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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 3: RESPONSES TO THE COOPRA 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.

This report provides a summary of the current LPSD PSAs in Member countries, covering the elements which make up the PSAs. This report: (1) identifies the uses of the LPSD PSAs; (2) summarizes current approaches, aspects, and good practices; (3) identifies and defines differences between methods and data in full power and LPSD PSAs; and, (4) identifies guidance, methods, data, and basic research needs to address the differences. The responses to the questionnaires are provided in the Appendixes.

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 2 of this series contains the responses from the CSNI/WGRisk Survey

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EXECUTIVE SUMMARY

The Cooperative Probabilistic Risk Assessment (COOPRA) research program is an international working group of senior research scientists that represent regulatory organizations on safe operation of commercial nuclear power plants.

This report provides a summary of the information shared by COOPRA members in the low power and shutdown (LPSD) working group, which is charged with the responsibility to assess their member country's plant operations at LPSD conditions. It is a status report of each country's activities in the area of LPSD delineated across specific topics. 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.

Probabilistic Risk Assessment/Probabilistic Safety Assessment (PRA/PSA) studies have provided insights that suggest that some operating modes during LPSD conditions could lead to risk that are comparable to some full power operating modes. Given: (1) the repeated events during LPSD modes, (2) changes made in plant operations in response to economic forces, and (3) ongoing initiatives to develop risk-informed, performance-based regulation, the international regulatory community elected to cooperate in sharing of analyses and data to assess the potential risk from LPSD operating conditions.

The following countries provided input to this status report:

- Canada
- Czech Republic
- France
- Germany
- Hungary
- Italy
- Japan
- South Africa
- South Korea
- Spain
- Chinese Taipei
- United States

Detailed country-specific discussions are contained in Chapters 4, 5, and 6. They provide information on LPSD uses, applications, analyses, results, and specific technical areas.

Chapter 4 provides a brief summary of how selected countries use LPSD programs to support applications. The specific applications discussed are: technical specifications, configuration control, risk determination, maintenance planning, and other programs. At this time, not all countries make extensive use of LPSD information, so this chapter does not have as many contributors as the others.

Chapter 5 provides a discussion of country activities in the general area of LPSD analyses and results. It also presents, in Table 5.1, a summary of published results covering the various LPSD PRAs conducted by member countries. These results include internal events that could potentially lead to severe core damage during various phases of a refuelling outage. Some of the PRAs include internal flooding and fire. The analyses cover different reactor types including the VVERs, Westinghouse PWRs, General Electric BWRs, and French PWRs. For internal events, typical CDFs were found to be approximately 4E-05 for VVERs, 1E-06 to 2E-05 for PWRs, and 1E-07 to 3E-06 for BWRs.

Primary-side LOCAs were frequently found to be the most significant events within the POSs that dominate risk in low power and shutdown operation. LOCA events were frequently induced by inadvertent human actions rather than hardware failures.

Although generalizations across countries is sometimes difficult, it appears that some important general findings were identified and modifications to operational practices were implemented, including:

- risk is the same order of magnitude as it is for power operations,
- new improved procedures were developed for loss of RHR,
- new improved procedures were developed for shutdown LOCAs, and
- steam generator water inventory is now required while plant operating conditions are susceptible to loss of RHR.

Chapter 6 provides member country input in the following technical areas:

Thermal Hydraulics - Thermal hydraulics studies pertaining to items such as homogeneous dilution of boron, primary system breaks, loss of RHR, and alternate cooling modes are discussed. They provide supporting information for establishing success criteria. Both simplified and complete models have been employed in these calculations and there is a need for additional code validation.

Core Physics - One of primary areas of concern appears to be risk of reactivity accidents caused by either accidents outside the vessel or inherent in-vessel rapid boron dilution. Another problem area identified is associated with the risk of cold overpressurisation. Still another is brittle fracture of the reactor vessel. It has been found to be caused, in the case of a RHR system break, by stresses introduced during two time frames. The first time frame occurs when the primary water is cooled due to the depressurization resulting from the break and the cold water injected by safety injection. The second occurs when pressure increases after the break is isolated. The combinations of these two events may introduce excessive thermal stresses on the reactor vessel during the cooling and mechanical stresses during the repressurisation.

Initiating Event Frequencies - The starting point for determining LPSD initiating events is usually the full-power list. This is then supplemented by those categories judged unique to LPSD. Estimating LPSD initiating event frequencies involves identifying the unique categories of events that have the potential to result in an initiating event for a plant operating state (POS) and then quantifying each unique category. These categories include internal causes and others, such as the erroneous change of the operational loop, erroneous draining of the operational loop, maintenance activities, erroneous system alignment, heavy load drop, and other interactions. The frequency for each category of event is quantified, and these frequencies are then combined to produce POS-specific initiating event frequencies for each initiating event.

Component Failure Rates - International databases have been accessed to obtain a large amount of data, such as the French Reliability Data Acquisition System and the Event File databases. Data was further verified via onsite investigations to allow for the inclusion of certain site-specific characteristics. Examples of component failure rates and of demand failures are provided. Unique aspects associated with LPSD were also considered such as unavailabilities for maintenance occurring during LPSD and data to be used for new components being installed during LPSD. Component failure rates were typically taken from the full power PSAs and analyzed the same way as they were in the full power PSA, i.e., generic data, specific component operating experience analysis, and Bayesian updating.

Common Cause - The approach used to quantify common cause frequencies (CCFs) most frequently is the β factor method. The β factor is defined as the ratio, for a particular component, of the number of common mode failures to the total number of failures. The method used to identify the CCF component groups and the group-related ß factors for the LPSD PSA is the same as the method used in the full power PSA. Common cause failures characterized by ß factors and the design-manufacturing-related coupling mechanism during outages were also considered to be the same as if at full power operation.

The values for equipment failures with insufficient feedback experience were estimated either by analogy with the preceding components or using engineering judgement, with allowance for the values used in probabilistic safety studies. Dependent failure categories that were identified during the analysis are listed.

Human Reliability Events - Human Reliability Analysis for LPSD is generally similar to the full power approach, although human intervention during LPSD has certain features that make modelling LPSD human actions using automated systematic processing very difficult, if not impossible. Both pre-accident and post-accident human actions are considered. To reduce subjectivity, the methodologies developed try to make maximum use of experience feedback, and they may make a number of relative assessments by keeping in mind similar situations.

The main sources of information are actual LPSD incidents, ad hoc interviews and investigations, and observations made during simulator tests. Specific LPSD human actions reported by Hungary include maintenance activities.

The pre-accident intervention category corresponds to errors during maintenance operations, tests, and normal operation liable to contribute either to the unavailability of an engineered feature system or the occurrence of an initiating event.

The post-initiator intervention category includes both diagnostic and decision-making errors and errors in taking action. In addition, distinctions are made among the roles of the operating team, the safety engineer (a French innovation) and the action of the emergency teams. The activity of the operating team is divided into two phases, diagnostic and operational. The diagnostic phase includes detection of the incident, making the diagnostic, and making a decision. Operation in accident situations can be represented by a list of key actions, whose success or failure modifies the accident sequence. In the event of failure of the diagnostic, it is generally considered that key actions are not performed. In the event of success of the diagnostic, the corresponding actions may fail. For each important action, the probability failure is evaluated. Some formulas are given.

Equipment Performance and Recovery - Estimating the ability of equipment to function in accident situations is one problem frequently encountered in the study of accident sequences.

POSs chosen were ideally selected based on the availability and changing redundancy of systems that can mitigate potential initiating events. The scope of the available mitigating systems are defined in each POS by taking into account the list of all the potential initiating events in the given POS. Possible operational modes for these systems during the shutdown period have been identified using reactor shutdown procedures as a basis, along with a number of existing shutdown schedule plans

It is recommended that some guidance on how to recover or even replace equipment should be incorporated into some type of plant operating (or emergency) procedures for the most risk significant scenarios.

The U.S.-based Licensee Event Report data used to establish the loss of support system failure frequencies have been used to estimate the recovery time distribution associated with each initiator.

Scope and Success Criteria - Current LPSD analyses are Level 1 PSAs for internal initiators only. An examination of available fire and internal flood risk analysis methodologies to determine whether enhancements can be made to allow these analyses to be performed in a more cost effective manner is needed, since these types of risk are believed to be significant and performance of these analyses should allow safety enhancements to be identified.

Success criteria for shutdown conditions were determined by reviewing various studies and performing plant-specific thermal-hydraulic analyses. The changing level of decay heat was accounted for by defining time windows after shutdown each with its own set of success criteria.

The resulting success criteria are somewhat similar to those of full power except that additional components, such as stored system energy and extent of fuel uncovery, are frequently considered.

One recommendation developed is that an examination of available fire and internal flood risk analysis methodologies should be made to determine whether enhancements can be made to allow the methodology to be used in a more cost effective manner, since these types of risk are believed to be significant and performance of these analyses should allow safety enhancements to be identified.

Another recommendation relates to the issue of risk from radionuclide release from sources other than the reactor vessel. Specifically, risk analysis of the refuelling phase is sometimes excluded based on the fact that there is no fuel in the reactor vessel, whereas it is located elsewhere in the containment or refuelling area and may still pose a threat.

Success Paths - Present LPSD results are typically for plant operational states occurring during a planned refuelling shutdown. Plans include extending LPSD PSAs to other types of outages, e.g. shutdowns that terminate at the hot standby state or shutdowns resulting from accident situation initiated from the nominal power operational state. Sequences were classified as successful according to the classic definition of the plant reaching a stable situation for the mission time assumed.

The means to consistently link success path sequences from full power event trees to event trees for LPSD should be developed. This would provide a complete PSA model that should be unique and coherent in its assumptions, codification, data, HRA input, dependency analysis, etc.

Screening Techniques - Core damage frequency is the central figure-of-merit for quantitative screening.

There is a need for a simplified, systematic approach for screening, one that would allow for a traceable process of selection of the most risk significant LPSD accident scenarios.

The COOPRA LPSD subcommittee is currently investigating ways to improve screening methodologies. Absent a systematic and technically plausible screening process, one that might need to be unique to LPSD PSA methodology, a full-scope LPSD PSA would be extremely costly.

LPSD Duration - Similar to screening, a methodology is also needed for selecting or identifying the duration of LPSD states. This should be supplemented by a methodology for developing sequences and crediting recovery beyond 24 hrs.

On the basis of operating experience with reactors in service, from 6 to 24 different operating states were identified and defined. Duration of each plant operational state was determined by a detailed review of shutdown schedule plans and past operational records. Since the duration of POSs differs, comparison of core damage risk among the states was made based on core damage probability - depending on the state duration rather than frequency. The decision was supported by the fact that some of the initiating events could be characterized by probability of occurrence rather than frequency. For example, some human errors that lead to a plant transient have the potential to be committed during special actions performed during shutdown, thus belonging to the group of initiating events characterized by probability of occurrence.

Average outage durations, based on duration times actually observed for the plant operational states, are used.

The effect of longer or shorter POS duration could be observed by means of sensitivity analysis or by means of on-line or off-line risk monitors or by some other system of "living PSA."

Software -Software currently being used for LPSD evaluations include both probabilistic and thermal hydraulic. The same software used in the full power PSA is typically used in LPSD PSAs. Examples of thermal hydraulic software include:

- The Risk Spectrum PSA Software Package
- The LESSEPS Computer System (used for specific quantifications including time dependencies)
- The NUPRA Computer Code (used to estimate core damage frequency)
- SAPHIRE (PRA)

Examples of thermal hydraulic codes employed include:

- MELCOR (thermal hydraulic)
- RELAP
- CATHARE
- ATHLET

To improve the efficiency of the user interface, preprocessor software needs to be developed. Future needs also include an application code for risk monitoring.

ABBREVIATIONS

ABWR	advanced BWR
AECL	Atomic Energy of Canada, LTD.
ANPA	Agenzia Nazionale per la Protezione Ambiente
ATWS	anticipated transient without scram
BWR	Boiling Water Reactor
CCFs	common cause failures
CDF	core damage frequency
CFD	computerised flow dynamics
COOPRA	cooperative probabilistic risk assessment
CSN	Consejo de Seguridad Nuclear
EOPs	emergency operating procedures
FMEAs	failure mode and effect analyses
GRS	Gesellschaft für Anlagen – und Reaktorsicherheit
HAZOP	hazard and operability
HEPs	human error probabilities
I&C	instrumentation and control
IPE	individual plant examination
ISPN	Institut de Protection et de Sûreté Nucléaire
JAERI	Japan Atomic Energy Research Institute
KAERI	Korea Atomic Energy Research Institute
KINS	Korea Institute of Nuclear Safety
KNPS	Koeberg Nuclear Power Station
LOCAs	loss of coolant accidents
LOOPs	losses of offsite power
LPSD	low-power-and-shutdown
NPP	nuclear power plant
NUPEC	Nuclear Power Engineering Corp
PORV	power-operated relief valve
POS	plant operational state
POSs	plant operational states
PRA	probabilistic risk assessment
PSA	probabilistic safety assessment
PWR	pressurized water reactor
RCS	reactor coolant system
RHR	residual heat removal
RHRS	residual heat removal system
RPV	reactor pressure vessel
RWST	refueling water storage tank
SDC	shutdown cooling
SG	steam generator
SPSA	shutdown probabilistic safety assessment
STUK	Sateilyturvakeskus Ydinturvallisuusosasto
VEIKI	Villamosenergiaipari Kutato Intezet, 18

1. INTRODUCTION

The international cooperative probabilistic risk assessment (COOPRA) research program for low-power-and-shutdown (LPSD) is an international working group of senior research scientists that represent regulatory organizations on safe operation of commercial nuclear power plants. The objective for establishing this working group is to provide a forum for identifying research needs and for sharing new information about on-going research activities and safety assessments of commercial nuclear power plants during LPSD conditions.

As the technology for performing probabilistic risk or safety analyses (PRA/PSA) advanced, the safety of nuclear power plant operations has improved to a point that risk at shutdown increased in relative terms. Limited PRA/PSA studies have provided insights that suggest that some operating modes during LPSD conditions could lead to risk that are comparable to some full power operating modes. Given the: (1) repeated events during LPSD modes, (2) changes made in plant operations in response to economic forces, and (3) ongoing initiatives to develop risk-informed, performance-based regulation, the international regulatory community elected to cooperate in sharing of analyses and data to assess the potential risk from LPSD operating conditions. The LPSD working group under COOPRA 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.

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3. GOALS AND OBJECTIVES OF THE WORKING GROUP

The goals and objectives of the working group on LPSD are as follows:

Assess the data and analyses available from member countries on LPSD operating conditions.

Assess the LPSD experience from the member countries.

Identify technical work required for quantifying the risk/safety of commercial nuclear power plants during LPSD conditions.

Identify cooperative programs that will assist the member countries to quantify the risk/safety during LPSD conditions.

Provide the tools and a forum for technical exchange that will enable the member countries make informed technical and regulatory decisions specific to their country’s needs.

4. LPSD PROGRAMS: USES AND APPLICATIONS

This section describes how LPSD programs have been, or are being, used to support various applications within the member countries. Each use or application is described in a separate subsection. At this time, not all member countries are making use of LPSD information; thus, each subsection may not have the same number of contributor countries.

4.1 Technical Specification

4.1.1 France

Several Technical Specifications improvements have been defined concerning the operation of the primary circuit, the operation of the residual heat removal system (RHRS) pumps, and the management of the containment.

4.1.2 Germany

Technical Specifications have been or will be improved in some NPPs. These plant-specific improvements are mainly related to measures to control an event, but also to prevent an event.

4.1.3 Hungary

The review of all current Technical Specifications is an ongoing activity in Hungary. However, one modification has been proposed based on the results of a systematic review of the boron dilution possibilities performed independently from the LPSD PSA study within a PHARE project whose purpose was to support the LPSD PSA. The proposed modification is concerned with the definition of a reserve loop.

4.1.4 South Africa

On the basis of LPSD PRA, specifically the risk of excessive draindown, technical specifications for draindown and refilling phases of shutdown were developed. The risk of boron dilution accidents, including fast dilution has led to modifications and improvements to technical specifications. The risk of station blackout, which includes LPSD states, has led to specific requirements (LCOs) for diesel generators.

4.1.5 Chinese Taipei

No Technical Specification has been defined, but procedures for shutdown operation have been improved.

4.2 Configuration Control

4.2.1 Hungary

The real possibilities of the configuration control management during LPSD 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 (see Section 4.5.2).

4.2.2 South Africa

As part of the periodic reassessment (approximately 10 year review) of the Koeberg pressurized water reactor (PWR) plant (twin 930 MWe), a comparison was made (Ref. 1) between the level 1 aspects of the Koeberg PRA with the French EPS-900 study (Ref. 2) for their 900 MWe units. Similarly, the level 2 aspects of the Koeberg PRA (Ref. 3) were compared with the North Anna Individual Plant Examination (IPE) study (Ref. 4). The focus of these comparisons was on methodology rather than risk quantification results. For this purpose, NUREG-1602 (Ref. 5) (draft) was used as a standard.

The findings and recommendations of references 1 and 3 are being followed up by the utility (Eskom) in a project designed to address the findings and to customize the Koeberg PRA for risk management and risk compliance purposes (Ref. 7).

Eskom has prepared a report (Ref. 6) on plant operational states for the LPSD aspects of the PRA.

The following is a brief summary of Reference 6.

Plant Operational States for Koeberg Nuclear Power Station (KNPS)

This study has been performed recently to develop the following:

- methodology for defining plant operational states (POSs) for PRA purposes,
- definitions of POSs for KNPS,
- estimated time fraction for each plant operational state (POS), and
- the relevant plant equipment configurations in each POS.

The KBG OTS defines eleven Standard Operating States. In addition, the Koeberg Shutdown Technical Specifications define two configurations for Maintenance Cold Shutdown and four configurations for Refuelling Shutdown.

To define the KNPS POSs, a combination of parameters from the OTS, SAR, CEA EPS-900 study (Ref. 2), and NUREG/CR-6144 were used.

New POSs were defined in terms of the following main parameters:

- ESF Actuation interlocks
- Heat Removal Configuration (SG or RRA)
- Reactor Core Characteristics (Power, Reactivity)
- Reactor Coolant Characteristics (Pressure, Temperature, Level)

- RCP Integrity (Open, Closed, Vessel head on/off)
- Process (Steady State or transient, Startup or Shutdown)
- Time after shutdown (Decay Heat Load)

The POSs defined in Table 4-1 represent the process or cycle from power operation to refuelling and back to power operation.

Table 4-1 Descriptions, durations, and annual fractions for POSs

POS	Description	Hours per year	Days per year	Fraction of year
A	Power Operation (with four sub-states based on RPR interlocks)	7455	311.6	8.511E-1
B	Cooldown with SGs (Automatic ESF Control) (with eight sub-states based on power level, feedwater configuration and main activities)	153	6.4	1.748E-2
C	Cooldown with Steam Generators (Manual ESF Control) (with 11 sub-states based on ESF configuration, feedwater configuration and pressure/temperature.	36	1.5	4.097E-3
D	Cooldown with RRA (with four sub-states based on number of RCP pumps in service, level and temperature)	59	2.4	6.689E-3
E	Draindown before Refuelling or Maintenance (with two sub-states based on different equipment configurations)	143	6.0	1.634E-2
F	Refuelling: Core Unload (with six sub-states based on different allowable equipment configurations)	148	6.2	1.689E-2
G	All fuel in Spent fuel Building (with two substrates based on heat load in SFP) This state is applicable to the Spent Fuel Building only.	303	12.6	3.454E-2
H	Refuelling: Core Reload (with four sub-states based on different equipment configurations)	68	2.4	7.747E-3
I	Draindown after Refuelling (with two sub-states based on different allowed equipment configurations)	65	2.7	7.449E-3
J	Heatup with RRA connected (with six sub-states based on RCP configuration, pressure and temperature)	156	6.6	1.785E-2
K	Heatup with Steam Generators (Manual ESF Control) (with three sub-states based on ESF configuration)	57	2.4	6.463E-3
L	Heatup with Steam Generators (Automatic ESF Control) (with six sub-states based on feedwater configuration, pressure, temperature and power level)	117	4.9	1.338E-2
	TOTAL	8760	365.0	1.00

The POS for forced outages are subsets of those for Refuelling Outages. For forced outages the process or cycle is shorter and involves fewer POSs. The duration of POSs during forced outages should be added to the duration of POS during refuelling outages.

Table 4.1 provides average durations and annual fractions of POSs during the last two fuel cycles of both KBG units.

The licensee is required to assess outage schedules in terms of defense-in-depth and risk using the plant-specific PRA. This is performed (three months) prior to each outage and submitted to the regulator as part of the request to commence the outage. Changes to the outage schedule during the outage are assessed using an accepted process.

4.23 Chinese Taipei

Some risk-significant configurations (called POSs) are defined. Maintenance activities are avoided in these POSs when possible.

4.3 Risk Determination

4.3.1 Czech Republic

The following risk measures are calculated for NPP Dukovany low power and shutdown operation below 55% of nominal power.

Core damage frequencies (CDFs) for all considered plant operational states (POSes) as well as overall CDF for low power and shutdown operation below 55% of nominal power.

Fuel damage frequencies (FDFs) for all considered plant operational states (POSes) as well as overall FDF for low power and shutdown operation below 55% of nominal power. The FDF covers, in addition to CDF, also risk from fuel in spent fuel pool.

Boiling frequency when reactor is open. It has been assumed that such boiling can have adverse impact on plant staff.

The contribution of particular initiators and systems to risk measures are determined as well.

Radiological releases and health effects have not been estimated since only Level 1 LP&SD PSA has been performed.

4.3.2 France

Core melt frequencies (average and instantaneous) have been assessed for all the operating modes, and the dominant contributions have been identified.

4.3.3 Germany

In the LPSD PSAs, frequencies of so-called system damage states (SDS) are determined, which endanger the core cooling. A system damage state occurs if the operational and safety systems for fuel cooling are not available. They do not consider accident management and repair measures.

4.3.4 Hungary

Core damage frequencies have been estimated for all the plant operational states of a refuelling outage. It should be noted however, that some initiating events could be characterized by probabilities rather than frequencies. Some human induced transients for example, can be related to such initiating events. In order to have a common baseline for the analyses, probabilities of these initiating events have been translated into frequencies taking into account the duration of the plant operational state in which they can occur. Thus, in some cases the core damage frequencies were fictitious ones, they could not be used for comparison. Therefore, dominant risk contributors (dominant plant operational states, dominant event sequences, etc.) have been identified based on the core damage probabilities rather than core damage frequencies.

In addition to the core damage, another end state, namely boiling in the core has been identified and analyzed in the plant operational states when the reactor is open for refuelling. It has been assumed that boiling can have on-site impacts on the plant personnel. Boiling frequencies have been estimated for all the appropriate plant operational states, and similarly to the core damage end state, dominant states and event sequences have been identified based on the boiling probabilities.

4.3.5 Italy

In 1987 after five years of commercial operation, the Caorso nuclear power plant, a boiling water reactor (BWR), was placed in cold shutdown. In 1998, a general report on radio-protection aspects related to the program of fuel discharge from Caorso was issued. The report found that the doses are under prescribed limits and that an emergency plan is not needed. In addition, the Safety Authority performed a preliminary risk evaluation related to the proposal of fuel transfer from the reactor to the pool. A qualitative analysis found advantages of having the reactor fuel discharged to the pool. These advantages included a reduction of personnel doses and reduction of systems/components required to be operable.

4.3.6 Japan

Core damage frequency (CDF) is estimated. No other risk measures such as radiological releases and health effects are estimated.

4.3.7 Korea

Core damage frequency is estimated. No other risk measures are estimated.

4.3.8 South Africa

Probabilistic Risk Assessment (PRA) has been part of the nuclear regulatory framework in South Africasince the early seventies. Fundamental safety standards, which include risk criteria, were established against which the safety of nuclear facilities are assessed. It is a regulatory requirement in South Africa for licensees of nuclear installations to develop and maintain a comprehensive plant-specific PRA for each facility or site to demonstrate compliance against these criteria.

Safety submissions to the regulatory authority relating to any change (modification, procedure change etc) impacting on safety must be supported by a safety assessment including a risk assessment where applicable.

Regulatory control of a nuclear installation is exercised by means of a nuclear licence issued by the regulator and incorporating the necessary conditions to ensure that the safety related aspects of the plant design and general operating rules are respected, and that a comprehensive current plant-specific safety analysis (including PRA) is maintained and used to demonstrate that the prevailing safety levels are acceptable.

A level 3 PRA study for the two 925 MWe Pressurised Water Reactors (PWRs) Koeberg units was performed in the late 1970s. This included the risk due to power and shutdown states and non reactor related accidents involving for example spent fuel storage, fuel handling and waste treatment related activities.

The Koeberg PRA was revised in 1991, the level 1 aspects being installed on software developed by the utility, Eskom. External peer reviews were conducted over the period 1994-1996.

A major review was conducted in conjunction with the 1996-1998 periodic review of Koeberg. For the PRA this involved a benchmark against the French EPS-900 study (level 1) and North Anna IPE (level 2). In June 2001, the utility submitted the revised PRA (levels 1-3, including LPSD) installed on Risk Spectrum.

4.3.9 Spain

Currently, the objective of the CSN requirement to perform LPSD PSAs for Spanish nuclear power plants is to estimate average risk (i.e., core damage frequency) without considering health effects.

The Spanish PSA Integrated Program has two objectives. The first is to analyze and assess the risk of nuclear power plant operation, improving safety weaknesses that are found during these analyses. The second objective involves using the PSA models and data to improve the effectiveness of different changes (e.g., plant design or procedure change) by comparing their safety impacts. Therefore, it is expected that LPSD PSAs will be used in the future for applications like configuration control, technical specification, and maintenance planning.

4.3.10 Chinese Taipei

Core damage frequency is estimated. No other risk measures such as radiological releases and health effects are estimated.

4.4 Maintenance Planning

4.4.1 Canada

Shutdown state PSA for the CANDU designs have typically addressed three plant operating states, viz: primary heat transport system full and pressurized, full and depressurized, and drained to the headers level. This work has provided important engineering insights and helped in optimization of maintenance programs. This has also led to development of provisions and procedures for rapid boiler bolt-up and heat transport system fill-up to enable recovery from loss of heat sink during the system drained state.

4.4.2 Hungary

The real possibilities of the risk-based maintenance planning in general have not been studied yet. However, an application of the LPSD PSA results has been made, results of which can be used for planning of maintenance activities in a more relaxed way (see Section 4.5.2).

4.4.3 Italy

The Caorso nuclear power plant, a BWR, was in commercial operation from December 1981 to 1986. The plant has been in cold shutdown since 1987. During the last refuelling shutdown in 1986, surveillance tests were performed to ascertain full operability and efficiency of all safety related systems. Reactor and safety related systems have been continuously verified according to the Plant Technical Specifications and the procedures in the operation surveillance manual. Plant systems have been maintained in preservation status and operational safety has been assured by plant activities that adhere to applicable technical prescriptions and surveillance procedures.

An independent assessment (OSART mission 1987) reviewed the maintenance program at Caorso in the areas of organization and staff, preventive and corrective maintenance, in-service inspections, work control system, workshops and warehouse. The overall assessment concluded that the maintenance programs are effective, the maintenance facilities are improving, the maintenance staff is adequate and competent, and that the standards of industrial safety need more attention.

Currently, there are numerous maintenance activities on-going at Caorso. These activities include surveillance tests related to the technical prescriptions for cold shutdown, surveillance tests related to plant preservation, preventive maintenance, and corrective maintenance. In addition, plant modifications are also implemented.

4.4.4 South Africa

The licensee is required to assess outage schedules, which includes maintenance activities, in terms of defense-in-depth and risk. This is performed (three months) prior to each outage and submitted to the regulator as part of the request to commence the outage. Changes to the maintenance schedule during the outage are assessed using an accepted process.

4.4.5 Chinese Taipei

No is no independent assessment for particular forced-shutdown maintenance planning. It is treated as a special application. For short shutdown maintenance planning, there are only the POSs before fuel shuffling, including the POSs on the way down, then back up. For long shutdown maintenance planning, there are all the other POSs except those involving fuel shuffling.

