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STATUS OF PSA PROGRAMMES IN MEMBER COUNTRIES

A Compilation of Contributions from Members of Principal Working Group No. 5

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CSNI

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and coordinate 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 amongst the OECD Member countries.

CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. it also reviews the state of knowledge on selected topics of 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 coordination of work in different Member countries including the establishment of co-operative research projects and international standard problems, and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences and specialist meetings.

The greater part of CSNI's current programme of work is concerned with safety 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 fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

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

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COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

Principal Working Group No. 5 on Risk Assessment

Status of PSA Programmes in Member Countries

This document is a summary of the current status of national PSA programmes in Member countries, prepared as an updated version of SIN/DOC(89)29 by including contributions and modifications which the Secretariat has received since the October 1990 meeting of PWG5.

BELGIUM

Abstract

In addition to the usual reliability analysis in the traditional licensing procedure, a re-analysis of safety systems reliability has been made for the safety reassessment of plants which have been in operation for 10 years. For Doel 3 and Tihange 2 this safety reassessment will take place in 1992: in order to prepare it, a PSA level 1 plus analysis is now being performed. A PSA is required for any new plant.

Programme Development

PSA applications to nuclear power plant safety in the past were restricted to traditional reliability analysis for selected issues. The safety practice is mainly based on the USNRC requirements. Some additional measures, e.g. supplementary bunkered safety systems, were taken for protection against improbable events with a potential for large consequences.

Belgian organisations took part in all CEC benchmark exercises: systems analysis, common mode failure, human errors, event sequence quantification, major hazard analysis (chemical plant).

Research and Development activities have been undertaken (and some are going on) for the adaptation or development of computer codes for PSA analysis, for fault tree construction, for Markov analysis or for technical specifications optimisation.

Status and Outlook

Participation in future CEC benchmark exercises (expert opinion ?) will be examined.

In accordance to the requirements of the operating licence, the safety of a nuclear power plant has to be reassessed, after 10 years of operation, on the basis of current knowledge and requirements. The next reassessment will be performed for Doel 3 and Tihange 2 in 1992.

As a help to the decision-making process, this reassessment will include a PSA of extended level-1, i.e. with some containment analysis. For the level 1 analysis, results are expected to be available for Doel 3 and Tihange 2 at mid-1992.

Another objective of this study is to get acquainted with PSA techniques to perform a PSA for the next Belgian nuclear power plant when it will be decided. The current requirements of USNRC will be applied, i.e. analysis of a high level of preventative safety and analysis of potential severe accident vulnerabilities.

BELGIUM

Plant	**Analysing Team/ Date of Starting Completion**	**Methods Used/ Procedure Guide**	**Goal of the Analysis Insights of Results Applications**
Doel 3 PWR, 3 loop 900 MWe Framatome under operation	Tractebel 1988-1992	Level 1+, including some containment analysis NUREG/CR 2815	Safety reassessment after ten years of operation getting acquainted with PSA techniques for future applications
Tihange 2 PWR, 3 loop 900 MWe Framatome under operation	Tractebel 1990-1992	Same as for Doel 3	Same as for Doel 3
Future NPP projects		PSA including severe accident considerations to USNRC practices	

CANADA

Abstract

Probabilistic assessments have always been required for CANDU licensing. Safety system unavailabilities and, in some cases, process system failure frequencies have been calculated using fault tree analysis methods to demonstrate compliance with licensing requirements. The SDM studies, introduced in 1975, developed descriptive event sequences where the mitigating systems, their chronology in accident sequences and the associated alarm indications were identified. Fault tree analyses are used in these studies to evaluate initiating event failure frequencies and mitigating systems unreliabilities. More recently, comprehensive PSAs have been prepared, or are in preparation for several CANDU reactors.

Programme Development

The licensing of CANDU nuclear power plants in Canada (Reference 1) requires that the sum of all serious process system failures shall not exceed a frequency of 0.3 events/year. In addition, the frequency of dual failures, where a serious process failure occurs together with an unavailability of a special safety systems, shall not exceed a frequency of 3 x 10^{-4} events per year. There are four special safety systems provided; two independent shutdown systems, emergency core cooling and containment and each must be shown to have an unavailability of less than 10^{-3} years/year. For each single process failure and dual failure (i.e., a process failure together with a failure of a special safety system) dose limits are applied and conservative assumptions are made in demonstrating compliance with these limits. This approach remains, up to the present, the basis for licensing of most Canadian reactors. However, for Darlington, the most recent reactor, postulated accidents are assigned to five classes based on predicted frequency and each class has a different dose limit (Reference 2).

In 1975, a series of studies was started by AECL and Ontario Hydro in which "best estimate" transient conditions were to be used. Known as Safety Design Matrix (SDM) studies, these studies attempted to assign failure frequencies to serious process failures by introducing initiating event fault trees. The accident sequences were developed using a descriptive event sequence format where the mitigating systems were defined and the chronology of the events retained, as far as possible, by including a time scale. Each accident sequence was developed to the point where either stable plant conditions could be shown, or to unstable plant conditions having the potential to cause radionuclide releases a credible frequencies or to sequence end point conditions less than 10^{-7} events per year (the sequence cut-off condition). Sequences where unstable plant conditions were shown were reviewed to identify possible plant design changes at the design and construction stages of the plant.

The predicted releases and associated frequencies for the accident sequences from these studies were compared with release acceptance criteria developed from the single-dual failure release limits. There was no summing of accident sequences in these studies to evaluate risk.

The descriptive event sequences of the SDM programme required the analysts to not only identify the mitigating system and associated probabilities, but to also add extensive description boxes detailing the plant accident conditions and the expected alarm indications available as the accident sequence proceeds. These diagrams were well received by plant operation staff but were not a convenient format for performing the probabilistic evaluations. At AECL the more conventional event tree format is now being used for probabilistic evaluations although the descriptive sequences are still prepared at the accident sequence definition stages. For licensing support purposes, it is again proposed to retain an event sequence frequency cut-off below which radiological consequences will not be evaluated.

Past studies:

1975 - 1976	Bruce A NGS. Safety Design Matrix (SDM) studies for a selected number of process failures.
1977 - 1981	SDM studies for 600 MW Candu reactors at Point Lepreau, Gentilly-2, Wolsung-1 (Korea)
1979 - 1982	SDM studies for Ontario Hydro reactors at Pickering-B (Units 5 - 8) and Bruce B (Units 5 - 8)

Peer reviews were performed on the SDM studies by utility and regulatory staff.

Status and Outlook

PRA/PSA Programmes

Atomic Energy of Canada Ltd. (AECL), as the designer of CANDU reactors is undertaking or planning a number of probabilistic safety assessment studies for reactor design and licensing support.

In the CANDU 3 Programme, AECL is using PSA as a design tool as well as for licensing. Major design and construction innovations are planned for this reactor to make the station concept more competitive with coal-fired units of a similar size. PSA is planned which will evolve with and influence both the conceptual and detailed design. It will provide a means of keeping the Atomic Energy Control Board (AECB), the Licensing authority for reactors in Canada, informed.

The CANDU 3 PSA programme is divided into four phases: Mini PSA, Conceptual PSA, Generic PSA, and Site Specific PSA. The first two phases have been completed, and work on the third is about to begin. The Mini PSA reviewed those areas which are expected, based on past experience, to have the most potential for design changes; the Ontario Hydro data base was used extensively for this assessment.

The Conceptual PSA assessed whether the CANDU 3 design has adequate redundancy and diversity, and it set reliability targets for systems. Thus, during the conceptual phase of the design process, reliability targets, and interface requirements were set. The interface requirements define the expected behaviour of a system, given a failure in a related or support system.

The Generic PSA will confirm whether the reliability targets are being met as the design develops, via detailed fault tree analysis.

The Site Specific PSA will modify the Generic PSA to account for differences between the site and the standard product design.

All phases of the CANDU 3 PSA programme will be peer-reviewed by AECL designers outside the CANDU 3 team, as well as by utility and regulatory staff.

Slowpoke Energy Systems

The research organisation within AECL is sponsoring PSAs of its Slowpoke reactors:

- Slowpoke Demonstration Reactor (2 MW pool reactor) situated at Whiteshell, Manitoba, Canada. The objective of this study is to model the plant response to a reactor regulation runway and to assess the adequacy of the shutdown systems. The methods proposed are the same as the CANDU 3 proposals. The data base used is generally the Ontario Hydro CANDU data base. This work has recently been completed.

- Slowpoke 10 MW reactor. This work is ongoing. Several initiating events have been examined and analyses are being used to compare alternative design configurations. Again, methods used are as for CANDU 3 and the data base used is the Ontario Hydro CANDU generic design data. Reliability analyses have been performed on the reactor protection system, and have been peer-reviewed by another set of independent designers at AECL.

Research Reactors - AECL Chalk River Nuclear Laboratories

A PSA (level 1) of Loss of Cooling Accidents for the NRU 135 MW research reactor was completed in 1985. A reliability study of the trip system of the Z-2 reactor, a zero energy research facility, is underway and is scheduled for completion by the end of 1987. Reliability studies of loss of off-site power and reactor diesel generators at Chalk River has been recently completed and reported. CRNL is presently involved in preparing a limited scope technical document for the IAEA on Research Reactor PSA.

A technical and safety re-assessment programme is now being planned for the NRU reactor. The safety re-assessment will update the existing deterministic licensing analyses, and will include probabilistic safety assessment also. These analyses are required to determine if the lifetime of the NRU reactor can be significantly and safely extended.

AECL-KEMA Collaborative Study

In the Netherlands, there is a concern with very severe "beyond design basis" events (for any reactor design). AECL has completed a study in co-operation with the Dutch authorities to look at siting a CANDU reactor in the Netherlands. This study is a limited risk analysis for a CANDU 6 reactor using fault trees and event trees, but using the Canadian SDM (Safety Design Matrix) information and operator model. Preliminary results indicate that existing CANDU 6 designs compare favourably with the safest currently available light water reactors (e.g. Sizewell B). Further work is now being planned over the next year to refine the risk estimates. Inherent CANDU features are being further evaluated to possibly demonstrate an even lower predicted risk. This study was extensively peer-reviewed by the Dutch authorities. It should also be noted that the work was based on SDM information which had been peer-reviewed previously by Canadian Utility and Regulatory Staff.

MAPLE-X

The research organization of AECL is sponsoring PSAs of its MAPLE-X reactor, an isotope producer. The PSA work will concentrate on the reliability of the reactor regulating and protection systems, using fault tree analyses.

Ontario Hydro Studies

In 1982, Ontario Hydro started comprehensive risk assessment of its Darlington reactor, then under construction. This risk assessment was completed in 1987 and, because of safety design review benefits obtained, a decision was made to extend the risk assessment programme to all Ontario Hydro operating reactors (i.e., Pickering A and B, Bruce A and B). The Pickering A risk assessment will be completed in 1991.

All of these studies have the following characteristics:

- Performed in-house.
- Level-3 PSAs without external events.
- Multiple levels of core damage considered.
- System fault tree models are very detailed.
- Risk assessments will be kept up-to-date throughout the life of the reactor.

To complement the risk assessment programme an enhanced programme of component fault data collection and analysis has been initiated. Where at all possible, station-specific data on initiating event frequencies, reactor state probability, component failure rate and restoration times, maintenance and test frequencies and durations, and human error probabilities will be generated and used.

The operational safety reliability programmes at the stations use system models developed under the risk assessment programme to set testing frequencies and to maintain the reliability of selected safety-related systems at required values.

In addition, Ontario Hydro has recently endorsed, for in-house trial use, a set of risk based safety goals. The risk assessments will be the primary vehicle to show compliance with these goals.

Use of PRA/PSA

System level reliability assessments are used regularly on all stations in design and operations to review design adequacy and set testing requirements. Reliability monitoring of special safety systems and other selected safety-related systems is performed in operation and compared to regulatory limits for unavailability (this is a regulatory requirement for the special safety systems).

References:

1) "Reactor Licensing and Safety Requirements", D.G. Hurst and F.C. Boyd, 72-CNA-102, CNA Conference, Ottawa, June 1972.

