-2	3E-2	5E-2
Window 4	2E-3	5E-3	9E-3	3E-2

NUREG/CR-6616 and 5718- The confidence is less than that of 1), because the study makes use of what was done in 1) without gathering more plant information and made more simplifying assumptions.

60. To what degree does the confidence in your results allow or impair risk trade-off between full power and shutdown states?

1) NUREG/CR-6144- At the time the PSA was completed, degree of confidence should not have prevented any risk trade-off between full power and shutdown states. Due to the fact that the plant practice may have changed, an update of the PSA is necessary. For example, the most significant change in plant practice started in the refuelling outage of unit one in 1992, during which mid-loop operation was totally avoided. Another way of reducing the risk is to carry out reduced inventory operation while the fuel in the core is removed during refuelling operation.

2) Screening analysis of NUREG/CR-6144- The screening analysis should not be used in any trade-off between full power and shutdown states.

3) NUREG/CR-6616 and 5718- Similar to 1), the study needs to be updated. In addition, the simplifying assumptions need to be evaluated. The cold over-pressurization scenarios should be further studied.

61. What insights have been obtained from the LPSD PSA?

1) Insights of mid-loop operation- NUREG/CR-6144

<u>Level–1 Internal events</u>

Operator Response- The dominant cause of core-damage was the operator's failure to mitigate the accident. (There is very large uncertainty in the human error probabilities used in this study.) In general, it would be

beneficial to have good training, procedures, and instrumentation to ensure that the utility's staff can respond to shutdown accidents.

Procedures for Shutdown Accidents - Very few procedures are available for accidents during shutdown; the procedure for loss of decay heat removal, AP 27.00, is the only one that was written specifically for the shutdown scenarios analyzed in this study. The procedure is conservative with regard to the equipment needed to establish reflux cooling and feed-and-bleed. In this study, the use of fewer than the number of steam generators specified in the procedure for reflux cooling was treated as a recovery action, and a more realistic success criteria was used for feed-and-bleed when the decay heat is high. In most cases, the information in the procedures for power operation is helpful, for shutdown accidents. Credit for this procedure was taken into account in this study. However, some procedures written for power operation would mislead the operator if followed during shutdown.

Instrumentation - The level instrumentation used during mid-loop operation, i.e., standpipe level instrumentation and ultra-sonic level instrumentation, has limited applicability during a shutdown accident. The standpipe system indicates the correct level only when there is no build-up of pressure in the system. The ultra-sonic level instrumentation only provides level indication when the level is within the reactor coolant loops, and therefore, may not be useful during a feed and bleed operation.

Supporting Thermal Hydraulic Analysis - The thermal hydraulic behaviour of the reactor coolant system is rather complex, mainly because the pressuriser is usually the relief path for coolant or steam, and the vessel head does not have a large vent. When performing thermal hydraulic analysis in support of the PRA effort, consideration must be given to longer term system behaviour, at least 24 hours into the accident. In this study, such calculations were done for feed-and-bleed operation using a charging pump, and with gravity feed from the RWST. Additional calculations would be helpful to better understanding the effectiveness of reflux cooling, and feed and bleed using a low pressure injection pump.

Maintenance Unavailability - A review of shift supervisor's log books and minimum equipment lists for three refuelling outages showed that the maintenance unavailabilities of equipment that can be used to mitigate an accident were very high. Generic letter 88-17, requires the plant to have one high-head pump and one low-head pump available. In our quantifications, it was assumed that charging pump A, charging pump cooling water pump A, and low head injection pump B are available. From the check list used for reduced inventory conditions, the study also assumed that the maintenance of diesel generators, 4 kv emergency buses, and stub buses is not allowed.

Maintenance unavailability is the dominant cause of equipment unavailability. In combination with human errors, maintenance of the charging pump cooling water pump, the charging pump, and the low head injection pump appear in the dominant cutsets for some of the core-damage sequences.

Isolation of Reactor Coolant Loops - Isolation of the RCS loops is an important contributor to core-damage frequency. Review of the plant shutdown experience indicated that the reactor coolant loops are isolated for extended periods in a refuelling outage, making the steam generators unavailable for decay-heat removal upon loss of RHR. In a cold shutdown condition, the steam generators are usually maintained in the wet lay-up condition with the secondary side filled with water. During mid-loop operation, the availability of the SGs makes reflux cooling a possible method of mitigating a loss of RHR; this might be the only mitigation function available in a station blackout.

Single Failures of the RHR System - The RHR system at Surry has no active safety function (i.e., it does not perform the safety injection function in scenarios initiated at full power). Consequently, many single component failures can cause loss of RHR.

Valve Arrangement of Auxiliary Feedwater System and Main Steam System During Shutdown - The auxiliary feedwater system has several MOVs in the flow path to the steam generators, that are normally closed during shutdown. They are difficult to locate during a station blackout. Similarly, the main steam non-return valves are normally closed during shutdown, and have to be opened to use steam dump to the condenser.

Potential for Plugging the Containment Sump When Recirculation Is Needed - Because of activities inside the containment, transient material and equipment are brought into it during shutdown. For example, large plastic sheets have been used to separate work areas from the rest of the containment. When an accident requiring recirculation from the containment sump occurs, as is the case in time windows 1 and 2, the material would increase the potential for plugging the containment sump.

Level–1 Internal Fires

No prevalence of fires at shutdown in the data base was noticed, as compared to power operation fires (after the construction events are taken out). It is true that greater potential exists for fires in certain categories (e.g., transient or welding igniting cables or other equipment fires). It is also true that possibility of some types of fires is reduced (e.g. de-energized equipment, oil dripping on hot piping). A fire at shutdown is liable to be detected much sooner and extinguished in its early phases, because of increased floor traffic. (Credit is taken for this by disallowing events that were discovered in the smoking stage (without flames) or early enough such that deenergizing equipment extinguished the fire.) Increased vigilance by licensees may play a part in this also. At Surry, a fire watch is in place at welding operations; fire doors are kept closed.

Human error events are not prominent contributors individually in terms of Fussell-Vesely importance range (a few percent). Part of the reason for lack of prominence of individual HEPs is that there are many HEPS, each applicable in a small fraction of sequences; another reason is in the values assigned to the HEPs; the third reason is that in many important scenarios hardware failures dominate due to heavy damage by fire.

Although the plant spends much less time in mid-loop, the core damage frequency is comparable to that of power operation. The main reason is that the routing of the cables of the equipment needed to support RHR operation or mitigate an accident during mid-loop operation is such that a single fire at a few critical locations can damage almost all the equipment need to mitigate the accident, while during power operation much fewer critical locations exist.