4.5 Other Programs

4.5.1 France

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

A probabilistic target of 10E-06 per reactor-year for the CMF related to shutdown conditions was set by the Safety Authority (considering in particular that during shutdown containment integrity is not guaranteed). This objective was not really a safety goal but an indicative value.

After the first results from the LPSD PSA became available, several plant modifications and backfits were defined to reduce the risk related to the dominant sequences, mainly involving loss of RHRS during mid loop operation and the inadvertent rapid boron dilution during hot shutdown:

Loss of RHRS during mid-loop operation:

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

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

- *Improved level monitoring* (especially ultrasonic device to measure the water level in the hot leg of the reactor coolant system),

- *Reduction in the number of instances of mid-loop operation*. At the request of the safety authority, EDF has agreed to begin reducing the number of instances of mid-loop operation. With new operating procedures for purification and venting before removing the reactor vessel head only 10% of cold shutdown need to reach mid-loop operation at the early outage.

- Improved technical specifications, improved emergency operating procedures and training

- *Implementation of a vortex alarm* (vortex detection system featuring measurement of the discharge pressure of the residual heat removal system pumps),

- *Implementation of an automatic water make-up in case of loss of RHRS.* This make-up is automatically activated by a signal triggered by very low flow at the residual heat removal system pumps or a signal triggered by low pump discharge pressure. It is performed by existing equipment in the installations (charging pump aligned to the borated water tank of the safety injection system in the 900 MW reactors and low-pressure safety injection system in the 1300 MW reactors)

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

Rapid boron dilution: Due to the potentially high consequences of this type of accident sequence (reactivity accident), an immediate corrective action has been taken by EdF--implementation of an automatic suction of the Chemical and Volume Control System (CVCS) pumps from the borated Refuelling Water Storage Tank (RWST) in case of reactor trip, in order to avoid the formation of an unborated water slug.

To perform an in-depth assessment of these kinds of sequences, complementary studies have been undertaken including physical calculations, experiments and more complete identification of boron dilution sources. These studies led to complementary improvements of plant operation (operating procedures, technical specifications...)

After a reassessment of risk, other complementary modifications have been discussed between the Safety Authority and the utility EDF. The most important modifications decided on the French plants are related to the risk of cold overpressurisation during shutdown.

A threat of violent cooling of the vessel followed by a strong repressurisation of RCS has been identified in case or RHR break (Residual Heat Removal system) on French PWRs (their effect on the vessel being similar to Pressurised Thermal Shock). PSA-based analysis of such sequences have been done both on 900 and 1300 MWe series: the most important sequence corresponds to an inappropriate isolation of the RHRS in case of LOCA during cold shutdown when the primary circuit is closed, which could lead to a rapid pressure increase with a low temperature in the primary circuit, and to a risk of reactor vessel rupture. This safety problem was due especially to weaknesses in the Emergency Operating Procedures.

The Safety Authority required EDF to improve the situation and to analyse in more detail the risk of overpressurisation. Following this analysis, the utility proposals are, for all the plants, improvements of the

Emergency Operating Procedures concerned. In particular, the Emergency Procedures will require to align the circuits in order to avoid a complete RHRS isolation.

Moreover, for the 900 MWe series, a modification of the PORVs setpoint during relevant plant states, in order to reduce significantly the risk of vessel rupture, is proposed.

4.5.2 Hungary

The current estimation of core damage risk in low power and shutdown states is significantly lower than the one obtained during the initial quantification of the LPSD PSA model in 1997. The reduction of annual core damage probability in low power and shutdown states to its current value is a manifestation of safety improvements and follow-up analyses performed as a response to the initial SPSA results and recommendations.

The annual core damage probability due to accidents in low power and shutdown operating modes can, in principle, be further reduced, if effective measures are implemented against a limited number of risk factors. The following improvements have been conceptualised to achieve such a substantial risk reduction:

1. Implementation of all measures defined in a recent safety evaluation of crane operations (lifting and moving heavy equipment) in the reactor hall so that the risk due to heavy load drops can be decreased

Implementation of administrative measures to reduce the risk of loss of secondary side coolant due to inappropriate system alignment or inadequate maintenance operation (similar to the provisions prescribed for the primary circuit pressure boundary in the Infobank 8.14 Appendix of the Reactor Operating Procedures: electrical disconnection of valves, locking of valves to remain closed, tagging with effective warning to avoid inadvertent valve opening)

Development of emergency operating procedures and training of operators for their use to help responses to loss of secondary side coolant accidents in those shutdown states when the primary circuit temperature is below 255 °C

Reduction of the likelihood of pre-initiator human errors by

- collection and analysis of available plant data on pre-initiator actions in the greatest possible extent for the purpose of identifying weaknesses, developing recommendations for error rate reduction through improving training, maintenance conditions, work organisation as necessary

- improving of post-maintenance system tests so that the function of each system component is effectively tested (e.g. interlock operation triggered by low water level in a tank is tested by reducing water level in reality rather than simulating level decrease through a test signal)

Implementation of measures to (1) ensure availability of the emergency and auxiliary emergency feedwater systems below 150 °C primary temperature and, at the same time, (2) avoid cold overpressurisation of the steam generator if water is injected from either the emergency or the auxiliary emergency feedwater system.

The dominant minimal cut sets indicate that human reliability is a key factor that greatly determines core damage risk in low power and shutdown states. This finding is very similar to that of the PSA for full power operation. As a necessary permanent safety improvement activity at the Paks NPP it is considered of ultimate importance to ensure and maintain favourable conditions for safety significant human interactions (normal operational, maintenance, and emergency actions) including human related, equipment (or technology) related and organisation related aspects.

The results of the LPSD PSA study have already been used to relax the maintenance program. The application was originally aimed at the reduction of the duration of the refuelling outages through evaluating risk of plant operational states with multiple safety systems unavailable (current shutdown procedures allow for one safety system to be unavailable due to maintenance); however, the reduction of the outage duration was reached by other measures. Thus, relaxation of the maintenance tasks, avoiding overload the maintenance personnel during some plant operational states, has become the new goal of the application. The risk related to some selected configurations in each plant operational states has been assessed for this purpose.

The LPSD PSA study has become part of the living PSA tools that are used at the plant for decision-making and, as such, is being updated on an annual basis.

4.5.3 South Africa

On the basis of LPSD PRA, specifically the risk of excessive draindown, modifications (improved level monitoring) have been implemented and technical specifications for draindown and refilling phases of shutdown were developed. The risk of boron dilution accidents, including fast dilution has led to modifications and improvements to technical specifications.

With regard to the installation of high density spent fuel storage racks at Koeberg, additional requirements (modifications and procedural changes) were identified. In this regard, and with respect to other non-reactor applications listed in section 6, the PRA has proven to be a powerful tool for assessment of design weaknesses where human actions play a significant role.

The Koeberg PRA has been used in support of the periodic review of the Koeberg Nuclear Power Station that was conducted by the South African utility Eskom. Differences between Koeberg and CP1 plants in France were screened and assessed on the basis of risk and defence-in-depth.

The utility Eskom initiated a project in 1998 called the "PRA Customisation Project" with the following objectives:

- Address the findings of the periodic review in respect of compliance with risk criteria. This phase has been completed for internal events.

- Implement risk ranking of components, systems and safety issues

- Perform risk balance analyses (risk monitor). A feasibility study has commenced.

- Implement risk-informed emergency measures

4.5.4 Chinese Taipei

For the Maanshan nuclear power plant (NPP), some improvements have been made for loss of residual heat removal (RHR) during mid-loop operation:

- reducing the mid-loop operation duration, and

- improving the operating procedure (including keeping a minimum of one steam-generator available during mid-loop operation by delaying the removal of the pressuriser's PSV).

5. LPSD ANALYSES AND RESULTS

Risk analyses estimating CDF have been performed for internal events (excluding flood and fire). A few analyses have included radionuclide release and health effects and addressing internal flood and fire, and seismic events. Most of the studies considered all the various plant operating states. The analyses cover different reactor types including VVER, Westinghouse PWRs, General Electric BWR, French PWRs. The CDFs range from approximately 1E-4/ry to 1E-7/ry.

The main objectives of the studies were to:

- 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, and
- develop recommendations for improving safety in low power and shutdown modes.

The LPSD PSA studies covered analysis of all internal initiators that can potentially lead to severe core damage during various phases of a refuelling outage.

Important insights and modifications to operational practices were identified and implemented, including:

- risk is the same order of magnitude as it is for power operations,
- new improved procedures developed for loss of RHR,
- new improved procedures developed for shutdown LOCAs, and
- steam generator water inventory is now required while plant operating conditions are susceptible to loss of RHR.

In applying the methodology, the following types of scenarios were selected from a previously identified list:

- LOCAs (via reactor coolant system [RCS]}, RHR or interfaces) and power-operated relief valve (PORV) openings,
- loss of RHR, during hot or cold shutdown or mid-loop operation,
- station blackout during various plant operational states,

- reactor trips during low power operation,
- low temperature overpressurization during hot shutdown,
- boron dilutions, and
- core misloadings.

Results indicated that LPSD risk is of the same order of magnitude as power operations. Also several design and operational modifications have been implemented for safety enhancement, including:

- nitrogen accumulators for some pneumatic valve actuators,
- improved operation procedures for shutdown LOCAs, recirculation after shutdown LOCAs, station blackout during cold shutdown and refuelling, and better administrative controls,
- required availability of one AFWS train and steam generator inventory during hot and cold shutdown, required availability of two steam generators while in mid-loop operation with fuel inside the core,
- improved maintenance scheduling to avoid simultaneous unavailabilities for some equipment during some POSs, and
- improved refuelling outage scheduling to reduce mid-loop operation time with fuel inside the core.

This section identifies LPSD PRA/PSAs performed by each member and summarizes results from each analysis. Table 5-1 presents this information in tabular form.

Table 5.1 Summary of LPSD results

Country	Plant	Type	Date of PRA	Scope: Level	Internal	Flood	Fire	Seismic	Other	Operating States Modelled	Method/ Approach Used	CDF	Release Frequency	Early/ Latent Health Effects	Comments
Canada	Typical CANDU 6 Design (700 MWe)	Two loop Pressurised Heavy Water design (AECL)	1995 currently being updated	1	Y	Y	Y	N	N	Nominal power & LPS (3 main POS for the shutdown state)	ET/FT	TBD	NA	NA	
Czech Republic	Dukovany (4 units)	VVER 440/V213	1997-1998	1	Y	Y	Y	N	Refuelling Pool	All below 55% Nominal Power	Small ET Large FT	TBD	NA	NA	Preliminary results only. Quantification not completed.
France	Standard 900 MWe	PWR (France)	1990	1	Y	N	N	N	N	All	ET/FT	1.6E-5	NA	NA	Update in progress.
Germany	Neckarwestheim, Unit 2	PWR Konvoi-Type	2000	1	Y	N	N	N	N	13	S-ET/L-FT	2.5E-6 (SDS-core cooling) 3.5E-8 (SDS-deboration)	NA	NA	SDS do not consider AM measures and repair. SDS (deboration) considers the forming of unborated coolant due to evaporation-condensation mode after loss of RHR chains.
Hungary	Paks 2	VVER 440/213	1997 updated 1999	1	Y	N	N	N	N	All with one or no turbines running (24 POSs)	ET/FT with POS specific maintenance unavailabilities	3.5E-5	NA	NA	Yearly CD probability for a planned refuelling outage
Italy	SBWR	Simplified BWR(USA)	1991	1	Y	N	N	Y	N	Nominal through LPS	ET/FT	Not Provided	NA	NA	Collaboration with General Electric
Italy	AP600	Advanced PWR/W	1996	3	Y	Y	Y	N	N	Nominal through LPS	ET/FT	Not Provided	Not Provided	Not Provided	Collaboration with NRC for design certification
Japan	Typical 1100 MWe BWR	BWR/ GE BWR 5	1994 1998: Updated	1	Y	N	N	N	N	All (7 POSs)	ET/FT THERP	1.0E-7/ RY	NA	NA	
Japan	Typical 1100 MWe PWR	PWR/W 4-loop	1994 1998: Updated	1	Y	N	N	N	N	All (19 POSs)	ET/FT THERP	1.3E-6/RY	NA	NA	

Table 5-1 Summary of LPSD results (Continued)

Country	Plant	Type	Date of PRA	Scope: Level	Scope: Internal	Scope: Flood	Scope: Fire	Scope: Seismic	Scope: Other	Operating States Modelled	Method/ Approach Used	CDF	Release Frequency	Early/ Latent Health Effects	Comments
Korea	KSNP 1000MW	PWR	1994-2000	1	Y	N	N	N	N	17	ET/FT	1.73E-6/Y (mid-loop)	NA	NA	
South Africa	Koeberg	900 Mwe PWR (Fr) 2 units	1979 Updated 1981, 2001	3	Y	N	N	N	Spent fuel, Fuel Handling, Waste Treatment	11 LPSD states with 54 substates	ET/FT	1.35 E-5/Y	4.9E-7 (LPSD LERF)	Under review	
Spain	Asco 1&2	PWR/W Large Dry	1997	1	Y	Not Yet	Not Yet	N	N	Screening analysis of all POSs	Thorough qualitative and partially quantitative screening. More thorough data and human reliability analyses.	2.2E-5/Y	NA	NA	Results are average per calendar year. The variability of conditional (i.e., instantaneous) risks during different POSs is discussed in the PSA. It does not provide numerical results.
Chinese Taipei	Chinshan	BWR4/GE	1996 1998 updated	1	Y	N	N	N	N	Refuelling shutdown only	ET/FT Modified ASEP	1.5E-6/Y	NA	NA	
Chinese Taipei	Kousheng	BWR6/GE	1996 1998 updated	1	Y	N	N	N	N	Refuelling shutdown only	ET/FT Modified ASEP	3.0E-6/Y	NA	NA	
Chinese Taipei	Maanshan	PWR/W 3-Loop	1996 1998 updated	1	Y	N	N	N	N	Refuelling shutdown only	ET/FT Modified ASEP	3.2E-5/Y	NA	NA	

Table 5-1 Summary of LPSD results (Continued)

Country	Plant	Type	Date of PRA	Scope: Level	Scope: Internal	Scope: Flood	Scope: Fire	Scope: Seismic	Scope: Other	Operating States Modelled	Method/ Approach Used	CDF	Release Frequency	Early/ Latent Health Effects	Comments
USA	Grand Gulf	BWR Mark III	1994	Phase 1: 1	Y	N	N	N	N	Phase 1: All POSs (Screening)	ET/FT with average plant specific maintenance unavailabilities and simplified HRA	Not Provided	NA	NA	
				Phase 2: 3	Y	Y	Y	Y	N	Phase 2: POS 5 (Cold Shutdown) during refuelling (Detailed)	ET/FT with plant and POS specific maintenance unavailabilities and detailed human reliability analysis	2E-6/Y Internal 2.3E-8/Y Flood >1E-8/Y Fire 7.1E-8/Y 2.5E-9/Y Seismic (1993 LLNL Hazard Curve) (EPRI Hazard Curve)	Not Provided	Early fatality risk: 1.4E-8/Y Total latent cancer risk: 3.8E-3/Y	
USA	Surry	PWR/W 3-loop Subatmos-pheric	1994	Phase 1:1	Y	N	N	N	N	Phase 1: All POSs (Screening)	ET/FT with average plant specific maintenance unavailabilities and simplified human reliability analysis	Not provided	NA	NA	
				Phase 2: 3	Y	Y	Y	Y	N	Phase2: 3 POSs (Midloop Operation during refuelling and drained maintenance outages)	ET/FT with plant and POS specific maintenance unavailabilities and detailed human rel. anal.	4.9E-6/y-Int 4.8E-6/y-FL 2.5E-5/y-FI 3.5E-5/y-S*	Not Provided	Early fatality risk: 1.6E-7/Y Total latent cancer risk: 5.5E-2/Y	* using 1993 LLNL Hazard Curve; 8.6E-8/Y seismic using EPRI Hazard Curve

5.1 Czech Republic

The LP&SD PSA for NPP Dukovany is being currently updated so changes of results are expected. Previous calculations revealed the following important contributors to the risk from LP&SD operation:

- Loss of natural circulation due to primary circuit causes (overdraining, gas accumulation)
- Loss of Decay Heat Removal system due to loss of secondary circuit RHR pumps or interruption of cooling circuit
- Cold overpressurization
- Man-induced LOCAs
- Heavy load drops

5.2 France

LPSD PSA has been included in the French PSAs since the very beginning of the studies. The first results were published in 1990, by IPSN for a 900 MWe PWR and by EDF for a 1300 MWe PWR. The results of both studies indicated a high contribution of LPSD to CMF (32% for the 900 MWe and 56% for the 1300 MWe), which was the same order of magnitude as full power operation. Due to these results, the Safety Authority required plant improvements which were decided and implemented on the French plants.

Since these plant modifications, the PSAs were updated by EDF and by IPSN. The updating indicates that plant modifications are efficient for reducing the risk related to the dominant sequences. However some complementary sequences were identified (especially cold overpressurization) which led to complementary plant improvements.

The ongoing LPSD PSA updates take into account recent plant modifications and fire initiating events.

5.3 Germany

Two LPSD PSAs were performed for GRS. An analysis for a BWR-72 was finished in 1998 and an analysis for a 1300 MWe PWR of the Konvoi-type was finished at the end of 2000. It is planned to perform a PSA for a BWR-69 type to evaluate the methodology for an older BWR also.

BWR: The objective of the BWR-72 study was the development of the methodology of LPSD PSAs and the application of the methodology by some examples. The scope of the PSA is limited to Level 1 and internal initiating events. For a detailed analysis, four initiating events were defined by using a set of criteria. The four initiating events include:

- Loss of shutdown cooling via modified shutdown line
- Cold overpressurisation of the reactor pressure vessel (RPV)
- Leak at the cavity seal liner, and
- Leak at the RPV bottom head

According to international experience, these initiating events contribute to a large extent to CDF during shutdown. The analysis of these events covered most of the particular problems of LPSD PSA.

PWR: The objective of the PWR-PSA was the evaluation of the PSA methodology for LPSD. The scope of the PSA is limited to level 1 and internal initiating events (IE). A set of initiating events was derived from different sources including operation experience and international studies. In a screening process for every initiating event the POSs with the highest demands on the system functions were established. In a second step the initiating events for detailed analysis were selected by expert judgement. These are events which may lead to a loss of RHR during mid-loop operation after shutdown and events which lead to an unintended injection of unborated water into the primary circuit.

The following IEs were analyzed:

- Loss of preferred power during mid-loop, RPV closed,
- Loss of preferred power during mid-loop, RPV open,
- Loss of RHR due to faulty level lowering,
- Loss of RHR due to faulty emergency core cooling signals,
- Loss of RHR-chains, mid-loop operation, RPV closed,
- Loss of RHR-chains, mid-loop operation, RPV open,
- Leak in the RHR-system, in the containment, mid-loop operation, RPV closed,
- Leak in the RHR-system, in the containment, mid-loop operation, RPV open,
- Leak in the RHR-system, in the annulus, mid-loop operation, RPV closed,
- Leak in the RHR-system, in the annulus, mid-loop operation, RPV open,
- Different events with an unintended injection of unborated water into the primary circuit.

The system damage states from these IEs were analysed. SDS endangers the core cooling due to failures of the operational and safety systems. AM measures and repair is not considered. Compared to full power, the frequency is in the same order of magnitude and about ¼ of an overall frequency (with the conditional probability of 1 outage per year).

Concerning the unintended injection of unborated water only the measures to prevent the injection of a large amount of water were considered. A deboration of the coolant occurs also as a result of the evaporation – condensation mode after a loss of the RHR chains. The nearly boron-free condensate accumulates in the steam generator tubes an d in the pump suction line. These effects are treated together with the loss of RHR. No substantial statements can yet be made regarding the mixing behaviour up to the passage through the core nor with respect to the criticality behaviour of the core.

As a result of the PSA a lot of improvements were performed in the plant. These are mainly more specified technical specifications and an increase of the availability of systems during subcritical cold conditions.

5.4 Hungary

As an extension to the initial level 1 PSA for full power operation, a low power and shutdown PSA has been performed for the Paks Unit 2 NPP by VEIKI Institute for Electric Power Research. 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, and
- develop recommendations for improving safety in low power and shutdown modes.

The LPSD PSA study covered analysis of all internal initiators that can potentially lead to severe core damage during various phases of a refuelling outage. The LPSD PSA process was similar to a typical PSA for full power operation. However, the characteristics of the reactor type and the low power conditions required the use of new or improved methods as compared to the full power PSA. Definition of plant operational states, identification of shutdown probabilistic safety assessment (SPSA) specific initiating events, modelling of state dependent accident sequences, human reliability assessment and computerization of the large SPSA model are examples of such improvements.

An analysis of plant/reactor operating procedures, maintenance and test rules and practices as well as past records of outages at Paks led to the definition of 24 POSs. Thus, the LPSD PSA for Paks integrates 24 stand-alone PSA models each representing a given POS.

Over 400 initiating events have been identified in total for the 24 POSs. In addition to the standard steps of a PSA procedure, development of accident sequences required an evaluation of the availability of mitigating systems as defined by the actual systems configuration and the maintenance status characteristic for a given plant operational state. Also, emphasis was place on analyzing outage specific scenarios, such as inadvertent boron dilution, drop of a heavy load, pressurized thermal shock and loss of natural circulation. As a result of these efforts around 3000 accident sequences have been modelled and quantified in the LPSD PSA for Paks.

The risk profile during an outage has been produced by quantifying all core damage sequences within the 24 POSs. Sensitivity analyses have been performed to identify dominant risk contributors. The LPSD PSA model, the underlying component and human reliability data are managed, and the quantified risk measures are stored in a single project within the Risk Spectrum PSA Code Package.

Taking into account the current status of plant systems, operating conditions, characteristics of emergency responses, and all plant modifications made prior to the end of 1998 the total annual core damage probability due to potential accidents in low power and shutdown operating modes of unit 2 of the Paks NPP is 3.53E-05. The results show that risk during an outage is dominated by those plant operational states when the reactor vessel is open for refuelling and the water level is low in the reactor. Plant states which follow boration and in which primary temperature is above 150 °C are the next in terms of risk significance. Other important POSs are characterised by open reactor vessel with refuelling water level, and POSs in which heat is removed from the reactor by means of natural circulation and the primary circuit is pressurised. This kind of distribution is mostly explained by two factors: the difference between the average lengths of the POSs and the reduction in automatic plant responses in some plant operational states.

According to the dominant minimal cut sets primary side LOCAs and losses of secondary coolant are the most significant accidental processes that dominate probability of core damage during a refuelling outage of the Paks NPP. In the dominant cases a primary side LOCA can be induced mostly by inappropriate maintenance actions (e.g. inadvertent opening of valves, dismantling of piping components) rather than hardware failures. Inadequate system alignments have been identified as the main causes of loss of secondary coolant. With respect to low power and shutdown specific accident sequences termination of natural circulation should be mentioned first. The principal cause to termination of natural circulation (excluding primary LOCA) is loss of secondary cooling dominated by inappropriate system alignment, drop of a heavy load in the turbine hall causing pipeline break in the secondary circuit, and loss of secondary decay heat removal pumps. Drop of a heavy load during certain refuelling crane operations can endanger core cooling, which is also a risk significant fault. In comparison with these potential core damage processes, other low power and shutdown specific scenarios (e.g. inadvertent dilution, cold overpressurization of reactor vessel) are less important due to some existing defenses.

5.5 South Africa

Table 5.2 Contribution to CDF

Initiating Event Group	At power	Shutdown (1)	RHR condition (2)	Total LPSD	Total
Secondary Line Break Accidents	8.14E-08	0.00E+00	0.00E+00	0.00E+00	8.14E-08
Secondary System Transients	4.82E-06	2.32E-07	0.00E+00	2.32E-07	5.05E-06
Loss of Compressed Air	2.82E-08	0.00E+00	0.00E+00	0.00E+00	2.82E-08
Primary System Transients	2.99E-06	5.95E-08	6.26E-08	1.22E-07	3.11E-06
Loss of Residual Heat Removal	0.00E+00	0.00E+00	4.70E-06	4.70E-06	4.70E-06
Loss of Ventilation	5.63E-08	0.00E+00	0.00E+00	0.00E+00	5.63E-08
Loss of Off-site Power	6.99E-06	1.07E-06	1.79E-06	2.86E-06	9.85E-06
Loss os Switchboards	5.79E-07	1.88E-08	1.14E-06	1.16E-06	1.74E-06
SGTR	2.35E-07	6.89E-08	0.00E+00	6.89E-08	3.04E-07
LOCAs	1.62E-06	4.07E-08	3.26E-06	3.31E-06	4.92E6
Loss of Component Cooling	4.44E-07	1.57E-08	9.92E-07	1.01E-06	1.45E-06
Total	1.78E-05	1.51E-06	1.20E-05	1.35E-05	3.13E-05

(1) Shutdown - Reactor shutdown with Residual Heat Removal System valved out (POS B5 -B8, C, K and L1 - L2) (Table 4-1)
(2) Reactor Shutdown with RHR conditioned (POS D, E, F1 - F2, I and J) (Table 4-1).

Table 5-3 Dominant Minimal Cutsets

Initiating event	Sequence	Frequency
Loss of RHR in POS F1-F6, H1-H4, I1-I2	Loss of RRA in maintenance / refuelling shutdown, operator fails to establish make-up from PTR and fails to use alternative make-up to the RCP	4.63E-06
Loss of RRI in POS F1-F6, H1-H4, I1-I2	Loss of RRI during maintenance / refuelling shutdown, no RRI recovery in 24 hours.	9.03E-07
Maintenance induced LOCA (intermediate) in POS E1-E2, J1-J2	Intermediate K-LOCA during draindown, Operator fails to establish manual LHSI	8.47E-07
Maintenance induced LOCA (small) in POS E1-E2, J1-J2	Small K-LOCA during draindown, Operator fails to establish manual LHSI	8.47E-07
Recoverable flow diversion via RRA in POS F1-F6, H1-H4, I1-I2	Recoverable flow diversion (H-LOCA) dominated by excessive draindown in RRA,. Operator fails to recover from this event (series of dependent operator errors).	6.23E-07
Maintenance induced LOCA (large) in POS E1-E2, J1-J2	Large K-LOCA during draindown, Operator fails to establish manual LHSI	4.24E-07
Loss of Offsite Power in POS F1-F6, H1-H4, I1-I2	Loss of Offsite Power when LGE switchboard in maintenance and LHQ fails to start. Accident transfers to a Station Blackout. Operator fails to establish cooling.	3.73E-07
Intermediate LOCA in RRA in POS E1-E2, J1-J2	Intermediate RRA LOCA in draindown, followed by dependent HEP of operator failing to isolate RRA from RCP, and operator failing to manually establish SI	3.16E-07
Loss of RRI with the reactor critical	Loss of RRI during power operation, reactor trip, RCP pumps tripped, Operator fails to establish cooldown and seal protection, RRI not recovered within 24 hours.	2.91E-07
Loss of Offsite Power (reactor above 15%)	House load initially successful but a failure to restore the grid (400kV) fails, and Acacia Line in maintenance. LHP and LHQ diesel generators fail while running, resulting in a SBO. RCP isolated, AG003PO startup, battery load shedding successful and also battery preservation successful. Crash cooling (ECA 0.0, step 16) successful and power restored in 17 hrs, but the RCP pump seals failed, allowing sufficient inventory loss to result in core damage when power is restored.	2.82E-07
Loss of LCB in POS F1-F6, H1-H4, I1-I2	Loss of LCB during RRA conditions, vessel head removed. LHA energized, operator fails to re-establish RRA cooling, resulting in a loss of RRA. Operator fails to establish PTR backup and longer term RCP cooling.	2.73E-07
Loss of LCA in POS F1-F6, H1-H4, I1-I2	Loss of LCA during RRA conditions, vessel head removed. LHA energized, operator fails to re-establish RRA cooling, resulting in a loss of RRA. Operator fails to establish PTR backup and longer term RCP cooling.	2.73E-07

5.6 Spain

According to the Spanish Integrated Program on PSA, each of the seven Spanish nuclear power plants will have a PSA, the first having begun in 1983. Plant specific PSA requirements are time and scope phased. All PSAs shall be revised to get to a common final scope for the seven plants. The final scope will consider all internal and external initiators for all modes of operation and will estimate characteristics and frequencies of fission product releases.