2) "Requirements for the Safety Analysis of CANDU Nuclear Power Plants" Consultative Document C-6, Atomic Energy Control Board, June 1980.

CANADA

Plant	Analysing Team/ Date of Starting Completion	Methods Used Procedure Guide	Goal of the Analysis Insights of Results Applications
CANDU 3 Task 1/2: Task 3: Task 4:	AECL Mar 87/Apr 89 Ongoing (Subject to project commitment)	Fault tree-event tree modified NUREG 2300 Ontario Hydro data	Task 1: Mini PSA Task 2: Conceptual PSA Task 3: Generic PSA Task 4: Site specific PSA
Slowpoke 2MW Slowpoke 10MW	AECL December 1986/ June 1987 AECL June 1987/ongoing	AS CANDU 300/ modified NUREG 2300 Ontario Hydro data base as above	Plant response to reactor regulation runway Comparison of alternative design configurations and reliability analysis on reactor protection systems
Darlington	Ontario Hydro July 1982/ December 1987	Fully integrated event tree-fault tree Ontario Hydro Procedure Guide	1. Safety design review 2. Procedure development 3 Licensing support
Pickering A Bruce B Pickering B	Ontario Hydro June 1988/ August 1991 Ontario Hydro January 1991/ December 1993 Ontario Hydro January 1992/ December 1993	As Darlington As Darlington As Darlington	As Darlington As Darlington As Darlington
Bruce A	Ontario Hydro September 199 / April 1995	As Darlington	As Darlington
NRU 135 MW	AECL/ April 1985/ October 1985	Fault tree-event tree AECL data base	Local analyses/ Level-1

CANADA (continued)

Plant	**Analysing Team/ Date of Starting/ Completion**	**Methods Used/ Procedure Guide**	**Goal of the Analysis Insights of Results/ Applications**
CANDU 6	AECL/KEMA September 1986/ June 1987 Phase 1 March 1987/ March 1988- Phase 2	Fault tree-event tree Developed from SDM operator model and procedure as basis	Phase 1 - releases at containment boundary level 2 internal events only phase 2 review of consequences analyses

FINLAND

Abstract

Although the formal licensing procedure has been based on deterministic rules and criteria, probabilistic safety assessment has played a supporting role in licensing. Reliability analyses at the system level, and a mini PSA for one initiating event of one plant type, have provided a review of the safety of the plants in operation.

At the end of 1984, utilities decided, and the authorities required that PSAs should be performed for all Finnish plants. For possible future plants, PSA will be an essential part complementing the deterministic safety analysis. To ensure sufficient safety of new plants, numerical safety standards have been included in the regulatory guide.

PSA is used in dealing with low probability events involving unacceptable performance of safety systems or containment. PSA is closely connected with the design, construction and operating phases of nuclear power plants.

Programme Development

Over 20 reliability analyses were performed during licensing and construction of the Loviisa NPP from 1972 to 1980. In addition, in a mini PSA, the large break LOCA was analysed in 1973 - 75. The accident sequences leading to the damage of the reactor core were resolved utilising the results of the system reliability analyses. The behaviour of the containment and the amount of radionuclide releases into the environment were assessed and categorised with the techniques used in WASH-1400. An integrated probabilistic analysis of Pressurised Thermal Shock to the reactor pressure vessel was carried out for the Loviisa Unit 1 in 1984 - 86. Scenarios for all overcooling events have been developed using an event tree formalism. Plant-specific data was used in quantifying event sequences. The Loviisa training simulator was widely applied to predict the thermal-hydraulic response of the plant to selected overcooling transients. The conditional probability of vessel failure was calculated with the OCA-P code.

In October 1984, the TVO management decided to perform a level-1 PSA for both Olkiluoto plants. The Finnish authority STUK also required such studies. The study concentrated on those issues recognised as being most important in the earlier Swedish studies: human factors and detailed identification of dependencies in the multi-redundant units. Besides the evaluation of core melt frequency and ranking of the most important core melt sequences, the most important goals of the study were related to mapping, ranking and optimisation of improvements in system design, education, training, technical specifications and procedures. To increase the familiarisation of the plant staff to the systems and safety features of the plant, the study was mainly performed by utility personnel.

A pre-study during 1985 preceded the PRA project for education and training of project staff by several consultants. Special approaches have been developed for effective dependencies identification, human error analysis (SHARP, shortened version of THERP, human cognitive reliability correlation and Human Reliability Handbook), and fault tree modelling (Swedish SUPERTREE code). System success criteria were determined mainly to be the best estimate calculations (Swedish BISON and GOBLIN codes) or the Final Safety Analysis Report (FSAR) criteria. The utility experiences of such an intensive study is purely positive.

The main PSA activities at present in Finland are as follows:

- low power and shutdown analyses and level 2 PSA for TVO 700 MW-BWR

- fire, flood, low power and shutdown analyses and level 2 PSA for Loviisa 440 MW-PWR

The fire and flood analyses are completed at the TVO and are underway at the IVO Power Company. The low power shutdown analyses are next to the fire and flood analyses at IVO.

a) TVO PSA level 1

Level 1 PSA of TVO comprises only internal initiating events, loss-of-offsite power and initiating events induced by human errors. PSA is to be extended later on to level 2 and possibly to level 3.

In TVO PSA small event and large fault trees have been used. Failure data are mainly acquired from the own plant but Swedish ATV-data are used to a limited extent as well.

The initiating events have been categorized in three LOCA and three transient divisions. ATWS events are included in each initiating event division. The coverage of the initiating events has been assured by the aid of initiating event lists of 8 PSA studies and NUREG/CR-2300. Best estimate criteria have been used as to the most important safety systems. Otherwise, FSAR criteria are used.

The most important initiating events are as follows:

Initiating event	Number of accident sequences	Contribution to core melt frequency %
- loss of condenser	27	53
- loss of offsite power	21	19
- small LOCA	9	17
- loss of main feedwater	23	8
- medium LOCA	9	4
- large LOCA	5	0.1

The regulatory review of TVO PSA is already completed and the power company is informed of the results of the review. The TVO PSA model is installed in SPSA code (STUK PSA code) and used as Living PSA both at TVO and STUK.

b) LOVIISA PSA level 1

Level 1 PSA of Loviisa plant comprises internal initiating events, loss-of-offsite power and human error induced initiating events.

In Loviisa study very small and general event trees and large fault trees are used. A majority of failure data comes directly from Loviisa data acquisition system and for small part general data have been used.

Over 70 initiating events have been identified in Loviisa study. Initiating events have been divided into 9 transient and 11 LOCA divisions.

Conservative FSAR criteria and assumptions have been used as a basis for the analysis. Using highly conservative assumption more extensive analyses have been avoided as to the phenomenology of the plant. This implies, however, that overconservative results have been received in some points of PSA.

The most important initiating events of Loviisa PSA are as follows:

Initiating event	Contribution to core melt frequency %
- loss of ventilation cooling of electrical and instrument rooms	73
- Medium reactor cooling pump seal LOCA/safety valve LOCA in pressurizer	9.5
- total loss of service water system	6.5
- loss of offsite power	3.0
- multiple steam generator tube rupture	2.5
- small LOCA	1.7
- steam generator collector break	1.3
- medium LOCA	0.8
- loss of DC power	0.7
- etc.	

The preliminary review of Loviisa PSA revealed in the fall 1989 that three design errors in safety systems resulted in CCFs that contributed about 90% of total core melt probability. This made the utility prepare the rapid backfitting plans for the systems such as:

- electrical and instruments room cooling system
- service water system
- minimum flow lines of ECCS.

The new designs made the core melt probability decrease almost one order of magnitude and changed the risk rank order of accident sequences radically.

Status and Outlook

A Finnish licensing authority guide requires probabilistic safety analyses complementing the traditional deterministic analyses. A construction permit for a future plant is only granted if a mini PSA has been completed. This is a level 1 PSA for the most important initiating events based on the design concept. It should reveal interconnections and interactions between various systems and supporting systems, as well as reasons for common cause failures and weak points at the function, system and redundancy levels. The mini PSA is essentially qualitative, but not intended for showing compliance with probabilistic objectives of safety functions of the Guide.

After the construction permit, a level 1 PSA is commenced, including consideration of containment by-pass chains. Before the operating licence is issued a level-2 study must be performed. The licensing authority requires the utility to update the PSA during design and construction. Furthermore, the PSA is required also to provide a tool for controlling and regulating the safety of a nuclear power plant all through its service life.

The PSA is qualitatively reviewed by the licensing authority. No fixed acceptance standard is prescribed for the probability of core damage. The unreliability of the most important safety functions must be below design objectives set by the authority. During operation and acceptable level of safety must maintained. The utility must be able to demonstrate this using PSA methods.

Both qualitative regulatory requirements and common understanding of the parties involved serve as a basis for the PSA programme related to the operating plants. The strict regulatory requirements, however, work as a basis for the PSA program of possible new NPPs.

To avoid severe reactor accidents and to mitigate their consequences, the PSA shall be utilised for training and operation. The operating personnel shall familiarise themselves with severe accidents by means of accident sequences identified in the PSA. Instructions for preventing and for mitigating severe accidents must be prepared. Simulator models must be developed and applied to the important accident sequences.

The following points are also worth mentioning:

1. Utilities running the NPPs are in charge of performing PSAs using their operating personnel as far as possible (in-house PSAs required). Contractors are to be used only to the limited extent, for ex. to special tasks such as CCF and human performance analyses.

PSA has an essential role in the licensing of possible new plants. A so-called Mini-PSA has to be completed before the construction permit can be issued. As well, a PSA study of level 2 has to be completed before the operating license can be granted.

2. The PSAs of level 2 are required both for operating plants and for licensing purposes of possible new plants. These analyses include accident sequences induced by internal initiating events, fires and internal floods. As concerns the methods, references are made to well-known PSA procedures guides and as far as the data base is concerned, the plant-specific data are preferred to generic data.

3. The regulatory body (STUK) is in charge of the independent peer review of the PSAs. In addition, the utilities use independent reviewers on their own for ensuring

the quality of PSA to be submitted to the regulatory body.

4. a) The independent peer review will be a document distinct from the PSA documentation.

b) In practice the independent peer reviews are exposed to the analysts responses even though no formal procedure has been fixed.

5. No formal procedures are used to make regulatory decisions related to the insights gained from PSAs of operating plants. Instead, in regard to the possible new NPPs, formal criteria are used in the licensing process as stated before.

References

1) Guide YVL 2.8, Probabilistic Safety Analysis in the Licensing and Regulation of Nuclear Power Plants. Finnish Centre for Radiation and Nuclear Safety, 1987.

2) Reino Virolainen, Seppo Vuori, "Finnish Experiences in the Risk Assessment and Reliability Analysis of Nuclear Power Plants", IAEA's workshop on Advances in Reliability Analysis and PSA, Budapest, Hungary, 7-11 October 1985.

3) Reino Virolainen, "The Uses of Probabilistic Safety Analysis in the Licensing and Regulation of Finnish Nuclear Power Plants", PSA'87. International Topical Conference on Probabilistic Safety Assessment and Risk Management, Zürich, August 30 - September 4, 1987.

4) A. Vuorinen et al, "Deterministic versus Probabilistic Based Safety and Licensing Decisions with Particular Emphasis on Severe Low Probability Events", International Conference on Nuclear Power Performance and Safety organized by IAEA, Vienna 28 September - 2 October 1987.