Level–1 Internal Floods

The plant -specific spatial arrangement of piping and equipment is an important consideration for the development of the flooding accident scenario and its risk significance

Level–1 seismic events

Core-damage frequency - The core-damage frequency for earthquake-initiated accidents during refuelling outages in POS 6 and POS 10 is found to be low in absolute terms, below 10^{-6}/year. The reasons for this are (i) Surry's seismic capacity in responding to earthquakes during shutdown is excellent, well above its design basis and similar to its ability to respond to earthquakes during full-power conditions; (ii) the Surry site enjoys one of the least seismically active locations in the United States; (iii) the Surry plant is only in POS 6 and POS 10 (combined) for an average (mean) of 6.6% of the time. The core-damage frequencies are also low relative to the frequencies during POS 6 and POS 10 for internal initiators.

The results are plant-specific - The results for Surry are highly plant-specific, in the sense that the seismic capacities, the specific sequences that are found to be most important, and the seismicity of the site are all difficult to generalize to other reactors elsewhere.

Shutdown seismic sequences are similar to full-power seismic sequences - Nevertheless, it is important to observe that all of the sequence types, components, and human errors that emerge in the key sequences in this analysis are similar or identical to sequences, components, and human errors that appear in typical full-power seismic PRAs. That is, nothing that has arisen as important in this study appears to be unique to earthquakes occurring during shutdown conditions. Whether this observation is able to be generalized to other reactors at other sites is unknown.

Sensitivities - Sensitivity studies reveal that if the Surry reactor were moved to the Zion site in Illinois (a typical mid-western site) or the Pilgrim site in Massachusetts (one of the most seismically active sites among all of the reactor sites in the eastern U.S.), the mean annual CDF from this study would increase by factors of about 1.8 and 10, respectively.

Uncertainties - While there are significant uncertainties in the numerical values of core-damage frequencies found in this study, the above conclusions are relatively robust --- they do not depend on the detailed numerical values found.

Level 2/3 insights

The main finding of the study is that during mid-loop operation the risk of consequence measures related to long-term health effects, latent cancer fatalities and population dose, are high, comparable to those at full power, despite the much lower level of the decay heat and the radionuclide inventory. The reason for this is that containment may be unisolated during mid-loop operation so the releases to the environment are potentially large and the radionuclide species which mostly contribute to long-term health effects (such as cesium) have long half-lives. Accident sequences involving failure to correctly diagnose the situation or take proper actions are the largest contributors to the integrated risk. Another finding of the study is that the risk of early fatalities is low despite the unisolated containment due to the decay of the short-lived radionuclide species such as iodine and tellurium which contribute to early fatality risk. The integrated risk estimates have a range of uncertainty extending over approximately two orders of magnitude from the 5th to the 95th percentile of the distribution.

Containment Status - The containment may be either unisolated or may not have full pressure retaining capability during mid-loop operation. This is particularly the case if the operators fail to diagnose the accident as it was judged unlikely that they would take action to isolate containment or could succeed in doing so within the available time frame. This factor played a significant role in influencing the risk estimates of mid-loop operation. During the course of the study, new procedures were made available for containment closure during mid-loop operation. However, it was difficult to assess the adequacy of these procedures in ensuring the pressure retaining capability of the containment within the time frame encompassed by this study. This feature contributed significantly to the uncertainty in containment status and the estimate of risk.

Availability of Containment Sprays - The containment sprays were not required to be available during shutdown. Plant records show that the spray systems could be inoperable because of maintenance. Spray availability was modelled as an uncertainty parameter in the integrated risk analysis. Since the sprays perform an important safety function in mitigating the effects of releases, spray unavailability contributed both to the risk and its uncertainty.

Possibility of Core Damage Arrest - The inclusion of the possibility of arresting the core degradation process before vessel failure is an important feature of this analysis as it was for the full power study. Termination of the accident in-vessel can significantly reduce some of the fission product releases and thus the risk. The potential for core recovery depends on the nature of the accident progression and is different for the various PDS Groups. Overall, the conditional probability of core damage arrest ranged from 0.23 (5th percentile) to 0.44 (95th percentile) with a mean of 0.35.

Comparison with Full Power Study - The mean core damage frequency for accidents during mid-loop operation is about an order of magnitude lower than the mean frequency of accidents caused by internal events at full power. However, the risk distributions obtained for comparable long term health consequences are very similar in the two studies. What this finding implies is that the lower decay heat and lower radionuclide inventory of the mid-loop operating state, compared with full power, is offset by the likelihood of containment being unisolated. Finally, the mean risk of early health effects is over two orders of magnitude lower for accidents during mid-loop operation than for accidents during full power operation. This is due to the natural decay of those radionuclide species which have the greatest impact on early fatality risk because accidents during mid-loop operation occur a long time after shutdown.

2) Insights of screening analysis of NUREG/CR-6144

The screening analysis identified a few potentially vulnerable configurations. They include mid-loop operation, use of temporary seals at the seal table, isolation of the steam generators, maintenance of redundant equipment, and the reactivity accident identified in the French PSA.

For fire risk, consideration of repair or replacement of cables (e.g., RHR or CCWW pump cables) versus cable separation, at shutdown, is important since repair or replacement may not be likely. Also, in case of a fire that causes the loss of the SG PORVs while at power, the alternative shutdown method employs the SG code safety valves. During shutdown, the set point of the safety valves is too high to be useful.

3) Insights of NUREG/CR-6616

NUREG/CR-6616 found that 1) level-1 analysis alone may not be adequate to assess the risks of preventative maintenance, 2) the time window at which preventive maintenance would yield the least risk depends on the selection of risk measures, 3) there is a slight variation of system ranking determined by the impact on CDF and on health risk, and 4) conclusions from point-estimate analysis remain applicable when the results from uncertainty analysis are used.

APPENDIX M - USE OF THE LPSD PSA

PRELUDE AND QUESTIONS

In the past PSAs were performed with the objective of evaluating the risk associated with a nuclear power plant (NPP) and identifying improvements to reduce plant risk. In several countries, PSA results are used in identifying tradeoffs and making decisions for reducing regulatory burden to NPP licensees. Furthermore, Risk Monitors have (or are) been developed and are used to better evaluate and manage the risk associated with the different configurations a plant undergoes during full, low, or shutdown operations. However, the rigor and the quality of the PSA needs to be sufficient in order use the PSA for these types of applications.