Risk from initiators during low power and shutdown operations was not included in the scope of the PSAs up to the sixth PSA, required by the CSN in 1990 for the Vandellos 2 nuclear power plant, a Westinghouse 3-loop PWR.

For Vandellos 2, a rough qualitative screening analysis was performed at the beginning of that LPSD PSA to select only two types of scenarios (defined as groups of initiating events in a plant operational state) for further evaluation. These two scenarios included:

- LOCAs in cold shutdown, and
- RHR losses during mid-loop operations.

Previous vendor studies were used to support this selection. Additionally, a very simplistic method, also proposed by the vendor and incongruous with the rest of the PSA, was used for human reliability analysis. CSN review objected to these aspects of the methodology, but work on the PSA continued and it was finally submitted to the CSN in 1995, together with the results from the full-power Level 2 analysis.

Important insights and modifications to operational practices were identified and implemented, including:

- risk is the same order of magnitude as it is for power operations,
- new improved procedures developed for loss of RHR,
- new improved procedures developed for shutdown LOCAs, and
- steam generator water inventory is now required while plant operating conditions are susceptible to loss of RHR.

The PSA for LPSD operations at the Asco nuclear power plant, another Westinghouse 3-loop PWR, started in 1994 after completing the analysis for power operation. This portion of the Asco PSA was developed, in principle, according to the IAEA LPSD PSA guide, which includes a more rigorous screening process. This LPSD PSA is now considered as a “pilot” in Spain for this subject, along with another soon to start LPSD PSA for the Garoña nuclear power plant, a General Electric BWR Mark I.

After a full CSN review, now ongoing for Asco, a decision will be made concerning the LPSD PSA methodology to be used for the other four plants during future revisions to their PSAs.

Applying the methodology used to perform the Asco LPSD PSA, the following types of scenarios were selected from a previously identified list:

- LOCAs (via RCS, RHR or interfaces) and PORV openings,
- loss of RHR, during hot or cold shutdown or mid-loop operation,

- station blackout during various plant operational states,
- reactor trips during low power operation,
- low temperature overpressurization during hot shutdown,
- boron dilutions, and
- core misloadings.

HRA methods were improved to be more state-of-the-art. Qualitative concepts from the ATHEANA methodology were applied, together with Success Likelihood Index, or multi-attribute, techniques.

The risk assessed was again in the same order of magnitude as for power operations. Also several design and operational modifications have been implemented for safety enhancement, including:

- nitrogen accumulators for some pneumatic valve actuators,
- improved operation procedures for shutdown LOCAs, recirculation after shutdown LOCAs, station blackout during cold shutdown and refuelling, and better administrative controls,
- required availability of one AFWS train and steam generator inventory during hot and cold shutdown,
- required availability of two steam generators while in mid-loop operation with fuel inside the core,
- improved maintenance scheduling to avoid simultaneous unavailabilities for some equipment during some POSs, and
- improved refuelling outage scheduling to reduce mid-loop operation time with fuel inside the core.

6. SPECIFIC LPSD TECHNICAL AREAS

This section presents information on various specific technical areas from the LPSD PRA/PSA performed by the member countries.

6.1 Thermal Hydraulics

Simplified, thermal hydraulic calculations have been employed to estimate the time to reach potential core damage, and to determine the effects of boron dilution. Three alternate cooling methods were investigated. Gravity feed was analyzed by using the MELCOR computer code for a few different decay-power levels. Feed-and-bleed by gravity and by steam generation were analyzed for a large range of initial operating conditions using simplified models.

Thermal hydraulics studies pertaining to homogeneous dilution of boron and primary breaks have also been performed. They provide supporting information for establishing success criteria. Detailed thermal hydraulic calculations, some using RELAP/Mod2, were also conducted for mid-loop operation to verify the appropriateness of the success criteria adopted.

Some planned activities include performing thermal hydraulic calculations to estimate the timing associated with accident sequences for typical advanced BWR (ABWR) plants. For PWR plants, thermal hydraulic calculations using more realistic decay heat conditions and calculations to confirm the effect of passive mitigating methods, such as injection from the refuelling water storage tank (RWST) using static water head between RWST and RCS and RCS cooling by steam generator (SG) (i.e., blow through main steam relief valve), are planned.

Identified needs include: Refinement and adaptation through appropriate validation and verification of thermal-hydraulic codes against available experimental separate and integral effect tests.

6.1.1 Canada

6.1.1.1 Currently Available Information

Information is currently available confirming the adequacy of the shutdown cooling (SDC) system for all shutdown configurations. The CANDU shutdown cooling system is capable of decay heat removal at full heat transport system pressure and temperature. Calculations have been performed for the time span from the failure of the SDC system to fuel failure when the heat transport system is cold and drained to the headers. The SDC impairments considered in these calculations include loss of circulation and loss of service water. Similar calculations have also been performed for the case when the heat transport system is full. These calculations have provided a realistic estimate of operator action time to initiate back-up heat sink (e.g. steam generators).

6.1.1.2 Ongoing Activities

None.

6.1.1.3 Planned Activities

None.

6.1.1.4 Future Needs

Confirmatory analysis for the intermittent buoyancy induced flow IBIF in the heat transport circuit during drained state and reflux condensation in the steam generators.

6.1.2 Czech Republic

6.1.2.1 Currently Available Information

Various sorts of thermal hydraulic analyses have been used in LP&SD PSA for NPP Dukovany to determine success criteria, time to reach potential core damage and other applicable outputs. The sources of such analyses are:

Thermal hydraulic analyses applicable for full power operation. Two kinds of such analyses are available:

- generic VVER analyses from IAEA-TECDOC-719 for LOCA,
- plant specific analyses performed for various purposes (SAR, support for EOPs, full-power PSA support)

Thermal hydraulics analyses performed specifically for NPP Dukovany LP&SD operation. They include loss of secondary circuit heat removal, loss of natural circulation and LOCAs during pressure tests.

Thermal hydraulics analyses performed for shutdown conditions of VVER-440/213 plants in Slovakia (NPP Bohunice V2) and Hungary (NPP Paks) in the frame of PHARE programs. They cover:

- LOCAs and RCS draindown, loss of secondary circuit heat removal, loss of natural circulation.

6.1.2.2 Ongoing Activities

NPP Dukovany specific analyses are being conducted on loss of natural circulation for selected boundary conditions.

6.1.2.3 Planned Activities

None.

6.1.24 Future Needs

The time extended thermal hydraulic analyses of LOCA into refuelling pool as well as realistic analyses of gas behaviour in primary circuit are the most needed analyses for LP&SD PSA of NPP Dukovany.

6.1.3 France

6.1.3.1 Currently Available Information

Currently available information includes results from thermal hydraulics studies related to: homogeneous dilution of boron, loss of heat removal, and primary breaks (i.e., LOCAs).

The loss of heat removal and the primary breaks calculations were used to:

a) Estimate the maximum time available to activate safety injection for break sizes that do not activate the automatic water makeup.

b) Assess the efficiency of the countermeasures (water make-up by CVCS, by the automatic make-up system and by the gravity make-up)

The calculations were performed for different configurations of the primary circuit (closed, small and large opening). The computer code used was the CATHARE code.

6.1.3.2 Ongoing Activities

Complementary thermal hydraulics calculations are ongoing in order to define the physical transients leading to cold overpressurisation of the primary circuit.

6.1.3.3 Planned Activities

Calculations will be carried out for some dominant sequences of the updated PSAs (especially boron dilutions).

6.1.3.4 Future Needs

None.

6.1.4 Germany

6.1.4.1 Currently Available Information

For both reference plants particular models for the thermal hydraulic analysis during LPSD were performed. The analyses were performed to establish the minimal success criteria.

Concerning the BWR, thermal-hydraulics calculations were performed to analyze the plant behaviour after a loss of heat removal during cold shutdown. Since the coolant inventory is still high at a level comparable to power operation, the time to core uncovery is quite long (approximately 7 hours).

For the PWR, calculations were performed at mid-loop operation while the RPV is closed (steam generator available). Special attention was paid to phenomenon of condensation of nearly boron-free coolant in the steam generator after a loss of the RHR chains. After about 4 h approximately 15 Mg of deborated coolant have accumulated in the steam generator and about 5 Mg in the pump suction line. This can be prevented, if the primary circuit will be flooded within the first two hours after the loss of the RHR chains. At the moment no substantial statements can be made regarding the mixing behaviour on the way through the core nor with respect to the criticality behaviour of the core.

6.1.4.2 Ongoing Activities

None.

6.1.4.3 Planned Activities

It is planned to perform thermal hydraulic calculations for a BWR-69 type.

6.1.4.4 Future Needs

A sound calculation with a verified code is needed to predict the mixing of the unborated coolant after the restart of the natural circulation. As a result of the calculation the distribution of the boron acid concentration below the core is needed to calculate the recriticality.

6.1.5 Hungary

6.1.5.1 Currently Available Information

Currently available information includes the following:

- results from approximately 100 thermal hydraulic calculations performed by KFKI/AEKI (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

- calculations performed by KFKI/AEKI (Hungary) in 1995 with ATHLET and RELAP5/mod3.1 codes to clarify some success criteria for LOCA event trees related to shutdown states of Paks NPP Unit 2 when (1) cool down is in progress, but hydro-accumulators are isolated from the primary circuit, and (2) the reactor is open, heat removal is ensured by the secondary side decay heat removal system, and there is natural circulation in the primary circuit

- calculations performed 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) aimed at the systematic identification and evaluation of boron dilution faults.

- results from approximately 130 thermal hydraulic calculations performed by RIMAN (Hungary) up to 1998 with the CRUISE code for the purpose of defining success criteria for the LPSD PSA assuming primary LOCAs of different sizes occurring in different POSs of natural circulation

6.1.5.2 Ongoing Activities

None.

6.1.5.3 Planned Activities

None.

6.1.5.4 Future Needs

None.

6.1.6 Japan

6.1.6.1 Currently Available Information

For a typical BWR-5 type plant, estimates of the time required to exceed a peak cladding temperature (PCT) limit of 1200 °C for each POS are available. These estimates were derived from simple hand calculations using a core decay heat curve, heat exchanger capacity, initial coolant inventory, etc.

For the 4-loop PWR plant, information from two types of thermal hydraulic calculations is available. The first type includes thermal hydraulic calculations used to set success criteria for mid-loop and non mid-loop operations. Simplified thermal hydraulic calculations were performed for mid-loop operation, and RELAP5/MOD2 calculations were made for non-mid-loop operational states. The simplified code has a water-steam-air piston model for thermal hydraulic calculation, which is quite similar to that of NUREG/CR-5855. This code can calculate reactor coolant system behaviour during mid-loop operation with and without steam generator nozzle dams installed until core uncovery, taking into account gravity and forced primary feed-and-bleed as well as steam generator feed-and-bleed. For example, the allowable time to core uncovery given the loss of RHR system during mid-loop operation with all steam generator nozzle dams in place is estimated to be 1.2 hours, given no mitigation system. Core damage is assumed to occur when the core uncovers, which differs from the definition used in the rated power PSA where a peak cladding temperature of 1200 °C is used as the criterion for core damage. Success criteria for the non mid-loop states (i.e., cold shutdown states with a water-filled reactor coolant system) were estimated by RELAP5/MOD2 calculations, taking into account available stand-by mitigation systems. For example, one AFW pump and one main steam relief valve are required during RCS cooling down state (POS3) at the loss of RHR system.

The second type of thermal hydraulic calculations provided supporting information used to set the success criteria described above.

These included:

- a boron dilution analysis by three dimensional fluid-dynamic code alpha-FLOW, and
- additional RELAP5/MOD2 calculations for mid-loop operations.

In the LPSD PSA, rapid reactivity insertion events during the reactor start-up operational state are neglected because the probability of the event leading to core damage is small. The boron dilution analysis was conducted to show that the reactivity insertion due to erroneous boron dilution would not occur in a short period after loss of off-site power under the realistic plant conditions. After verifying and validating the alpha-FLOW code for this phenomenon through the CREARE-1/5 experimental analyses, which were pressurized thermal shock experiments, the boron dilution phenomena occurring as a result of the loss of off-site power were analyzed by the alpha-FLOW code. Results indicated that injected non-borated water would be well-mixed within the reactor coolant system piping as well as in the inlet plenum of the reactor vessel; thereby producing no substantial reactivity insertion.

In addition, detailed thermal hydraulic RELAP5/MOD2 calculations were conducted for mid-loop operation to verify the appropriateness of the success criteria adopted. Two specific cases, with no mitigation system available, were analyzed for the loss of RHR at mid-loop operation. These included:

- steam generator nozzle dams installed, and
- steam generator nozzle dams not installed.

The times to core uncovery and core damage were estimated to be 6.6 hours and 8.0 hours for the case with no steam generator nozzle dams installed, and 1.8 hours and 2.5 hours for the case where the steam generator nozzle dams are installed.

6.1.6.2 Ongoing Activities

On-going activities include estimating the time for typical BWR-4 type plants using the same method that was used for the BWR-5 plant. Further, supporting thermal hydraulic calculations for the 4-loop PWR plant are being conducted. Detailed thermal hydraulic calculations by RELAP5/MOD2 are on-going for mid-loop operation under the condition that primary system feed and bleed by gravity and steam generator feed and bleed are available. The objectives of the calculations are to identify the allowable timing for actuation of mitigation systems and to identify plant configurations required for mitigation systems. Finally, activities are underway to set the success criteria for 2-loop PWR plants through simplified calculations.

6.1.6.3 Planned Activities

Planned activities include performing thermal hydraulic calculations to estimate the timing associated with accident sequences for typical ABWR plants. For PWR plants, thermal hydraulic calculations using more realistic decay heat conditions and calculations to confirm the effect of passive mitigating methods, such as injection from RWST using static water head between RWST and RCS and RCS-cooling by SG-blow through main steam relief valve, are planned. In addition, JAERI plans to conduct some experiments for the above phenomena during January 1999.

6.1.6.4 Future Needs

None.

6.1.7 Korea

6.1.7.1 Currently Available Information

For the Yonggwang 5 and 6 NPP, four thermal hydraulic calculations were performed to determine need of containment spray recirculation cooling as ultimate heat sink using RERLP5/MOD3. There are two types calculations, one is base case which is simulate reactor phenomena after loss of SCS (Shutdown Cooling System). The other is calculation of reactor coolant system behaviour with forced primary feed-and-bleed. Each type of calculation results are available on POS 3 and 4A. The results indicated that CSS cooling is not required for mitigate the loss of SCS event in the POS 4A.

6.1.7.2 Ongoing Activities

On-going activities involve performing base calculation on all shutdown POS excluding POS 7, 8, and 9 at which the refuelling cavity is full.

6.1.7.3 Planned Activities

Planned activities include performing thermal hydraulic calculations to estimate the timing associated with accident sequences for YGN 5 and 6.

Future Needs

None.

6.1.8 South Africa

6.1.8.1 Currently Available Information

A plant-specific model of the Koeberg plant was run on MAAP-4 to analyse the initiating events:

- Large LOCA (Intermediate shutdown, cold shutdown and midloop operation)
- Small LOCA (Intermediate shutdown, cold shutdown and midloop operation)
- Nozzle dam failure (midloop operation)

All systems assumed failed to determine worst case time to boiling/core uncovery, and minimum makeup requirements.

The purpose of the analyses was to determine the timing of boiling/core uncovery for human reliability analysis, and to determination of minimum makeup requirements.

6.1.8.2 Ongoing Activities

None.

6.1.8.3 Planned Activities

None.

6.1.8.4 Future Needs

None.

6.1.9 Spain

6.1.9.1 Currently Available Information

In the Vandellos 2 LPSD PSA, the thermal-hydraulic considerations are based on two previous generic Westinghouse reports (1988 and 1991), in principle adapted to Vandellos 2 NPP singularities, as included in another Westinghouse technical report (1992). The same generic reports are referenced by the Asco LPSD PSA.

In 1996, a CSN-sponsored project carried out by a Spanish university, performed some thermal-hydraulic calculations by means of the RELAP5 code for some scenarios analyzed in the Vandellos 2 LPSD PSA. Those scenarios included two LOCAs (medium and small) in cold shutdown operation and a SBO in mid-loop operation. According to the calculations, some success criteria for the Vandellos 2 LPSD PSA (related to gravity driven injection of the RWST into the reactor core) should be changed. This may not be the case of Asco, due to a different physical layout of the RWST in the plant.

6.1.9.2 Ongoing Activities

As a continuation of the previous CSN project, a new project is ongoing (1998-99) at the same Spanish university to try to overcome some difficulties that were identified during the use of the RELAP5 to model certain thermal-hydraulic characteristics specific to shutdown operation.

6.1.9.3 Planned Activities

As a part of the new project, an identification of new scenarios to be analyzed by means of the same or improved tools is planned. The ongoing CSN review of the Asco LPSD PSA will identify the new scenarios and new calculations will be performed for the Asco LPSD PSA. Plans also call for a LPSD PSA of the Garoña facility, a BWR Mark I plant.

6.1.9.4 Future Needs

More realistic thermal-hydraulic calculations are needed for low power and, above all, shutdown transient or accident scenarios to minimize the potential conservatism being introduced into current LPSD PSAs for some sequences. This lack of knowledge may also lead to optimistic models in other cases. Both cases are negative, because, on one hand risk might be underestimated and, on the other hand, a LPSD risk overestimation, due to conservatism, could lead to a wrong counterbalance with full power risk and, therefore, to a wrong PSA application, if this risk comparison is used.

Refinement and adaptation through appropriate validation and verification of thermal-hydraulic codes against available experimental separate and integral effect tests are needed.

6.1.10 Chinese Taipei

6.1.10.1 Currently Available Information

A “simple boiled-off model” was used for thermal hydraulic calculations. For Maanshan NPP, a further project is in progress for more detailed calculation by another group (not PRA group).

6.1.10.2 Ongoing Activities

For the Maanshan NPP, a project for more detailed calculation by another group (not PRA group) was in progress.

6.1.10.3 Planned Activities

None.

6.1.10.4 Future Needs

None.

6.1.11 United States

6.1.11.1 Currently Available Information

Information is currently available from the Phase 1 and 2 LPSD PRAs for Grand Gulf and Surry. For the Phase 1 Grand Gulf analysis, simplified thermal hydraulic calculations were performed for selected plant operational states and thermal hydraulic information from the NUREG-1150 analyses were used for the remaining plant operational states. In the Phase 2 analysis, several MELCOR, Version 1.8.0, computer code calculations were performed to support the development and quantification of the level 1 PRA models. For these calculations, the parameters of interest included the time to:

- reach various pressure setpoints,
- top-of-the-active-fuel (i.e., core uncovery), and
- core heat and clad failure.

The scenarios included:

- an open main steam isolation valve or valves,
- low pressure boiloff;
- high pressure boiloff with a closed reactor pressure vessel head vent,
- high pressure boiloff with an open reactor pressure vessel head vent,
- large break LOCA, and
- station blackouts.

NUREG/CR 6143, Vol. 2, Part 2, Appendix F summarizes calculations performed to support the success criteria and event tree development for the Phase 2 analysis.

For the Phase 1 Surry analysis, simplified thermal hydraulic calculations were performed for selected plant operational states and thermal hydraulic information from the NUREG-1150 analyses was used for the remaining plant operational states. In the Phase 2 analysis, more detailed, but still simplified, thermal hydraulic calculations were used to estimate the time that the plant might reach potential core damage, and to determine the effects of boron dilution. The greatest effort was expended in modelling the thermal-hydraulic behaviour in the reactor coolant system during mid-loop operation after the loss of RHR. Three alternate cooling methods, i.e., gravity feed from the RWST, reflux cooling, and feed-and-bleed were investigated. Gravity feed was analyzed by using the MELCOR computer code for a few different decay-power levels between 1 day to 29 days after reactor shutdown. Feed-and-bleed operation was analyzed for a large range of initial operating conditions using simplified models developed for this low power and shutdown program. Details of the calculations are discussed in NUREG/CR-6144.

6.1.11.2 Ongoing Activities

None.

6.1.11.3 Planned Activities

None.

6.1.11.4 Future Needs

While both studies indicated the need for improved thermal hydraulic analysis capability, the Surry study identified alternate cooling methods that could prevent core damage following the loss of RHR. These alternate cooling methods may require additional detailed feasibility studies. Secondly, the mixing of boron with water should be experimentally verified to confirm the preliminary findings.

6.2 Core Physics

One of primary areas of concern appears to be risk of reactivity accidents caused by either accidents outside the vessel or inherent in-vessel rapid boron dilution. Another problem area identified is associated with the risk of cold overpressurisation. Still another is brittle fracture of the reactor vessel. It has been found to be caused, in the case of a RHR system break, by stresses introduced during two time frames. The first time frame occurs when the primary water is cooled due to the depressurization resulting from the break and the cold water injected by safety injection. The second occurs when pressure increases after the break is isolated. The combinations of these two events may introduce excessive thermal stresses on the reactor vessel during the cooling and mechanical stresses during the repressurisation.

Uncertainty exists in some scenarios about the possibility of insufficient mixing between non-borated and borated water and the formation of a slug of non-borated water that may move into the core and cause prompt criticality. Appropriate computerized flow dynamics (CFD) codes will be used for modelling mixing phenomena and will be adapted to the specific scenarios of fast boron dilutions due to slug formation.

In the U.S., three criticality events were identified:

- Rod withdrawal error
- Refuelling accident
- Instability event

6.2.1 Canada

6.2.1.1 Currently Available Information

Reactivity accidents need not be analyzed in detail for the CANDU LPSD PSA. Unique CANDU design features such as the use of two independent and diverse shutdown systems and guaranteed shutdown state provisions provide assurance that criticality accidents can be excluded from the analysis. The boron dilution type of events leading to criticality does not apply to CANDU design.

6.2.1.2 Ongoing Activities

None.

6.2.1.3 Planned Activities

None.

6.2.1.4 Future Needs

None.

6.2.2 Czech Republic

6.2.2.1 Currently Available Information

Two sorts of thermal hydraulic analyses have been used in LP&SD PSA for NPP Dukovany to determine effects of reactivity accidents and other applicable outputs. The sources of such analyses are:

- Plant specific analyses performed for SAR
- Boron dilution analyses performed for shutdown conditions of VVER-440/213 plants in Slovakia (NPP Bohunice V2) and Hungary (NPP Paks) in the frame of PHARE programs.

6.2.2.2 Ongoing Activities

None.

6.2.2.3 Planned Activities

None.

Future Needs

The boron dilution analyses following primary-to-secondary breaks are the most needed analyses for LP&SD PSA of NPP Dukovany.

6.2.3 France

6.2.3.1 Currently Available Information

a) Boron dilution: Studies of the risk of heterogeneous boron dilution have been carried out (neutronic calculations and experiment for study of water mixing). The objective of the neutronics calculations was to evaluate the critical size if an unborated water slug which could lead to a reactivity accident when entering into the core. The aim of the experiment was the study of the mixing of an unborated water slug with the primary water before reaching the core.

b) Cold over pressurization: In the case of a residual heat removal system break, stresses are introduced during two time frames. The first occurs when the primary water is cooled due to the depressurization resulting from the break and the cold water injected by safety injection. The second occurs when pressure increases after the break is isolated. The combination of these two events introduces two kinds of stresses: (1) thermal stresses on the reactor vessel during the cooling and (2) mechanical stresses during the repressurisation. These can provoke brittle fracture of the reactor vessel. The core damage sequences, in the first stage, has been quantified with conservative assumptions.

6.2.3.2 Ongoing Activities

In addition to the thermal hydraulics calculations (see Section 6.1.3.2), mechanical studies are in progress for some representative transients of cold overpressurization.

6.2.3.3 Planned Activities

Examination of these issues should be pursued to demonstrate that the safety of the reactor vessel (i.e., the mechanical strength of the reactor vessel that depends notably on the NDTT and the size of the defects) is sufficient.

6.2.3.4 Future Needs

Boron dilution: studies of the risk of criticality due to natural primary circulation blockage and the formation of cold water slugs (i.e., boron dilution) are needed.

Cold over-pressurization: studies are needed on structural mechanics issues. The objective is to appreciate in more realistic conditions the importance of the risk related to cold over-pressurization transients.

6.2.4 Germany

6.2.4.1 Currently Available Information

Information is available from studies of scenarios potentially leading to dilution accidents. External dilutions (e.g. umborated water injection by CVCS) and internal dilution during reflux-condenser mode after loss of heat removal during mid-loop operation have been identified. Evaluation of minimum slug volumes leading to prompt criticality has been performed. Generic analysis of plant behaviour during dilution accidents were performed by coupled thermal hydraulic and 3D neutronics codes.

6.2.4.2 Ongoing Activities

The application of coupled of thermal-hydraulics codes with 3D neutronics models for realistic analysis of boron dilution transients is ongoing. Calculations by such codes still need assumptions about the degree of mixing. Therefore, qualification of CFD codes for mixing phenomena in the primary circuit is ongoing also.

6.2.4.3 Planned Activities

An analysis of specific accident conditions by using a coupled thermal-hydraulic and 3D neutronic code is still planned.

6.2.4.4 Future Needs

A consistent methodology based on experimental results and validated analytical methods are needed. This requires (e.g., experimental investigations for re-establishing natural circulation after stagnant flow conditions) experimental investigations of flow mixing in the primary circuit and reactor vessel for qualification of CFD codes and modelling of mixing processes and the implementation of mixing models in coupled thermal-hydraulic and 3D neutronic codes for realistic analysis of boron dilution accident conditions.

6.2.5 Hungary

6.2.5.1 Currently Available Information

Currently available information includes the following:

- results from the 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 in the ´90s, and

- calculations performed by AEA Technology (UK) with the CFX-4 code, and by KFKI/AEKI (Hungary) in 1998 with ATHLET, coupled with 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) aimed 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.

6.2.5.2 Ongoing Activities

None.

6.2.5.3 Planned Activities

None.

6.2.5.4 Future Needs

None.

6.2.6 Spain

6.2.6.1 Currently Available Information

Reactivity accidents are not considered in the current version of the Vandellos 2 LPSD PSA. CSN review objected to this. Boron dilution events are considered in the Asco LPSD PSA; however, they are not identified as significant risk contributors for either slow dilutions or fast dilutions.

Uncertainty exists in some scenarios about the possibility of insufficient mixing between non-borated and borated water and the formation of a slug of non-borated water that may move into the core and cause prompt criticality.

6.2.6.2 Ongoing Activities

Spain (i.e., CSN) is making a decision on participation in a CSNI PWG2 International Standard Problem (ISP-43) on the mixing issue. In any case, a follow up of the issue is being pursued.

6.2.6.3 Planned Activities

Appropriate CFD codes will be used for modelling mixing phenomena and adapted to the specific scenarios of fast boron dilutions due to slug formation.

6.2.6.4 Future Needs

Mixing experiments may be needed to test and verify the correct modelling of the mixing phenomena.

6.2.7 United States

6.2.8.1 Currently Available Information

In the Grand Gulf LPSD PRA, three criticality events were identified for consideration. They include:

- rod withdrawal error,
- refuelling accident (rod or fuel misposition), and
- instability event.

The rod withdrawal accident is a design basis accident for full power operation. Thus, for significant fuel damage to occur, other failures (such as control rod pattern errors) must exist. The analysis concluded that the likelihood of significant fuel damage from a rod drop event at full power is □1.0E-8/Y. Thus, the event was not developed further in the LPSD PRA.

The refuelling accident can be of two types. The first type involves incorrect placement of a fuel bundle in the core during refuelling. This is a design basis accident and does not lead to unacceptable consequences. The second type is associated with local loading of fuel when more than one control blade is removed. Grand Gulf procedures suspend all fuel loading operations when control blades are out. Based on this procedural control and prior estimates of the frequency (approximately 1E-7/Y) without this control, this event was screened from further consideration.