FINLAND

Plant	Analysing Team/ Date of Starting Completion	Methods Used/ Procedure Guide	Goal of the Analysis Insights or Results Applications
Loviisa 1-2 2x440 MWe PWRs Operating since 1977	Technical Research Centre of Finland (VTT) 1972-1979 (additional studies in 1980)		
Olkiluoto 1-2 2x660 MWe BWR Operating since 1979	Technical Research Centre of Finland (VTT), ASEA-ATOM - 1979	Separate analysis of all safety systems, quantification of the reliability of main safety functions using DBA assumptions for success criteria	Verification of the overall reliability of the safety systems in licensing optimisation of the test and repair arrangements and limiting conditions of operation
Concept design of SECURE 200 MWe thermal heating plant	VTT, Studsvik (Sweden) - 1978	Identification of potential accident sequences, comparison of design alternatives, containment and offsite consequence analysis for 3 selected accidents	Time dependence of accidents, possibility of recovery actions. Design optimisation and systematic design review
Olkiluoto ASEA-ATOM 710 MWe BWR Operating since 1979	Industrial Power Company (TVO) 1985 - 1988	Level 1 PSA Initiating events selected using NUREG/CR-2300, WASH-1400, Millstone Limerick, Forsmark 3, Baresbäck 1	Evaluation of core melt frequency, ranking of most important core melt sequences. Mapping ranking and optimisation of improvement in system design, education, training and procedures.

Plant	Analysing Team/ Date of Starting Completion	Methods Used/ Procedure Guide	Goal of the Analysis Insights or Results Applications
	1991 complete 1991 complete 1992 complete	NUREG/CR-2728 (partly) and reference studies Forsmark 3, Barsebäck 1 Flood analysis Fire analysis Low power and shut down analysis	Optimisation of technical specifications. Familiarisation of the plant staff to the systems and safety features. Results should be applicable to higher level PSAs.
Loviisa-1 VVER-440 MWe PWR Operating since 1977	Imatra Voima Power Company 1985 - 1988 1992 complete 1992 complete 1992 complete 1993 complete	Level 1 PSA Initiating events: EPRI-list, plant specific frequencies NUREG/CR-2728 (IREP) NUREG/CR-2300 Low power and shutdown analysis Fire analysis Flood analysis Level 2 PSA	

FRANCE

Abstract

From the beginning of the French nuclear programme in 1975, probabilistic methods were developed to study the reliability of safety related systems. Reliability studies were performed for safety systems of the standardised French PWR plants. In 1977, the Ministry of Industry (Safety Authorities) set a probabilistic target (overall probability of unacceptable consequences below 10^{-6} per year) and specified a framework for the PSAs performed by the utility.

Two level-1 PSAs have been performed, one by the utility EDF for a standardised 1300 MWe PWR plant, and one by the CEA/IPSN, for a generic 900 MWe PWR.

These studies were completed in 1990. Some major insights have been drawn, especially the benefit related to the implementation of emergency procedures and the importance of risk during shut down situations.

"A living PSA" approach was developed to control changes in data and knowledge.

Programme Development

In France, the design of plant is based on deterministic rules, by studying the consequences of a limited number of situations classified into four categories according to their expected frequencies. For each class, technical requirements and limits are assigned.

A probabilistic approach was first used to decide if certain external initiators (e.g. airplane crash) have to be considered within the design base or not.

In 1977, the Ministry of Industry set a probabilistic target (not to be considered as a formal licensing requirement) that the overall probability of unacceptable consequences should not exceed 10^{-6} per year for one PWR unit and 10^{-7} per year for a family of events. In 1978, the safety authority specified the framework for the PSAs performed by EDF. PSAs should be performed for the greatest possible number of families of events, without implying that the safety of a pressurized water reactor be demonstrated through an extensive probabilistic analysis.

At first (1976 - 1978), the reliability of all safety related systems was studied for the Fessenheim plant, commissioned in 1977 and representative for the series of about thirty CP1 and CP2 standardised 900 MWe PWRs. In 1981 - 1983 similar studies were performed for the Paluel 1300 MWe PWR, representative for about twenty plants of the P4 series. These studies were incorporated in the regulatory process and examined by the safety authority before plant start up.

The reliability analysis of some redundant systems, frequently or permanently used, showed the necessity of complementary provisions to ensure a satisfactory level of safety for certain situations not included in the list of conventional design basis conditions: ATWS, total loss of ultimate heat sink, total loss of electrical power supplies. To face such "beyond design basis" conditions, now called in France "complementary situations" additional means were defined and then implemented. Operating procedures, H procedures, were specified for the use of these measures.

For the new project N4 (1400 MWe PWR), a probabilistic approach was used to demonstrate the efficiency of the H procedures, taking a probabilistic target into account (10^{-7} per year for a family of events as a limit for core melt probability).

In parallel, probabilistic approaches are used as a support for defining technical specifications in the case of partial unavailability of safety-related systems.

Status and Outlook

Two level-1 PSAs have been performed, one since 1983 for a generic 900 MWe reactor by CEA/IPSN and one for a 1300 MWe reactor, since 1986 by EDF.

These studies were completed in 1990. A crossed external review was performed by CEA and EDF before the final phase of the studies.

The main objective of the CEA/IPSN PSA was to provide the Safety Authorities with a tool for safety analysis of the 900 MWe series. The objectives of the EDF study were to verify the overall safety level, to check possible weaknesses of design and operation, and to improve the safety of the 1300 MWe plants.

The main specific features of the studies are the account for all the states of the plants, an extensive use of French experience feedback (data, human factors), a detailed modelling of recovery actions and emergency procedures, and a computerized system for a "living PSA" approach (LESSEPS software).

Some major insights have been drawn from these studies, for instance:

- the contribution of states other than full power is high (about 30% for the 900 MWe and 50% for the 1300 MWe).
- particular sequences requiring immediate measures were identified: sequences initiated by a spurious dilution, and sequences during cold shutdown and mid-loop operation.
- the benefit related to the procedures developed in France for certain beyond design basis situations, and to the human redundancy due to the safety engineer, is significant.

For the near future the PSA will be used in the following areas:

- the periodic reassessment of 900 and 1300 MWe plants safety,
- the safety assessment of the future N4 series (1400 MWe),
- the emergency operating procedures,
- the optimisation of technical specifications,
- the improvement of equipment reliability.

Moreover, these studies will be kept alive, continually introducing recent operating experience feedback and new safety study results.

FRANCE

Plant	Analysing Team/ Date of Starting Completion	Methods Used/ Procedure Guide	Goal of the Analysis Insight of Results Applications
Fessenheim 900 MWe PWR Operating since 1977 (repre-sentative for about 30 CP1 & CP2 plants)	EDF 1976 - 1978	FMEA, fault trees Markov graphs	Study of the reliability of all safety-related systems
Paluel 1300 MWe PWR (repre-sentative for about 20 P4 plants)	EDF 1981 - 1983		Study of all safety-related systems (altogether 15)
Generic 900 MWe PWR	CEA/IPSN 1983 - 1988	Level 1 PSA without external events. Including all operating states of the plant and long term post-accident situations. Living PSA model (LESSEPS software)	Living PSA model which will be the basis of a permanent tool for safety analysis. Main insights: importance of shutdown states, benefit due to emergency procedures, specific sequences requiring plant modifications (dilutions - loss of RHRS)

FRANCE (continued)

Plant	Analysing Team/ Date of Starting Completion	Methods Used/ Procedure Guide	Goal of the Analysis Insight of Results Applications
Paluel 3 1300 MWe PWR	Three EDF directorates FRAMATOME 1986 - 1988	Living PSA of level 1, without external events, but including different operating states, i.e. shutdown states, specific software (LESSEPS, EXPRESS) and data banks (SRD, CONFUCIUS)	Verification that the core melt frequency is significantly reduced due to the specific French approach. Balanced design, identification of weak points, re-evaluation of technical specifications. Living PSA model for continuous control and monitoring of the overall plant safety. Main insights similar to CEA/IPSN PSA
N4 Project 1400 MWe PWR start of operation 1992	1983	PSA of accident sequences related to the total loss of 4 different redundant safety systems, including detailed human reliability analysis, common cause failure analysis, equipment or systems repairs, except failures within the reactor building	Analysis of the risk due to the total loss of 4 redundant safety systems and of the benefit of additional means or procedures

FEDERAL REPUBLIC OF GERMANY

Abstract

PSA activities in the Federal Republic of Germany evolved mainly from two different approaches:

- systematic reliability analyses for licensing purposes (sufficient reliability, balanced design);

- full scope level 3 PSA for a 1300 MWe-PWR (GRS-A) and level-2 PSA for a 1300 MWe BWR (now underway) for research purposes.

For the past applications of lower level PSA within the licensing procedures, the respective requirement is contained in the BMU Safety Criteria for Nuclear Power Plants, Criterion 1.1. Furthermore, a number of PSA requirements are contained in guidelines and technical rules.

For new LWRs - they are all PWRs - plant-specific level 1 PSA information is available. This, combined with results from the German risk study phase B, allows to extrapolate towards level 2 (and to some extent also level 3 PSA) assessments.

For all operating power plants, extended reliability analyses are available which for specific problems have been, and will be, supplemented by additional analyses. In connection with accident management considerations for some BWRs, beyond design base accident sequences have been analysed - including probabilistic assessments.

For the feedback of operational experience, a plant specific precursor study has been performed.

For future plants, vendors and research institutions use PSAs for integral safety assessments and design optimisations.

The current programme is based on a decision by the Supreme Federal Regulatory Authority, the BMU, that in future utilities should perform periodic reassessments, every ten years of operation, including a PSA. This requirement has also been included in some operation licenses. The reactor safety commission (RSK) was asked to give advice on specific requirements, how to perform these reassessments. The RSK recommendations are contained in the Final Report, Results of the Safety Review of Nuclear Power Plants in the Federal Republic of Germany, November 23, 1988.

As recommended by the Reactor Safety Commission in November 1988, a living level 1 + PSA (level 1 plus active containment related systems) should be part of future safety reassessments.

With respect to the quantitative outcomes of a PSA current practice is that implicit, qualitative regulatory requirements (safety criteria) and case by case and decisions by the Supreme Regulatory Authority are used.

Programme Development

Probabilistic analysis of the reliability of safety systems has been employed during licensing review for about 15 years. The first detailed fault tree analyses were performed for the ECCS and the shutdown system mostly restricted to large LOCAs. With growing experience and maturity of methods and data, the scope and depth of PSA for licensing purposes were extended in order to ascertain sufficient reliability and the well-balancedness of design, as required by the safety criteria for Nuclear Power Plants. The unavailability of primary and secondary heat removal systems during small, medium and large LOCA, loss of off-site power and loss of main feedwater, for example, were analysed probabilistically. The scope of the PSA was restricted to a number of important initiating events. Formal numerical criteria have not been established. Results were evaluated using relative assessment principles and by comparison with earlier reliability analyses.

During 1976 - 1979, the German Risk Study, Phase A (GRS-A) - a level 3 PSA - was performed under sponsorship of the Federal Ministry for Research and Technology. Using WASH-1400 methodology the risk of a PWR under the specific German conditions was investigated. In Phase B of the GRS during 1981 - 1989, improved methodology and new results of safety research were employed to consider a broader spectrum of events using more realistic modelling assumptions and plant specific data.

Experience and insights gained from these risk studies and other investigations influenced reliability and safety considerations in regulation and licensing. Different modifications of design and operation were carried out.

Status and Outlook

After the German Risk Study Phase A for the Biblis Nuclear Power Plant Unit B, some lower level PSAs for operating plants were initiated to transfer safety relevant results. A plant specific Precursor Study has been carried out for the 2 Units of the Biblis Nuclear Power Plant to integrate operational experience into PSAs.

In June 1989, the results of the German Risk Study Phase B were published.

PSA methods were used in this study to evaluate event sequences that can lead to plant hazard states, to identify vulnerabilities and to evaluate possible means and procedures for safety improvement. For plant conditions beyond the design basis (plant hazard states), safety reserves were analyzed. Accident management procedures to prevent and mitigate severe accidents have been identified and evaluated. Recent results of reactor safety research have been used for better modelling of accident scenarios.

The results of the German Risk Study Phase B (GRS-B), have been transferred to other comparable plants to identify relevant severe accident sequences, including those leading to early containment failure, and also to develop appropriate accident management measures by flexible use of existing systems of by adequate additional equipment. Special consideration is given to long term control of successfully managed beyond-design-base events.

For BWRs, no full scope PSA is available. Nevertheless, some accident sequences considered to be risk-dominant have been identified and analysed using the Source Term Code Package (STCP). Preventive measures (inerting the containment, controlled filtered containment venting) have been introduced.