63. What were the objectives of carrying out the LPSD PSA?

a) Have all these objectives been met?

64. How has the LPSD PSA been used?

a) What role has the LPSD PSA played in design reviews, decision making, addressing regulatory requirements, etc.?

b) What further uses are being planned and considered?

65. How did these objectives and intended uses of the LPSD PSA influence the approach that was adopted for the analysis (methodology, level of detail of the analysis, etc.)

66. Has the LPSD PSA been (planned to be) used as a basis of a risk-monitor tool? If so:

a) What is the current status of the Risk Monitor for LPSD conditions?

b) What was the reason for developing Risk Monitor?

c) What changes were made to the basic LPSD PSA so that it could be used for a LPSD Risk Monitor? (Please addresses any modifications or enhancements that were made to the basic LPSD PSA)

d) What operational safety criteria were used to define the regions of low / moderate / high / unacceptable risk for LPSD conditions? How were these derived?

e) What are the results, experiences and lessons learned for the use of the Risk Monitor for LPSD conditions?

RESPONSES

Belgium

63. What were the objectives of carrying out the LPSD PSA?

To have an assessment of the level of safety of the plant in the Periodic Safety Review.

a) Have all these objectives been met?

Yes.

64. How has the LPSD PSA been used?

Only as part of the Periodic Safety Review.

a) What role has the LPSD PSA played in design reviews, decision making, addressing regulatory requirements, etc.?

The shutdown PSA was for some plants a major argument to modify some operational practices (e.g. opening of the pressuriser manhole during midloop operation).

b) What further uses are being planned and considered?

None.

65. How did these objectives and intended uses of the LPSD PSA influence the approach that was adopted for the analysis (methodology, level of detail of the analysis, etc.)

The scope of the LPSD PSA has been determined by the Periodic Safety Review objective.

66. Has the LPSD PSA been (planned to be) used as a basis of a risk-monitor tool? If so:

No

Germany

63. What were the objectives of carrying out the LPSD PSA?

The objectives were the determination the safety level of a PWR plant in low-power and shutdown operating modes, the identification of the main contributors to risk and the evaluation of the PSA methodology for LP&SD-conditions.

a) Have all these objectives been met?

Yes.

64. How has the LPSD PSA been used?

a) What role has the LPSD PSA played in design reviews, decision making, addressing regulatory requirements, etc.?

Some modifications of the availability of safety systems during LP&SD have been introduced. The moment of the periodical test of the mid-loop-level-measurement has been changed. Some procedures for human actions have been introduced or improved.

It became obvious that also for LP&SD states procedures are necessary to cope with initiating events.

Concerning regulatory requirements, a LP&SD PSA of Level 1 will be required in the new PSA-guideline.

b) What further uses are being planned and considered?

The PSA will be used as an example for LP&SD PSAs of other plants.

65. How did these objectives and intended uses of the LPSD PSA influence the approach that was adopted for the analysis (methodology, level of detail of the analysis, etc.)

It was the main objective of the PSA to develop the methodology of a LP&SD PSA.

66. Has the LPSD PSA been (planned to be) used as a basis of a risk-monitor tool?

The use of a risk monitor is not planned.

Hungary

63. What were the objectives of carrying out the LPSD PSA?

The main objectives of the study were to meet the following requirements defined by the Hungarian Atomic Energy Authority for Periodic Safety Reviews of the plant:

- quantify core damage risk when the plant is operated at low power or is shut down
- identify dominant contributors (initiating events, accident sequences, malfunctions, equipment and human failures) to risk
- develop recommendations for improving safety in low power and shutdown modes.

a) Have all these objectives been met?

Yes, the core damage risk has been quantified for a refuelling outage, dominant contributors to risk have been identified, and safety enhancement measures have been recommended and prioritized.

64. How has the LPSD PSA been used?

Requirements defined by the Nuclear Safety Directorate of Hungarian Atomic Energy Authority for Periodic Safety Reviews of the plant have been fulfilled.

A number of upgrading measures have been proposed and implemented (some of them are still underway) based on the results of the LPSD PSA. These measures include improvements in plant systems (crane operations), procedures (EOPs), and administrative controls (provisions on access to equipment) as well.

One modification has been proposed based on the results of a systematic, PSA based review of the boron dilution possibilities performed separately from the LPSD PSA study within a PHARE project. The proposed modification is concerned with the definition of a reserve loop.

Selected configurations have been analyzed in each plant operational state for the purpose of relaxation of the maintenance program.

a) *What role has the LPSD PSA played in design reviews, decision making, addressing regulatory requirements, etc.?*

See above.

b) *What further uses are being planned and considered?*

- configuration control, risk monitor
- precursor event studies for events occurring at LPSD conditions

65. How did these objectives and intended uses of the LPSD PSA influence the approach that was adopted for the analysis (methodology, level of detail of the analysis, etc.)

Similarly to the full power PSA the model developed for the LPSD states is very detailed. No screening and grouping was applied during the POS definition in order to be able to (1) distinguish potential high risk states even if they are short, (2) use the model later for analysis of events that may increase the core damage risk of POSs having low (or even negligible) average risk.

66. Has the LPSD PSA been (planned to be) used as a basis of a risk-monitor tool? If so:

a) *What is the current status of the Risk Monitor for LPSD conditions?*

Under consideration

Japan

63. What were the objectives of carrying out the LPSD PSA?

a) *Have all these objectives been met?*

In Japan, the LPSD PSA is conducted for the typical schedule of the maintenance outage, under the periodic safety review. The risk level of the plant is confirmed to be sufficiently low.

64. How has the LPSD PSA been used?

a) *What role has the LPSD PSA played in design reviews, decision making, addressing regulatory requirements, etc.?*

In the periodic safety review, the countermeasures for those plant operation states (i.e., POS) were identified to enhance the plant safety, which were risk dominant in the LPSD duration. These

countermeasures include the preparation of the emergency operation procedure, the diversification of the reactor water level sensors, etc.

b) What further uses are being planned and considered?

The LPSD PSA continues to be conducted under the periodic safety review. In the future, if the online maintenance is applied to Japanese LWR plants, the LPSD PSA may be used to confirm the risk increments would be small or negative for both shutdown and full power conditions.

65. How did these objectives and intended uses of the LPSD PSA influence the approach that was adopted for the analysis (methodology, level of detail of the analysis, etc.)