The flow instability event refers to operation of the core with a local high power-to-mass-flow ratio. The concern is that with the instability, should a design basis event take place, the high power regions of the core can be damaged due to the occurrence of phenomena such as critical heat flux or excessive fuel enthalpy. Prior studies indicated early fuel damage is not a concern for events occurring given operation with instabilities. While no detailed studies were performed for the LPSD study, instabilities should be less likely than at full power; thus, these events were screened from further analysis.

In the Surry LPSD PRA, several categories of potential reactivity events during shutdown were examined. These include:

- addition of diluted accumulator water,
- addition of diluted RWST water,

- boron dilution due to maintenance problems,
- uncontrolled boron dilution from CVCS,
- boron dilution via the RHR,
- startup of RCP after improper boron dilution,
- rod ejection accident,
- misloading of fuel assemblies, and
- uncontrolled bank withdrawal.

Prior analyses determined that the first, second, fourth, sixth, and eighth categories may be relatively risk significant, thereby requiring more detailed analyses. Thus, the Surry LPSD PRA focused mainly on the accident sequences belonging to these five categories.

Applying the Surry refuelling outage frequency of 6.0E-1/Y, the expected frequency of inadvertent criticality was calculated to be 2.2E-6/Y.

6.2.8.2 Ongoing Activities

None.

6.2.8.3 Planned Activities

None.

6.2.8.4 Future Needs

Issues associated with mixing of boron should be investigated in more detail to determine whether slugs of "pure" water can occur and be transported through the core.

6.3 Initiating Event Frequencies

The starting point for determining LPSD initiating events is usually the full-power list. This is then supplemented by those categories judged unique to LPSD. Estimating LPSD initiating event frequencies involves identifying the unique categories of events that have the potential to result in an initiating event for a plant operating state (POS) and then quantifying each unique category. These categories include internal causes and others, such as the erroneous change of the operational loop, erroneous draining of the operational loop, maintenance activities, erroneous system alignment, heavy load drop, and other interactions. The frequency for each category of event is quantified, and these frequencies are then combined to produce POS-specific initiating event frequencies for each initiating event.

6.3.1 Canada

6.3.1.1 Currently Available Information

Table 6-1 identifies the events considered in the CANDU shutdown state PSA. The frequencies given in Table 6-1 are based on CANDU operating experience and/or fault tree analysis.

Table 6.1 Typical CANDU 6 LPSD PSA

Initiating Event	Frequency (per year)	Comment
Loss of shutdown cooling system	3.43E-3/yr - primary system full 1.57E-3 - primary system drained to headers level	loss of circulation
Loss of service water	7.18 E-3/yr - primary system full 2.32E-3 - primary system drained to headers level	loss of closed loop recirculating water or loss of the raw service water system
Loss of Offsite Power	5.61E-3/yr - primary system full 1.81E-3 - primary system drained to headers level	
Loss of Regulation	1.64E-3/yr - primary system full	This event needs to be considered only for the primary system full case when the reactor may not be in guaranteed shutdown state.
Heat Transport System Leaks	3.45E-3/yr - primary system full	This event is being considered for CANDU 6 LPSD PSA update. Additional LOCA events will be considered as appropriate - such as breaks equivalent to feeder size.

6.3.1.2 Ongoing Activities

Ongoing activities include a review of plant outage heat sink manuals to help define a complete set of shutdown state configurations.

6.3.1.3 Planned Activities

Planned activities include updating initiating event frequencies as well as defining a further breakdown for the shutdown state configurations.

6.3.1.4 Future Needs

None.

6.3.2 Czech Republic

6.3.2.1 Currently Available Information

The frequencies of IEs in LP&SD PSA for NPP Dukovany are based primarily on NPP Dukovany operational experience and on VVER-440 type operational experience taken from IAEA PRIS database. The NPP Dukovany operational history is considered preferably, then type VVER-440/213 and finally all VVER-440 types are taken into account.

The LOCA and secondary side break frequencies are based on EPRI Interim Report TR-100380 which specifies failure rates per piping segment. So the number of segments has been determined for each LOCA category and for each secondary side break from NPP Dukovany design specifications and schemes.

6.3.2.2 Ongoing Activities

IE frequencies for LP&SD PSA of NPP Dukovany are currently being updated based on recent operational experience.

6.3.2.3 Planned Activities

None.

6.3.2.4 Future Needs

Analysis is needed to determine the frequencies of LOCAs in low power and shutdown states (when RCS pressure is lower than nominal).

6.3.3 France

6.3.3.1 Currently Available Information

The values adopted for LOCAs and total blackout are given in Table 6-2 and Table 6-3. The large RHRS break case uses the same hourly rate as for RCS breaks. Pressuriser breaks and RHRS breaks due to the stuck open of a safety valve were evaluated on the basis of the probabilities of demand and non- reclosure or spurious opening of valves.

For PSA 900, the principal sources of information used for identifying the initiating events were:

- similar probabilistic assessments,
- French and worldwide experience feedback,
- the safety analysis report (design situations),
- ad hoc studies carried out in the field of safety (for example post-Chernobyl studies).

For the quantification of Initiating Events, Insofar as possible, priority is given to experience feedback. The following methods were used:

- for events of observable frequency, the values used are specific ones derived from operating experience with Electricité de France power plants (e.g. loss of off-site electrical power supplies, inadvertent drop of level water during mid-loop operation),

- for less frequents events, the values used are derived from worldwide feedback experience (e.g. small primary breaks),

- for extremely rare events, the values have been estimated by expert judgement, allowance being made in particular for the absence of observation anywhere in the world and the values used in foreign studies (e.g. large LOCAs),

- when the initiating event is the loss of a system, quantification is based on a reliability study (e.g. electrical power blackout, loss of the component cooling system and loss of the service water system).

Table 6-2 Frequency of LOCAs

Reactor State	Initiating event	Frequency per reactor year
B	Large breaks Intermediate breaks Small breaks Break at pressuriser safety valves	5E-7 1.5E-6 1E-5 2.4E-4
C	Large RHRS break Intermediate RHRS break Small RHRS break Break at RHRS safety valves Break at pressuriser safety valves	3.5E-6 1.1E-5 7.1E-5 6.7E-6 2.3E-4
D	Small RHRS break	1.22E-4
E	Small RHRS break	5.8E-5

Table 6-3 Probability of Total Blackout

Type of Initiating Event	Reactor State	Probability of Initiating Event per Reactor-year
Short circuit of bus LHA and failure of diesel set LHQ	Unit on AFW under RHRS conditions	1.98E-4
	Unit under normal cold shutdown conditions	9.62E-6
	Unit in cold shutdown condition for maintenance	1.66E-5
	Unit in cold shutdown condition for refuelling	7.87E-6
Loss of off-site then on-site electrical power supplies	Unit on AFW under RHRS conditions	1.08E-4
	Unit under normal cold shutdown conditions	5.18E-6
	Unit in cold shutdown condition for maintenance	8.94E-10
	Unit in cold shutdown condition for refuelling	4.24E-6
Short circuit of busses LHA and LHB	Unit on AFW under RHRS conditions	1.41E-8
	Unit under normal cold shutdown conditions	6.34E-10
	Unit in cold shutdown condition for maintenance	1.10E-9
	Unit in cold shutdown condition for refuelling	5.19E-10

The others initiating events that can be mentioned are:

- the inadvertent boron dilution:
  - -progressive boron dilution during cold shutdown: 1.6 E-2/r-y,
  - -heterogeneous boron dilution during hot shutdown: 1.2 E-6/r-y,
- loss of the RHRS pumps by drop of primary water level during mid-loop operation : 1.E-2/r-y,
- total loss of service water system, total loss of component cooling system, combined failures of service water system and component cooling system leading to the loss of cold heat sink, total loss of cold heat sink : 8E-6/r-y,
- total loss of RHRS system: 1.3E-3/r-y.

Identification and quantification of initiating events 'progressive boron dilution' and 'loss of the RHRS pumps by drop of primary water level during mid-loop operation' are deduced from operating experience.

6.3.3.2 Ongoing Activities

The updating of the initiating event frequency is in progress. Moreover, complementary initiating events are being introduced notably those which can be induced by fire during shutdown states.

6.3.3.3 Planned Activities

In the future, reassessments are planned, notably in the frame of the periodic safety review.

6.3.3.4 Future Needs

There are no particular needs for the estimation of the initiating event frequency when this frequency is deduced from the operating experience. But for the less frequent initiating events, notably the LOCAs, specific studies permitting to get more realistic values are needed.

6.3.4 Germany

6.3.4.1 Currently Available Information

In the BWR-study only the frequency for the four analyzed initiating events was determined. In the PSA for the PWR, a more systematic analysis of the frequency of the initiating events was performed (see table below).

BWR:	Frequencies (1/R-Y)
• Loss of shutdown cooling via modified shutdown line	5.2E-2
• Cold overpressurisation of RPV	4.0E-3
• Leak at the cavity seal liner	1.0E-3
• Leak at the RPV bottom head	1.3E-4

The frequencies for these events were derived by the following different approaches:

- The frequency for the loss of shutdown cooling was derived from a fault tree analysis. The modelling of the RHR-train within the fault trees was necessary because of dependencies in the controlling of the event.

- The frequency for an inadvertent injection into the RPV as an initiator for a cold overpressurisation was derived from the operating experience as a probability of 3.2E-3. Related to 1.25 Shutdowns/Year, this leads to the above frequency.

- The frequency for the leak in the cavity seal liner is based on worldwide operating experience.

- The leak at the RPV bottom head assumes a failure during the withdrawal of the pump shaft of an internal recirculation pump. Hence, the frequency consists of two factors, the frequency of the withdrawal of the pump shaft (1.0E-1/R-Y) and the probability of a failure in the communication (1.3E-3).

PWR:

The reported events of German PWRs during low power and shutdown were evaluated. Those events identified as initiating event in this evaluation were assigned to event groups. Frequencies for these events are established on the basis of the German operating experience. Such events are marked in the table below with “OE” for occurred event. For very rare events like LOCAs the international operating experience was considered also. The frequency of events which have not occurred in the operation experience are determined with a fault tree analysis. Such events are marked in the table below with "FT" for fault tree.

Table 6-4 Frequencies of initiating events for 14-day outage of the reference PWR-plant

Initiating Event and Plant Operation State		**p/Out**	**P**
Transients			
Loss of preferred power-external (20 h mid-loop)	T1.1, 1B2	4,8E-4	OE
Loss of preferred power-external (20 h mid-loop)	T1.1, 1C	4,8E-4	OE
Loss of preferred power-internal	T1.2, D,E	2,5E-2	OE
Loss of main heat sink without loss of main feedwater	T3, 1A1	6,6E-3	OE
Loss of main feedwater with loss of main heat sink	T4, 1A1	3,9E-3	OE
Loss of residual-heat removal due to	T7		
- faulty level lowering	T7.1, 1C	4,9E-6	FT
- failure of residual-heat removal chains	T7.2, 1B2	4,8E-5	FT
- failure of residual-heat removal chains	T7.2, 1C	4,8E-5	FT
Faulty emergency cooling signals	T8, 1B2	3,6E-2	FT

Initiating Event and Plant Operation State (continued)		p/Out	P
Coolant losses			
Inadvertently open pressuriser RV due to maintenance fault	S6, 1A1	6,7E-3	OE
Steam generator tube leak < 2 A	S7, 1B1	5,0E-4	OE
Leak in the residual heat removal system < 25 cm²	S8	5,0E-4	OE
- S8 in the containment, RPV closed	S8.1, 1B2	1,25E-4	
- S8 in the containment, RPV open	S8.1, 1C	1,25E-4	
- S8 in the annulus, RPV closed	S8.2, 1B2	1,25E-4	
- S8 in the annulus, RPV open	S8.2, 1C	1,25E-4	
Leak in the volume control system < 25 cm²	S9, 1B1	5,0E-5	OE
Leak in the reactor cavity / settling pond	S10, 1D	4,0E-5	FT
Leak into a connected system	S11, 1B2,1C	<1,0E-7	FT
Deboration			
Leaks from systems carrying unborated water	D1		
- Steam generator tube leak	D1.1, 1B2,1C	5,0E-4	OE
- Leak in the RHR cooler	D1.2, 1B2,1C	5,0E-4	OE
- Leak in a bearing seal	D1.3, 1B2,1C	9,5E-3	OE
- Inadvertent injection into the primary system via relief tank / pressuriser	D1.4, 1B2	4,6E-6	FT
Borating fault during shutdown	D5, 1A1	2,3E-4	FT
Inadvertent deboration during start-up following a failure of all reactor coolant pumps	D6/T1, 2A1	3E-8	FT

p/Out = Probability per outage P = Probability
OE = Operation Experience FT = Fault Tree

6.3.4.2 Ongoing Activities

None.

6.3.4.3 Planned Activities

The determination of frequencies of initiating events for a BWR-69 is planned.

6.3.4.4 Future Needs

None.

6.3.5 Hungary

6.3.5.1 Currently Available Information

The initiating event list from the PSA study for nominal power operational mode was used as a starting point for identifying the initiating events for low power and shutdown states. Events on this list were removed and new events added by taking into account the specifics of the different plant operational states. Events were removed from the list 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 review, events having no possibility of occurrence during a POS owing 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:

- inadvertent boron dilution,
- cold over-pressurization of the reactor vessel,
- termination of the natural circulation, or
- events initiated by heavy load drop.

The final list of initiating events is given in Table 6-5, where the validity of the initiating events in the different plant operational states is also indicated.

Estimating the initiating event frequencies involved identifying the unique categories of events that have the potential to result in an initiating event for a POS and then quantifying each unique category. These categories included internal causes and other causes, such as the erroneous change of the operational loop, erroneous draining of the operational loop, maintenance activities, erroneous system alignment, heavy load drop, and other erroneous interactions. The frequency for each category of event was quantified, and these frequencies were then combined to produce POS-specific initiating event frequencies for each initiating event. A total of 447 frequency values have been defined. This final list of initiating events and their frequencies is not included owing to this large number.

6.3.5.2 Ongoing Activities

None.

6.3.5.3 Planned Activities

None.

6.3.3.4 Future Needs

None.

Table 6-5 Initiating Events in Different Plant Operating

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 6-5 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
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 6-5 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				+	+	+													+		+	+		

6.3.6 Japan

6.3.6.1 Currently Available Information

Eleven initiating events are considered for each POS for BWR-5 type plants. These frequencies are derived from the Japanese operating experiences except for the LOCAs. Typical frequencies are given in Table 6-6.

Table 6-6 Typical BWR-5 frequencies

Initiating Event	Frequency (1/ry) (As of March 1997)	Comment
Loss of 1 train RHR (front line)	2.8E-3	one failure assumed
Loss of 1 train RHR (support line)	2.8E-3	one failure assumed
Loss of Offsite Power	4.5E-3	
Large LOCA	2E-6	WASH-1400 method
Medium LOCA	8E-6	WASH-1400 method
Small LOCA	2.4E-5	WASH-1400 method

Five initiating events are considered for each POS for PWR 4-loop plants. These frequencies are derived from Japanese operating experiences except for the LOCA. Table 6-7 presents the frequencies.

Table 6-7 Typical 4-loop PWR frequencies

Initiating Event	Frequency (1/ry)
Loss of RHR (loss of coolant at midloop operation)	8.7E-4
Loss of RHR (1 train failure)	3.2E-3
Loss of coolant (not at midloop operation)	8.7E-4
Loss of Offsite Power	1.3E-3
Reactivity Insertion Accident	4E-6

6.3.6.2 Ongoing Activities

Ongoing activities include updating initiating event frequencies for BWR-4 and PWR 2-loop plants.

6.3.6.3 Planned Activities

Updating the initiating event frequencies for ABWRs and PWR-3 loop plants are planned for the future.

6.3.6.4 Future Needs

None.

6.3.7 Korea

6.3.7.1 Currently Available Information

For YGN 5 and 6, initiating events were identified by similar methodology of full power PSA. The plant activities were intensively reviewed using plant document such as, operation log book. We selected similar IE groups to NUREG/CR-6144:

- transients,
- LOCAs,
- loss of shutdown cooling function, and
- reactivity incidents, and
- specific IE (loss of supporting system).

6.3.7.2 On-going Activities

The initiating event frequencies being estimated, but the most of them will be used generic data.

6.3.7.3 Planned Activities

None.

6.3.7.4 Future Needs

None.

6.3.8 South Africa

6.3.8.1 Currently Available Information

The Koeberg PRA has always recognized both full power and low power and shutdown risk. The following methodology has been used to develop initiating events for Koeberg.

Initiating events are classified as follows:

- LOCA;
- secondary breaks (steam line breaks, steam header rapture, feedwater line breaks);
- loss of support systems (total loss of RHR during shutdowns, loss of component cooling, power supplies, air conditioning, ventilation; and
- transients, all events other than those classified above.

For the identified initiating events, the following studies have been reviewed:

- Koeberg SAR,
- CEA EPS-900,
- NUREG/CR-6144,
- Koeberg load profile and shutdown diagrams, and
- Koeberg incident procedure.

Primary LOCA sizes are based on plant response as well as the required operator response

List of actual events that caused reactor trips, NUREG-4550 is being used. This list is supplemented by any events unique to Koeberg from load profile diagrams. Systematic screening of support systems is done to determine whether an automatic reactor trip, manual reactor trip or controlled shutdown results from loss of the system.

Plant safety functions are identified as in NUREG-1602:

- reactor subcriticality,
- core heat removal,
- RCS integrity, and
- containment overpressure suppression.

The screening and grouping of initiating events is based on NUREG-1602 (sections 2.1.2.2 and 2.1.2.3 respectively).

As an example, a summary of initiating events, other than generic transients that have occurred at Koeberg, is given in Table 6-8. Transient initiating events that occurred at Koeberg are provided in Table 6-9.

Table 6-8 Initiating events other than generic transients

Classification	Number	Frequency (per year per unit)*
Controlled OTS Shutdown (APG)	9	3.1E-1
Controlled OTS Shutdown (Batteries)	3	1.0E-1
Controlled OTS Shutdown (EVC)	1	3.4E-2
Controlled OTS Shutdown (PTR)	1	3.4E-2
Controlled OTS Shutdown (RCP)	5	1.7E-1
Controlled OTS Shutdown (RIS)	3	1.0E-1
Controlled OTS Shutdown (RPN)	4	1.4E-1
Controlled OTS Shutdown (RRA)	1	3.4E-2
Controlled OTS Shutdown (SEC)	1	3.4E-2
Controlled OTS Shutdown (VVP)	4	1.4E-1
Controlled Shutdown (unknown reason)	1	3.4E-2
Controlled Shutdown for repairs to BOP	3	1.0E-1
Fuel Conservation Shutdown	1	3.4E-2
Grid Transient	7	2.4E-1
Loss of 400kV	11	3.8E-1
Loss of LCB	1	3.4E-2
Loss of LDA	1	3.4E-2
Loss of Lni	1	3.4E-2
Minor secondary system pipe rapture	1	3.4E-2
Refuelling shutdown	17	5.9E-1
Shutdown for modifications	1	3.4E-2

*The frequency is based on approximately 14.5 calendar years from startup of Unit 1 (April 84) up to (January 98)

Table 6-9 Transient initiating events at Koeberg

NUREG 4550 Generic Transients	Number	Frequency (per year
Auto Trip (no transient)	10	3.4E-1
Condenser leakage	3	1.0E-1
CRDM mechanical problems and/or rod drop	3	1.0E-1
Full or partial closure of MSIV	2	0.069
FW instability	3	1.0E-1
Generator trip	10	3.4E-1
Inadvertent safety injection signal	3	1.0E-1
Increase FW flow (all loops)	1	3.4E-2
Increase FW flow (one loop)	7	2.4E-1
Leakage in primary system	4	1.4E-1
Loss of all CEX pumps	2	6.9E-2
Loss of Condenser Vacuum	1	3.4E-2
Loss of Offside Power	1	3.4E-2
Loss of power to necessary plant system	5	1.7E-1
Loss of reduction in Feedwater flow (one loop)	9	3.1E-1
Manual trip, no transient	19	6.6E-1
Misc. leakage in secondary system	6	2.1E-1
Steam generator leakage	1	3.4E-2
Total loss of FW flow	3	1.0E-1
Turbine Trip – EHC problem	21	7.2E-1
Load reduction following loss of CEX pump	1	3.4E-2

*All data presented are available in reports of Koeberg PRA Customization Project

Table 6-10 Initiating Event Frequencies

Accidents	Initiating Event Frequency				
	SDH	**SDL**	**RR1**	**RR2**	**RR3**
Plant Operating States	B5-B8 & L	C&K	D&J3-J6	E&J1-2	F,I&H
Secondary line breaks					
Large					
Inside cont.	2.670E-06				
Small					
Outside cont., downstream of MSIVs	1.330E-04				
Feedwater line break					
Large, inside containment	2.670E-06				
LOCAs					
Large LOCA					
RCP LOCA	2.640E-06		9.900E-06		
RRA LOCA			2.790E-05		
Intermediate LOCA				1.180E-04	
Pressuriser LOCA		2.860E-05	1.800E-04		
RCP LOCA	1.030E-05	1.740E-06	1.980E-05		
RRA LOCA			1.430E-04		
Interfacing Systems LOCA			8.640E-08		
via PTR	2.310E-08				
via RCV	1.230E-09				
via RIS	4.350E-11				
Small LOCAs					
RCP LOCA		3.370E-06	1.980E-05		
RRA LOCA			1.430E-04		
Vessel LOCA	2.410E-09				
H-LOCA			1.180E-04	9.860E-05	3.920E-03
K-LOCA					
Large				1.190E-05	
Intermediate				2.380E-05	
Small				2.380E-05	

Loss of ventilation systems					
Loss of DVG	2.590E-04				
Loss of DVH	2.590E-04				
Loss of RRM	2.590E-04				
Primary system transients					
Spurious PRZR heater demand	1.160E-03				
Seal injection line rupture	5.460E-04				
Spurious dilution	1.310E-03	2.680E-03	4.000E-04	6.610E-03	8.330E-03
Spurious depressurisation		2.620E-04			
Spurious PRZR sprays	6.820E-05				
Spurious SI	7.330E-04				
Secondary system transients					
Increase in feedwater flow	1.680E-03	7.930E-04			
MSIV	1.820E-03				
GCT to atm. In advert. Open	5.280E-02				
Turbine trip	5.540E-04				
Loss of Electrical Supplies					
Loss of LBA	2.600E-05		2.690E-05	2.240E-05	4.380E-05
Loss of LBB					4.380E-05
Loss of LCA	6.220E-05		6.430E-05	5.5350E-05	1.050E-04
Loss of LCB	6.220E-05				1.050E-04
Loss of LCC	6.030E-05				1.010E-04
Loss of LDA	2.940E-04				4.970E-04
Loss of LHA	5.220E-05		5.400E-05	4.500E-05	8.800E-05
Loss of LHB					8.800E-05
Loss of LNA					8.800E-05
Loss of LNB					8.800E-05
Loss of LNC					8.800E-05
Loss of LND					8.800E-05
Loss of LNE					8.800E-05
Loss of Offsite Power	4.070E-02		4.200E-02	3.500E-02	6.840E-02
Overpressure Accident			1.420E-01		

Loss of RRA			1.100E-03	9.120E-04	1.780E-03
LOSS OF RRI	2.410E-06		2.490E-06	2.070E-06	4.050E-06
Loss of SAR	9.160E-05				

6.3.8.2 Ongoing Activities

None.

6.3.8.3 Planned Activities

None.

6.3.8.4 Future Needs

None.

6.3.9 Spain

6.3.9.1 Currently Available Information

The process to identify initiators during LPSD operation for LPSD PSA is similar to process used for the full power PSA for Vandellos 2 and Asco. Generic and plant specific operating experience, failure mode and effect analyses (FMEAs), other PSAs, qualitative applicability analysis of full power initiators, analysis of plant specific operational practices and design features, etc., are the techniques used for this identification.

Initiators are identified for each plant operational state modelled in the PSA. Definition and selection of these POSs are important tasks. Related to this is the issue of screening scenarios (i.e., groups of initiators in a POS) to be analyzed in detail. Quantitative screening criteria should not be based only on duration of the corresponding POS. Conditional risks during the POS should somehow be considered.

The applicability of initiators that have occurred in specific POSs should be analyzed to develop consistent statistics before normalization of the frequencies to a "per calendar year" basis occurs. For instance, likelihood of LOOP, or other transient types, might be different for different POS, according to plant design or operational practices. Therefore, LOOPs that have occurred should be assigned to a specific POS when estimating the statistics (number of POS-specific LOOPs divided by the total time in a POS in years), rather than doing a general estimation (total number of LOOPs in total number of calendar years) and multiplying by the mean fraction of the year in a POS.

6.3.9.2 Ongoing Activities

Currently, every LPSD PSA project is involved with the above task and, although each project faces the difficulties mentioned above, task procedures are expected to improve from project to project.

6.3.9.3 Planned Activities

Planned activities include performing this task for each of the remaining LPSD PSA Projects.

6.3.9.4 Future Needs

A standardized process is needed to make this part of the analysis more cost effective, while maintaining the capabilities to accurately estimate POS-specific initiating event frequencies and to select the more important scenarios for detailed analysis. Data banks initially designed to collect information for full power operation should be adapted to collect relevant information about LPSD operational incidents.

Such information would help support better understanding of operating experience and allow for more efficient initiating event frequency estimation.

6.310 Chinese Taipei

6.3.10.1 Currently Available Information

Generic data from the Grand Gulf and Surry analyses were used. This data was then updated using Bayesian techniques for loss of RHR events and FMEA events, where FMEA events were considered only for loss of RHR events.

6.3.10.2 Ongoing Activities

None.

6.3.10.3 Planned Activities

None.

6.3.10.4 Future Needs

None.

6.3.11 United States

6.3.11.1 Currently Available Information

For Grand Gulf, accident initiating events were classified into four groups after a systematic search of actual events, as well as identified accident scenarios was performed. The four groups include:

- transients,
- LOCAs,
- loss of decay heat removal, and
- special events involving reactivity excursions.

Hazard events, namely internal fires and flooding, were also considered. However, no systematic search of all possible accident scenarios was performed using techniques such as failure mode and effect analysis, hazard and operability analysis (HAZOP), master logic diagrams, and heat balance fault trees.

These initiating events are listed on Table 4.1.1, Vol.2, Part 1A of NUREG/CR 6143. The three special events associated with reactivity excursions were reviewed, but were screened from further analysis. Since Grand Gulf is a BWR, with no need for a chemical shim to maintain shutdown margin when the rods are in, there were no issues associated with reactivity excursions caused by boron dilution.

For Surry, the approach used to identify initiating events that may occur when the plant is operating at low power or shutdown conditions is summarized in Table 4.1-1 of NUREG/CR-6144, Vol. 2, Part 1a.

.Just as with Grand Gulf, a systematic search of actual events, as well as identified accident scenarios was performed. Again, no systematic search of all possible accident scenarios was attempted

Table 4.1-4 of NUREG/CR-6144, Vol. 2, Part 1a lists each initiating event and its frequency.

6.3.11.2 Ongoing Activities

None.

6.3.11.3 Planned Activities

None.

6.3.11.4 Future Needs

None.

6.4 Component Probabilities

International databases were accessed on a major scale in order to obtain a large amount of data, such as the French Reliability Data Acquisition System and the Event File databases, as well as files of statistical data. To make the studies more realistic, it frequently is necessary to support the data collection by performing onsite investigations to allow for the inclusion of certain site-specific characteristics.

Examples of component failure rates and of demand failures are listed in this section.

The failure data are mainly based on generic data. Typical failure rates for various components are provided.

Component failure rates are normally taken from the full power PSAs and analyzed the same way as in the full power PSA (e.g., generic data, specific component operating experience analysis, and Bayesian updating).

Component maintenance outages can be determined by a maintenance schedule. This has raised an issue related to POS definition. Should different POSs be defined for each "deterministic" component unavailability?

A simple and better way of considering maintenance unavailabilities, connected with the POS definition and screening analysis methodologies, should be developed. New or adapted software to consider the maintenance schedule part of the plant outage schedule in a LPSD PSA may be necessary.

6.4.1 Canada

6.4.1.1 Currently Available Information

Typically the component failure data used so far was from the operating experience of CANDU plants in Ontario. The test and maintenance intervals were based on the operating practice of CANDU 6 plants in Canada and Korea.