A PSA is now underway for the Gundremmingen 1300 MWe BWR, under contract of the Federal Research Ministry for Research and Technology (BMFT). The first part of this PSA shall be finished in early 1992. A level 1 PSA shall be performed for the Philippsburg 900 MWe BWR under contract of the Federal Ministry for Environmental Protection, Nature Conservation and Nuclear Safety (BMU), the results of this PSA are expected in 1993. Furthermore, some regulatory authorities have already asked for first mini-PSAs for other BWRs.

Utilities have started their own programmes to fulfill current requirements and future needs. Special working groups have been established. It has been decided by the supreme regulatory authority that for all operating nuclear power plants, periodic safety reassessments have to be performed every ten years of operation. A level 1 + PSA should be part of these reassessments, and should be maintained current, using operating experience, insights gained through incidents and accidents, and the progress of science and technology. A time schedule for the studies has been fixed. It has been recommended to collect plant-specific data at least for the components of safety important systems. Utility organisations have started developing common approaches to data collection.

A first version of a PSA-procedure guide has been published by the supreme regulatory authority in October 1990. This guide should be used for the required PSAs.

With respect to living PSA approaches, a computerized Safety Analysis and Information System (SAIS) is being developed. SAIS - implemented on a workstation - consists of a plant specific PSA Level 1+ including PSA models, data and computer tools for the modification and re-evaluation of the event and fault trees. Thus it can be used for a living PSA. System and component data and graphics are also part of SAIS so as to provide supporting information for plant engineers.

The main parts of the SAIS data bank and the PSA analysis tools are already available and will now be implemented for the reference plant NPP-Brokdorf (PWR).

For the further development, a major investigative programme has been defined. The programme will start at the end of 1991. Main topics are:

- Current state-of-the-art and results of PSA studies
- Characterizations of the state of safety technology with the help of qualitative and

quantitative PSA results

- PSA review
- improvements of PSA procedure guide
- improved methods for special issues like human factors dependent failures, non full power operation, uncertainties
- development of a precursor programme
- PSA for fire and external events
- PSA for containment performance related to severe accidents
- qualification of PSA-Codes
- PSA for passive or inherent safety features
- Contribution to an integral risk management approach

FEDERAL REPUBLIC OF GERMANY

Plant	Analysing Team/ Date of Starting Completion	Methods Used/ Procedure Guide	Goal of the Analysis/ Insights of Results Applications
Brunsbüttel KKB 770 MWe BWR	Vendor/Utility TÜV-Nord-Deutschland/GRS 1976	Fault tree analyses	Analysis of system reliability, identification of weakpoints and determination of test intervals and allowable repair times within licensing process
Krümmel KKK 1260 MWe BWR	Vendor/Utility TÜV-Nord-Deutschland/GRS 1981	Fault tree analyses	Analysis of system reliability, identification of weakpoints and determination of test intervals and allowable repair times within licensing process
Unterweser KKU, 1230 MWe PWR	TÜV Rheinland/GRS 1975/1976 Vendor/TÜV Nord-Deutschland 1980	Fault tree analyses	Analysis of the emergency cooling system Fuel element storage pool cooling system
Mülheim-Kärlich 1300 MWe PWR	BBR TÜV Rheinland/GRS 1978	Equivalent to WASH 1400 Reliability analyses, e.g. for emergency coolant injection system	Analysis of plant safety and reliability of safety systems equivalent to level 1 PSA Primary and secondary side accident initiators and related safety system functions selective backfitting measures

FEDERAL REPUBLIC OF GERMANY (continued)

Plant	Analysing Team/ Date of Starting Completion	Methods Used/ Procedure Guide	Goal of the Analysis/ Insights of Results Applications
Biblis-B KWB/B 1300 MWe PWR in operation "German Risk Study Phase A (GRS-A)"	GRS & others 1979	Mainly analogous to WASH-1400 Success criteria in general defined according to requirements of the licensing procedure.	Research, determination of risk from a variety of PWRs in the Federal Republic of Germany
Grohnde, KWG Philippsburg 2, KKP II Brokdorf, KBR Philippsburg 2, KKP II 1300 MWe PWRs then under construction	KWU Vendor/Utility TÜV Norddeutschland 1981 - 1982 TÜV Südwestdeutsch-land for KKP II 1981	Common fault tree analyses due to similar plant design Mainly analogous to GRS-A expert review	Check of well-balancedness of safety systems in the licensing process and determination of test intervals
Neckarwest-heim 2, GKN II Emsland, KKE, Isar 2, KKI 2 1300 MWe PWR Konvoi-plants under construction	KWU 1982 (and additional analysis by TÜV/GRS) TÜV Südwestdeutsch-land for GKN II 1982 GRS 1987	Level-1 analyses anologous to GRS-A differences in CMF and HE modelling Mainly anologous to GRS-A except for specialities in common mode failure rates analogous to GRS-B	Check of well-balancedness of safety systems in the licensing process and determination of test intervals Reliability Analyses of selected system

Plant	Analysing Team/ Date of Starting Completion	Methods Used/ Procedure Guide	Goal of the Analysis/ Insights of Results Applications
Neckar-westheim-2 GKN II, PWR	KWU 1988	Reliability analysis	Within the scope of the licensing procedure
Neckar-westheim-2 GKN II, 1300MWe PWR then under construction	TÜV Stuttgart 1983	Mainly analogous to GRS-A but less detailed	Check for well-balancedness of safety systems in the licensing procedure
SNR-300 330 MWe FBR under construction	GRS & others 1982	Level 3 PSA generally analogous to GRS-A except for design specific features	Research and political decision-making, comparison with risk from PWR (GRS-A)
Biblis-KWB/A&B "German Precursor Study"	GRS 1982 - 1986	Success criteria in general defined according to requirements of licensing procedure. Evaluation of operational experience with respect to frequencies of initiating events and unavailabilities of system functions	Check of suitability of probabilistic evaluation of operational experience with respect to: - increasing safety relevant insight; - identification of weakpoints
Obrigheim KWO, 350 MWe PWR	GRS 1984 - 1989	Level 1 PSA analogous to GRS-A with respect to initiating events, details in fault tree	First PSA to examine the safety of an older plant

FEDERAL REPUBLIC OF GERMANY (continued)

Plant	Analysing Team/ Date of Starting Completion	Methods Used/ Procedure Guide	Goal of the Analysis/ Insights of Results Applications
Biblis-B KWB/B (see GRS-A) "German Risk Study Phase B (GRS-B)"	GRS 1985 - 1989	Plant-specific failure rates, plant-specific redundancy dependent common mode failure rates, best estimate calculations for success criteria, inclusion of accident management measures.	Evaluation of event sequences, identification of vulnerabilities and safety improvements, investigation of safety reserves for event sequences exceeding the design limits, and evaluation of accident management measures
Obrigheim KWO 350 MWe PWR	GRS KWO/Seimens 1990 - 1992	Report on safety status. Current plant description transient and event analysis, probabilistic safety analysis, evaluation of operational experience and events.	Evaluation of technically important components and systems, backfitting measures, realization of RSK recommendation of November 1988
Philippsburg I KKP I BWR 865 MWe	KWU 1989 TÜV Südwest-deutschland/GRS 1991	PSA Level 1 without instrumentation and control systems, PC programmes RISA, REED, TREEMOD, VARDA Expert Review of the level 1 PSA	PSA, quantitative evaluation of safety conception, part of 865 the periodical safety inspection
WWER-440 B1, 1 - 4 W - 230	Energiewerke NORD AG 1989 - 1991	Limited level 1 PSA program FAULT TREE	Plant status, event analysis on partially simplifying assumption back-fitting measures

Plant	Analysing Team/ Date of Starting Completion	Methods Used/ Procedure Guide	Goal of the Analysis/ Insights of Results Applications
Biblis-A KWB/A 1146 MWe PWR	RWE/KWU 1988 - 1991 TÜV Bayern/GRS 1991	Level-1 PSA analogous to GRS-B Expert review	Safety status of the plant, evaluation of relevant courses of events, determination of design reserves of the plant with regard to keeping of protective goals, accident management measures, PSA within the scope of the periodical safety inspection
Gundrem-mingen KRB II, 1300 MWe BWR in operation	GRS 1988 - 1992 first phase	Similar to GRS-B	Similar to GRS-B
Philippsburg I, BWR	GRS/TÜV Rheinland 1991 -	Analogous to GRS-B PSA including with instrumentation and control systems	PSA within the scope of the periodical safety inspection, backfitting measures
Gemein-schaftskern-kraftwerke Neckar I, GKN I PWR 785 MWe	KWU Seimens 1989 - 1990 report to authority, PSA to be continued TÜV Südwest-deutschland 1991	Level 1+ PSA according to guide Expert review	PSA within the scope of the periodical safety inspection, safety status of the plant, courses of events, backfitting measures

FEDERAL REPUBLIC OF GERMANY (continued)

Plant	Analysing Team/ Date of Starting Completion	Methods Used/ Procedure Guide	Goal of the Analysis/ Insights of Results Applications
Isar I KKI 1 BWR 870 MWe	KWU 1990 TÜV Bayern	Level 1+ PSA according to PSA-guide, using PC programme RISK SPEKTRUM Module for common cause failures Expert review	PSA within the scope of the periodical safety inspection
Brunsbüttel KKB BWR 770 MWe	HEW 1990 TÜV Norddeutschland GRS	Level 1+ PSA according to PSA-guide, RISA-reliability programme Expert review	
Stade KKS PWR 660 MWe	KWU 1990 TÜV Norddeutschland GRS	PSA according to PSA-guide and analogous to risk study GRS-B Expert review	Safety status of the plant, evaluation of relevant courses of events, backfitting measures, PSA within the scope of the periodical safety inspection
Würgassen KWW BWR	TÜV Rheinland GRS 1990	Reliability analysis	Analysis emergency coolant injection system
Unterweser KKU PWR 1230 MWe	GRS 1990	Discussion of weakpoints anlogous to risk study GRS-B	Backfitting measures analogous to KWB/B
WWER-1000	GRS 1991	Safety evaluation in cooperation with Soviet experts	Plant status, evaluation of events, proposals for backfitting measures, statements of Soviet experts

ITALY

Abstract

With the growing maturity of PSA methods, the probabilistic approach has found a wide use in Italian safety practices. PSAs were at first requested by the Italian regulatory body for new plant safety optimisation during design and construction. The studies were performed by utility (ENEL) and industry. Meanwhile, the Italian regulatory body has extended the requirements for level 1 PSAs to existing plants in order to assess their actual safety levels and to identify, possible backfitting actions. For core melt a frequency of 10^{-5} to 10^{-6} per reactor year is used as a target. At present an additional probabilistic target is being used on trial for the fission products releases. If core damage occurs, the releases of the most volatile (I - Cs) fission products should exceed 0.1 per cent of core inventory with a conditional probability of less than 5 per cent. This also requires the development of level 2 PSA.

After the suspension of Nuclear Plant operation, following the Chernobyl accident and the national referendum, interest has been focused on new passive and simplified reactors.

Programme Development

Italian safety practices were in the past mainly based on deterministic criteria and review guides. This also included reliability analysis for important safety functions. A more systematic use of PSA techniques was at first performed for the safety assessment of new plants in order to provide assurance that accidents which cause plant degradation are very unlikely to occur.

Practical applications have been extended with the growing maturity of data and methods. For the Alto Lazio nuclear power plant a safety reliability analysis (ALSRA) was performed for assessing the generic adequacy of plant protective functions. The request of the study was attached to the construction permit of the plant, but without specified numerical targets. The study made use of the current PRA methodology in which conservative calculations of transients and ECCs were made with licensing hypotheses in order to derive success criteria. A number of design changes, in order to meet the generic reliability requirements, have been taken so as to improve the reliability of the relatively weakest plant systems. The requirements are expected to be cleared up before the issue of the operating license. Recently the study has been reviewed by using realistic calculations for the success criteria and was extended to the re-estimation of the core meltdown probability which was lower than 10^{-5} per year.