We consider that the LPSD PSA should be improved to have the same level of the accuracy as that of the full power PSA. However, we don't think the intended uses has influenced the methodology or level of detail of the analysis. We have tried to choose best method as we could use under the constraints of time and manpower so that the results can be used as extensive as possible. Therefore the uses of the results of the present LPSD PSA would be made with careful considerations of applicability of the results for intended uses. Especially, when LPSD is used for specific decision making, the applicability of models and data specifically related to that decision has to be examined.

Two additional questions / responses were included:

What areas of your analysis approach are, in your opinion, most in need of improvement?

The methodology should be improved to evaluate the frequency of those initiating events that are induced by human errors.

What are the areas that would most benefit from further research?

The method should be clarified to estimate the total CDF through the CDF of the LPSD and the full power. Usually, the CDF of the LPSD is estimated as the probability during the plant maintenance outage. On the other hand, the CDF of the full power is explained as the annual probability of the core damage. In viewpoint of the total risk evaluation, the fire PSA, the seismic PSA and the internal flooding PSA should be also addressed during the LPSD condition.

Korea

63. What were the objectives of carrying out the LPSD PSA?

a) Have all these objectives been met?

Yes, the design objective, i.e., risk-informed design enhancement, is achieved.

64. How has the LPSD PSA been used?

a) What role has the LPSD PSA played in design reviews, decision making, addressing regulatory requirements, etc.?

See the response to Q63.

b) What further uses are being planned and considered?

No further uses are planned.

65. How did these objectives and intended uses of the LPSD PSA influence the approach that was adopted for the analysis (methodology, level of detail of the analysis, etc.)

Because the analysis was used in plant design preparation, some conservative assumptions and criteria were adopted.

66. Has the LPSD PSA been (planned to be) used as a basis of a risk-monitor tool? If so:

No.

The Netherlands

63. What were the objectives of carrying out the LPSD PSA?

a) Have all these objectives been met?

64. How has the LPSD PSA been used?

a) What role has the LPSD PSA played in design reviews, decision making, addressing regulatory requirements, etc.?

b) What further uses are being planned and considered?

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 contributes 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.

Major modification and backfitting programmes were announced at around the same time, 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 licenses 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 licenses 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 Midloop 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 cutsets. In both cases about 40 cutsets 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 offsite 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 cutsets (rank 3, 4, 7 and 8 of the total TCDF cutset list (cutset 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 cutsets 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 cutsets almost disappeared from the sequence cutset 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 midloop 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 depressurization. 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 (Midloop 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.

After the PSA relating to the modification project was completed, the focus shifted towards 'Living PSA' (LPSA) applications. Even, the new license 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 optimization 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, use of the risk monitor, optimization 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.

Spain

63. What were the objectives of carrying out the LPSD PSA?

As explained before, the Spanish PSA program has the aim of characterize the nuclear risk of the facilities independently of its origin, initiating cause or operational condition. This program is phased and the LP&SD PSA for internal events is one of this stages.

a) Have all these objectives been met?

Even though the main objective of the program is characterize the risk, a subsequent one is to use the PSA in applications and decision making. This second objective is still far to be accomplished.

64. How has the LPSD PSA been used?

Initial and immediate applications have been implemented: small design changes, more accurate procedures and outage scheduling, etc. Insights from the studies have also been used in qualitative analysis, outage programme assessment, for instance.

a) What role has the LPSD PSA played in design reviews, decision making, addressing regulatory requirements, etc.?

It is not yet fully used

b) What further uses are being planned and considered?

The aim of the studies is to be used in all future PSA applications that require a shutdown assessment, moving from current qualitative to more quantitative analysis. Risk monitors are also expected and probably used in outage planning and follow-up.

65. How did these objectives and intended uses of the LPSD PSA influence the approach that was adopted for the analysis (methodology, level of detail of the analysis, etc.)

As the objectives and the methods and accuracy (level of detail, truncation, etc.) for full power PSA have also been used in the LP&SD PSA, no additional impact of the objectives on the methodology was expected.

66. Has the LPSD PSA been (planned to be) used as a basis of a risk-monitor tool? If so:

In Spain, the development of a risk monitor is not a CSN requirement. Several plants have developed one for full power operation.

a) What is the current status of the Risk Monitor for LPSD conditions?

Currently, no LP&SD risk monitor has been developed in Spain. However, some plants are planning to develop it to complement the full power existing ones.

b) What was the reason for developing Risk Monitor?

The reasons for full power risk monitors were Maintenance Rule, risk management, on-line maintenance and PSA applications in general.

c) What changes were made to the basic LPSD PSA so that it could be used for a LPSD Risk Monitor? (Please addresses any modifications or enhancements that were made to the basic LPSD PSA)

Not applicable.

d) What operational safety criteria were used to define the regions of low/moderate/high/unacceptable risk for LPSD conditions? How were these derived?

Not applicable.

e) What are the results, experiences and lessons learned for the use of the Risk Monitor for LPSD conditions?

Not applicable.

Sweden

63. What were the objectives of carrying out the LPSD PSA?

a) Have all these objectives been met?

CSNI Question number	Answer from Swedish plant	Answer
Q63	F1/2	Risk verification.
	R1	Make risk estimate of LP & outage mode.
	R2-4	R2, Risk estimates of LP and outage modes
	B2	To identify areas of risk

	O1	See answer to question Q2. Answer only valid for low power PSA, not outage period.
	O2	To assess the complete risk. To compare risk contribution from staying in power operation with components unavailable, with risk contribution from shutting down
	O3	To assess the complete risk. To compare risk contribution from staying in power operation with components unavailable, with risk contribution from shutting down.
SKI REMARK	The primary objectives have of course been to evaluate the risks and find weaknesses.	
Q63 a	F1/2	The LPSD has been used as a part of a PSA report to SKI (Nuclear Inspectorate).
	R1	To the date, yes.
	R2-4	R2, Most
	B2	To some extent
	O1	The analysis is not completed yet so it is not possible to answer the question. Answer only valid for low power PSA, not outage period.
	O2	They will be
	O3	They will be
SKI REMARK	This work is an ongoing process since not all plants have finalised the LPSD studies yet.	

64. How has the LPSD PSA been used?

a) What role has the LPSD PSA played in design reviews, decision making, addressing regulatory requirements, etc.?

b) What further uses are being planned and considered?