6.4.1.2 Ongoing Activities

None.

6.4.1.3 Planned Activities

None.

6.4.1.4 Future Needs

Assess the suitability of the existing component failure data for the shutdown state plant modelling. Current data pertains mostly to the plant nominal state of full power.

6.4.2 Czech Republic

6.4.2.1 Currently Available Information

Plant specific data have been used for the component probabilities in LP&SD PSA for NPP Dukovany. Especially pumps have their own data specific for each system. There are just a few exceptions like check valve failures for which generic data are applied. Much of the component data were taken from full power PSA since the same components are modelled. However, attention has been paid to test intervals which are sometimes different from the intervals used in full power PSA.

Failure rates or probabilities per demand for the most important components used in PSA for NPP Dukovany are as follows:

Motor driven pump (general range):

failure to start: between 7.4×10^{-7}/hr - 2.7×10^{-5}/hr

failure to run: between 1.7×10^{-6}/hr - 1.7×10^{-3}/hr

LPI ECCS pump:

failure to start: 1.8×10^{-6}/hr

failure to run: 1.7×10^{-4}/hr

HPI ECCS pump:

failure to start: 5.5×10^{-6}/hr

failure to run: 8.4×10^{-4}/hr

Confinement Spray pump:

failure to start: 2.0×10^{-6}/hr

failure to run: 1.2×10^{-3}/hr

Main FW pump:

	failure to start:	2.7×10^{-5}/hr
	failure to run:	1.3×10^{-5}/hr

Auxiliary FW pump:

	failure to start:	3.6×10^{-6}/hr
	failure to run:	1.2×10^{-4}/hr

Emergency FW pump:

	failure to start:	3.6×10^{-6}/hr
	failure to run:	1.1×10^{-3}/hr

Essential Service Water pump:

	failure to start:	2.2×10^{-6}/hr
	failure to run:	1.7×10^{-6}/hr

Diesel generator:

	failure to start:	3.0×10^{-5}/hr
	failure to run:	1.7×10^{-3}/hr

Circuit breaker:

6 kV pumps	failure to close:	4.9×10^{-6}/hr
0.4 kV pumps	failure to close:	1.7×10^{-6}/hr

Valve:

MOV	failure to change position:	2.3×10^{-6}/hr
AOV	failure to change position:	2.3×10^{-6}/hr
check valve	failure to open	1.0×10^{-6}/hr (generic)

PORV:

Main	stuck in the open position	2.0×10^{-2}/d
Remote controlled	failure to close	1.0×10^{-2}/d

6.4.2.2 Ongoing Activities

A software for PSA component database is currently being developed for NPP Dukovany PSA.

6.4.2.3 Planned Activities

Identification and quantification of software failures for new I&C equipment is planned.

Future Needs

A detailed methodology is needed to identify and quantify software failures.

6.4.3 France

6.4.3.1 Currently Available Information

With few exceptions, all the reliability data and operating data used in PSA 900 and PSA 1300 studies were the result of EDF's experience in operating PWRs. This approach has been made possible by the mass of operational feedback from a standard series of PWRs representing more than 200 reactor years, unparalleled throughout the world.

The French series of PWRs is, in fact, characterized by its uniform design, operation and maintenance. Even if differences do exist between ranges, such as between the 900 MWe and the 1300 MWe series in relation to safeguard systems, the equipment making up these systems is virtually identical except for a few exceptions.

The studies required a detailed analysis of the operating data. The database was build and has been critically analyzed by the IPSN as an external control organization.

The following paragraph describes the data collection.

National databases were accessed on a major scale in order to obtain a large amount of data, that is, the Reliability Data Acquisition System and the Event File databases as well as the file of statistical data on the operation of French nuclear power stations were analyzed. To make the studies more realistic, it became necessary to support the data collection by performing onsite investigations to allow for the inclusion of certain site-specific characteristics. Finally several foreign databases were accessed to back up and compare some specific data.

Examples of component failure rates include:

- pumps - between 5.5E-6 /h and 5.5E-5 /h
- AFWS pumps - 3.2E-4 /h
- AFWS turbines and diesel generators - between 3E-3/h and 1E-2/h
- operation of passive components - between 1E-7/h and 1E-6/h
- or electronic components
- motors (independent of their output) - around 4E-6/h.

Examples of demand failures include:

- motors - 7.5E-6/d
- AFWS turbine - 8.6E-3/d
- diesel generators - 3.4E-3/d
- valves - between 1E-4/d and 1E-3/d
- check valves - 1E-5/d
- various pumps - between 2E-5/d and 1E-3/d

6.4.3.2 Ongoing Activities

The updating of the values of the failure rates is in progress.

6.4.3.3 `Planned Activities

In the future, reassessments are planned notably in the frame of the periodic safety review.

6.4.3.4 Future Needs

None.

6.4.4 Germany

6.4.4.1 Currently Available Information

Concerning the reliability data of components in general, the plant-specific values from the full power PSA have been used. Information from comparable German plants is considered if the observation time for a particular component is too short. For new components, the reliability data were derived in the same way as for the full power PSA. Attention has been paid to the test intervals, which may be different from the nominal test frequency during full power because of operational demands during the plant shutdown.

6.4.4.2 Ongoing Activities

None.

6.4.4.3 Planned Activities

Component probabilities for a BWR-69 will be determined.

6.4.4.4 Future Needs

None.

6.4.5 Hungary

6.4.5.1 Currently Available Information

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 using the following steps:

- First, data from eight different, internationally accessible sources (e.g. IAEA-TECDOC-478, EIREDA, Swedish T-Book, IEEE database) 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 process, 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.

To validate the data for the low power and shutdown states, the source databases were reviewed to determine 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 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.

6.4.5.2 Ongoing Activities

A multi-purpose data collection system has been set up at the Paks NPP. The data gathered can be used for reliability studies - PSAs in particular, among others. The plant personnel are involved in the continuous task of to collect and classify operational data of different types, including mechanical failures of the equipment, human errors, etc. A review of the data collection system is currently going on at Paks.

6.4.5.3 Planned Activities

Reliability data will be updated every year as part of the framework of regular updates of the whole PSA model and assumptions. Such updates are necessary because of the number of safety enhancement measures currently being introduced that should be reflected in the PSA models.

6.4.5.4 Future Needs

None.

6.4.6 Japan

6.4.6.1 Currently Available Information

The component failure probabilities used are the same as for power operation. The failure data are mainly based on generic data from the USA (e.g. LER data for mechanical components, IEEE std-500 for electrical components). Typical failure rates for various components are as follows:

Motor driven pump	fail to start	3.0E-3/d
	fail to keep running	1.4E-5/h
Motor operated valve	fail to operate	2.6x10-6/h
	Plug	3.2x10-8/h
Relay	Loss of function	1.0x10-7/h

6.4.6.2 Ongoing Activities

None.

6.4.6.3 Planned Activities

NUPEC plans to replace the data with the Japanese experience data.

6.4.6.4 Future Needs

None.

6.4.7 Spain

6.4.7.1 Currently Available Information

Component failure rates are normally taken from the full power PSAs. Only data for new components, i.e., those not modelled in the full power PSA, are added to the PSA component database. These new data are obtained and analyzed the same way as in the full power PSA (e.g., generic data, specific component operating experience analysis, and Bayesian update).

Component maintenance unavailability data require additional analysis because full power data are not applicable. In the Asco LPSD PSA, a thorough analysis was done, using data recorded and kept in a computerized plant system for maintenance management. Those data were used to calculate total component unavailability duration in the applicable POSs and to estimate averages for these unavailabilities.

Component maintenance outages can be determined by a maintenance schedule. This raises an issue related to POS definition. Theoretically, different POSs should be defined for each “deterministic” component unavailability; however, depending on the level of detail associated with the system models, this may prove too difficult or require more resources that are available.

6.4.7.2 Ongoing Activities

Currently, every LPSD PSA project is involved with the above task and, although each project faces the difficulties mentioned above, task procedures are expected to improve from project to project.

6.4.7.3 Planned Activities

Planned activities include performing this task for each of the remaining LPSD PSA Projects.

6.4.7.4 Future Needs

A simple and better way of considering maintenance unavailabilities, connected with the POS definition and screening analysis methodologies, should be developed. New or adapted software to consider the maintenance schedule part of the plant outage schedule in a LPSD PSA may be necessary.

6.4.8 Chinese Taipei

6.4.8.1 Currently Available Information

Component failure data are normally taken from the full power PSAs. Component maintenance unavailability is considered only as component maintenance outage and can be determined by the maintenance schedule.

6.4.8.2 Ongoing Activities

None.

6.4.8.3 Planned Activities

None.

6.4.8.4 Future Needs

None.

6.4.9 United States

6.4.9.1 Currently Available Information

For Grand Gulf, the major source of information used came from the NUREG/CR-4550 analysis of Grand Gulf. Much of the component specific data came directly from this analysis. Plant specific data were used in determining estimates for maintenance unavailabilities and fractions of time spent in each POS. The data analysis described in NUREG/CR-6143 summarizes the experience at Grand Gulf Unit 1 with respect to:

- operating modes and plant configurations,
- transitions between modes, and
- reconfiguration of plant safety systems.

For Surry, maintenance unavailability of components was estimated for each plant operational state using data extracted from the shift supervisor's log books and the minimum equipment lists. Ten years of log books, five years for each unit (from 1985 to 1989), plus the 1990 refuelling outage of unit 1, were obtained from Virginia Power.

The following four types of information were of particular interest:

- maintenance/repair on safety and support systems,
- important events associated with important systems or components,
- potential fire, and flood-related events, and
- periodic tests.

Surry specific component unavailabilities are provided in NUREG/CR-6144.

6.4.9.2 Ongoing Activities

None.

6.4.9.3 Planned Activities

None.

6.4.9.4 Future Needs

None.

6.5 Common Cause

Rules for common cause failures (CCFs) are provided in this section.

The approach used to quantify CCFs is typically based on the β factor method. The β factor is defined as the ratio, for a particular component, of the number of common mode failures to the total number of failures. However, to cover cases of redundancy of a factor of more than 2, this method was made universal by using factors β_k^n representing the proportion of common modes affecting k components with redundancy of a factor of n.

The β_2^2 factors were estimated using reactor operating experience feedback, that is, the method used to identify the CCF component groups and the group-related ß factors for the LPSD PSA is the same as the method used in the full power PSA. The values for equipment with insufficient feedback experience are estimated either by analogy with the preceding components or using engineering judgement, with allowance for the values used in probabilistic safety studies.

6.5.1 Canada

6.5.1.1 Currently Available Information

In the current LPSD PSA, CCF was not modelled.

6.5.1.2 Ongoing Activities

None.

6.5.1.3 Planned Activities

Include CCF modelling in LPSD PSA. The CCF methodology to be used will be Unified Partial Method, developed by SRD in the United Kingdom.

6.5.1.4 Future Needs

None.

6.5.2 Czech Republic

6.5.2.1 Currently Available Information

Generic Beta factors based on NUREG/CR-5801 have been generally applied in LP&SD PSA for NPP Dukovany. Those factors have been modified for triple and more redundant CCF groups of components. For the most important CCF groups Alfa factor model has been implemented. This approach is the same as used in full power PSA for NPP Dukovany.

6.5.2.2 Ongoing Activities

An incorporation of Alfa factor for other components in NPP Dukovany PSA is underway.

6.5.2.3 Planned Activities

Effort on systematic CCF data collection is planned at NPP Dukovany.

6.5.2.4 Future Needs

CCF statistics from others NPPs is needed.

6.5.3 France

6.5.3.1 Currently Available Information

In the French PSAs, the common points identified (for example the common support systems) were explicitly introduced into the fault trees or event trees, and were therefore not considered to be common modes failures.

Common cause failures were considered using the following rules:

- components which are identical in terms of their geometrical and technical properties (however the AFW pumps represent a special case in which CCFs are only considered for the pump section),

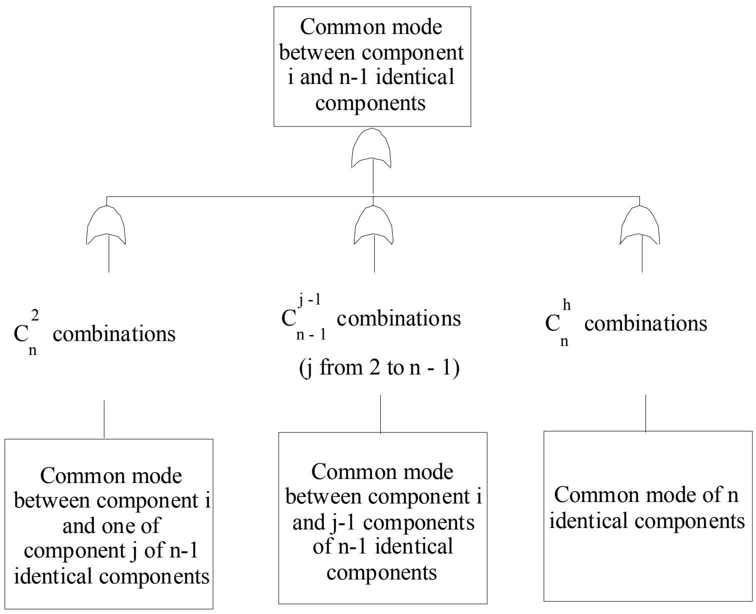
- components in different systems were not considered, except for certain sets of check valves such as those in the safety injection system and the reactor coolant system which are identical in the cold legs of the injection lines or accumulators,

- for each component for which CCFs are postulated, allowance is made for all the failures which can affect the component and one or several other identical components (see Figure 6-1), and failures, either on demand or in operation, are considered. Furthermore, for certain cases (pumps and compressors) it was assumed that a common mode could simultaneously cause failure to run of the component in service and failure to start of the standby component.

The approach used to quantify CCFs is based on the β factor method. The β factor is defined as the ratio, for a particular component, of the number of common mode failures to the total number of failures.

However, to cover cases of redundancy of a factor of more than 2, this method was made universal by using factors β_k^n representing the proportion of common modes affecting *k* components with redundancy factor of *n*.

Figure 6-1 Common Cause Failure Combinations



Using the Atwood method parameters, it is possible, to calculate the β_k^n set using the three basic values which are β_2^2, β_3^3, and β_4^4.

It can be stated that:

$$\beta_2^2 = \frac{\mu P^2 + \omega}{\lambda}$$

$$\beta_3^3 = \frac{\mu P^3 + \omega}{\lambda}$$

$$\beta_4^4 = \frac{\mu P^4 + \omega}{\lambda}$$

Where:

μ = non-lethal shock rate,
P = probability of component failure, given that a non-lethal shock has occurred,

ω = lethal shock rate,
λ = failure rate, for all modes, of the component in question, the value found in databases.

Using Atwood terminology, a lethal shock causes failure of all redundant components lethal shock causes the failure of one component with a conditional probability P.

The above formulas were used to derive the values of parameters P, μ and ω. It is then to calculate, for each type of component, values of:

β_k^n, where *k* # *n* corresponds to the failure of *k* components in a set of *n*.

$$\beta_k^n = \frac{\mu P^k (1-P)^{n\cdot k}}{\lambda}$$

The β_2^2 factors (n = 2 to 4) were estimated using Électricité de France reactor operating experience feedback using data in the SRDF reliability data collection system and the event file for the following equipment:

- sensors and instrumentation,
- check valves,
- contactors and circuit breakers,
- reactor trip breakers,
- pumps,
- steam isolation valves,
- motor-operated valves, and
- pneumatically-operated valves.

The values for equipment with insufficient feedback experience were estimated either by analogy with the preceding components or using engineering judgement, with allowance for the values used in probabilistic safety studies in other countries.

6.5.3.2 Ongoing Activities

The updating of the reliability data related to common cause is under progress.

6.5.3.3 Planned Activities

Though not specific to the shutdown state, it was decided to take into account the common modes across the boundary of the system.

6.5.3.4 Future Needs

None.

6.5.4 Germany

6.5.4.1 Currently Available Information

The newly developed coupling model is used for common-cause data for the full power PSA and for LPSD PSA, too. Some LPSD-specific events are not included in the data set, since it had been developed for the full power PSA. Special attention was paid to the failure detection time that might be different from the normal inspection frequency because of additional demands of the components during the plant shutdown.

6.5.4.2 Ongoing Activities

None.

6.5.4.3 Planned Activities

Common cause data for a BWR-69 will be determined with the same methodology.

6.5.4.4 Future Needs

There is a need for common cause data in accommodation to the specific operational conditions during low power and shutdown.

6.5.3 Hungary

6.5.5.1 Currently Available Information

The following dependent failure categories were identified during the analysis - the same as in the PSA study for the nominal power operational mode:

- functionally dependent events
  - time dependent events
  - structurally dependent events
  - human related events
- common cause failure events

Dependent failures of the first category have been modelled either explicitly (time dependent events and structural dependence) by building in separate basic events in the logic structure, or during the definition of a human error and evaluation of the human error probabilities.

The common cause failure events identification task is complex - involving detailed root-cause analysis, as well as a search for those coupling mechanisms that were necessary to induce roughly simultaneously multiple component failures. Common cause failures were considered for:

- design and manufacturing
- actions related to operation, maintenance, and test
- equipment location

The CCFs related to design and manufacturing were considered using a parametric model of the common cause failures. Groups of mechanical, I&C, and electrical components susceptible to common cause failure were identified. Using the ß-factor model, the independent parameters and the ß parameters were quantified using the single independent failures (from combination of the generic and Paks NPP specific data) and multiple failures (from generic data). Redundancy and test strategy were considered during the calculation of ß factors.

The method used to identify the CCF component groups and the group-related ß factors for the LPSD PSA was the same as the method used in the nominal power PSA. Common cause failures characterized by ß factors and the design, manufacturing related coupling mechanism during outage were considered to be the same as at full power operation.

Common cause failures related to human interactions or equipment locations were considered by direct numerical estimation or by using fault tree/event tree models as follow:

- during the estimation of the initiating frequencies (e.g. heavy load drop)
- during the human reliability analysis (e.g. pre-initiator actions as CCFs)
- during event sequence modelling (e.g. further equipment failures induced by appropriate initiating events).

6.5.5.2 Ongoing Activities

None.

6.5.5.3 Planned Activities

None.

6.5.5.4 Future Needs

None.

6.5.6 Japan

6.5.6.1 Currently Available Information

The CCF rates used are the same as for power operation. The data values are based on NUREG-1150. The values used are shown in Table 6-11.

Table 6-11 Common cause values used for LPSD

Plant Type	Component	Value
BWR	Scram contactor	Beta 6=0.016
	SLCS Pump fail to start	Beta 2=0.21
	ADS fail to open	Beta 5=0.1
	LPCI Motor operated pump fail to start	Beta 3=0.11
	Motor operated valve loss of function	Beta 3=0.054
	RHRSW Motor operated pump fail to start	Beta 4=0.0096
	Motor operated valve loss of function	Beta 2=0.088
PWR	NPI Pump fail to start	Beta 2=0.21
	RHR Pump fail to start	Beta 2=0.15
	AX feed water pump fail to start	Beta 2=0.056
	CCW pump fail to start	Beta 2=0.026, Beta 3=0.014, Beta 4=0.0096

6.5.6.2 Ongoing Activities

None.

6.5.6.3 Planned Activities

None.

6.5.6.4 Future Needs

None.

6.5.7 Spain

6.5.7.1 Currently Available Information

Common cause failure events for the LPSD PSAs used the same models and data as were used in the full power PSAs

6.5.7.2 Ongoing Activities

Currently, every LPSD PSA project involves this task. Although each specific PSA project attempts to solve the general difficulties in modelling common cause events, task procedures are expected to improve from project to project.

6.5.7.3 Planned Activities

Perform this task for each of the remaining LPSD PSA Projects.

6.5.7.4 Future Needs

Better available data are needed. The use of recent developments in CCF databases and analysis systems may be explored.

6.5.8 Chinese Taipei

6.5.8.1 Currently Available Information

The Beta factor model is used for common-cause failure.

6.5.8.2 Ongoing Activities

None.

6.5.8.3 Planned Activities

None.

6.5.8.4 Future Needs

None.

6.5.9 United States

6.5.9.1 Currently Available Information

For both Grand Gulf and Surry, direct functional dependencies that can lead to failure of multiple components were incorporated directly into the fault or event trees. For those dependent failures that are classified as CCFs, Grand Gulf and Surry included common-cause failure events in the system fault trees. Both Grand Gulf and Surry used the beta factor model to represent these failures.

As noted in NUREG/CR-6143, the estimates for the beta factors for Grand Gulf were obtained either directly from NUREG/CR-4550, Vol. 6 for those failures that were the same or from the beta factor approach described in NUREG/CR-4550, Vol. 1 for those failure that were new. Example beta factors include those for:

- the diesel generator system,
- diesel driven pumps,
- motor operated valves in the containment spray system, and
- motor operated valves in the suppression pool cooling and suppression pool makeup systems.

For Surry, considerable effort was devoted to generating Beta factors for multiple failures using recent common-cause analytical methods.

6.5.9.2 Ongoing Activities

None.

6.5.9.3 Planned Activities

None.

6.5.9.4 Future Needs

None.

6.6 Human Reliability Events

Human Reliability Analysis (HRA) for LPSD is generally similar to the full power approach. But human intervention in LPSD has certain features that make modelling human actions using automated systematic logical processing very difficult, if not impossible. Examples are given.

To reduce subjectivity, the HRA methodology makes maximum use of experience feedback, and makes a number of relative assessments by comparing different situations.

The main sources of information were actual incidents, ad hoc interviews and investigations, and observations made during simulator tests.

The human factor has frequently been found to play an implicit role in the quantification of common modes, initiating events and even certain items of reliability data. When there are several human actions in the same accident sequence dependencies can be introduced among the actions.

The pre-accident intervention category corresponds to errors during maintenance operations, tests, and normal operation liable to contribute either to the unavailability of an engineered feature system or the occurrence of an initiating event. Some examples are listed. These errors are introduced in the fault trees at the level of the components. However there are frequently important correlations between errors of this type. Recovery of pre-accident errors during the accident is introduced in exceptional cases, where justified by the context and the weight of the event.

The post-initiator intervention category includes both diagnostic and decision-making errors and errors in taking action. In addition, distinctions are made among the roles of the operating team, the safety engineer and the action of the emergency teams. This provides a mechanism to represent the manner in which operation in an accident situation is organized, particularly the human redundancy provided by the safety engineer, and allows for a different quantification process for each role. Human intervention in accident situations was identified during preparation of the event trees by analyzing the operating procedures and making allowance for actual incidents and observations made on simulators.

The activity of the operating team is divided into two phases, diagnostic and operational.

The diagnostic phase includes detection of the incident, making the diagnostic, and making a decision. These operations are quantified using curves that give the probability of failure as a function of the time available to the operators to take action. These curves, which respectively correspond to an easy, an average, and a difficult situation, were prepared on the basis of simulator tests, supplemented by engineering judgement, making special use of Swain's work.

Operation in accident situations can be represented by a list of key actions, whose success or failure modifies the accident sequence. In the event of failure of the diagnostic, it is generally considered that key actions are not performed. In the event of success of the diagnostic, the corresponding actions may fail. For each important action, the probability failure is evaluated. Some formulas are given.

"Safety engineer" intervention, employed in French reactors, allows for recovery of degraded situations. The recovery failures of the safety engineer, the probabilities of failure, and the probabilistic statements of absence are listed.

6.6.1 Canada

6.6.1.1 Currently Available Information

Current HRA analysis was primarily based on ASEP & AECL models for errors of diagnosis.

6.6.1.2 Ongoing Activities

None.

6.6.1.3 Planned Activities

Planned activities include detailed HRA using ASEP/THERP.

6.6.1.4 Future Needs

Future needs include an analysis of recovery actions and a methodology for incorporating recovery into LPSD PSAs.

6.6.2 Czech Republic

6.6.2.1 Currently Available Information

THERP procedure as specified in NUREG/CR-1278 has been used for pre-accident human errors quantification in LP&SD PSA for NPP Dukovany.

New symptom-oriented EOPs implemented in NPP Dukovany developed primarily for full power operation are the basis for human error definition and quantification, if applicable. Accommodation of decision tree method and ASEP methods as specified in NUREG/CR-4772 have been used for quantification of post-accident human interventions. Human errors from full power PSA have been adopted as much as possible. They were adjusted for longer time windows and different boundary conditions if necessary. However, currently implemented EOPs are not applicable for possible actions used to mitigate most of accidents during shutdown operation, especially during reactor cooldown and refuelling. So it has been necessary to add several human actions into the LP&SD model which are credited just based on operational practices and training of plant staff, as well as on enough available time to determine applicable action.

6.6.2.2 Ongoing Activities

Post accident human errors are currently being reassessed to reflect those new full power EOPs implemented at NPP Dukovany which are applicable also for LP&SD operation.

6.6.2.3 Planned Activities

Assessment of post accident human errors to reflect new shutdown specific symptom-oriented EOPs is expected in the future. The planned activities also include accommodation of some principles of new HRA methods (ATHEANA, CREAM).

Future Needs

Analysis of recovery actions and methodology for incorporation of recovery actions into LP&SD PSA are needed.

6.6.3 France

6.6.3.1 Currently Available Information

Human intervention has certain features that make modelling human actions using automated, systematic, logical processing very difficult, if not impossible. For example:

- the scope of human intervention is virtually unlimited,
- for a given action, success or failure depends to a very large degree on the context (information, stress, etc.),
- human errors are generally recoverable, and
- human intervention is to a large extent mutually dependent.

To account for these highly specific aspects, the French approach includes the following steps:

- identification of potential human intervention,
- estimation of impact and selection of potentially significant intervention,
- detailed analysis,
- quantification, and
- integration into the probabilistic safety assessment.

Furthermore, depending on the results and their relative weights, iteration (i.e., backtracking) among these steps may be necessary.

Great importance is attached to experience feedback, the only proper basis for realistic analysis. Experience feedback is used to:

- identify potential errors,
- quantify their probability by direct statistics when the situation studied corresponded to observed cases, and
- create models making it possible to extrapolate from experience to other situations.

The main sources of information were actual incidents, ad hoc interviews and investigations, and observations made by Électricité de France during simulator tests.

Simulator tests constituted a particularly rich source of information. A particular test program, so-called << MSR >> simulator tests, was specially organized by EDF to observe the behaviour of operators under abnormal circumstances. Two hundred and four tests were carried out using 900 MWe and 1300 MWe control room simulators. Seventy-eight teams of operators participated in these tests.

Human intervention explicitly included in the probabilistic safety assessment can be divided into two main categories:

- pre-accident (pre-initiator) intervention, those that contribute to the unavailability of engineered safety feature systems or give rise to initiating events, and
- post-initiator intervention (diagnostics, execution of procedures, and actions outside the scope of procedures), those that influence the sequence of events.

It should also be noted that the human factor frequently plays an implicit role in the quantification of common modes, initiating events and even certain items of reliability data.

The pre-accident intervention category corresponds to errors during maintenance operations, tests, and normal operation liable to contribute either to the unavailability of an engineered feature system or the occurrence of an initiating event.

Examples include:

- incorrect positioning of an actuator or its control,
- incorrect adjustment of a sensor, and
- omissions or mistakes in the application of a normal operating procedure.

They are identified during studies of system reliability or actual incidents.

The pre-accident errors are introduced in the fault trees at the level of the components. However there are frequently important correlations between errors of this type (adjustment of a number of sensors, testing of a number of valves etc.). In that case, a common mode failure is added at the top of the fault tree. Estimation of this common mode failure is done using Swain’s methodology as represented by:

P(common mode) = P(independent pre-accident error) *P_1 *P_2* ⌡ P_n.

Where:

P_1 = conditional probability of a second error,
P_2 = conditional probability of a third error,
!
P_n = conditional probability of n+1 error, and

P_i = 0.15 in case of medium dependency,
P_i = 0.5 in case of high dependency, or
P_i = 1 in case of complete dependency.

Quantification of pre-accident errors is accomplished using the following formula:

$$P = P_b \text{ H } P_{NR}$$

Where:

P_b = a basic value estimated at 3E-2 on the basis of Électricité de France experience feedback, and
P_{NR} = the probability of non-recovery.