For the new Italian reference standard plant (PUN), the Italian Regulatory Body issued General Design Criteria which include the application of a probabilistic safety study (PSS) with the use of numerical targets for the probability of core melt accidents. This should be in the range from 10^{-6} to 10^{-5} per year. The methodology described in

NUREG/CR-2300 was used in the study, in which realistic calculations of transients and ECCS were made for the success criteria. Design improvements were identified throughout the study in order to meet the probabilistic targets.

The studies have been submitted to the Regulatory Body of review. A systematic review procedure has been established. The main issue is the extent of the upgrading of plant design which is required in the light of the analysis results.

Status and Outlook

The increased need of more plant safety for protection of public health led the Italian regulatory body to also extend the application of PSA to operating plants to assess their actual safety levels and identify possible improvements. Level 1 PSAs have been performed for all plants. Modifications and procedures were taken for further reduction of the core damage frequencies which were assessed through the PSA.

The analysis of containment performance under severe accident conditions and of external releases is difficult because of the still limited understanding of some important phenomena and uncertainties concerning accident phenomena. Level 2 PSAs are also foreseen for all Italian plants, but they are used with great care. A trial use is made of the probabilistic target for external releases which is a conditional probability of 5 per cent for exceeding a 0.1 per cent I-Cs release, given a core damage. In order to meet this target, in spite of the uncertainties in accident phenomena, catastrophic early containment failures (by steam explosions) are excluded from a detailed consequence analysis, since these very uncertain events are considered to be extremely incredible. On the basis of the present status of knowledge it is believed that other events which are potentially capable to threaten the containment integrity, can be mitigated an that level 2 PSA can be useful to identify improving measures for the containment system.

At present, since the regulatory body required that PSAs should be interactive with the design, some of these studies (Caorso, Alto Lazio, Pun) have been turned into living PSAs on Personal computers.

Moreover, the activities are now concentrated on the study of the new passive and simplified reactors like SBWR, AP600 and Prism.

Living PSAs for SBWR and AP600 are being developed since 1989. The SBWR study has been updated according to changes in the design in 1991. The AP600 updating is presently ongoing and is foreseen to be completed by the end of the year.

Plant	Analysing Team/ Date of Starting Completion	Methods Used/ Procedure Guide	Goal of the Analysis/ Insights of Results Applications
Latina 150 MWe Magnox Operating since 1964	ENEL/NNC 1986	Level-1 PSA	Probability to exceed safety constraints, practical improvement to reduce core damage frequency
Trino Versellese 260 MWe PWR	ENEL/Westing-house 1989	Level 1 PSA and severe accident analysis including specific items (H2)	
Caorso 860 MWe BWR 4 Mark II Operation since 1981	ENEL/NUS 1986	NUSSAR PC Workstation Level 1 PSA. Preliminary containment event trees	Assurance of high preventive safety. Design changes for decreasing the core melt probability (10^{-5} - 10^{-6})
	ENEL 1986	Individual plant examination for source terms (MAAP3)	Identification of severe accident mitigating features
	ENEL 1986-1987	Level 2 PSA	0.05 probability to exceed 0.1% I-Cs, given core damage
Alto Lazio 2 units 900 MWe BWR 6 Mark III	ENEL/ANSAL-DO/GE 1984	Level 1 (ALSRA)	Identification of design improvements. Generic adequacy of the plant protective functions Core melt probability in the range 10^{-5} - 10^{-6}
under construction	ENEL 1980	Level 1 (AL-PSS)	0.05 probability to exceed 0.1% I-Cs, given core damage
	1987 - 1988	Level 2	

Plant	Analysing Team/ Date of Starting Completion	Methods Used/ Procedure Guide	Goal of the Analysis/ Insights of Results Applications
PUN Italian Reference Standard Plant 2 units per site 900 MWe PWR	ENEL/NIRA/ Westinghouse - 1984 (first phase)	Level 1 NUREG/CR-2300	Systematic assessment of plant behavious under accident conditions, balance assessment of plant defenses, support of practical improvements during design Core melt probability in the range of 10^{-5} - 10^{-6}
	ENEL 1986 - 1987	Individual plant examination for source terms (STCP)	Identification of severe accident mitigating features
	ENEL 1987 - 1988	Level 2	0.05 probability to exceed 0.1% I-Cs, given core damage
SBWR 600 MWe	ENEL/ENEA/ ANSALDO/GE 1989 (1st phase)	Level 1&2	Identification of design improvements Adequacy of the plant protective functions
	1991 (2nd phase)	Updating	
AP600 600 MWe	ENEL/ENEA/ ANSALDO/ WESTING-HOUSE	Level 1&2 in progress	Identification of design improvements. Adequacy of the plant protection functions

JAPAN

Abstract

Most of the current licensing procedures in Japan are based on the deterministic approach. The probabilistic approach has been used to supplement regulatory decisions. Recently, the usefulness of PSA has been more and more widely recognized. Research organizations have been developing a series of methodologies for system reliability analysis, operational data analysis, core melt accident analysis, environmental consequence analysis and seismic risk analysis. Electric utilities and research organizations are carrying out PSAs for nuclear power plants.

Programme Development

Most of the current licensing procedures in Japan, with a few exceptions, are based on the deterministic approach. In licensing applications, the probabilistic approach has been used to supplement the regulatory decisions, but will not replace the deterministic approach in the near future.

Since the usefulness of PSA has been widely recognized in Japan, research institutes have been engaged in the development of methodologies. For example, the Japan Atomic Energy Research Institute (JAERI) has been developing a series of methodologies for PSA of LWRs. Also, the Power Reactor and Nuclear Fuel Development Corporation (PNC) has been developing methodologies through the PSA of a prototype FBR plant "Monju".

According to the progression of methodology development, its application to practical problems has become more popular. Utilities and research organizations are carrying out PSA for Japanese NPPs.

According to the progression of methodology development, its application to practical problems has become more popular. Utilities and research organizations are carrying out PSA for Japanese NPPs.

Status and Outlook

Probabilistic Approach in Licensing

Licensing procedures in Japan are principally deterministic. In some cases, however, the probabilistic approach is followed to define the scope of safety evaluation. For example, a probabilistic study revealed that the frequencies of station blackout and multiple failures in process systems are low enough that such accidents can be precluded from the scope of deterministic safety evaluation of NPPs in Japan. The probabilities of turbine missiles and airplane crashes are estimated in the course of design examination of NPPs.

PSA Methodology Development

Level 1 PSA:

JAERI provided code packages REFT and GO-UA for system reliability analysis based on the fault tree and GO methods, respectively, and has developed PC-based programmes PC-CREFT and ETAP for constructing fault trees and analyzing event trees. JAERI is carrying out a FT benchmark exercise to cooperate with IAEA in its coordinated research programme "Reference Study on Probabilistic Modelling of Accident Sequences".

PNC developed a PC-based interactive PSA code "QUEST", a fault tree construction programme based on the modular method "MODESTY" and event tree construction programme using an expert system "ETAAS" as well as large code packages for system reliability analysis based on the fault/event tree method. Also, PNC has been developing a PC-based living PSA system "LIPSAS".

The Ship Research Institute developed the GO-FLOW code.

The Nuclear Power Engineering Test Center (NUPEC) and Industry Groups established their own methodologies for system reliability analysis, mostly based on the fault tree method.

Component Reliability Database

Component reliability data are collected by the utilities and sent to the Central Research Institute of Electric Power Industry (CRIEPI) and (NUPEC) and these are analyzed and evaluated there statistically. A computerized reliability database, RECORD, was developed at JAERI as a supporting tool for PSA. Abnormal occurrence data are not only analyzed by the relevant utilities but also by CRIEPI and NUPEC. Human reliability data are also collected by the utilities using plant simulators.

As for FBR, PNC has been compiling and integrating the operating experience of its facilities in the FREEDOM/CREDO database and exchanging them with those in the USA. PNC has developed a new management system for the CREDO database based on a relational database AIM/RDB. The CREDO database provides the reliability, availability and maintainability measures such as failure rates and repair times. PNC developed a component failure rate analysis code using the Bayesian technique "BAYES", a failure rate trend analysis code using the multiple regression "TREND", and an aging factor analysis code using the linear aging model "AGE".

Human Factor Analysis

The importance of human factor research is highly recognized and many new programmes have been started in Japan. The programmes at JAERI, NUPEC and CRIEPI are comprehensive ones and cover various aspects - such as human reliability analysis, man-machine interface research, operational management, training, utilization of artificial intelligence for operational aid and collection and analysis of human reliability data. Utilities are collecting and analyzing human behavior data at their training centers. For human reliability analysis, JAERI developed a DeBDA methodology based on Detailed Block Diagram Analysis method. PNC has developed a PC-based interactive human reliability analysis code HURASS/SHERI principally based on the THERP method.

Level 2 PSA:

In order to estimate the source terms, JAERI developed the THALES/ART code package for analyzing progression of a core melt accident and fission product release and transport behaviour. The integration of the code package into a single code THALES 2 has been performed. The utilities are using and modifying the MAAP code developed in the IDCOR Program and NUPEC is using the STCP code introduced from USNRC. As for FBR, PNC has been developing codes; SAS, SIMMER, SSC, BBC, PLUG, DEBRIS-MD, APPLOHS, CONTAIN, etc. for analyzing hypothetical core disruptive accident sequences.

Level 3 PSA:

In order for assessing the off-site radiological consequence of nuclear accidents, JAERI has developed the OSCAAR computer code package, which consists of interlinked computer codes to predict (1) Transport of radionuclides through the environment to man, (2) Subsequent dose distributions, and (3) Health effects in the population. PNC improved the CRAC2 code so as to take into account long-term FP release and the effects of undulating topography.

Seismic Risk Analysis

Several organizations are now eager to develop the methodologies for seismic risk analysis, intending to use the analysis results mainly in seeking more balanced seismic designs or regulations. JAERI and PNC have established whole sets of methodologies for seismic risk analysis of LWR and FBR respectively. Among those tasks required for the seismic risk analysis, the seismic hazard evaluation has progressed, where long history of earthquake records and active fault data are utilized with the experts' interpretations.

Value Impact Analysis

PNC has developed a value impact analysis code "VIA" with a view to examining the rationalization of safety design policy, which can compare among NPPs through quantitative comparisons of the attributes of the base plant and alternative plants. In this analysis, the risk to the public and the cost are selected as the plant attributes.

PSA for Nuclear Power Plant

JAERI started level 2 PSAs for a model BWR/5 Mark II and a model PWR in 1986 and 1990, respectively, aiming at demonstrating the applicability and usefulness of the PSA methodologies developed at JAERI. As part of this programme, a level 1 PSA for the BWR on earthquake induced events was initiated in 1991.

The Tokyo Electric Power Company (TEPCO) and vendors used the event tree/fault tree technique in the design of an advanced BWR (A-BWR) in order to find out the best safety design of the plant. TEPCO and vendors are also carrying out level 1 PSAs for BWR/3 Mark I, BWR/4 Mark I and BWR/5 Mark II and level 2 PSAs for BWR/4 Mark I and BWR/5 Mark II and II Modified. They also started study of level 1 shut-down phase PSA for BWR/4 and BWR/5.

The Japanese PWR Industry Group carried out a level 1 and 2 PSAs for the Japanese 4-loop PWRs of large dry containment type and of ice condenser containment type.

NUPEC has carried out level 1 and 2 PSAs for 1100MWe class PWR and BWR plants under sponsorship of MITI since 1987. Then level 2 PSAs were started in 1988 to understand the spectra of accident sequences. NUPEC also started a level 1 PSA for an advanced BWR which is now under construction permit stage and for 800MWe BWR and PWR.

PNC has been carrying out level 3 PSA of the Monju plant since November, 1982, aiming at constructing a probabilistic model to be used in evaluating the overall safety of the plant.