CSNI Question number	Answer from Swedish plant	Answer
Q64	F1/2	The LPSD has been used as a part of a PSA report to SKI (Nuclear Inspectorate).
	R1	Make risk estimate of LP & outage mode.
	R2-4	R2, Risk estimates of LP and outage modes
	B2	As input to various discussions
	O1	See answer to question Q63a. Answer only valid for low power PSA, not outage period.
	O2	It hasn't yet
	O3	It hasn't yet
SKI REMARK	In some cases changes in the procedures during SD have been introduced.	
Q64	F1/2	More restriction on the lower containment door". More emphasis about operability of "as many systems possible" during refuelling. Shortening of the maintenance time for the RHR system. Availability and connection to the power grid as high as possible.
	R1	One of many tools in the decision making.

	R2-4	R2, A tool for identifying needs for plant modifications and need for improvements of instructions
	B2	To a limited extent
	O1	See answer to question Q63a. Answer only valid for low power PSA, not outage period.
	O2	Not applicable
	O3	Not applicable
SKI REMARK	See Q64	
Q64 b	F1/2	An update of the LPSD is planned.
	R1	The future will hold the answer. We don't see any limits.
	R2-4	R2, No answer
	B2	No answer
	O1	See answer to question Q63a. Answer only valid for low power PSA, not outage period.
	O2	None so far
	O3	None so far
SKI REMARK	Nordic joint project.	

65. How did these objectives and intended uses of the LPSD PSA influence the approach that was adopted for the analysis (methodology, level of detail of the analysis, etc.)

CSNI Question number	Answer from Swedish plant	Answer
Q65	F1/2	
	R1	N. A.
	R2-4	No answer
	B2	No answer
	O1	In order to give a fairly accurate measurement of the safety level of the plant it is of importance to be as detailed as possible. It is however of no use to be more detailed in the low power PSA than in the full power PSA in order to be able to determine the actual safety level for a whole year of operation. The level of detail is of course also determined on the resources given for the analysis. The main focus during the analysis has therefore been to identify possible hazards during low power conditions with a conservative approach. It has not been decided upon yet (since the analysis is not competed) if the level of detail needs to be enhanced or not. Answer only valid for low power PSA, not outage period.
	O2	No comment
	O3	The objectives and intended uses were the starting points for designing the analysis.
SKI REMARK	This is an objective for the joint Nordic project.	

66. Has the LPSD PSA been (planned to be) used as a basis of a risk-monitor tool? If so:

a) What is the current status of the Risk Monitor for LPSD conditions?

b) What was the reason for developing Risk Monitor?

c) What changes were made to the basic LPSD PSA so that it could be used for a LPSD Risk Monitor? (Please addresses any modifications or enhancements that were made to the basic LPSD PSA)

d) What operational safety criteria were used to define the regions of low / moderate / high / unacceptable risk for LPSD conditions? How were these derived?

e) What are the results, experiences and lessons learned for the use of the Risk Monitor for LPSD conditions?

CSNI Question number	Answer from Swedish plant	Answer
Q66	F1/2	No
	R1	No
	R2-4	R2, No
	B2	No
	O1	The low power PSA model is prepared for instantaneous risk applications. Answer only valid for low power PSA, not outage period.
	O2	Eventually, in the future
	O3	Eventually, in the future
SKI REMARK	This is an objective for the joint Nordic project.	
Q66 a	F1/2	Risk monitors are not considered as a "qualified tool" yet. Because of uncertainties in the analysis
	R1	N. A.
	R2-4	R2, No such application in the near future
	B2	No application in sight
	O1	A risk monitor is under evaluation but is not in operation for the day to day work. Answer only valid for low power PSA, not outage period.
	O2	Not started
	O3	Not started
SKI REMARK	No plant has yet used the LPSD PSA as a basis for risk monitor tools.	
Q66 b	F1/2	
	R1	N. A.
	R2-4	R2, Not applicable
	B2	No answer
	O1	Our reason to buy a risk monitor was that we want the plant operators to more quickly understand the effects of for example taking a pump out of service but also top use the tool to better optimise the plant operation

		safety.
		Answer only valid for low power PSA, not outage period.
	O2	Not applicable
	O3	Not applicable
SKI REMARK	See Q66a.	
Q66 c	F1/2	None (better qualitative methods first)
	R1	N. A.
	R2-4	R2, Not applicable
	B2	No answer
	O1	The Risk Monitor chosen by OKG for evaluation (RiskSpectrum® RiskWatcher) does not require any changes to be made in the PSA model. If a low power Risk Monitor will be taken into service at OKG1 the developed will need no changes.
		Answer only valid for low power PSA, not outage period.
	O2	Not applicable
	O3	Not applicable
SKI REMARK	See Q66a.	
Q66 d	F1/2	
	R1	N. A.
	R2-4	R2, Not applicable
	B2	No answer
	O1	Since no Risk Monitor has been taken into service at OKG1 it is not possible to give an answer to this question.
		Answer only valid for low power PSA, not outage period.
	O2	Not applicable
	O3	Not applicable
SKI REMARK	See Q66a.	
Q66 e	F1/2	
	R1	N. A.
	R2-4	R2, Not applicable
	B2	No answer
	O1	Since answer to question Q66d.
	O2	Not applicable
	O3	Not applicable
SKI REMARK	See Q66a.	

United Kingdom

63. What were the objectives of carrying out the LPSD PSA?

a) Have all these objectives been met?

The original objective was to provide a 'complete' PSA for licensing purposes. The more recent changes to the LPSD PSA were effected to incorporate improvements to operating practices and to better align the

POSs with the PSA. In aligning the LPSD PSA it is better able to aid decision making during different shutdown modes. It has also been refined in response to the station needs.

64. How has the LPSD PSA been used?

a) What role has the LPSD PSA played in design reviews, decision making, addressing regulatory requirements, etc.?

It has played an essential part of the design and licensing and continues to form part of the decision making process.

b) What further uses are being planned and considered?

Current initiatives in risk informed decision making are being considered.

65. How did these objectives and intended uses of the LPSD PSA influence the approach that was adopted for the analysis (methodology, level of detail of the analysis, etc.)

The initial use as part of the licensing process, effectively made the choice to use an average annualised maintenance model. If a risk monitor tool had been the aim, then an all plant available model would have been developed. However this does not preclude the use of the current model to model specific outage states and equipment operability.

66. Has the LPSD PSA been (planned to be) used as a basis of a risk-monitor tool? If so:

The current LPSD PSA is not planned to be used as a risk monitor.

United States

BWR

63. What were the objectives of carrying out the LPSD PSA?

The primary objective was to perform a detailed analysis of potential accidents that could occur at Grand Gulf while the plant is in POS 5.