The latter was estimated by dividing the situations into four categories on the basis of the elements conducive to recovery (see Table 6-12). This table was prepared using engineering judgement and information obtained from other operators.

Table 6-12 Probability of non-recovery of errors prior to accident

Class	Elements Favouring Recovery	P_{NR}
1	Category 1 alarm window	0
2	Large change in the value of a parameter recorded during each shift Requalification enabling the anomaly in question to be effectively detected Administrative lock-out	0.01
3	Periodic test of a frequency of one month or less Anomaly detectable by verifications planned during standard state changes (depending on instructions GP, GS, E, etc.) Indication of position in control room Alarm of category other than 1 (on the screen)	0.1
4	None of the above factors	1

In certain cases, it is also possible to introduce the possibility of recovery of pre-accident errors during the accident. Such recovery is only introduced in exceptional cases, where justified by the context and the weight of the event. Here quantification is carried out using the principles described in the preceding paragraph.

The post-initiator intervention category includes both diagnostic and decision-making errors and errors in taking action. In addition, distinctions are made among the roles of the operating team, the safety engineer and the action of the emergency teams. This provides a mechanism to represent the manner in which operation in an accident situation is organized, particularly the human redundancy provided by the safety engineer, and allows for a different quantification process for each role. Human intervention in accident situations was identified during preparation of the event trees by analyzing the operating procedures and making allowance for actual incidents and observations made on simulators.

The activity of the operating team is divided into two phases (i.e., diagnostic and operation).

The diagnostic phase includes detection of the incident, making the diagnostic, and making a decision (selection of a procedure or policy). These operations are quantified using curves that give the probability of failure as a function of the time available to the operators to take action (see Figure 6-2). These curves, which respectively correspond to an easy, an average, and a difficult situation, were prepared on the basis of simulator tests, supplemented by engineering judgement, making special use of Swain's work. The easy curve corresponds, for instance, to cases where the symptoms are clear and unambiguous and there is a suitable procedure. The difficult curve corresponds to cases where the symbols are unclear and do not correspond and there is no suitable procedure (combination of accidents for example).

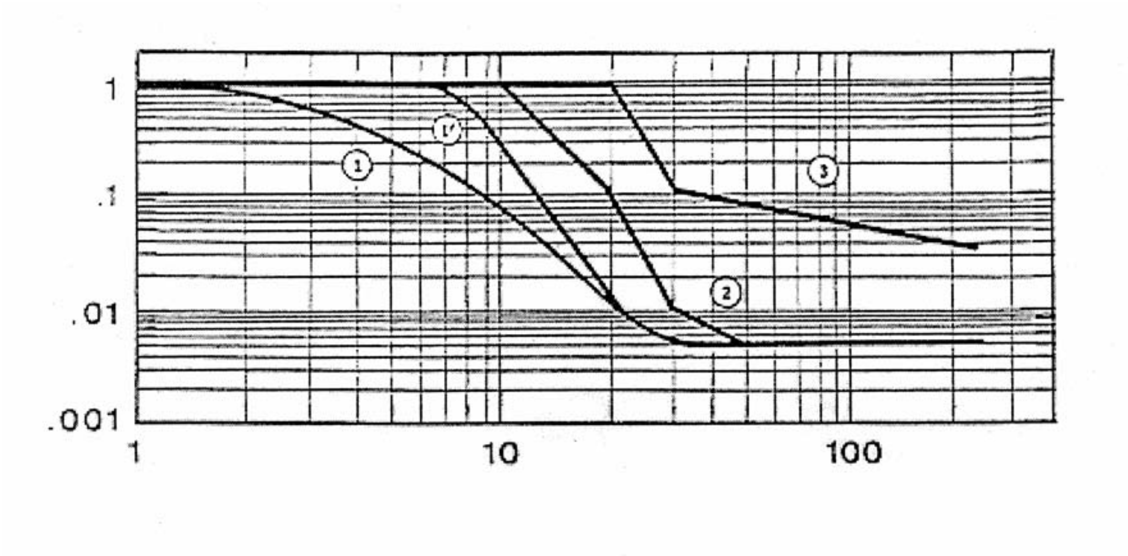


Figure 6-2 Diagnostic Model — Probability of Diagnostic Failure

Operation in accident situations can be represented by a list of key actions, whose success or failure modifies the accident sequence. In the event of failure of the diagnostic, it is generally considered that key actions are not performed. In the event of success of the diagnostic, the corresponding actions may fail (omissions, mistakes or unsuitable action). For each important action, the probability failure is evaluated as using the following formula:

$P = P_b \ H \ K_i \ H \ P_{NR}$

Where:

P_b = the basic probability estimated at 6E-2 (test on simulator),

K_i = context factor equal to 1/3, 1 or 3 depending on whether circumstances are favourable, average or unfavourable, and

P_{NR} = probability of non-recovery, estimated using elements favourable to recovery. (See Table 6-13)

It should be noted that these models are only used where there are no direct statistics specifically applicable to the situation being analyzed.

Safety engineer intervention represents recovery of degraded situations by application of procedures SPI and U1. It is assumed that the safety engineer does not play a role in the diagnostic and the actions taken by the operating team. The recovery failure of the safety engineer is the sum of two terms:

- the probability that the safety engineer will be absent from the control room, and
- the probability of failure to apply procedure SPI and U1 actions.

Table 6-13 Probability of non-recovery of errors in accident situations

<table>
<tr><th>Description of Error</th><td rowspan="2"></td><th colspan="2">P_{NR}</th></tr>
<tr><td>Irreversible errors non-detectable from the control room</td><td colspan="2">1</td></tr>
<tr><td rowspan="5">Errors recoverable from the control room</td><th>Recovery Factor</th><th>Time < 30 min</th><th>Time > 30 min</th></tr>
<tr><td>No recovery factor</td><td>0.6</td><td>0.6</td></tr>
<tr><td>Explicit redundancy</td><td>0.3</td><td>0.1</td></tr>
<tr><td>Signal</td><td>0.3</td><td>0.03</td></tr>
<tr><td>Signal and explicit redundancy</td><td>0.1</td><td>0.03</td></tr>
<tr><td rowspan="2">Errors recoverable locally</td><td rowspan="2"></td><th>Total time[1] < 30 min</th><th>Total time[1] > 30 min</th></tr>
<tr><td>1</td><td>3E-2 (upgraded on a case-by-case basis)</td></tr>
</table>

[1] Detection time plus execution time.

The probability of absence is estimated by Électricité de France using the following probabilities:

- paging of safety engineer (simulator tests), and
- reliability of means of paging and the time taken to reach the control room (on-site investigation).

The probability of failure in applying procedure SPI and U1 actions are as follows:

- SPI action: P = 5E-2,

- U1 action: available time > 30 min, P = 1E-1, and
- U1 action: available time < 30 min P = 3E-1.

It was postulated that the local emergency team would be operational four hours after the start of the accident (first symptoms of a severe accident situation). Policy errors beyond this deadline are not considered. However, allowance is made for non-recoverable execution errors.

When there are several human actions in the same accident sequence (in the same branch of the event tree) dependencies can be introduced among the actions. Quantification of dependent actions is based on Swain's model. Depending upon the success or failure of a given action A, this model generates the conditional probability of an error B in accordance with the estimated independent probability of errors A and B given a specified level of dependency between the events.

Despite efforts in this field, allowance for the human factor in the French probabilistic safety assessments still remains somewhat subjective. To reduce this subjectivity, the methodology makes maximum use of experience feedback, and makes a number of relative assessments by comparing different situations. Nevertheless, the human factor, which is predominant in the results, remains one of the major causes of uncertainty.

For shutdown situations the human factor is particularly important, because many automations are unavailable and the actuation of safety systems relies on operator intervention. Moreover, the emergency operating procedures were less developed for shutdown modes than for full power operation.

Although no specific model was developed for the LPSD PSA, several particular problems appeared during the course of the study.

For example:

- The problem of omission or commission errors in the application of normal operating procedures. This problem appears especially for boron dilution scenarios.
- The problem of long delays for recovery actions. What is the effect of a delay of several hours compared to a delay of one hour?
- The problem of dependencies between human errors. Several sequences include more than one human intervention, and the effect of dependencies is critical.
- There are very few simulator experiments representative of shutdown situation, and consequently, very few reliable data.

6.6.3.2 Ongoing Activities

In the framework of the PSAs updating, the updating of human factor probability assessment is in progress using the same methodology as in the past studies. In particular, it can be noted that the HRA updating takes into account the new symptom-oriented emergency procedures.

6.6.3.3 Planned Activities

For the PSA carried out for the N4 plants, EDF develops a new HRA methodology. This advanced HRA method (MERMOZ method) is not specific to shutdown situations, but will be used for all the HRA problems in the future PSA.

6.6.3.4 Future Needs

Improvement of HRA method is needed for taking into account the problems specific to shutdown situations. As already indicated above, these problems include in particular:

- Omission or commission errors in the application of normal operation procedures
- The recovery actions with important intervention delays
- The dependencies between human actions
- The lack of representative simulator experiments

6.6.4 Germany

6.6.4.1 Currently Available Information

The German PWR LPSD PSA considers human actions, which are:

- procedure-driven or
- memorised with sufficient occasions for refreshment (i.e., during normal operation or re-qualification).

Consequently only those actions were considered, which would be classified as skill- or rule-based behaviour. Available methods to evaluate knowledge-based behaviour are not sufficiently developed and proved.

Taking into account this limitation, operator actions were investigated which:

- initiate an event as a result of an error,
- actuate active safeguards after the occurrence of an event and the recovering of failed active safeguards.

Emphasis is given to qualitative assessment supported by interviews, walk-/talk-through, task modelling and task analysis.

The THERP method was used to quantify human reliability.

6.6.4.2 Ongoing Activities

Presently, the following activities are ongoing:

- Establishing a quantitative human reliability data base derived from available operational experience (development and application of the method.)

- Models for prediction and evaluation of cognitive errors.

6.6.4.3 Planned Activities

Assessment of human actions will be performed within the planned LPSD PSA for BWR-69.

6.6.4.4 Future Needs

Extension of HRA methodology taking into account knowledge-based behaviour, cognitive errors and data based on operational experience.

6.6.5 Hungary

6.6.5.1 Currently Available Information

A classification of the human interactions was made as follows:

- Type A--Human actions occurring 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 occurring after the start of an accident sequences (post-initiator actions) are actions performed in the response to an accident by the plant personnel to operate standby equipment or to recover equipment that can be used to mitigate the accident. These types of human errors cover the mitigation of the accident caused by an initiating event.

For Type A human actions, the modelling and quantification approach consisted of two main tasks:

1. Determination of preliminary HEP values by using screening values, and

2. Use of some feedback from plant experience to enhance the preliminary HEP values.

To start, the HRA analysts constructed a model of the maintenance and test process during plant outages. Discussions were held with plant personnel regarding this model to ensure that it reflected the maintenance process at the Paks NPP. It included, in its first phase, the system of work planning, work orders, and work permits. The next phase was maintenance error generation, which leaves the equipment in a degraded state. As a result of inspections, the number of maintenance errors is reduced. After the maintenance operation, operational staff carried out functional tests to check the equipment. As a result of these functional tests, the number of latent errors present is reduced considerably. Following the functional tests, other tests are organized during the start up phase, again reducing the number of latent errors. Each of these tests and checks can result in recovery actions, reducing the failure probability.

A composite preliminary value of the HEP was produced for each system train modelled in the PSA. Completed data sheets for the various systems were used to identify, from POS to POS, what component failure modes could be caused by human errors. Table 6-14 provides an example for the identification of the maintenance actions and tests for one train of the LP ECCS. The composite number was 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.

Table 0-14 Example for the identification of the maintenance actions and tests for equipment failure modes during outage

Basic Event		POS´s																								
Identifier	Description											Change of the Safety System Train						Startup Tests			LSL-2 Test			LSL-1 Test		
		0	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24
20TJ10B001-RPO	Water Storage Tank Rupture						E	E	E	E	E	M														
20TJ10L101	"LP tank level is smaller than 500mm" signal not occured; interlock TJ2.1						E	E	E	E	E	K														
20TJ10L501	"LP tank level is smaller than 500mm" signal not occured; interlock TJ2.1						E	E	E	E	E	K														
20TJ10L502	"LP tank level is smaller than 500mm" signal not occured; interlock TJ2.1						E	E	E	E	E	K														
20TJ10P000-RP-	Pipes Rupture						E	E	E	E	E	M														
20TJ10S201	left closed						E	E	E	E	E	Ta						Tr						Ta		
20TJ10SQ01-CCS	Check Valve Fails to Open (TI=24h)						E	E	E	E	E	Ta						Ta						Ta		
20TJ11N001-PG-	Filter plugged						E	E	E	E	E															
20TJ11S201-CCS	MOV Fails to Open						E	E	E	E	E	K						Tvr								
20TJ11W001-PG-	Heat Exchanger Plugged						E	E	E	E	E	M														
20TJ11W001-RPI	Heat Exchanger-Pipe Rupture						E	E	E	E	E	M														
20TJ11W001-RPT	Heat Exchanger-Tank Rupture						E	E	E	E	E	M														
20TJ12D001-FRO	LP ECCS Pump Fails to Continue Running						E	E	E	E	E	K														
20TJ12D001-FSS	LP ECCS Pump Fails to Start						E	E	E	E	E	Tv						Tv			Tv			Tl1		
20TJ13F501	interlock TJ 1.2 is not operated						E	E	E	E	E	K														
20TJ13F502	interlock TJ 1.2 is not operated						E	E	E	E	E	K														
20TJ13S801-CCS	Air Operated Valve Fails to Open						E	E	E	E	E	K						Tvr			Tv					
20TJ13S802-CCS	Air Operated Valve Fails to Open						E	E	E	E	E	K						Tvr			Tv					
20TJ13S803-CCS	Air Operated Valve Fails to Open						E	E	E	E	E	K						Tvr			Tv					
20TJ13S804-CCS	Air Operated Valve Fails to Open						E	E	E	E	E	K						Tvr			Tv					
20TJ13SQ01-CCS H	Check Valve Fails to Open						E	E	E	E	E	K									Ta					
20TJ13SQ02-CCS H	Check Valve Fails to Open						E	E	E	E	E	K														
20TJ14SQ01-CCS H	Check Valve Fails to Open						E	E	E	E	E	K														
20TJ15S201	recirculation test line left opened						E	E	E	E	E	Ta						Tj			Ta			Ta		
20TJ16S201-OOS H	MOV Fails to Close						E	E	E	E	E	K						Tvr								
20TJ16S202-OOS H	MOV Fails to Close						E	E	E	E	E	K						Tvr								
20TJ40S201	left opened						E	E	E	E	E	M						Tr								
20TJ41S201	left opened						E	E	E	E	E	M														

During the system analysis process, four basic types of maintenance or test related tasks were identified that are susceptible to a human error. Evidence from a number of sources indicated that the HEP range for latent errors under these conditions ranged from 1E-2 to 1E-3. The effect of checking and testing was to reduce the error rate to the lower HEP value. From discussions with Paks personnel, it was concluded that the HEP 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. Thus, the base numbers were increased by a factor of 3. This value was chosen by experienced 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 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 the system line-up.
- There is no re-configuration after a system alignment because
  - there is no procedure for re-alignment
  - 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. They are as follows:

- The main focus was on the maintenance and test of mechanical equipment; thus, preliminary HEP values for associated I&C and electrical faults were conservative. However, it was noted that a number of tests are performed for I&C and electrical part that can reveal maintenance errors, especially for errors that can cause shared equipment unavailability (e.g. loss of a bus-bar). Even though there may be a greater potential for making an I&C or electrical maintenance error, the potentials for detecting such errors were seen to be better than for those associated with the mechanical equipment, since a higher the level of dependence increases the chances are for detecting an I&C or electrical fault during subsequent tests of multiple components and interlocks. Nevertheless, in the long term further investigation is needed to justify this assumption. Also, I&C and electrical maintenance must be addressed in both the event analysis and in the construction of a general decision tree framework.

- If some of the human errors can be detected by subsequent test or checks during the outage, a recovery factor was used. If appropriate, the effect of recovery was taken into account at component level. 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 related to low power and shutdown modes that involved plant personnel were analyzed. Results of the analysis were used to correct the estimation of the probability values of the four main error types.

The Type B human actions are discussed in the section describing determination of initiating events and their frequencies. Type C human actions are discussed in the following paragraphs.

For the analysis of post-accident (operator) errors, a so-called decision tree approach was used. This approach was based on a generalization of the decision tree method developed and applied in the PSA for the 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 different from the full power accidents, which required extensions to the existing decision tree model. Examples of some of these 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 10 hours. Such extra long response times have not been addressed in the full power PSA.

- In general, the emergency operating procedures are not 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 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 the influences in order of importance and select the most important ones.

4. Select the number of branches per performance influence and draw a 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 the HEP.

7. Calculate the HEP distribution.

8. Check the 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 Paks Unit 2,
- interviews with plant operators, and
- an analysis of 156 safety related events that occurred during outages at the four units of Paks.

The HRA model development process consists of four important steps. These are:

10. Identification of the most important scenario dependent performance influences,

11. Choice of anchor values for calculating human error probabilities,

12. Construction of the decision tree, and

13. Checking the HEP estimates.

During responses to accidents starting from a low power or shutdown state, the most important performance influences have been found to include:

- time available to take action,
- crew knowledge (level of training) of the situation,

- workload (distraction),
- quality of man-machine interface, and
- availability and quality of emergency operating procedures.

The definition of the so-called anchor HEP value is an important element in calculating the 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-4, i.e. the reference value 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 time.

Considering the assumptions and reasoning outlined above, a decision tree has been drawn and the associated HEP values have been calculated for each pathway in the tree. The decision tree is composed of five headings as defined by the performance influences, and the number of pathways in the tree is 144. 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. For such situations, 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.

6.6.5.2 Ongoing Activities

New, symptom-based emergency operating procedures are being developed the Paks NPP. They should cover a wider scope of accident situations, including those starting from shutdown states.

6.6.5.3 Planned Activities

Since the risk associated with the maintenance activities appeared to be significant, a more detailed, specific analysis of human factors and human reliability during shutdown will be performed. The main objectives of the analysis will be to:

- evaluate the safety influencing factors related to the human activities performed during the shutdown, refuelling, and start-up phases based on real operational data

- decrease the possibility of events with the involvement of plant personnel

- develop methods of reporting, investigating, evaluating, and documenting events with the involvement of plant personnel

- re-evaluation of the role of human factors using PSA.

6.6.5.4 Future Needs

The effect of introducing symptom-based emergency operating procedures on the risk should be evaluated.

6.6.6 Japan

6.6.6.1 Currently Available Information

Human error probabilities are estimated using the THERP method in NUREG/CR-1278. The techniques used to estimate human error probabilities during LPSD are the same as those used for rated power operation analyses. Human errors are divided into two parts as shown in Table 6-15.

Table 6 – 15 Human errors

Human Actions	Examples	Probability
Pre-accident	Miscalibration of instrumentation	6.4E-4/d
	Failure to restore equipment after maintenance and/or test	6.5E-5/d
Post-accident	Plant diagnostic	1E-1/d to ~1E-4/d
	Manual initiation of ECCS	Less than 1E-7/d

Dependency between operating personnel is treated as follows:

- Operator - assistant shift supervisor High dependence
- Operator - shift supervisor Moderate dependence

6.6.6.2 Ongoing Activities

On-going activities include estimating human errors for BWR-4 type plants using the same method that was used for the BWR-5 plants. Further, human reliability analysis is being conducted based on THERP for typical PWR 2-loop plants similar to the analysis conducted for the PWR 4-loop plant. However, some recovery factors for the diagnosis error are given credit for considering some obvious changes in the accident progression, such as secondary alarm and the actuation of RHR relief valve.

6.6.6.3 Planned Activities

New methods are being sought for estimating plant diagnostic error. ATHEANA will be introduced for use in LPSD.

6.6.6.4 Future Needs

None.

6.6.7 Korea

Currently Available Information

THERP(NUREG/CR-1278) and ASEP HRA Procedure(NUREG/CR-4772) are used for quantifying human actions modelled in Low Power/Shutdown PSA of YGN 5&6 NPPs. ASEP HRA Procedure is used for only determining the stress level of human actions. In POS 1, 2, 14, 15 of YGN 5 & 6 NPPs, which are similar to full power operation, the results of HRA used in full power PSA of YGN 5 & 6 NPPs will be used. ASEP HRA Procedure is used for HRA of full power PSA of YGN 5 & 6 NPPs.

Work sheets are developed to increase the plausibility and credibility of the quantification process of human actions. And, decision trees for quantifying dynamic human actions are developed to allow a human reliability analyst to perform a systematic and consistent HRA. The performance shaping factors considered in HRA for Low Power/Shutdown PSA are available time, stress level, level of operator training/experiences, existence of hesitancy, type of task, failure of makeup at mid-loop POS, and etc.

Three kinds of dependencies are considered as follows:

1) Dependencies between the activities that make up an action,

2) Dependencies between operators,

3) Dependencies between various parallel human actions in accident sequences; (e.g. fails to shutdown cooling operation using standby shutdown cooling pump or containment cooling water system).

The level of dependencies between the activities is evaluated as complete or zero. The dependencies between operators are explicitly considered in the quantification of recovery probability of execution error. There is no general rule in THERP to treat the dependencies of multiple human actions in accident sequences. The dependency level between multiple human actions having same goal in accident sequences is determined on mainly the cues of operator actions, time difference between two human actions, structures of procedures, and interviews with operators.

Ongoing Activities

The HRA for Low Power/Shutdown PSA of YGN 5 & 6 NPPs is being conducted.

6.6.7.3 Planned Activities

ATHENA will be applied to the quantification of a few commission errors during Low Power/Shutdown operation for sensitivity study.

6.6.7.4 Future Needs

None.

6.6.8 South Africa

Currently Available Information

The Koeberg PRA Human Reliability Analysis Methodology is based on NUREG/CR-1278, and updated sections in NUREG/CR-4772. In the applying the methodology special emphasis is placed on the effects of dominant estimation variables. Past experience has shown that in many instances most of the spread and variability of probability estimates evident in human reliability analysis results are attributable to a few dominant variables. Of these possibly the most dominant is that of uncertainty.

Human Error Probabilities (HEP) typically have large associated uncertainties. To make matters worse generic point estimate values do not account for uncertainties. From a statistical viewpoint some of these uncertainties are explained and accounted for by assuming an underlying lognormal distribution where the standard deviation from the central value effectively becomes an expression of uncertainty. The range between the upper and lower bound HEP values in NUREG/CR-1278, for example, reduces the uncertainty that true HEPs may fall outside estimated values. For PRA HEP estimation where human errors potentially have major consequences the upper bound or another similarly conservative value may be more appropriate than a central value depending upon the degree of risk tolerance or confidence that is required.

Plant specific HEPs derived from simulator observations of initial senior reactor operator licensing examinations are found to be significantly different from corresponding generic values used previously. The combined effect of several of the former values in accident sequence evaluations is expected to affect overall PRA results significantly. In principle underlying lognormal distributions can also be assigned to plant specific HEPs, although at present small sample sizes precludes substantiation of distribution characteristics. HEP range values are used rather than point estimates to limit uncertainty.

6.6.8.2 Ongoing Activities

Further development of plant specific HEPs.

6.6.8.3 Planned Activities

None

6.6.8.4 Future Needs

None

6.6.9 Spain

6.6.9.1 Currently Available Information

The Vandellos 2 LPSD PSA involved a very simple consideration of human failure events. Only two values for human error probabilities were considered, according to whether the human actions were considered to be "simple" or not. This approach was taken from the vendor studies.

The Asco PSA involved a much more thorough analysis. 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 does the full power PSA (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, as proposed by consultants. The main consultant was part of the team that was developing the ATHEANA framework at that time. Some objections to this approach were raised by the CSN review, mainly related to the calibration and consistency between both models. However, it is recognized that substantial progress has been achieved in developing this framework since the first LPSD PSA.

6.6.9.2 Ongoing Activities

This task is part of every Spanish LPSD PSA project and, although each specific case tries to solve the difficulties involved with the task, task procedures are expected to improve from project to project. A research project is being started in Spain, co-sponsored by the CSN and the joint utility organization (UNESA), with the objective of developing and validating methodologies for the treatment of human errors of commission. Eventually, progress from this project will be used for future LPSD PSAs in Spain, as the impact of this type of errors is particularly important during LPSD operation.

6.6.9.3 Planned Activities

Perform this task for each of the remaining LPSD PSA Projects. Use products from the ongoing research project, as they become available.

6.6.9.4 Future Needs

This is an area where research is most clearly needed, since LPSD risk is dominated by human errors, consequently HRA methodological shortcomings become important.

6.6.10 Chinese Taipei

6.6.10.1 Currently Available Information

The modified ASEP method is used for HRA.

6.6.10.2 Ongoing Activities

None.

6.6.10.3 Planned Activities

None.

6.6.10.4 Future Needs

None.

6.6.11 United States

6.6.11.1 Currently Available Information

For Grand Gulf, a detailed human reliability analysis was performed for the Phase 2 study of POS 5 (i.e., cold shutdown during refuelling). The Accident Sequence Evaluation Program Human Reliability

Analysis Procedure ASEP HRAP. (NUREG/CR-4772) was the general methodology used for conducting and determining the human error probabilities for the identified human actions.

The ASEP HRAP was closely adhered to for determining basic human error probabilities for each human action and was adjusted according to the rules for applying performance-shaping factors described within the procedure. Deviations from the prescribed methodology were taken only when it was felt that the low power and shutdown environment created a situation that was not well handled by the ASEP HRAP procedure. For example, long-term scenarios occurring during shutdown were rarely assessed "extremely high" stress by the human reliability analyst, even when the procedure may have called for it, as in the case of more than two primary safety systems having failed.

For Surry, two types of post-accident human errors were modelled either in the fault trees or as the top events of the event trees. The two types included failure to diagnose and failure to carry out the needed action given successful diagnosis. To evaluate the actions, the event scenario, required actions, important factors affecting operator performance, and the consequences of the action being unsuccessful were qualitatively defined. It was assumed that, given failure to diagnose, the operator would fail to perform the needed actions; therefore, core damage would result. The same basic event representing failure to diagnose was used in all fault trees for a given event tree. Failure to carry out the action given successful diagnosis, on the other hand, would only fail the specific top event of the event tree.

6.6.11.2 Ongoing Activities

None.

6.6.11.3 Planned Activities

None.

6.6.11.4 Future Needs

None.

6.7 Equipment Performance and Recovery

Estimating the ability of equipment to function in accident situations is one problem frequently encountered in the study of accident sequences.

Some guidance on how to recover or even replace equipment should be incorporated into some type of plant operating (or emergency) procedures for the most risk significant scenarios.

The US based Licensee Event Report data have been used to establish the loss of support system failure frequencies and to estimate the recovery time distribution associated with each initiator.

6.7.1 Czech Republic

6.7.1.1 Currently Available Information

The incorporation of recovery action into LP&SD model for NPP Dukovany has been done in accordance with IAEA Safety Series No. 50-P-4. So only manual realignment of a valve to heal the most important minimal cut set has been modelled. The possibility of equipment repair which put it into operation has not been considered so far.

New symptom-oriented EOPs implemented at NPP Dukovany have been developed primarily for full power operation. They are not applicable for possible actions used to mitigate most of accidents during shutdown operation, especially during reactor cooldown and refuelling. So it has been necessary to add several remedy actions into the LP&SD model which are credited just based on applicability of action to mitigate the accident, on operational practices and training of plant staff, as well as on enough available time to determine such decision.

6.7.1.2 Ongoing Activities

Post accident human errors are currently being reassessed to reflect those new full power EOPs implemented at NPP Dukovany which are applicable also for LP&SD operation.

6.7.1.3 Planned Activities

Assessment of post accident human errors to reflect new shutdown specific symptom-oriented EOPs is expected in the future.

6.7.1.4 Future Needs

Analysis of recovery actions and methodology for incorporation of recovery actions into LP&SD PSA are needed.