JAPAN

Plant	Analysing Team/ Date of Starting Completion	Methods Used/ Procedure Guide	Goal of the Analysis/ Insights of Results Applications
ABWR 1300 MWe	TEPCO, BWR Vendors 1984 - 1988	Level 1 PSA	To find out best design concept
BWR/3 MKI BWR/4 MKI BWR/5 MKII	TECO, BWR Vendors 1984 - 1988	Level 1 PSA	To evaluate the difference of system configuration
BWR/4 MKI BWR/5 MKII BWR/5MKII modified	TEPCO, BWR Vendors 1984 - 1990	Level 2 PSA	To provide supplemental information
4 loop PWRs large dry containment type, ice condenser containment type	Japanese PWR Group 1984 - 1990	Level 1 PSA Level 2 PSA	To evaluate safety margin and understand the characteristics of the plants To provide supplemental information
BWR/5 MKII model plant	JAERI 1987 - 1989	Level 2 PSA	To verify model applicability and usefulness of JAERI methodologies
MONJU Prototype FBR	PNC 1982 - 1992	Level 1,2&3 Internal and external events	To evaluate overall PSA safety and to use for assisting in operational management
1100 MWe BWR/5 MK II	NUPEC 1987 - 1989 (Phase - 1) 1990 - 1991 (Phase - 2)	Level 1&2 PSA	To supply probabilistic safety information to the regulatory authorities

JAPAN (continued)

Plant	Analysing Team/ Date of Starting Completion	Methods Used/ Procedure Guide	Goal of the Analysis/ Insights of Results Applications
1300 MWe ABWR 800 MWe BWR/4 MKI PWR 3 loop dry contain-ment	NUPEC 1988 - 1990 NUPEC 1989 - 1992	Level 1 PSA Level 1&2 PSA	Backup for licensing procedure To supply probabilistic safety information to the regulatory authorities
BWR/4 BWR/5	TEPCO, BWR Vendors 1991 - 1992	Level 1 Shut-down phase PSA	To evaluate safety of annual outage
1100 MWe PWR 4 loop large dry containment type	NUPEC 1987 - 1989 (Phase - 1) 1990 - 1991 (Phase - 2)	Level 1&2 PSA	To supply probabilistic safety information to the regulatory authorities

The following points are also worth noting:

a) Current Reference Documents:

The outlines of the PSA-related activities in Japan were introduced in the PSA'89-International Topical Meeting: Probability, Reliability and Safety Assessment held at Pittsburgh in 1989 (ref. 1). That paper overviewed the status of research and development (R&D), applications in industries and probabilistic approaches in licensing. A special emphasis is placed on the recent progress of R&D at JAERI and PNC, where systems analysis, consequence analysis and seismic risk analysis are carried out for PSA's of LWR and FBR, respectively.

Then a lot of information was presented in the CSNI Workshop on Applications and Limitations of Probabilistic Safety Assessment held at Santa Fe in 1990. The current policy and status on the PSA applications in the licensing process were summarized in Ref. 2. The results of level 1 PSAs carried out NUPEC and industry groups were introduced in Ref. 3. PSA applications in designing ABWR and APWR were in Ref. 4.

(Ref. 1) K. Sato; T. Tobioka, K. Abe and K. Aizawa "Current Status on PSA-related Activities in Japan", PSA '89-Intl Top. Mtg: Probability, Reliability and Safety Assessment, Pittsburgh, (1989).

(Ref. 2) S. Kondo an T. Tobioka, "Application of Probabilistic Safety Analysis to Licensing in Japan".

(Ref. 3) M. Hirano, M. Sugawara and H. Fujimoto, "Recent results of Level 1 PSA for Nuclear Power Plants in Japan".

(Ref. 4) A. Yamada and T. Nakamura, "Utilization of PSA Method for Nuclear Power Plant in Japan".

b) Basis of PSA Programme

Licensing procedures in Japan are principally deterministic and there is no regulatory requirements on the probabilistic basis. The governmental safety research in Japan is carried forward according to the five year safety research plan (Ref. 5). The current covers 1991 - 1995 where the research programmes are categorized into six fields, including "Research on Probabilistic Safety Assessment". The research programmes in the PSA field are carried forward mainly by the Japan Atomic Energy Research Institute (JAERI) and the Power Reactor and Nuclear Fuel Development Corporation (PNC) and partially by the Ship Research Institute. The progress of these research programmes is annually reviewed by the governmental review committee.

(Ref. 5) Nuclear Safety Commission/Committee of Safety Research on Nuclear Facilities, "Annual Programme of Safety Research in Japan-Fiscal Year 1991 to 1995".

c) 1. Who has to perform PSAs?

2. Scope, depth and methods of PSAs.

Nuclear Safety Commission of Japan encourages industry, research organizations and regulators to perform PSA and study the phenomenology of severe accidents. As for the PSA of LWRs, JAERI, NUPEC and the utilities are performing level 1 and level 2 PSA of typical plants. As for the PSA of FBR, PNC is performing level 3 PSA of the prototype FBR, Monju.

The results of the PSAs are expected to be used in improving operation procedure for accident so as to flexibly utilize the safety margins to terminate the progression of Beyond DBEs or mitigate their effects, though this is not the part of regulatory requirement in Japan.

(See Table A-1)

3. Reviews of the Results of PSAs

Level 1 and 2 PSAs of the typical plants of BWRs and PWRs have been conducted by the Japanese BWR Group represented by TEPCO and by the Japanese PWR Group respresented by the KANSAI Electric Power company (The KANSAI), respectively. NUPEC has also conducted level 1 and 2 PSAs of BWRs and PWRs. The reviews of the results of the PSAs are conducted primarily by the analysts within each organization. After those intra-group reviews, the utilities and NUPEC had submitted voluntarily the results of their level 1 PSA to Nuclear Safety Commission of Japan for peer review. Nuclear Safety Commission conducted peer review of those PSA results in Japan and issued the review report as a part of the interim report on the fundamental principles on severe accidents, review on PSA results and accident management and future task for examination in February 1990.

d) 1. The results of the Independent Peer Review

The integral core damage frequencies for all the typical BWR and PWR plants are lower than 10^{-5}/ry which is the reference value for new type reactors proposed in "Basic Safety Principles for Nuclear Power Plants" by the INSAG of IAEA. This is mainly attributable to the fact that the frequency of transients is relatively small in Japanese plants due to the proper operation management.

2. The Analysts' Responses

The results of level 1 PSAs for the typical plants provide insights into the safety characteristics of design and operation of the present LWRs.

Table A-1 Current Status on PSA Studies in Japan

Organization	Plant	Scope & Depth of PSAs	Methods of PSAs
TEPCO BWR Vendors	BWR/3 MK I BWR/4 MK I BWR/5 MK II BWR/4 MK I BWR/5 MK II BWR/5 MK II mod.	Level 1 PSA, estimate uncertainty distribution as well as point values, in-plant initiators Level 2 PSA point values	FT/ET methods, refer NUREG/CR-2300, -2728 and -1278 Operator recovery as well as human errors and common cause failures are considered.
The Japanese PWR Industry Group	4 loops PWRs, large dry contain-ment type and ice condenser containment type	Level 1&2 PSA ditto	FT/ET methods, refer NUREG/CR-2300 and -1278 ditto
NUPEC	1100 MWe PWR (4 loop, dry) 1100 MWe BWR (BWR/5 MK II)	Level 1&2 PSA ditto	
	1300 MWe A-BWR 800 MWe BWR/4 MK I 800 MWe PWR 3 loop dry containment	Level 1 PSA ditto	ditto
JAERI	BWR/5 MK II Model plant	Level 2 PSA, estimate uncertainty distribution as well as point values, in-plant initiators	ditto
PNC	Prototype FBR	Level 1,2&3 PSA estimate uncertainty distribution as well as point values, internal & external events, estimate location-dependent failure effect	ditto use location transformation method

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Apart from the traditional reliability analyses, most of the current regulating process is based on a deterministic approach. Recently, the usefulness of PSA has been recognized. Before Chernobyl, a level 3 PSA was foreseen to be part of the siting procedure for the at the time proposed new nuclear power plants. After Chernobyl, at least a level 1 PSA was recommended for the existing nuclear power plants with the purpose to optimize plant improvements. The bid specification for the PSA of the Dodewaard BWR is in preparation and for the Borssele PWR the actual PSA-work (level 2 minus) has already started.

Programme Development

Beside the traditional reliability analyses within the existing regulating process, a PSA programme has been set up. Before Chernobyl, two new nuclear power plants were foreseen in the Netherlands. To show compliance with the at that time newly postulated environmental safety goals for new hazardous installations, a level 3 PSA was foreseen to be part of the siting and licensing procedure. Within this framework Electrowatt Engineering Services UK (Ltd.) was asked to develop a PSA Procedures Guide for the Dutch government. This procedures guide was, although heavily relying on NUREG/CR-2815 "PSA Procedures Guide" and NUREG/CR-2728 "IREP Procedures Guide" tailored to the Dutch situation (to show compliance with the safety goals).

After Chernobyl, the decision to expand the nuclear power capacity was postponed. The government decided to reconsider the nuclear option. Several studies were initiated to help them with this decision-making process. One of these studies was about the possible accident management measures of the nuclear power plants Borssele end Dodewaard. This study, performed by GRS, recommended to perform at least a level 1 PSA for both plants with the purpose to optimize plant improvements. For Borssele, a 472 MWe KWU-PWR, both the licensee and the licensing authorities agreed with this proposal. This resulted in a bid specification for a level 2 minus PSA. This PSA-project was awarded to the combination KWU and NUS. The actual PSA-work started 1st September 1989, and will last approximately 2 years.

Early 1990, bid specifications for a level 2 minus PSA of the Dodewaard plant (58 MWe BWR) were sent out by GKN (utility which operates the Dodewaard nuclear power plant) and KEMA. The study was awarded to Science Applications International Corp. (SAIC) from the USA. In April 1990, SAIC started with the analyses; plant familiarization and plant-specific data analysis.

One of the main purposes of both the PSAs is to identify forgotten scenarios with respect to the GRS-study on accident management studies. The latter was based on insights gained from the German Risk Study-Phase B (although not identical, Biblis-B is a larger sister of the Borssele power plant). Other purposes are identification of the dominant accident sequences leading to core-melt, estimation of the contribution of operator errors to the core-melt frequency, estimation of containment response to severe accidents, identification of dominant pathways in the containment event tree and, last but not least, guidance on backfit decisions and accident management measures. Because it is intended that both the PSAs will be used as an operational tool (backfitting, technical specs, etc.) the PSA is to be maintained as a so-called living PSA.

For the Borssele PSA the PRA Procedures Guide (NUREG/CR-2300) and the PSA Procedures Guide (NUREG/CR-2815) will be used as guidance for the PSA.

External events will be treated in a more qualitative way. For the seismic events a plant-walk-through is foreseen. If necessary a more quantitative analysis will be performed in a later stage. Internal hazards (area events) will be quantitatively analyzed in detail. For the fire initiating events an analysis with COMBRN is foreseen. The human error analysis will be structured around the SHARP methodology. Both HCR model and THERP will be used within this framework. The system modelling uses the small event tree/large fault tree approach. For fault trees the NUPRA-code will be used.

As far as possible, plant specific data will be used for both plants. If these are not available, generic best estimate data from other sources will be used. For Borssele, approximately 100 component-types plant specific failure data will be collected. Uncertainty analyses will be carried out.

The already existing thermal-haudraulic transient analyses (e.g. LOCA analyses performed with RELAP V) will be used for further analyses, like the definition of realistic success criteria. In case these analyses are not available, conservative assumptions from the safety report will be used, but in case these assumptions are too unrealistic, simple best estimate transient analyses will be performed.

For Borssele, the first phase of a peer review by the IAEA took place in the last week of August 1989. This review involved the scope of the project and how this scope was translated into a project proposal by the contractor. The review was conducted by a team of PSA specialists under the IAEA's recently initiated International Peer Review Services (IPERS). The peer review is split into three phases. The second phase has been conducted in June 1990, approximately halfway the project. The last phase will take place after 90% completion of the PSA. In combination with this peer review, a training course on the review of PSAs was given by the team members of the IPERS-team for staff and consultants of the Dutch regulatory authorities.

For Dodewaard, the first phase of an IPERS Peer Review took place in May '91 after approximately 60% completion. The second and last phase will take place in February 1992, after 100% completion.