Other objectives included the following. (1) Compare the results of this study to the results of the full power analysis for Grand Gulf (NUREG/CR-4550, Vol. 6). (2) Develop a methodology for performing PRAs for nuclear power plants in conditions other than at full power. (3) Provide an analytical tool with which the NRC can evaluate the potential benefits of proposed changes in regulations affecting the required operability of equipment when a plant is in a condition other than full power.

a) Have all these objectives been met?

The primary objective was met, and objective (1) of the other objectives was met. For other objective (2), a methodology was developed; however, additional refinement to the methodology would prove useful. As for other objective (3), the results from the LPSD analysis, along with other work, provided a mechanism for the NRC to evaluate proposed changes.

64. How has the LPSD PSA been used?

a) What role has the LPSD PSA played in design reviews, decision making, addressing regulatory requirements, etc.?

See answer to Q2 and Q56.

b) What further uses are being planned and considered?

See answer to Q2 and Q56.

65. How did these objectives and intended uses of the LPSD PSA influence the approach that was adopted for the analysis (methodology, level of detail of the analysis, etc.)?

Since the objectives were to perform a detailed analysis of important LPSD conditions and to compare the results with those from full power, this required that the LPSD analysis be equivalent to the full power analysis; thus, the LPSD analysis used essentially the same methods and level of detail as did the full power analysis.

66. Has the LPSD PSA been (planned to be) used as a basis of a risk-monitor tool? If so:

No.

PWR

63. What were the objectives of carrying out the LPSD PSA?

See answer to Q2.

a) Have all these objectives been met?

1) NUREG/CR-6144- Yes.

2) Screening analysis of NUREG/CR-6144- Yes; however, as noted in Q10, development of methods for screening POSs would benefit from further research.

3) NUREG/CR6616 and 5718 - Yes.

64. How has the LPSD PSA been used?

a) What role has the LPSD PSA played in design reviews, decision making, addressing regulatory requirements, etc.?

See answer to Q2 and Q56.

b) What further uses are being planned and considered?

See answer to Q2.

65. How did these objectives and intended uses of the LPSD PSA influence the approach that was adopted for the analysis (methodology, level of detail of the analysis, etc.)

1) NUREG/CR-6144- The study is the first PSA that analyzed mid-loop operation in detail. The approach was to make it as realistic as we could, subject to limitations on information from the plant and available resources.

2) Screening analysis of NUREG/CR-6144- The screening objective and timing requirement of the study required that many simplifying assumptions be made. It does lay down the foundation for further studies.

3) NUREG/CR-6616- The objective of performing sensitivity calculations for risk comparison requires that the PSA model development to be simplified.

66. Has the LPSD PSA been (planned to be) used as a basis of a risk-monitor tool? If so:

None by the NRC.

APPENDIX N - AREAS FOR RESEARCH/DEVELOPMENT

PRELUDE AND QUESTIONS

As stated in the beginning, the purpose of this questionnaire is to identify the differences between methods (and associated data) used in full power and LPSD and determine data and method development needs to better address LPSD assessments. For each of the areas discussed above (scope, success criteria, system modelling, data, human reliability, quantification/sensitivity/studies/uncertainties, plant damage stages, containment performance) please respond to the following questions.

67. What are the strengths and weaknesses of the analysis you have carried out?

68. What areas of your analysis in your opinion need improvements?

69. What areas would benefit from further research?

RESPONSES

Germany

67. What are the strengths and weaknesses of the analysis you have carried out?

The PSA has shown the importance of initiating events during LP&SD. It has led to significant improvements (procedures, availability of redundancies, …) and led to insights on weak points during LP&SD.

Weaknesses may be seen in the completeness with regard to IEs in POSs after refuelling, in other than the analyzed outage and a systematic analysis of possible human induced IEs.

68. What areas of your analysis in your opinion need improvements?

A systematically consideration of repair and accident management measures to analyze the core damage frequencies would improve the PSA.

69. What areas would benefit from further research?

Further research is needed to analyze the consequences of deboration accidents in more detail. The effect of an internal boron dilution on the recriticality has not been finally analysed and rated. Since the frequency of deboration accidents is in the same order of magnitude as for transients, further research is necessary to evaluate these scenarios.

Hungary

67. What are the strengths and weaknesses of the analysis you have carried out?

The main strength of the analysis carried out is the very detailed model that allows the use of the model and results for practically any applications. The main weakness is related to the same, namely the detailed model requires rather long calculation time.

68. What areas of your analysis in your opinion need improvements?

Determination of probability/frequency of specific erroneous human interactions (by plant staff or maintenance personnel) that can result in an initiating event. The uncertainty of the present numbers is high, at the same time the results are sensitive to these values.

There is also a need for more data support to human reliability analysis of pre-initiator and initiator type human actions.

69. What areas would benefit from further research?

See Q68.

Korea

67. What are the strengths and weaknesses of the analysis you have carried out?

The detailed classification of POS and related initiating events can be one of major strength of the analysis. The lack of detailed modelling regarding realistic LPSD conditions will be major weaknesses.

68. What areas of your analysis in your opinion need improvements?

The detailed modelling development regarding realistic LPSD conditions is needed.

69. What areas would benefit from further research?

The on-line maintenance and/or risk configuration control program, such as a risk monitor, can be developed from further LPSD PSA research.

Spain

67. What are the strengths and weaknesses of the analysis you have carried out?

The whole process followed is considered a strength of the analysis: accuracy, level of detail, review process. This process provides the model with a confidence that allows it to be used.

68. What areas of your analysis in your opinion need improvements?

The following areas should be improved: T/H analyses, man induced initiating events, internal fire and floods and Level 2 analyses.

69. What areas would benefit from further research?

See answer to question Q68.

Sweden

67. What are the strengths and weaknesses of the analysis you have carried out?

CSNI Question number	Answer from Swedish plant	Answer
Q67	F1/2	Strengths: A "systematic" identification of IE's and following sequences. Weaknesses: see Q62
	R1	No major weakness identified in regard to the goal.
	R2-4	
	B2	The study has addressed important issues but was not covering the whole spectrum of IEs, PODs and consequences
	O1	The strength is that the complete low power period (from full power → cold shutdown and back again from cold shutdown → full power) is covered by the analysis and that a very thorough study in order to identify all relevant operator actions has been conducted. Another strength here is not only those actions that been included have been documented; also those that have not been included are covered. The main weakness with the low power PSA that is being conducted is that some conservative assumptions have been necessary to make (e.g. about system requirements). Answer only valid for low power PSA, not outage period.
	O2	No comment
	O3	The analysis has only recently been performed; therefore the strengths and weaknesses of the analysis have not yet been identified.
SKI REMARK	The strengths are that weaknesses during LPSD have been/will be found, for a total view of the reactor safety. The weaknesses in the studies are that they yet not have reached the level of detail that the power studies have.	