6.7.2 France

6.7.2.1 Currently Available Information

Estimating the ability of equipment to function in accident situations is one problem frequently encountered in the study of accident sequences. The positions adopted in the probabilistic safety assessment are described as follows:

- Any item of equipment operating outside its qualification conditions is considered to be faulty, except if there is sufficient evidence (tests or studies) to qualify this hypothesis. For example, when temperature and pressure conditions in a containment exceed the qualification limits of the instrumentation, the latter is considered to be lost. However, for the primary pump seals, a more realistic model has been adopted.

- Furthermore, all equipment supposed to be qualified is considered to be so, and its reliability parameters are the same as under normal conditions. No allowance is made for the effect of incomplete or inadequate qualification. This assumption may be investigated in subsequent applications of the probabilistic safety assessment.

To make sure the scenarios are realistic, particularly in long-term sequences, the possibility of recovery was allowed. Recovery may consist of repairing the system (particularly when the initiating event is the failure of the system) or human intervention to apply a procedure or implement a remedial strategy. The following three cases describe how recovery was considered:

- For repair of a system, recovery is treated by introducing a mean repair time for the repairable components and by using an exponential law. The quantification method used in the PSA makes this possible.

- For recovery of a human error, recovery by the operators or by the safety engineer is treated with the human factor events as described in the preceding section.

- For recovery of degraded situations, recovery implies a human intervention to apply a procedure or implement a remedial action. In general, the success or failure of recovery has been introduced into the event trees in the same manner as the success or failure of the engineered safety feature systems.

6.7.2.2 Ongoing Activities

None.

6.7.2.3 Planned Activities

None.

6.7.2.4 Future Needs

None.

6.7.3 Germany

6.7.3.1 Currently Available Information

In the PWR analysis recovery was considered in the case of a loss of RHR due to faulty level lowering. The faulty level lowering leads to a suction of nitrogen of the operational RHR trains. As a recovery action the de-aeration of the affected trains was considered in case of a failure of the redundant trains.

6.7.3.2 Ongoing Activities

None.

6.7.3.3 Planned Activities

Recovery actions will be considered in the PSA for the BWR-69.

6.7.3.4 Future Needs

In the future LPSD PSA the repair of components should be considered.

6.7.4 Hungary

6.7.4.1 Currently Available Information

Twenty-four plant operational states were identified for the low power and shutdown operational modes of the Paks NPP. The POS 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 by taking into account the list of all the potential initiating events in the given POS. Possible operational modes for these systems during the shutdown period were identified using the Reactor Shutdown Procedure as a basis, along with 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. The true unavailability of the safety related systems—owing to maintenance works performed on them, as well as, their changing unavailability caused by partial or full tests following the maintenance—were also identified. Changes in the automatic operation of equipment were also considered.

The initial scope of the post-initiator human interactions was determined based on the existing Emergency Operating Procedures. Owing to the fact that these procedures cover the accidental situations starting from shutdown states only to a very limited degree, they could not be considered sufficient for the identification of post-initiator actions. Thus, the scope of the post-initiator human interactions was supplemented by a number of recovery actions whose credibility was determined by taking into account the training and the operational practice of the operating personnel, the available time for the mitigation of serious consequences after an initiating event, and the potential to call in other personnel (e.g. shift supervisor). The information used for the identification of these actions included the following:

- some results from the human reliability analyses performed within the PSA studies for the nominal power operational mode,
- knowledge about the operational practices and the training of the operating personnel gained during earlier PSA studies, and
- results from discussions with the operating and maintenance personnel.

The errors associated with the identified recovery actions were then analyzed (i.e., modelled and quantified) using the same approach, namely the so-called decision tree approach, as used for the post-accident errors.

6.7.4.2 Ongoing Activities

New, symptom-based emergency operating procedures are being developed the Paks NPP. They should cover a wider scope of accident situations, including those starting from shutdown states.

6.7.4.3 Planned Activities

None.

6.7.4.4 Future Needs

The effect of introducing symptom-based emergency operating procedures on the risk should be evaluated.

6.7.5 Japan

6.7.5.1 Currently Available Information

Recover of offsite power and diesel generators are taken into account for PWRs. The recovery credits for core inventory makeup systems were considered for a BWR-5 plant. Such systems are LPCI, LPCS, and MUWC. The repair times for offsite power is based on the Japanese operating experience and that for failed components are based on the WASH-1400 data. For a PWR 4-loop plant, only the offsite power recovery was considered.

6.7.5.2 Ongoing Activities

The same method is used for the estimation of typical BWR-4 /3 type plants as for a BWR-5 plant. For PWR 2/3-loop type plants, same method is used as for a PWR 4-loop plant.

6.7.5.3 Planned Activities

If the mitigating methods discussed in 6.1.4.3 are judged effective, they will be accounted by recovery factors.

6.7.5.4 Future Needs

None.

6.7.6 Korea

6.7.6.1 Currently Available Information

None.

6.7.6.2 Ongoing Activities

None.

6.7.6.3 Planned Activities

In Korea, the approach described in NUREG/CR-4550 was used for recovery analysis of full power PSA of PWR. Therefore, same approach will be used for recovery analysis of Low Power/Shutdown PSA.

6.7.6.4 Future Needs

None.

6.7.7 Spain

6.7.7.1 Currently Available Information

No recovery of equipment is modelled in any of both Spanish LPSD PSAs up to now. Pump cavitation is considered as a cause of RHR losses for this type of scenarios and cavitation is considered as certain for scenarios with sufficient water level decrease in the RCS. The different effects of continuing operation or stopping the pump are not analyzed yet.

6.7.7.2 Ongoing Activities

No special activities are being carried out in this area

6.7.7.3 Planned Activities

There are no planned activities in this area. Recovery is considered as the final step in the model. Any modifications to decrease risks from scenarios should first be preventative in nature.

6.7.7.4 Future Needs

Some guidance on how to recover or even replace equipment should be incorporated into some type of plant operating (or emergency) procedures for the most risk significant scenarios.

6.7.8 Chinese Taipei

6.7.8.1 Currently Available Information

For recovery of loss of RHR events, the recovery time is considered as the time that the RCS water level goes down to the level for RHR successful running. And, for a successful recovery, it is also required to recover water level to normal position.

For recovery of FMEA events, the recovery time is considered as the time to RCS boiling. The data of mean time to repair is used to estimate the failure probabilities of recovery.

For recovery of electric power, recovery of electric power, including the recovery of offsite power and diesel generator failure, and the backup of gas turbine and DG-5, is almost the same way used in full power PSAs. The recovery time was considered up to 48 hours.

6.7.8.2 Ongoing Activities

None.

6.7.8.3 Planned Activities

None.

6.7.8.4 Future Needs

None.

6.7.9 United States

6.7.9.1 Currently Available Information

Grand Gulf used the same process described in NUREG/CR-4550 to estimate recovery of offsite power, taking into account LPSD events appropriate for the POS being analyzed. NUREG/CR-6143 documents results for the following:

- recovery of offsite power,
- recovery of diesel generator failures (i.e., fail to start, fail to run),
- recovery of offsite power and diesel generator failures, and
- recovery of dc bus failures.

Surry used the same computer code for determining recovery of offsite power as was used in the NUREG/CR-4550 study. The Licensee Event Report data used to establish the loss of support system failure frequencies also were used to estimate the recovery time distribution associated with each initiator. The recovery curves were derived from all events for which recovery data existed. A mean recovery curve was obtained by assuming that the recovery time is log distributed.

6.7.9.2 Ongoing Activities

None.

6.7.9.3 Planned Activities

None.

6.7.9.4 Future Needs

None.

6.8 Scope and Success Criteria

An examination of available fire and internal flood risk analysis methodologies to determine whether enhancements can be made to allow these analyses to be performed in a more cost effective manner is needed, since these types of risk are believed to be significant and performance of these analyses should allow safety enhancements to be identified.

Current LPSD analyses are Level 1 PSAs for internal initiators only. Some information is available regarding the status of containment main locks, but only from a thermal-hydraulic point of view. Since an open containment can be a characteristic of LPSD scenarios, this allows the more risky scenarios, from a radionuclide release point of view, to be easily identified.

The issue of risk from radionuclide release from sources other than the reactor vessel warrants attention. Specifically, risk analysis of the refuelling phase is sometimes excluded based on the fact that there is no fuel in the reactor vessel.

The success criteria for shutdown conditions were determined by reviewing various studies and performing plant-specific thermal-hydraulic analyses. The changing level of decay heat was accounted for by defining time windows after shutdown each with its own set of success criteria. Three functions must be successfully provided to mitigate a LPSD accident:

- reactivity control,

- level control, and
- energy removal.

In general, whenever the success criteria for one system or mitigating function change, a new time window should be defined.

6.8.1 Czech Republic

6.8.1.1 Currently Available Information

LP&SD PSA for NPP Dukovany analyzes very broad scope of risk sources, outages as well as initiators. All LP&SD states of NPP Dukovany below 55% of nominal power are covered. They include:

- refuelling outage (both partial and complete refuelling)
- planned maintenance outage (cooling & service water pump station inspection)
- unplanned outage and power reduction (including Technical Specification request)

The following sources of risk have been analyzed:

- reactor core
- spent fuel pool
- fuel transport (screened out)
- radwaste processing facilities (screened out)

The following risk measures limited to Level 1 PSA have been selected for quantification:

- core damage
- boiling in open reactor
- fuel damage (it includes, in addition to core damage, the damage of fuel in spent fuel pool)

The risk from the following categories of initiating events have been analyzed:

- internal events (including LOSP)
- internal fires and floods
- heavy load drops and fuel drops during transportation

The success criterion for core or fuel damage is defined in term of the maximum fuel cladding temperature which shall not exceed 1200°C. However, in accordance with IAEA-EPB-WWER-09 the prevention of core uncovery has been used especially for shutdown states during cooldown and refuelling. It assures the compatibility with available thermal-hydraulic analyses for LP&SD states since up to this time point the accident scenarios are usually calculated. It is expected that after the core uncovery the fuel cladding temperature would rapidly increase to 1200°C. In addition to this, the subcriticality must be assured for reactivity transient as well as fuel integrity for heavy load drops to reactor or spent fuel pool as specified in e.g. IAEA-EPB-WWER-09.

6.8.1.2 Ongoing Activities

None.

6.8.1.3 Planned Activities

External events are expected to be incorporated into the NPP Dukovany PSA (including LP&SD PSA). The level 2 PSA should be extended into LP&SD states as well.

Future Needs

A methodology is needed for selecting ant treating external events in PSA.

6.8.2 France

6.8.2.1 Currently Available Information

The scope of PSA 900 and 1300 was limited to internal events. The shutdown situations were treated with an approach less sophisticated than that of the power state.

The purpose of PSA 900 and PSA 1300 is to evaluate the probability of core meltdown. In practice, the accident sequences have not been systematically analyzed to the point of core meltdown, but only to the point where core integrity can no longer be guaranteed. Examples of this simplification include the following conditions:

- prolonged dry-out of the core with no means of injection available,
- clad temperature above 1204 ^{o}C,
- exceeding primary system test pressure, and
- stored energy greater than 200 cal/g.

These simplified characterizations of core meltdown introduce a degree of conservatism.

6.8.2.2 Ongoing Activities

In the first stage, the study of sequences during shutdown was simplified. For example, the updating of sequences at mid-loop operation is in progress, taking into account the new water make up automatism and quantifying the long term sequences. Moreover, the level 1 PSA being completed includes other initiators, notably those induced by fire.

6.8.2.3 Planned Activities

It is foreseen to take into account the sequences that can occur during shutdown the state in the level 2 PSA.

6.8.2.4 Future Needs

None.

6.8.3 Germany

6.8.3.1 Currently Available Information

The objective of the BWR study was the development of a LPSD PSA methodology and the scope was limited to the analysis of relevant internal initiating events through Level 1. The success criteria are derived from thermal hydraulic calculations or engineering-based estimations. In the case of core cooling by steaming, nearly all available systems have enough capacity to feed the RPV. In case of a large leak in the RPV bottom head, only the RHR-trains have the capacity to overfeed the leak.

The objective of the PWR study was the systematic analysis of initiating events during LPSD conditions and the evaluation of the PSA methodology. The scope was also limited to internal initiating events and level 1. The determination of success criteria was performed with plant specific thermal hydraulic and hand calculation. A system damage state results from the failure of the operational and safety systems to ensure the fuel cooling. Accident management measures and repair of components is not considered. The fuel cooling may be also endangered by a large volume of unborated coolant, hence a second kind of system damage states results from the failure of the measures to prevent the accumulation of a large volume unborated water into the primary circuit.

6.8.3.2 Ongoing Activities

None.

6.8.3.3 Planned Activities

A Level 2 PSA for a BWR-69 is planned. The scope of the LPSD PSA will be limited to internal initiating events.

6.8.3.4 Future Needs

None.

6.8.4 Hungary

6.8.4.1 Currently Available Information

The main goals of the LPSD PSA study, in accordance with the Guidelines of the Periodic Safety Review, were the following:

- determination of the risk of core damage and boiling (in the states when the reactor open) originating from non-nominal power operational states,
- identification of the dominant factors affecting risk by sensitivity and uncertainty analyses, and
- identification of safety enhancement measures.

The scope of the study included the following:

- Unit 2 of the Paks NPP was selected as the reference unit - applicability of the results for other units was evaluated by comparing studies.
- Plant operational states analyzed included those that occur during a planned shutdown for refuelling (a short description of the characteristics of the plant operational states identified is given in Table 6-16).
- Internal initiating events, except internal fires and floods, resulting from hardware failures or human errors were studied.
- The reactor core is the potential source of the radioactive release.

Three end states have been identified and analyzed in the LPSD PSA study:

(1) success,

(2) core damage, and

(3) boiling in the core.

The latter relates to plant operational states where the reactor is open. The success criterion for this end state, avoiding boiling the core given atmospheric pressure, is obvious. The success criteria that must be met to avoid core damage were the same as those use in the PSA for the nominal power operational mode, namely:

- Emergency core cooling:
  - the maximum fuel rod cladding temperature does not exceed 1200 °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, and
  - changes in the core geometry are such that the core remains amenable to cooling.
- 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.

Table 0-16 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
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

Table 6-16 Plant operational states during the shutdown period (Continued)

Plant Operational State					
No.	**Description**	**Start**	**End**	**Estimated duration**	**Characteristics**
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
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

6.8.4.2 Ongoing Activities

None.

6.8.4.3 Planned Activities

The present results are for plant operational states occurring during a planned refuelling shutdown. Plans include extending the LPSD PSA study to other types of outages, e.g. shutdowns that terminate at the hot standby state or shutdowns resulting from accident situation initiated from the nominal power operational state.

6.8.4.4 Future Needs

Internal fires and floods as well as external hazards such as earthquakes should be analyzed for the low power and shutdown states.

6.8.5 Japan

6.8.5.1 Currently Available Information

The scope for BWR-5 type plants covers seven POSs from breaking condenser vacuum (POS-S) to plant start up (POS-D). The success criteria for each POS are decided based on the thermal hydraulic calculations. The POSs are described in Table 6-17. Typical success criteria for POS-S are presented in Table 6-18.

The scope for PWR-4 loop plant covers 19 POSs. The success criteria for each POS are based on thermal hydraulic calculations. The POSs used are shown in Table 6-19. Typical success criteria for RHR2 (loss of inventory at mid loop operation) are presented in Table 6-20.

6.8.5.2 Ongoing Activities

None.

6.8.5.3 Planned Activities

None.

Table 6-17 BWR-5 POSs

POS Identifier	Status (Period)	Water level	RPV head
S	Main condenser vacuum break to HPCS out of service	Normal	On
A	POS S to PCV/RPV open	Normal	On
B1	POS A to CUW out of service	Well full	Off
B2	POS B1 to RHR train switch (A–>B)	Well full	Off
B3	POS B2 to end of fuel shuffling	Well full	Off
C	POS B3 to PCV leak test	Normal	On
D	POS C to CRD withdrawal	Normal	On

Table 6-18 Typical success criteria for POS S

Initiating Event	Decay Heat Removal	Core Cooling
Loss of A train RHR	B train of RHR	HPCS, DEP & (one of LPCS, LPCI-B, LPCI-C, or MUWC)
Loss of Offsite Power	one of two RHR trains	HPCS, DEP & (one of LPCS, LPCI-A, LPCI-B, LPCI-C, or
Large LOCA	N/A	HPCS, LPCS, LPCS-A, LPCS-B, LPCI-C, or MUWC

Table 6-19 PWR 4-Loop POSs

POS No.	Plant Operating State Description	Duration Time (h)
1	Low power operation and reactor shutdown	9
2	Cool down with SG	7
3	Cool down with RHR	3
4	Cool down with RHR (Pressuriser solid)	97.5
5	Drain RCS to midloop	28.5
0.25	Midloop operation (SG nozzle dams are not installed	1
6AB	Midloop operation (All of SG nozzle dams are not installed)	77
6B	Midloop operation (All of SG nozzle dams are installed)	21.5
7	Fill for refuelling	14.5
8	Refuelling	307.5
9	Drain RCS after refuelling	14.5
0.41667	Midloop operation (All SG nozzle dams are not installed)	7
10AB	Midloop operation (All of SG nozzle dams are not installed)	79
10B	Midloop operation (All of SG nozzle dams are not installed)	65
11	Refill RCS completely	379
12	Cool down with RHR connected (Pressuriser solid)	14
13	Cool down with RHR connected	2
14	Cool down with SG	14
15	Low power operation and reactor shutdown	223

Table 6-20 Typical front-line success criteria for initiating event RHR2 (loss of inventory at mid loop operation)

Plant Operational States	RCS Integrity (RCSIN)	RCS Makeup (RCSMU)	Restore RHR (RSTRH)	SG Feed and Bleed (SGFNB)	Feed and Bleed by Gravity (FNBG)	Feed and Bleed by Force (FNBF)
POS 6A (Midloop operation without SG nozzle dams)	Upper branch: intact Lower branch: loss of inventory	1/2 HPIP Allowable time: 10.6 hours	(1) If running RHR pump successfully stopped: restart the stopped RHR pump (2) If running RHR pump is not stopped: start standby RHR pump Allowable time: 10.6 hours	1/4 SG and RCS closed and no SG nozzle dams and 1/2 AFWP (1/4 MSRV or 1/5 MSSV) Allowable time: 10.6 hours	Impossible due to closed RCS Allowable time 10.6 hours	1/2 HPIP and 1/2 PORV and 1/2 RHRRV Allowable time: 10.6 hours
POS 6B (Midloop operation without SG nozzle dams)	Upper branch: intact Lower branch: loss of inventory	1/2 HPIP Allowable time: 1.2 hours	(1) If running RHR pump successfully stopped: restart the stopped RHR pump (2) If running RHR pump in not stopped: start standby RHR pump Allowable time: 1.2 hours	Impossible due to installation of all SG nozzle dams	Impossible due to closed RCS	(1) 1/2 HPIP and 1/3 PZSV (2) 1/1 CHP and 1/3 PZSV Allowable time: 1.2 hours

Table 6-20 Typical front-line success criteria for initiating event RHR2 (loss of inventory at midloop operation) (Continued)

Plant Operational States	RCS Integrity (RCSIN)	RCS Makeup (RCSMU)	Restore RHR (RSTRH)	SG Feed and Bleed (SGFNB)	Feed and Bleed by Gravity (FNBG)	Feed and Bleed by Force (FNBF)
POS 10A (Midloop operation without SG nozzle dams)	Upper branch: intact Lower branch: loss of inventory	1/2 HPIP or 1/2 CHP Allowable time: 2.6 hours	(1) If running RHR pump successfully stopped: restart the stopped RHR pump (2) If running RHR pump in not stopped: start standby RHR pump Allowable time: 2.6 hours	Impossible due to installation of all SG nozzle dams	Impossible due to closed RCS	(1) 1/2 HPIP and 1/3 PZSV (2) 1/1 CHP and 1/3 PZSV Allowable time: 2.6 hours
POS 10B (Midloop operation without SG nozzle dams)	Upper branch: intact Lower branch: loss of inventory	1/2 HPIP or 1/2 CHP Allowable time: 26 hours	(1) If running RHR pump successfully stopped: restart the stopped RHR pump (2) If running RHR pump in not stopped: start standby RHR pump Allowable time: 26 hours	1/4 SG and RCS closed and no SG nozzle dams and 1/2 AFWP (1/4 MSRV or 1/5 MSSV) Allowable time: 26 hours	Impossible due to closed RCS Allowable time: 26 hours	(1) 1/2 HPIP and 1/2 PORV (2) 1/1 CHP and 1/2 PORV Allowable time: 26 hours

6.8.5.4 Future Needs

None.

6.8.5 Korea

6.8.5.1 Currently Available Information

For YGN 5 & 6 LPS PSA involved Level 1 probabilistic safety assessment of all low power and shutdown plant operational states for internal initiating event only. Because the plant is constructing and operating experience of reference plant is too short, the scope is limited to planed refuelling outage.

6.8.5.2 Ongoing Activities

None.

6.8.6.3 Planned Activities

The success criteria for all POSs will be determined based on the thermal hydraulic calculations.

6.8.6.4 Future Needs

None.

6.8.7 Spain

6.8.7.1 Currently Available Information

Current LPSD analyses are Level 1 PSAs for internal initiators only. Fire and internal flood initiated accidents will be the subject of future research plans. A Level 2 analysis is also foreseen. Some information is available regarding the status of containment main locks, but only from a thermal-hydraulic point of view. Since an open containment can be a characteristic of LPSD scenarios, this allows the more risky scenarios, from a radionuclide release point of view, to be easily identified.

Success criteria have been discussed in the thermal-hydraulics and core physics sections (i.e., 6.1.7 and 6.2.7).

6.8.7.2 Ongoing Activities

Planning is underway to include a project to analyze fire and internal flood risks during LPSD operation for a "pilot" plant in Spain. This project will be included in the Research Agreement between UNESA Spanish utilities association) and the CSN.

6.8.7.3 Planned Activities

Start the project being planned (see above).

It is believed that the issue of risk from radionuclide release from sources other than the reactor vessel warrants attention. Specifically, risk analysis of the refuelling phase is sometimes excluded based on the fact that there is no fuel in the reactor vessel. For this reason, risk analyses for other sources, mainly for spent fuel pools, is another topic that needs addressing. Next year begin planning a research project for a "pilot" risk analysis of other sources of radioactivity inside a nuclear power plant. This project would subsequently be included in Research Agreement between UNESA and CSN.

6.8.7.4 Future Needs

An examination of available fire and internal flood risk analysis methodologies to determine whether enhancements can be made to allow projects to be perform in a more cost effective manner is needed since these types of risk are believed to be significant and performance of these analyses should allow safety enhancements to be identified.

Current PSA methodologies should be examined to determine whether they can be used to estimate risk from other sources of radioactivity. If necessary, new methodologies should be developed.

6.8.8 Chinese Taipei

6.8.8.1 Currently Available Information

Core uncovery is assumed conservatively as core damage.

Scope is limited to refuelling shutdown. For the BWRs (Chinshan and Kuosheng), the refuelling shutdown is divided into 11 POSs, and only POS 3 to 10 are analyzed (POS 1, 2 and 11 are assumed as power operation conservatively).Most maintenance activities are in POS 6, while the water level is full in refuelling cavity. For the PWR (Maanshan), the refuelling shutdown is divided into 15 POSs, and all POSs except POS 8 are analyzed. POS 5 and 10 are in mid-loop operation, and POS 8 is core-empty.

There are two success criteria considered for BWR and PWR NPPs:

1. bleed and feed operation (flooding), and

2. feed and bleed by steaming operation (steaming).

6.8.8.2 Ongoing Activities

None.

6.8.8.3 Planned Activities

None.

6.8.8.4 Future Needs

None.

6.8.9 United States

6.8.9.1 Currently Available Information

For Grand Gulf and Surry, Phase 1 of the Low Power and Shutdown Accident Frequencies Program involved a coarse screening Level 1 probabilistic risk assessment of all low power and shutdown plant operational states. Based on the results from the Phase 1 work, detailed analyses were performed in Phase 2. For Grand Gulf , POS 5 (Cold Shutdown) during a refuelling outage was examined. For Surry, three midloop POSs where examined. This scope of the Phase 2 analysis included internal events analysis, internal fire, and internal floods.

The success criteria for shutdown conditions were determined by reviewing various studies and performing plant-specific thermal-hydraulic analyses. The changing level of decay heat was accounted for by defining time windows after shutdown (three for Grand Gulf and four for Surry), each with its own set of success criteria. In general, whenever the success criteria for one system or mitigating function changes, a new time window should be defined; however there is a trade-off between the accuracy of the model and the level of effort needed to arrive at a solution. Thus, the number of time windows ultimately defined may involve compromise.

Three functions must be successfully provided to mitigate an accident-initiating event:

- reactivity control,
- level control, and
- energy removal.

For Grand Gulf, the rods are fully inserted, and it is extremely unlikely that reactivity excursions from this subcritical condition can occur. Consequently, it was assumed that reactivity control is a function that is always present. For Surry, which uses boron to maintain subcriticality, reactivity control was not assumed to be always present (see Section 6.2.8). As stated previously, success criteria for the other two functions were determined by performing various plant-specific thermal-hydraulic calculations.

6.8.9.2 Ongoing Activities

None.

6.8.9.3 Planned Activities

None.

6.8.9.4 Future Needs

None.

6.9 Success Paths

Present LPSD results are typically for plant operational states occurring during a planned refuelling shutdown. Plans include extending LPSD PSAs to other types of outages, e.g. shutdowns that terminate at the hot standby state or shutdowns resulting from accident situation initiated from the nominal power operational state. Sequences were classified as successful according to the classic definition of the plant reaching a stable situation for the mission time assumed.

The means to consistently link success path sequences from full power event trees to event trees for LPSD should be developed. This would provide a complete PSA model that should be unique and coherent in its assumptions, codification, data, HRA input, dependency analysis, etc.

6.9.1 Czech Republic

6.9.1.1 Currently Available Information

LP&SD PSA for NPP Dukovany adopted the approach used in full power PSA. Event tree sequences have been therefore generally developed until stable long-term conditions is established (e.g. ECCS recirculation, hot shutdown with FW makeup, cold shutdown with secondary circuit RHR operation). LOCA outside confinement boundary requires specific safe states when cold shutdown is needed to mitigate the accident successfully.

6.9.1.2 Ongoing Activities

None.

Planned Activities

None.

Future Needs

An approach or methodology is needed to allow for the efficient modelling full-power-PSA success paths in LP&SD PSAs. In addition to this, a methodology is needed to determine what constitutes an appropriate LP&SD safe state.

6.9.2 France

6.9.2.1 Currently Available Information

Analysis of the sequences has been conducted either to the point of core damage or to a state in which the risk can be considered to be negligible. The latter condition has resulted in making allowance for post-accident situations of long durations, particularly for the case of primary system breaks, for which we have studied a long-term phase lasting as long as one year. Due to the fact that the long-term scenarios have been studied, it was not necessary to consider the success branches in the LPSD studies.

Ongoing Activities

None.

6.9.2.2 Planned Activities

None.

6.9.2.4 Future Needs

None.

6.9.3 Germany

6.9.3.1 Currently Available Information

Success paths are paths where the initiating event is controlled by the operational and/or safety systems and core cooling is not endangered and the plant is in a state. This includes also sequences with evaporation of the coolant in the containment (while the RPV is open) if feeding of the steam flow is established.

In the BWR study sequences which are characterized by evaporation of coolant without feeding and with a duration of more than 40 hours are judged to be negligible due to the large number of available systems with sufficient capacity.

In the PWR analysis a success path requires also the lack of a large volume of unborated water in the primary circuit.

6.9.3.2 Ongoing Activities

None.

6.9.3.3 Planned Activities

None.

6.9.3.4 Future Needs

None.

6.9.4 Hungary

6.9.4.1 Currently Available Information

Event sequences with success end states have been described within the level 1 PSA studies for nominal power operational mode. Thus, the kind of shutdown required for the different situations that develop after different initiating events is known.

6.9.4.2 Ongoing Activities

None.

6.9.4.3 Planned Activities

The present results are for plant operational states occurring during a planned refuelling shutdown. Plans include extending the LPSD PSA study to other types of outages, e.g. shutdowns that terminate at the hot standby state or shutdowns resulting from accident situation initiated from the nominal power operational state.