The Dutch regulatory body asked the utilities to expand their PSAs with an analysis of the non-power states. Also they requested a special study to identify those human errors of commission which may cause plant degradation. Especially, sequences of 'wrong' human errors in the cognitive domain area of interest.

Recently, it has been decided by the Dutch government to ask for a full scope level 3 PSA from both nuclear power plants. Therefore, the PSAs under 'construction' will be expanded to a level 3 PSA. Both these studies should be finished in 1993. Backfitting as foreseen for both NPPs will be included in these PSAs.

Because the state-of-the-art of PSAs is changed since the aforesaid Electrowatt PSA-guide, a new PSA procedures guide is in preparation. Recently comparisons of existing PSA guidelines and state-of-the-art techniques for level 1 and 2 PSAs have been completed. In a guiding document special attention is given to the process of showing compliance with probabilistic risk criteria, how to deal with uncertainties, arguments and guidance for cut-off frequencies, and other possible applications of PSAs.

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Plant	PRA Program	Analysing Team/ Starting date/ Duration	Peer Review	Scope/ Methods Used/Procedure Guides Codes Used	Goal of the Analysis/ Insights or Results/ Applications
BORSSELE KWU PWR 472 MWe special features: bunkered primary & secondary side reserve supplet. systems	Utility	KWU/NUS Sept 1989/ 2 years.	Peer Review by IAEA (IPERS-progr.); in 3 phases: 1st at project initiation, 2nd at 50% completion, 3rd at 90% completion. Detailed review by regulating authorities+ detailed 100% review by KEMA on behalf of the utility.	Level 2 minus/NUREG/CR-2300, NUREG/CR-2815, SHARP (HCR+THERP) Plant walk-through for external event NUPRA STCP; WESHSL; RELAP V; COMBRN IV; for fire analysis. All power states will be analysed. Uncertainty analysis in included.	Identification of weaknesses and forgotten scenario's with respect to the GRS-study on possible accident management measures. Evaluation of core melt sequences. Guidance for backfitting. Development of accident management procedures. PSA will result in living PSA structured around NUPRA.
	Utility	Unknown 1991/1992 (bid specs. have been sent out in early 1991)	Detailed review by Dutch regulatory body.	Analysis of non-power states	Assessment of importance of non-power states for total risk. Special attention for reactivity incidents.
	Utility	unknown 1992/1993		Extension to complete level 3	Comparison with PSC Influence of Accident Management

Plant	PRA Program	Analysing Team/ Starting date/ Duration	Peer Review	Scope/ Methods Used/Procedure Guides Codes Used	Goal of the Analysis/ Insights or Results/ Applications
DODEWAARD GE BWR 58 MWe Special features: natural circulation emergency condensor 100% turbine bypass capacity	Utility	SAIC/April 1990 16 months	Detailed review by Dutch regulatory body + detailed review by KEMA. Peer review by IAEA IPERS-progr.; 2 phases: 1st at 50% completion, 2nd at 100% completion.	Level 2 minus. Electrowatt-PSA procedures guide, NUREG/CR-2300, NUREG/CR-2815/SHARP (HCR)/SLIMM AUD for HRA. Multiple Greek letter for CCFs. Generic B- factor for screening CCF. Detailed internal flooding analysis. Screening of other external hazards. Critical safety function in EOPs are used to const. functional event sequence diagrams (FESDs). FESDs are used as a basis for Event Trees	Identification of weaknesses. Evaluation of core melt frequency and ranking of most important sequences. Identification of weaknesses in operating, test, maintenance and emergency procedures. Technol. transfer for living PSA applications. Living PSA structured around CAFTA & ETA for level 1 part and around STCP for level 2 part.
	Utility	unknown 1991/1992	Detailed review by Dutch regulatory body	Analysis of non-power states	Assessment of importance of non-power states for total risk. Special attention for reactivity incidents.
	Utility	unknown 1992/1993		Extension to complete level-3	Comparision with PCS influence of Accident Management

SPAIN

Abstract

An Integrated Programme on PSA realisation and use in Spain has been established by the Nuclear Safety Regulatory Agency, CSN. A specific PSA for each nuclear power plant in Spain will be done in a phased fashion, with utility-wide participation. This Programme will mean a major review of the safety of the plants. The models developed for each plant can be the basis for future applications in many areas of nuclear power plant performance.

Programme Development

The Spanish Nuclear Safety Regulatory Agency, Consejo de Seguridad Nuclear (Nuclear Safety Council, CSN) required in 1983 from the utility a plant specific PSA for Santa Maria de Garoña nuclear power plant for analysing the overall safety and helping to decide about engineering modifications after thirteen years of operation.

Before extending this type of safety analysis to the rest of the seven operating Spanish nuclear power plants, CSN prepared a proposal on the needs, benefits and possibilities of such a PSA programme. Preliminary results of the pioneer study in Garoña were used.

The general improvement of the plant safety got from the analysis, the increasing utilisation of probabilistic techniques for safety analysis all over the world and the future applications foreseen for the probabilistic modelling of each plant, were the basis for the proposal to implement an Integrated Programme to carry out specific PSAs of each Spanish nuclear power plant.

The Proposed Integrated Programme was commented by those with interest in the nuclear industry field and received fairly general support. The Integrated Programme was approved by the CSN in June 1986.

In its first part is an analysis of the needs, benefits and possibilities of such a programme, as was requested by the CSN. The second part, the Programme itself, consists of seven points: Probabilistic Safety Analysis Requirements, Data Bank Development, Rule Making and Guidance, Research Plans, International Relations, Promotion of Technological Progress, Recruiting and Training of Personnel. The main goal and the initial motivation of the Programme is to analyse in depth the safety of the Spanish nuclear power plants and to have a logic-probabilistic model of each plant, to be used in future applications.

The main features are:

- The specific requirements to each nuclear power plant will be done in a time-phased fashion, to optimise the use of national resources.

- The initial level of the PSAs will be 1, as defined in NUREG/CR-2300.

- Each PSA will be revised periodically and the revision made at the same scope of the latest PSA requirement.

- Personnel from the utilities staff shall be a part of the teams performing the studies and personnel from the CSN technical staff will be assigned in parellel to the project, to make a continious and interactive evaluation and get a final study review almost at the same time the study is presented to the CSN.

- One of the most significant other points is the development of a data bank at plant specific and national levels, to accumulate information on operational events and system and component malfunctions, to be used for the reliability data base construction in PSA studies. Other PSA applications for improvements in licensing, management and other aspects of nuclear power plant performance, for instance to improvements in the so-called technical specifications, are under consideration.

Status and Outlook

The final report for the first version of the Garoña PSA was issued after some improvements in several areas of the study had been made. The analysis was done following the IREP Procedures Guide methodology and has been the source of many design and procedures small changes for reducing in a considerable amount the core melt frequency.

The second PSA was required for the Almaraz nuclear power plant, a plant with two PWR units. A very general procedures guide, that of NUREG/CR-2815, Revision 1, was chosen as a minimum, for the utility being free to decide about basic methodology to reliability analysis. Utility staff was included in the team by CSN requirement. CSN staff performed an interactive and continuous evaluation of the project. Final report for this PSA first version was approved in 1991.

The third PSA was required for Asco, another two PWR units again using the NUREG/CR-2815, Revision 1, as the basic and minimum guide. Another external event, internal floods this time, was chosen to be added to the scope required to the Almaraz PSA. The project is near completion and is being evaluated in the same interactive way.

The fourth PSA was required by the CSN to the Cofrentes NPP, a BWR Mark III. Methodology and organization are basically the same as preceding PSAs, but more utility personnel is involved. The scope was enlarged to include external flooding. CSN personnel is doing the review in the usual interactive way and some improvements are being done to the preliminary version of the final report before approval.

The Asco and Cofrentes PSAs were reviewed by a peer review team that mainly consists of personnel from the utility of S.M. Garoña NPP. The comments accepted by the projects are incorporated into the PSA partial reports before the CSN's review.

The fifth PSA was required by the CSN to the José Cabrera NPP, a Westinghouse PWR that is the oldest plant in operation in Spain. This PSA is the first Level 2 study that will be performed in Spain. The Project was started in 1990 with a similar organisation to the preceeding PSA.

The sixth PSA was required by the CSN to the Vandellos 2 NPP a Westinghouse PWR with four years of operating experience. This PSA will be already a Level 1 and Level 2 study including all the external events and considering also the risk from non full power and shutdown operation modes. The project organised similarly to the others has started in 1991.

Finally, the seventh PSA has been required to the Trillo NPP, a KWU PWR with four years of operating experience. The scope will be the same as Vandellos 2 and the utility is preparing a proposal to the CSN on the project organization and technical basic options of the study. The project is planned to start in 1992.

The joint Spanish utilities organisation UNESA, prepared a proposal for the development of data banks which was approved by the CSN. Two data banks on operational events and on systems and components malfunctions, will be developed and implemented at each Spanish nuclear power plant and co-ordinated at the national level, to be operative in 1989. Both banks are already in operation for a trial period.

The regulatory framework is presently the individual requirements being done by the CSN to each utility.

The approval and implementation of the Integrated Programme has created a great demand on PSA-trained personnel in Spain. This has obliged the elaborate training plans for future personnel, either by means of courses, or by participation in specific development projects or tasks.

Reference

Programa Integrado de Realizacion y Utilizacion de los Analisis Probailistas de Seguridad en España. Consejo de Seguridad Nuclear. Agosto, 1986.

SPAIN

Plant	Analysing Team/ Date of Starting/ Completion	Methods Used/ Procedure Guide	Goal of the Analysis/ Insights or Results Application
Santa Maria de Garona GE BWR Mark I 460 MWe Under operation	Personnel from the utility and a Spanish engineering company. US consultants. Directed by utility personnel. 1984 - 1987	Level 1 analysis IREP - Procedures Guide NUREG/CR-2728	Reanalysis of overall safety after 13 years of operation, many small changes of design or procedures for reducing core melt frequency, models for future applications.
Almaraz Westinghouse PWRs, 2 units 930 MWe each Under operation	Personnel from the utility and 2 Spanish engineering firms. US consultants. Directed by utility personnel. 1987 - 1990	Level 1 analysis including fire and containment systems reliability NUREG/CR-2815, Rev. 1	Integrated Programme
Asco Westinghouse PWRs 2 units 930 MWe each Under operation	Personnel from the utility and a Spanish engineering association. US consultants. Peer review by S.M. Garona NPP utility personnel. Directed by utility personnel. 1988 -	Level 1 analysis including fire, internal flooding and containment systems reliability NUREG/CR-2815, Rev. 1	Integrated Programme
Cofrentes GE BWR Mark III 1015 MWe Under operation	Personnel from the utility & Spanish Engineering Association. US constulants only as advisors. Peer review by S.M. Garona NPP personnel. Directed by utility personnel. 1989 -	Level 1 analysis including fire, internal and external flooding and containment systems reliability NUREG/CR-2815, Rev. 1	Integrated Programme

Plant	Analysing Team/ Date of Starting/ Completion	Methods Used/ Procedure Guide	Goal of the Analysis/ Insights or Results Application
José Cabrera Westinghouse PWR 190 MWe Under operation	Personnel from the utility & Spanish Engineering Assoc. US consultants only for Level 2. Directed by utility personnel. 1990	Level 1+2 analysis including fire and flood risk analysis NUREG/CR-2815, Rev. 1 & GL 88-20	Integrated Programme
Vandellos 2	Personnel from the utility & Spanish Engineering Assoc. Directed by utility personnel. 1991	Level 1+2 analysis including all external events. Non full power & shutdown operation modes also considered	Integrated Programme
Tallo	Utility is preparing a proposal to the CSN. 1992	Same as Vandellos 2	Integrated Programme

SWEDEN

Abstract

Plant specific PSAs has during the 1980's constituted a major part of the Swedish periodic safety reassessment program (as operated Safety Analysis Report, ASAR-80). The studies are performed by the utilities and reported to the Swedish Nuclear Power Inspectorate (SKI) for review. The Inspectorate reports the results to the Swedish government. Level 1 PSAs have been completed for all Swedish plants. The studies performed before 1987 have been comparitively reviewed in the SUPER-ASAR project.