68. What areas of your analysis in your opinion need improvements?

CSNI Question number	Answer from Swedish plant	Answer
Q68	F1/2	See Q62 Identification of: "Man made" initiating events Area events External events (hazards) Level 2 Fault trees (system modelling and dependencies) Developing of methods (qualitative and repeatability).
	R1	No plans.
	R2-4	R2, Will be evaluated when current full power PSA update is finished
	B2	The study from 1995 must be redone from scratch
	O1	System requirements during low power conditions. Answer only valid for low power PSA, not outage period.
	O2	Shut down period risk assessment

	O3	Refuelling period risk assessment
SKI REMARK	This is the same question that was asked in Q9!	

69. What areas would benefit from further research?

CSNI Question number	Answer from Swedish plant	Answer
Q69	F1/2	"Man made" initiating events Area events External events (hazards)
	R1	Not different from operation.
	R2-4	R2, Will be evaluated when current full power PSA update is finished
	B2	There are many areas that could be developed. One objective should be to find methods that make the analysis cheap to perform and maintain. The deterministic analysis for other plant states than full power could be improved. Basic approaches for system requirements and success criteria could be defined. A simplified, and stable method for HRA is wanted. Exchanges of studies.
	O1	System requirements during low power conditions. Answer only valid for low power PSA, not outage period.
	O2	All
	O3	All
SKI REMARK	This is the same question that was asked in Q10!	

United Kingdom

67. What are the strengths and weaknesses of the analysis you have carried out?

The main strength is the comprehensive nature of the review of shutdown operation and the advice it has provided back to operations during shutdown. A weakness has been the use of average maintenance, which does not cover any 'specific' outage and requires data changes in order to model specific POSs. However, other specific outage planning tools are used which allow a specific outage to be considered.

68. What areas of your analysis in your opinion need improvements?

Non melt contributions to risk are calculated outside the event tree/ fault tree model and it would be better to integrate resulting plant damage states contributions within the model. PSA Level 2/3 factors are also combined utilising a spreadsheet approach and would benefit from an event tree/ fault tree approach in order to assign importances for PSA Level 3 risk metrics.

69. What areas would benefit from further research?

Areas to consider would include extended time factors for human factors evaluation.

United States

BWR

67. What are the strengths and weaknesses of the analysis you have carried out?

Strengths include:

- the detailed analysis of the POS 5 initiating events actually analyzed using event and fault trees (i.e., the use of multiple systems to respond to the initiating event),
- the detailed consideration of dependency among human actions within sequences, and
- the performance of various sensitivity calculations to understand the impact of various modelling changes.
- Weaknesses (from a general viewpoint) include:
- not modelling some types of initiating events (e.g., maintenance induced LOCAs),
- the inability to modelling the complete set of known pre-existing conditions relevant for all outage types,
- the modelling of a single outage type (i.e., refuelling),
- the lack of modelling transitions between POSs, and
- the lack of details relevant to the progression of accidents and the subsequent development of the source term for conditions unique to shutdown.

68. What areas of your analysis in your opinion need improvements?

The detailed analysis could be improved by inclusion of other POSs, other types of outages, and the inclusion of the complete set of known pre-existing conditions relevant for all types of outages. It might be improved by more subdivisions of the POSs and by modelling the transition between POSs. In addition, the accident progression and source term analysis could benefit from a better understanding of the unique shutdown-specific conditions and phenomena (e.g., core damage with an open reactor vessel and the subsequent accident progression with an open containment).

69. What areas would benefit from further research?

Modelling of the transition between POSs and development of a method to deal with the complete set of known pre-existing conditions relevant for all types of outages would benefit from further research. In addition, sequences with an open reactor vessel and the progression of accident sequences where the containment is open would benefit from additional research.

PWR

67. What are the strengths and weaknesses of the analysis you have carried out?

1) NUREG/CR-6144- The consideration of different outage types and the time window approach are the strength of the study. The potential weakness of the study is that closer interactions with the plant would ensure that the mitigating strategies modelled are consistent with the plant's understanding.

2) Screening analysis of NUREG/CR-6144- The screening analysis involved significant amount of effort in understanding the configurations of the plant and plant activities during shutdown. It includes reviewing operating procedures and shift supervisor's log books, visits to the plant, and the talk through format of the event tree development process. It established a good foundation for further development, even though the screening PSA includes many simplifying assumptions.

3) NUREG/CR-6616- The strengths of the study is that the PSA model was developed with much less resources. The weakness is that the simplifying assumptions remain to be further evaluated.

68. What areas of your analysis in your opinion need improvements?

In general, a PSA should represent a realistic model of the plant. The configuration of the plant at shutdown should be correctly defined to provide an initial condition for the shutdown PSA. The plant's response to an initiating event should be adequately analyzed using appropriate thermal hydraulic codes. The operator responses to accidents should be consistent with the realistic behaviour of the plant and guided by procedures and training. Plant specific data should be collected to support all aspects of the study.

Closer interaction with plant operators in creating the LPSD model would improve on the choice and modelling of the POSs.

The Surry LPSD analysis also could be improved by updating the analysis to ensure it corresponds with the current operation of the plant.

69. What areas would benefit from further research?

Comparison of plant designs using PSAs - A frequently quoted weakness of PSAs is that different assumptions and simplifications are the major causes of the differences in the results. It is therefore necessary to compare the PSAs of similar plants to determine if the differences in the results are based on real difference in design and operation of the plants. Such a comparison would also identify unique accident scenarios analyzed by one plant but not the others.

Improve the treatment of internal fire and flood and seismic initiators - Previous LPSD analyses have shown that risk from these initiators can be important contributors to specific plants (e.g., the analysis of Surry in NUREG/CR-6144 indicated that fire was a dominant contributor to CDF). As such, it is important to identify the shutdown-specific conditions and activities (i.e., issues) that affect internal fire flood and seismic analyses. Once done, issues that are already adequately examined using current fire, flood, and seismic methodologies should be identified. The effects on CDF and risk of the remaining issues should be prioritized. For those issues deemed a high priority, develop or enhance current techniques to incorporate them into an analysis of shutdown conditions.