6.9.4.4 Future Needs

An extension of the level 1 internal fires and floods PSA studies for the nominal power operational mode is has been completed. Success paths for the event trees developed during these analyses should be continued to evaluate the longer-term processes associated with the shutdown condition.

6.9.5 Spain

6.9.5.1 Currently Available Information

No information is currently available because these full power PSA sequences have yet to be examined in detail. Till now, these sequences were classified as successful if the plant reaches in a stable situation for the mission time assumed, usually is 24 hours. LPSD scenarios arising from full power initiators might prove to be worse than some scenarios being modelled in current LPSD PSAs.

6.9.5.2 Ongoing Activities

No activities related to this issue are currently ongoing.

6.9.5.3 Planned Activities

Capturing this interface between full power and LPSD PSAs should be done in the future when LPSD PSAs become available for all plants. For one full power PSA, where these considerations were especially important owing to a peculiar design for the emergency core cooling recirculation function, a discussion of this issue is already included. Resolution of the issue has been postponed till the LPSD PSA becomes available.

6.9.5.4 Future Needs

Future updates of full power PSAs will consider these aspects as development of LPSD PSAs continues. The means to consistently link success path sequences from full power event trees to event trees for LPSD should be developed. This would provide a complete PSA model that should be unique and coherent in its assumptions, codification, data, HRA input, dependency analysis, etc.

6.9.6 Chinese Taipei

6.9.6.1 Currently Available Information

The mission time is assumed as 24 hours. The mission time is used to estimate the failure probability of component running failure, and not applied to screen out the sequences that take longer than 24 hours to reach core melt. Sequence recovery before core uncovery (may be longer than 24 hours) was generally not considered in this study. Sequence recovery before containment failure for loss of NSCW of Maanshan NPP was considered for its special scenario.

6.9.6.2 Ongoing Activities

None.

6.9.6.3 Planned Activities

None.

6.9.6.4 Future Needs

None.

6.9.7 United States

6.9.7.1 Currently Available Information

While this issue was identified in both the Grand Gulf and Surry LPSD analyses, LPSD sequences arising from such conditions were not analyzed due to program limitations. The difficulty lies in the fact that each successful sequence may represent a unique configuration; thus the level of effort needed to model each possible configuration can be significant.

6.9.8.2 Ongoing Activities

None.

6.9.8.3 Planned Activities

None.

6.9.8.4 Future Needs

None.

6.10 Screening Techniques

Core damage frequency is the central figure-of-merit for quantitative screening.

There is a need for a simplified, systematic approach for screening, one that would allow for a traceable process of selection of the most risk significant LPSD accident scenarios.

The COOPRA LPSD subcommittee is currently investigating ways to improve screening methodologies. Absent a systematic and technically plausible screening process, one that might need to be unique to LPSD PSA methodology, a full-scope LPSD PSA would be extremely costly.

6.10.1 Czech Republic

6.10.1.1 Currently Available Information

Screening process has been used in LP&SD PSA for NPP Dukovany just for initiating event selection. Some equipment failure modes and some scenarios have been also eliminated based on their expected negligible contribution to the risk.

The target of the screening was to preserve all important contributors to CDF. So events or scenarios with contribution less than 0.1% of the expected CDF (1 x 10^{-4}/y) were intended to eliminate. Two stage screening, qualitative screening and quantitative screening, has been applied for IE selection.

Qualitative screening was based on expected likelihood of occurrence and on ease of mitigation. Events with expected low frequency (i.e. multiple events) and expected mild consequences have been screened out.

Quantitative screening was based on threshold value for IE frequency. Events with the following frequency have been screened out:

- frequency less than 1 x 10^{-6}/y for events with “reasonable” (not harsh) consequences
- frequency less than 1 x 10^{-7}/y for events with harsh consequences (RPV rupture etc.)

Since the particular type of event has been usually split into several subtrees according to different system/equipment availability (due to different break location etc.), their particular frequencies have been treated together for the purpose of quantitative screening. The same approach has been applied as much as reasonable for the frequencies of the particular initiating event distributed into POSes.

So-called “barrier” approach has been applied for reduction of large amount of possible scenarios leading to Man Induced LOCAs. It was determined how many relatively independent “barriers” (administrative control, leakage indication, check valve, inadvertent opening of valve, inadvertent start of pump) have to be breached or have to fail to cause primary circuit draindown. The scenarios with two (small leakages) or three (middle and great leakages) such barriers have been screened out. The “barrier” approach allows to select efficiently the most important scenarios which represent Man Induced LOCAs. Due to large amount of possible scenarios their rough quantification (even based on bounding values) to screen out the negligible ones would not result in sufficient scenario reduction.

6.10.1.2 Ongoing Activities

The more detailed identification of the potential initiating events currently defined just generally is underway. It includes also potential screening of the identified events or scenarios. This project covers:

- identification of gas accumulation scenarios, especially nitrogen penetration,
- identification of credible LOCAs due to pressure tests following maintenance,
- identification of additional man–induced IEs resulting in loss of secondary circuit decay heat removal.

6.10.1.3 Planned Activities

None.

6.10.1.4 Future Needs

None.

6.10.2 France

6.10.2.1 Currently Available information

Screening techniques were not used. However sequences with low contribution have been grouped with other sequences.

6.10.2.2 Ongoing activities

None.

6.10.2.3 Planned Activities

None.

6.10.2.4 Future Needs

None.

6.10.3 Germany

6.10.3.1 Currently Available Information

A qualitative screening was used for every initiating event to identify the POSs where the event is most relevant. In the screening the probability for the event, the availability of systems and the demands on the functions to control the event was considered.

In the PWR analysis a further screening was performed to identify the initiating events for a detailed analysis. The screening based on a quantitative estimation under the consideration of the international experience.

6.10.3.2 Ongoing Activities

None.

6.10.3.3 Planned Activities

None.

6.10.3.4 Future Needs

None.

6.10.4 Hungary

6.10.4.1 Currently Available Information

Generally, no screening techniques have been used. All identified initiating events along with their appropriate event sequences were evaluated. The only exception was the approach followed within the PSA based review of boron dilution faults within the PHARE project PH2.08/95. The screening technique used in this project consisted of the following:

Dilution scenarios leading to reactivity transient, or ultimately to core damage were defined systematically.

Scenarios whose total core damage probability remained below 1 % of the total yearly core damage probability originating from low power and shutdown states were screened out.

6.10.4.2 Ongoing Activities

None.

6.10.4.3 Planned Activities

None.

6.10.4.4 Future Needs

None.

6.10.5 Japan

6.10.5.1 Currently Available Information

Screening techniques were not used. All accident sequences for each POS are calculated.

6.10.5.2 Ongoing Activities

None.

6.10.5.3 Planned Activities

None.

6.10.5.4 Future Needs

None.

6.10.6 Korea

6.10.6.1 Currently Available Information

POS 7, 8 and 9 were screened out due to large RCS inventory or no fuel in the RCS.

Ongoing Activities

None.

6.10.6.3 Planned Activities

None.

6.10.6.4 Future Needs

None.

6.10.7 Spain

6.10.7.1 Currently Available Information

As discussed in Section 5, a rough qualitative screening analysis was performed for the Vandellos 2 LPSD PSA and a more thorough screening was performed for the Asco LPSD PSA. Nevertheless, the CSN review still consider the Asco screening process as incomplete, because of the belief that core damage frequency, even roughly estimated, should be the central figure-of-merit for quantitative screening. This lack of a systematic approach for screening, that allows for an easy and easily traceable process of selection of the most risk significant LPSD accident scenarios, is considered as one the most important and, therefore, among the most needed of research activities. This need exists because some kind of technically plausible screening process is needed to make LPSD PSAs more cost effective, allowing plant safety to be improved while minimizing associated costs.

6.10.7.2 Ongoing Activities

This task exists of each LPSD PSA project and, although difficulties exist for each project, task procedures are expected to improve from project to project.

6.10.7.3 Planned Activities

There are no specific plans regarding this issue, other than just performing a screening for each PSA using, at least, similar methods from preceding PSAs. International activities related to the development of some kind of standard screening method for LPSD scenarios would be welcome, and CSN participation would be likely, although current research programs would require adaptation to any such standard.

6.10.7.4 Future Needs

As already mentioned, this aspect is needed of some kind of standardization in methodology, because is a key to make LPSD PSA more cost effective and consequently attractive. Without a systematic and technically plausible screening process, that would be perhaps unique for LPSD PSA methodology, the scope, number of possibilities and detail needed would make LPSD PSA performance for all plants too costly. It is believed, based on experience, that this analyses, like it was the case for FP PSA previously, must be plant specific to really catch the plant peculiarities where safety can be enhanced.

6.10.8 Chinese Taipei

6.10.8.1 Currently Available Information

Screening techniques are not used. All accident sequences for each POS are calculated.

6.10.8.2 Ongoing Activities

None.

6.10.8.3 Planned Activities

None.

6.10.8.4 Future Needs

None.

6.10.9 United States

6.10.9.1 Currently Available Information

The Phase 1 analysis of Grand Gulf and Surry represented a screening analysis to identify the more important LPSD POSs. Through various simplifications, results for all LPSD POSs were determined. These results were then examined using simplified differentiators to characterize the accident sequences for each POS. This characterization provided a means of identifying the more important POSs. For Grand Gulf, POS 5 (consisting mainly of the cold shutdown operating condition) was chosen. For Surry, three midloop POSs were identified. These POSs were then analyzed in detail using various means for eliminating unimportant sequences, including eliminating sequences where core damage would not occur until after some specified mission time or if the sequence frequency fell below some specified truncation level.

6.10.9.2 Ongoing Activities

None.

6.10.9.3 Planned Activities

None.

6.10.9.4 Future Needs

None.

6.11 LPSD Duration

Similar to screening, a methodology is also needed for selecting or identifying the duration of LPSD states. This should be supplemented by a methodology for developing sequences and crediting recovery beyond 24 hrs.

On the basis of operating experience with reactors in service, from 6 to 24 different operating states were identified and defined. Duration of each plant operational state was determined by a detailed review of shutdown schedule plans and past operational records. Since the duration of POSs differs, comparison of core damage risk among the states was made based on core damage probability - depending on the state duration rather than frequency. The decision was supported by the fact that some of the initiating events could be characterized by probability of occurrence rather than frequency. For example, some human errors that lead to a plant transient have the potential to be committed during special actions performed during shutdown, thus belonging to the group of initiating events characterized by probability of occurrence.

Average outage durations, based on duration times actually observed for the plant operational states, are used.

The effect of longer or shorter POS duration could be observed by means of sensitivity analysis or by means of on-line or off-line risk monitors or by some other system of "living PSA."

6.11.1 Czech Republic

6.11.1.1 Currently Available Information

The 24 hour time duration is used to limit the extent of the LP&SD PSA for NPP Dukovany in accordance with full power PSA approach. Recovery actions made by plant personnel to make available systems needed to mitigate accidents are expected to be completely successful beyond that period.

6.11.1.2 Ongoing Activities

None.

6.11.1.3 Planned Activities

None.

6.11.1.4 Future Needs

A methodology is needed for selecting or identifying the duration of LP&SD states. In addition, a methodology for developing sequences and crediting recovery beyond 24 hrs is needed as well.

6.11.2 France

6.11.2.1 Currently Available Information

Probabilistic studies carried out in France have shown that the risk of a core meltdown is not negligible in reactor shutdown states. As a result, the PSA 900 dealt with all reactor states.

On the basis of operating experience with reactors in service, six different operating states were identified.

State A, duration 7488 hours per year, covers the following operating phases:

- Reactor under power: power > 2% nominal

  Average temperature: between 286 °C and 304 °C

  Pressure: 155 bar absolute.

- Reactor on hot standby: power < 2% nominal

  Average temperature: 286 °C

  Pressure: 155 bar absolute.

- Reactor on hot standby and sub-critical

  Average temperature: 286 °C

  Pressure: 155 bar absolute.

- Reactor on intermediate shutdown to the P11 (139 bar) and P 12 (286 °C) thresholds:

  Average temperature: between 280 and 286 °C

  Pressure between 139 and 155 bar absolute.

This state is characterized by the fact that in all the corresponding phases, the safety injection system, which enables the water inventory in the core to be preserved in an accident situation, is activated by a low primary-system pressure signal.

State B, duration 38 hours per year, covers the remainder of intermediate shutdown up to the point where RHRS implementation conditions are reached:

- Average temperature: between 177 °C and 280 °C
- Pressure: between 30 and 139 bar absolute.

State C, duration 264 hours per year, covers the phases where primary system cooling is provided by the residual heat removal system and the primary system is full. This state includes the footing three phases:

- Temperature between 90 °C and 177 °C and pressure equal to 24 bar (duration: 37 hours)
- Temperature between 60 °C and 90 °C (duration: 110 hours)
- Normal cold shutdown when the system PTR can remove the residual heat, necessitating a primary system escape (duration: 117 hours).

State D, duration 456 hours per year, covers phases in which the primary system is partly drained or open to the point where the primary pipes are half full. A number of specific cases are recognized depending on the location of the opening in the primary system. The RHRS (and possibly system PTR) is in service.

State E, duration 216 hours per year, covers cold shutdown for refuelling with the reactor cavity full and with at least one fuel element in the reactor vessel.

State F, duration 288 hours per year, covers all primary system configurations where all fuel has been unloaded from the reactor vessel.

6.11.2.2 Ongoing Activities

A new definition of the reactor operating states during shutdown is under progress. The duration of each operating phase is being updated.

The definition of Plant Operating State (POS) has to be sufficiently detailed in order to reflect correctly the functional analysis. However some grouping and simplification are necessary to avoid a too complicated study. The difficulty is to find the best balance between these two objectives. In France, a limited number of POS has been defined but, for particular initiating events, a subdivision of POS has been necessary.

6.11.2.3 Planned Activities

None.

6.11.2.4 Future Needs

None.

6,11.3 Germany

6.11.3.1 Currently Available Information

BWR:

Some major inspections with refuelling were evaluated. The time from leaving the desired power operation until the start up was split into plant operation states. The duration of the POSs is very different from inspection to inspection due to varying actions. The duration varies by a factor of 3 to 5 and in one POS by a factor of 10. The average duration of a major inspection is 45 days.

PWR:

In the reference plant, 5 types of major inspections are defined which have a definite time schedule. For PSA purposes, a "normal" inspection with a duration of 14 days was chosen. The duration of the POSs varies from 3 hours (power reduction) to 143 hours (core unloaded). The following table gives an overview of the definition and duration of the POSs for a 14-day outage of the reference plant.

Table 6-21 POSs of a 14-days outage of the reference PWR-plant

Identification	Duration (h)	**Physical characteristics /** System characteristics
(1) A0	**3**	**Power reduction to the condition subcritical hot /** Reactor protection signals and availability of the safety systems as during power operation
(1) A1	**19**	**Subcritical hot; shutdown via steam generators down to primary system pressure 31 bar and primary system temperature 120 °C /** All reactor protection systems still available
(1) B1		**Primary system cooldown to the condition depressurised cold /** Start-up of the residual-heat removal (RHR) system at 120 °C; accumulators and high-pressure pumps are disconnected
(1) B2	**21**	**Level lowering to mid-loop, mid-loop operation /** Core within the RPV, primary system pressure-tight closed
(1) C	**18**	**Opening the RPV closure head, mid-loop operation /** Core within the RPV, primary system not pressure-tight closed, refuelling hatch between settling pond and fuel pool closed
(1) D	**66**	**Flooding of the reactor cavity, unloading of the fuel elements /** Core wholly or partly within the RPV, refuelling hatch open
E	**143**	**Emptying of reactor cavity and partly the RPV /** Core fully unloaded, refuelling hatch closed, work performed at lower-edge loop
(2) D	**73**	**Refilling of the reactor cavity, loading of the fuel elements /** Core wholly or in part within the RPV, refuelling hatch open
(2) C	**30**	**Level lowering to mid-loop, closing of the RPV closure head /** Core within the RPV, primary system not pressure-tight closed, refuelling hatch closed
(2) B2	**31**	**Evacuation and refilling of the primary system /** Core within the RPV, primary system pressure-tight closed
(2) B1		**Primary system heat-up with the reactor coolant pumps /** All reactor protection systems available
(2) A1		**Coolant deboration and taking the reactor to critical condition /** Withdrawal of control rods or / and deboration
(2) A0	**4**	**Power increase up to specified level /** Reactor protection signals and availability of the safety systems as during power operation

(1) = Plant operation states during shutdown

(2) = Plant operation states during restart

6.11.3.2 Ongoing Activities

None.

6.11.3.3 Planned Activities

None.

6.11.3.4 Future Needs

None.

6.11.4 Hungary

6.11.4.1 Currently Available Information

Duration of each plant operational state identified was determined by a detailed review of shutdown schedule plans and past operational records. Since the duration of POSs differs, comparison of core damage risk among the states was made based on core damage probability - depending on the state duration rather than frequency. The decision was supported by the fact that some of the initiating events could be characterized by probability of occurrence rather than frequency. For example, some human errors that lead to a plant transient have the potential to be committed during special actions performed during shutdown; thus, belonging to the group of initiating events characterized by probability of occurrence.

6.11.4.2 Ongoing Activities

None.

6.11.4.3 Planned Activities

None.

6.11.4.4 Future Needs

None.

6.11.5 Japan

6.11.5.1 Currently Available Information

The typical LPSD duration times used for BWR-5 calculations are shown in Table 6-22.

Table 6-22 Typical duration times for BWR-5

POS	Duration (Hour)
S	24
A	168
B1	408
B2	288
B3	192
C	480
D	11

6.11.5.2 Ongoing Activities

NUPEC is filing duration times for typical BWR-4/3 type plants and for PWR 2/3-loop type plants.

6.11.5.3 Planned Activities

Plan is to file duration times for an ABWR plant

6.11.5.4 Future Needs

None.

6.11.6 Korea

6.11.6.1 Currently Available Information

The typical LPSD duration time used YGN 5 & 6 are shown in Table 6-23. The typical LPSD duration time is calculated using reference plant, YGN 3 & 4, refuelling outage experience.

Table 6-23 Typical POS Duration Times for YGN 5 & 6 O/H

POS	POS Description	Duration
1	Low power operation and Rx shutdown	6
2	Cooldown using SG	21.3
3	Cooldown using SCS (RHR)	46.6
4A	Drain without large vent	10
4B	Drain with large vent	25.5
5	Midloop operation	27.5
6	Fill for refuelling	136.9
7	Refuelling – fuel move out	112.5
8	Refuelling – no fuel in RCS	387.3
9	Refuelling – fuel loading	147.3
10	Drain after refuelling	32
11	Midloop operation after refuelling	54
12A	Refill with large vent	12
12B	Refill without large vent	142.1
13	Heatup with SCS	43
14	Heatup with SG	84.2
15	Rx startup and low power operation	69.2
Total	Refuelling outage	1360

6.11.6.2 Ongoing Activities

None.

6.11.6.3 Planned Activities

Plans are to calculate POS duration of maintenance outage.

6.11.6.4 Future Needs

None.

6.11.7 Spain

6.11.7.1 Currently Available Information

Average outage durations, based on duration times actually observed for the plant operational states, are being used.

The effect of longer or shorter POS duration could be observed by means of sensitivity analysis or by means of on-line or off-line risk monitors or by some other system of "Living PSA." If refuelling frequency were changed, for instance, a LPSD PSA maintained in this way could be updated without too much difficulty. Of course, supporting thermal-hydraulic and core-physics analysis should be updated too.

6.11.7.2 Ongoing Activities

No specific activities are ongoing.

6.11.7.3 Planned Activities

There are no research plans.

6.11.7.4 Future Needs

This depends on what kind of applications will be supported by LPSD PSAs. If they are used in the same context as the full power PSA, then durations for specific POSs will would simply become another parameters inside the PSA model, and any effect associated with changes could be followed up like other aspects of plant operation.

6.11.8 Chinese Taipei

6.11.8.1 Currently Available Information

Estimated POS durations are provided in Table 6-24.

Table 6-24 Estimated POS durations

POS*	Chinshan (hr)	Kuosheng (hr)	Maanshan (hr)
1	2	5	13
2	10	9	16
3	3	2	15
4	20	56	21
5	50	67	10
6	854	981	88
7	88	80	76
8	121	34	779
9	13	43	22
10	355	231	17
11	72	88	98
12			70
13			89
14			65
15			17

*Refuelling schedules for all three NPPs are reduced from about 65 days to about 45 days recently.

6.11.8.2 Ongoing Activities

None.

6.11.8.3 Planned Activities

None.

6.11.8.4 Future Needs

None.

6.11.9 United States

6.11.9.1 Currently Available Information

Estimated plant operational state durations for Grand Gulf and Surry are provided in Tables 6-25 and 6-26. These estimates were derived from plant-specific outage information and represent averages for each POS. To account for the variability of the decay heat within a POS, the time window approach was developed. These time windows allow a more detailed representation of the plant state and may have their own set of different of success criteria.

6.11.9.2 Ongoing Activities

None.

6.11.9.3 Planned Activities

None.

6.11.9.4 Future Needs

None.

Table 6-25 Fraction of time spent in each Grand Gulf POS

Plant Operational States	Fraction of Year in POS
1D	0.008
2D	0.0066
3D	0.0066
4D	0.0052
5D	0.053
6D	0.013
7D	0.034
7U	0.008
6U	0.028
5U	0.023
1U	0.022

Table 6-26 Estimated durations of Surry POSs

Plant Operational State	Refuelling	Drained Maintenance	Non-Drained Maintenance (w/RHRS) (N1)	Non-Drained Maintenance (w/o RHRS) (N2)
1	0.56	0.7	0.1	0.56
2	22.3	15.1.	12.3	15
3	10.7	13.6	16.8	
4	154.4	196.3	127.9	
5	45.5	20.2		
6	183	202		
7	374			
8	810.8			
9	206			
10	107			
11	118	44.1		
12	1840	175		
13	34.4	10.3		
14	69	40.4	21	
15	56.1	12.7	18.6	9.93

6.12 Software

Software currently being used for LPSD evaluations include both probabilistic and thermal hydraulic. The same software used in the full power PSA is typically used in LPSD PSAs. Examples of thermal hydraulic software include:

- The Risk Spectrum PSA Software Package
- The LESSEPS Computer System (used for specific quantifications including time dependencies)
- The NUPRA Computer Code (used to estimate core damage frequency)
- SAPHIRE (PRA)

Examples of thermal hydraulic codes employed include:

- MELCOR (thermal hydraulic)
- RELAP
- CATHARE
- ATHLET

To improve the efficiency of the user interface, preprocessor software needs to be developed. Future needs also include an application code for risk monitoring.

6.12.1 Canada

6.12.1.1 Currently Available Information

CAFTA (supplier SAIC, California) has been used for the PSA work

6.12.1.2 Ongoing Activities

Investigation is in progress to assess the suitability of other codes.

6.12.1.3 Planned Activities

None.

6.12.1.4 Future Needs

Future needs include an application code for risk monitoring.

6.12.2 Czech Republic

6.12.2.1 Currently Available Information

Risk spectrum is the software currently used.

6.12.2.2 Ongoing Activities

None.

6.12.2.3 Planned Activities

None.

6.12.2.4 Future Needs

Future needs include an application code for risk monitoring.

6.12.3 France

6.12.3.1 Currently Available Information

There were two goals in producing a quantitative evaluation of the frequency of core melt:

- First, making the fullest possible allowance for the extremely high degree of modelling detail and the complex operating modes of the systems.

- Second, creating a "living" probabilistic safety assessment, i.e., an analysis that could easily be adapted to make sensitivity studies and to incorporate updates as data and knowledge evolve.

For this reason the LESSEPS computer system, developed by Électricité de France, was chosen as the most suitable tool for quantification.

The probability of a sequence is calculated by combining the probabilities of the events constituting the sequences. However, the calculation rarely involved simple multiplication, as the occurrences may not be independent. Dependence may be functional (common sections or common support systems) or temporal (a system is only activated in the event of failure of another, its operating time depends on the repair time of a failed stem etc.).

There is no method of calculation that can be used to cover absolutely all the forms of dependence in a given sequence. The principles used in the LESSEPS software are the following:

- functional dependence is only taken into consideration in a given sequence. This means that for quantification, the event trees must be made sufficiently detailed to ensure that each sequence only consists of independent functional occurrences;

- temporal dependence, which in some cases has a considerable effect on the result, is treated in depth (e.g., Markov graphs are used).

Although LESSEPS is a powerful tool for the treatment of complex systems operation modelling, it appeared that, for current applications, the LESSEPS software was not easy to use. For that reason, in order to facilitate the future applications of probabilistic studies, it was decided to transfer also the level 1 PSA model on the software Risk Spectrum.

6.12.3.2 Ongoing Activities

The updating of the study is being carried out using the Risk Spectrum software.

6.12.3.3 Planned Activities

None.

6.12.3.4 Future Needs

None.

6.12.4 Germany

6.12.4.1 Currently Available Information

For the determination of the minimal cut sets the PSA-code "Risk Spectrum PSA Professional" was used. The probability distributions and the fractiles were calculated with a simulative method developed by GRS.

6.12.4.2 Ongoing Activities

None.

6.12.4.3 Planned Activities

None.

6.12.4.4 Future Needs

None.

6.12.5 Hungary

6.12.5.1 Currently Available Information

Recently Risk Spectrum PSA Professional for Windows (version 1.20) is being used. All current models, including the LPSD PSA, have been transferred to this version.

6.12.5.2 Ongoing Activities

None.

6.12.5.3 Planned Activities

None.

6.12.5.4 Future Needs

None.

6.12.6 Japan

6.12.6.1 Currently Available Information

The NUPRA computer code was used to estimate core damage frequency. NUPRA has the capability of handling up to 30,000 accident sequence minimal cut sets.

6.12.6.2 Ongoing Activities

To improve the efficiency of the user interface, preprocessor (graphical user interface) software is being developed.

6.12.6.3 Planned Activities

None.

6.12.6.4 Future Needs

Because the NUPRA source code is not available, future plans involve a search for new codes.

6.12.7 Korea

6.12.7.1 Currently Available Information

KIRAP (KAERI Integrated Reliability Analysis Code Package) is used for YGN 5,6 FP and LPS PSA. The code package is consists of window base ET editor and FT editor, cutset generator, uncertainty calculator and FT conversion utility. The code has two good feature to model and quantify.

- Solve logical loop.
- Rule-Based Recovery.

6.12.7.2 Ongoing Activities

None.

6.12.7.3 Planned Activities

None.

6.12.7.4 Future Needs

None.

6.12.8 Spain

6.12.8.1 Currently Available Information

The same software used in the full power PSA was used in both LPSD PSAs. This software is currently available.

6.12.8.2 Ongoing Activities

No specific activities are ongoing.

6.12.8.3 Planned Activities

There are no research plans in this area.

6.12.8.4 Future Needs

No specific need is foreseen in this area, except for better consideration of specific refuelling or other types of outage schedules, as mentioned in Sections 6.4 and 6.11. However, the software used for full power PSA development or for some PSA applications, like risk monitors, might be applicable for LPSD PSAs.

6.12.9 Chinese Taipei

6.12.9.1 Currently Available Information

The NUPRA computer code was used to estimate core damage frequency. TIRM (Taipower Integrated Risk Monitor) software for power operation is in the V&V stage (beta version available), while for refuelling shutdown, the top-logic fault tree model development is in progress.

6.12.9.2 Ongoing Activities

None.

6.12.9.3 Planned Activities

None.

6.12.9.4 Future Needs

None.

6.12.10 United States

6.12.10.1 Currently Available Information

Computer codes used in the low power and shutdown study of Grand Gulf and Surry included SAPHIRE (PRA) and MELCOR (thermal hydraulic).

6.12.10.2 Ongoing Activities

Both codes are continually being enhanced and modified to improve usability.

6.12.10.3 Planned Activities

None.

6.12.10.4 Future Needs

None.

7. WORKING GROUP RESEARCH ACTIVITIES

Topic	Essentiality	Feasibility	Resources	Time	Information	Priority
Item ‘A’	5	3	5	1	3	225
Item ‘B’	5	5	5	5	5	3125
Item ‘C’	3	5	5	3	5	1125
Item ‘D’	0	5	5	5	5	0

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