Improve the treatment of unplanned outages - Generally, LPSD analyses have not examined the risk from unplanned outages. To enhance the completeness of information used in risk-informed decision-making,

unplanned outages should be examined. To accomplish this, licensing requirements should be examined to identify conditions that require a plant to perform an unplanned shutdown. Those that do not involve equipment used to mitigate accidents should be eliminated. For the remaining equipment, an efficient method for analyzing the risk associated with the unplanned outage should be developed.

Improve the treatment of transition risk - To enhance the completeness of LPSD analyses, the degree to which transition risk is currently accounted for in shutdown analyses should be identified and the issues associated with transition risk should be investigated and prioritized. For those issues that are deemed a high priority, develop or enhance current techniques to incorporate them into an analysis of shutdown conditions.

Improve the treatment of fast-acting reactivity insertions - At the Sizewell B facility, this issue was found to contribute about 20% of the total fuel damage frequency during shutdown. To help ensure the adequate treatment of this issue, the significance of the following set of issues should be investigated: (1) potential pathways for un-borated water injection, (2) mitigative effects of mixing in the core region and mixing in the piping, and (3) maximum damage that could be expected if a slug of un-borated water moves through the core region. For those issues that are determined to be important, develop or enhance current techniques to incorporate them into an analysis of shutdown conditions, including a quantification methodology.

Improve HRA is used for LPSD conditions- Both the NRC and industry analyses have identified the importance of human actions. To help ensure that human actions are adequately analyzed during LPSD conditions, LPSD issues that can affect an HRA should be identified and prioritized. For those issues deemed a high priority, develop or enhance existing HRA techniques to incorporate into LPSD analyses. Ascertain whether errors-of-commission are important to LPSD risk. If important, develop or enhance current techniques to efficiently model and incorporate errors-of-commission into LPSD analyses.

Improve treatment of crane failures associated with heavy load lifts inside containment. Because accidents resulting from crane failures are not usually included in LPSD analyses, develop or enhance existing techniques to model crane failures during LPSD conditions.

Update crane failure frequencies for heavy load lifts inside containment - To support the improved treatment of crane failures associated with heavy load lifts inside containment, the drop frequencies of nuclear grade crane drops should be updated to account for the past 20 years of nuclear grade crane operating experience.

Establish a LPSD baseline model - To support the use of LPSD risk assessment information in risk-informed regulatory decision-making, a baseline model for LPSD conditions should be developed. The model should account for forced and unplanned outages–at a minimum, accounting for historical forced and unplanned outages.

Improve the current understanding of draindown events and update frequencies - Because past analyses have found that these types of events can be important, past draindown events should be examined to identify factors that influence their occurrence. Initiating event frequencies and uncertainties used in industry and NRC-sponsored LPSD risk analyses should be updated to include post-1990 data. The frequencies should be adjusted to account for the factors that influence their occurrence.

Develop failure data for extended operations - To increase the realism necessary for risk-informed regulatory decision-making, currently available data should be examined to ascertain whether or not it is sufficient to produce failure rate estimates for components that experience extended periods of operation

during shutdown conditions. If sufficient information is available, appropriate failure rates should be developed for the components.

Examine current thermal-hydraulic tools to ensure their efficient operation - Thermal-hydraulic calculations play an important roll in determining both success criteria and the time available for operators to respond to events. To increase the usefulness of thermal-hydraulic calculations, developers of the tools used to perform these calculations should be questioned to ascertain whether the tools function efficiently for shutdown conditions. Selected calculations should be performed to verify efficient operation during selected shutdown conditions. If necessary, areas where efficiency is lacking should be addressed by performing code modifications.

Investigate potential for spent fuel pool fuel misleading- Since plants are increasing the storage capacity of their spent fuel pools beyond their original limits and fuel misloading is not usually analyzed as part of a shutdown analysis, the current understanding of risk resulting from these activities should be improved. To accomplish this, the process used to develop new spent fuel reracking schemes to achieve additional storage capability should be examined to determine whether or not it is sufficient to prevent potential fuel damage resulting from fuel misloading.

Develop guidance on establishing success criteria - Because establishing success criteria is vitally important to correct accident sequence development, LPSD conditions should be examined to identify those that would affect the determination of success criteria. Guidance should then be developed to facilitate the appropriate consideration of these LPSD conditions.

Establish minimum requirements for defining a plant operational state - Defining plant operational states is one task in a LPSD analysis. Currently, different approaches are used, resulting in some analyses having a small number (e.g., 10 to 20) and some having a large number (i.e., much more than 20). To ensure that important parameters are considered when defining plant operational states for LPSD analyses, a workshop should be conducted to develop the minimum set of requirements necessary for defining plant operational states.

Enhance the streamlining of accident sequence analysis - Because LPSD risk has been found to be as important as full-power risk, and because resources should be used effectively to identify the more risk-significant conditions, current information should be examined to identify sets of conditions that generally make a plant state more risk-significant, or conversely, make a plant state less risk-significant. Characteristics of initiating events that can make them important even in a less risk-significant state should also be identified. Once identified, guidelines should be developed to determine which conditions and initiating events should be analyzed. To further enhance the wise use of resources, guidance should be developed on how many plant systems (capable of mitigating an accident sequence) to include in the accident sequence development process.

Develop guidance on how to use full-power models in LPSD analyses - Because of the importance of systems analysis to probabilistic risk assessment (PRA) and because conversion of full-power models for use in LPSD analyses is an efficient use of resources, guidance for converting full-power system models into models for use during LPSD analyses should be developed.

Develop guidance on the correct application of full-power common-cause failure (CCF) models to LPSD conditions - Because correct use or implementation of any PRA model is important, guidance should be developed on what common-cause factors should be reviewed to account for LPSD conditions. Furthermore, specific guidance should be developed on how to adjust full-power CCF models to account for LPSD conditions.

Provide guidance on simplified thermal-hydraulic calculations - Because thermal-hydraulic calculations play an important roll in determining both success criteria and the time available for operators to respond to events, a workshop should be conducted to identify whether simplified thermal-hydraulic calculations are sufficient. If simplified calculations are deemed appropriate, then the minimum set of thermal-hydraulic modelling requirements for these simplified calculations should be identified.

Develop guidance on incorporating uncertainty and sensitivity analysis techniques into LPSD analyses - To enhance the usefulness of risk information from LPSD analyses and to provide a more complete understanding of what can be important to risk, guidance on using full-power uncertainty and sensitivity techniques as part of LPSD analyses should be developed.

Develop guidance on cold overpressurisation analysis- To enhance the completeness of LPSD analyses, guidelines for incorporating adequate cold overpressurisation models should be developed.

Identify and study possible Markov process applications for LPSD analyses.