In the next reassessment program, ASAR-90, the scope of the PSAs will be expanded to cover level-2, external events and all types of operating modes of the plant.

The PSAs are considered as living documents and should be used in daily safety work at the plants.

Programme Development

In the early 1980s a programme aimed at thoroughly reviewing each Swedish nuclear power plant at least three times during its technical lifetime was proposed by the government and ratified in parliament. A thorough plant specific systems reliability analysis (level 1 PSA) constitutes a major part of these "As-Operated Safety Analysis Report" (ASAR) required from the Swedish utilities for each plant every eight to ten years of operation. Level 1 PSAs will be completed during the period 1982-1991. The utilities are responsible for carrying out the analyses and the Swedish Nuclear Power Inspectorate (SKI) carries out the review of the studies.

The main objectives of this systematic reliability evaluation programme were very detailed, plant specific fault and event trees developed and reviewed in close co-operation with senior plant personnel providing a detailed map of system functions and interdependencies and identifying sequences that are the main contributors to core damage. Computer graphic techniques to facilitate detailed documentation, modification and analysis were used extensively. To the greatest possible extent actual component data from the Swedish reliability data bank, which is continuously updated and compiled in the "Swedish Reliability Handbook" (T-book), were used. The studies are continously used as a basis for a systematic evaluation of operating experience when analysing distribution and incidents, keeping track of component and system reliability and their effect on plant safety, for planning and reviewing plant modifications and for training personnel in system functions and interdependencies facilitating their awareness of the safety significance of various operational and maintenance tasks.

Many plant improvements were performed during the time of the PSA and core melt frequency was reduced.

During the 1980s, computerised documentation and analysis systems (DORISK, SUPER-TREE, CADREE, RELTREE) have been developed and used. For the 1990s, a new computerised system has been developed (RISKSPECTRUM) and is now used by SKI and all the utilities.

The review of the earlier PSAs indicated significant differences. A broad spectrum of methods and assumptions was used, which complicates a comparision. In 1986, SKI therfore initiated the SUPER-ASAR project, to survey and compare the PSAs to facilitate the use of the studies in the process of decision-making and to supply background for the establishment of priorities for research projects within the area of probabilistic safety analysis.

The qualitative part of the project reviewed: scope and limitations of the PSAs, selection and definitions of initiating events, modelling of accident sequences, systems analysis, data, treatment of dependencies, treatment of human interactions.

The qualitative results constitute the basis for in-depth sensitivity studies, which were performed to facilitate the use of PSA in daily safety work.

For the 1990s, the goals for the ASAR program has been expanded. For the PSA part of the ASAR program the goal is to cover level-2, external events and all types of operating modes of the nuclear plant.

TABLE 1:

Survey of Swedish PSA related safety studies (oct 91), level 1.

Rev. Oct-91 /CKN	Type of Analysis								
Unit	IE (LOCA and transients)	Internal Fire	Internal Flooding	Seismic	Aircraft Crash	Internal Missiles	Fall of Heavy Objects	External Pipe Break	RPV rupture
B1/2	Finished (84) 87	B1 Finished 89. B2 Ongoing, planed to be finished 91	Finished 83, planed revision 92	Ongoing, planed to be finished 93	Finished 89	Finished 81	Finished (89) 90	Finished 84	Finished 88
F1/2	Finished 88	Not started, planed to start ? (11)	Not started, planed to start ? (11)	Not started, planed to start ? (11)	Finished 84	Not started, planed to start ? (11)	Not started, planed to start ? (11)	Finished 88	Finished 88 (3)
F3	Finished 85	Finished 85 (8)	Finished 85 (8)	Finished 85 (8)	Finished 84	Not started, planed to start ? (11)	Not started, planed to start ? (11)	Finished 89 (10)	Finished 85 (3)
O1	Finished (86) 91	Ongoing, planed to be finished 91	Not started, planed to start 92	Ongoing, planed to be finished 94	Not started, planed to start ?	Not started, planed to start ?	Not started, planed to start ?	Ongoing, planed to be finished 92	
O2	Finished 87	Finished 91	Not started, planed to start 92	Ongoing, planed to be finished 94	Not started, planed to start ?	Not started, planed to start ?	Not started, planed to start ?	Not started, planed to start 92	
O3	Finished 86	Finished 85 (8)	Finished 85 (8)	Finished 85 (8)	Not started, planed to start ?	Finished 85 (8)	Not started, planned to start ?	Not started, planed to start 93	
R1	Finished 83	Ongoing, planed to be finished 92	Finished 85, update 92	Ongoing, planed to be finished 92, (Finished 85(1))	Finished 84	Not started, planed to start ? (11)	Ongoing, planed to be finished ? (2)	Finished 83	Finished 83
R2	Finished 83	Finished 87	Finished 87	Ongoing, planed to be finished 93	Finished 84	Not started, planed to start ? (11)	Not started, planed to start ? (11)	Finished 83	Finished 83 (3)
R3/4	Ongoing, planed to be finished 91	Not started, planed to start 92	Not started, planed to start ? (11)	Not started, planed to start ? (11)	Finished 84	Not started, planed to start ? (11)	Not started, planed to start ? (11)	Ongoing, planed to be finished 92	Ongoing, planed to be finished 91 (3)

(*) See TABLE 1: (cont.)

TABLE 1: (cont.)

Survey of Swedish PSAs (oct 91), level 1.

Rev. Oct -91 /CKN	Other PSA level 1					Not power operation and Core Melt as end state		
Unit	CCI analysis	Sensitivity analysis	Uncertainty analysis	Other IE	Implementation of containment filter venting system in PSA level-1.	Power Reduction	Overhaul Period	Power Increase
B1/2	Finished 85	Finished 84	Finished 84 [10]			Ongoing, planed to be finished 93	Ongoing, planed to be finished 93	Ongoing, planed to be finished 93
F1/2	Finished 88	Finished 88	Finished 88		Finished 88	Not started, planed to start ? [11]	Ongoing, planed to be finished 89	Not started, planed to start ? [11]
F3	Finished 85 [12]	Finished 85	Not started, planed to start 92			Not started, planed to start ? [11]	Not started, planed to start ? [11]	Not started, planed to start ? [11]
O1	Not started, planed to start ?	Finished 91 [13]	Not started, planed to start 93			Not started, planed to start 92	Ongoing, planed to be finished 91	Not started, planed to start 92
O2	Not started, planed to start ?	Not started, planed to start 92	Not started, planed to start 93			Finished 89	Not started, planed to start 92	Finished 89
O3	Not started, planed to start ?	Not started, planed to start 93	Not started, planed to start 93			Not started, planed to start 93	Not started, planed to start 93	Not started, planed to start 93
R1	Finished 84 [4]	Ongoing, planed to be finished 91	Not started,		Ongoing, planed to be finished 91	Not started, planed to start ? [11]	Not started, planed to start ? [11]	Not started, planed to start ? [11]
R2	Ongoing, planed to be finished ? [11]	Ongoing, planed to be finished ? [11]	Finished 83		Ongoing, planed to be finished 91	Not started, planed to start ? [6] [11]	Not started, planed to start ? [5] [11]	Not started, planed to start ? [11]
R3/4	Ongoing, planed to be finished ? [11]	Ongoing, planed to be finished ? [11]	Not started, planed to start ? [11]		Ongoing, planed to be finished 91	Not started, planed to start ? [11]	Not started, planed to start ? [11]	Not started, planed to start ? [11]

Notes

1. A pilot study is finished, full scope analysis ongoing.

2. A quantitative analysis of Spent fuel gasket is finished 89.

3. Assumptions as WASH 1400.

4. Partly finished.

5. Cold over pressure of reactor vessel finished 87.

6. Stem line rupture in power decrease operating mode is analyzed.

7. Results is based on a qualitative comparison with R2 and F3.

8. Safety analysis done as a requirement in FSAR, not quantified in PSA.

9. -

10. Related to IE.

11. Replanning of time schedule ongoing.

12. Mainly qualitative analysis.

13. Simplified, included in updating.

SWITZERLAND

Abstract

Probabilistic Safety Assessment (PSA) methods have been used for regulatory decision-making in Switzerland since 1977, when an adaptation of the Phase-A German Risk Study to Gösgen (a Siemens-KWU PWR with large, dry containment) was used as the basis for refining the emergency planning for nuclear power plants. A similar study was also performed as part of the start-up licensing procedure for the Leibstadt (a GE PWR6 with MARK III containment) nuclear power station.

In 1986, the Swiss Federal Nuclear Safety Inspectorate (HSK) required the performance of full scope level 1 and level 2 PSA (including external events) studies for all Swiss nuclear power plants. At the end of 1990, HSK also required an extension of these studies to include start-up, shutdown and outage modes of operation.

The Mühleberg (a GE BWR4 with MARK I double torus containment) and Beznau (a Westinghouse PWR with large, dry containment) PSA studies are an integral part of the licensing documentation. These two plants need permanent operating licenses, which are expected to be issued by the end of 1992 for Mühleberg, and by end of 1993 for Beznau. Furthermore, the PSA studies are intended to (1) assist in monitoring the continued improvements in plant operation, (2) improve plant maintenance, and (3) assess potential modifications in normal and emergency operating procedures. Therefore, these PSAs are required by HSK to be maintained as living documents.

An independent regulatory review process has been developed and implemented at HSK (see attachment A). These reviews include a comprehensive assessment of PSA models, assumptions, and results, often relying on complete reanalyses using HSK methods and computer codes with comparison of PSA insights to other recent studies for similar plants.

Status and Outlook

Table I summarizes the current status of Swiss PSAs and their HSK review. At present the level 1 and level 2 studies for Mühleberg and Beznau are completed and submitted to HSK for review. The level 1 study for both plants were done by PLG using their standard support state methodology (small fault trees/large event trees). The level 2 study for Mühleberg was performed by PLG & RMA using the BWRSAR/CONTAIN computer codes to support quantification of the split fractions of the Containment Event Tree (CET), and the resulting radiological releases. The Beznau level 2 study was performed by Westinghouse using the MAAP computer code.

The PSAs for Gösgen and Leibstadt are currently underway and will be finished in 1993.

At present, the joint HSK and Energy Research, Inc. (ERI) review of the Mühleberg Safety Assessment has been completed (the final review report is expected to be issued by 1992). As part of this review, a detailed reanalysis was performed. For the level 1 part of the analysis, a fault tree linking technique was used; while, the level 2 portion of the PSA was performed using the MELCOR and ERPRA computer codes. Detailed CETs were quantified using

HSK/ERI accident progression analyses to arrive at the containment failure probabilities for the dominant plant damage states.

A qualitative review of the Beznau level 1 PSA study was completed by HSK and ERI by October 1991. The qualitative review of the level 2 study is currently underway and will be finished by end of 1991. For the detailed review/reassessment of the Beznau PSA a complete reanalysis of the level 1 and level 2 studies will be performed. The result of this reassessment will become an integral part of the HSK Safety Evaluation Report for Beznau.

The review of the Gösgen PSA will initiate as soon as the study is submitted to HSK. The HSK/ERI review of the Leibstadt PSA study will be performed in phased fashion following the progress of the study on a step-by-step basis. Leibstadt review is expected to provide considerable insights into (1) progress and methods of Living-PSA and (2) into regulatory aspects of the process. This review proces will start in spring 1992.

Through this detailed review and evaluation of the plant-specific PSAs, considerable insights are gained by the HSK staff into operation, management and performance of the Swiss NPP.

The on-going review process is providing an excellent vehicle for the regulatory staff to determine unique features, and vulnerabilities of the plant design, performance and operation, especially during potential accident conditions. These insights are by far the most important attributes of any PSA study, naturally complementing the tradional deterministic approach of reactor licensing. Living PSAs are becoming an important ingredient of nuclear regulatory process at HSK.

SWITZERLAND

Plant and Type	Contractor Organization Date of Completion Level 1	Level 2	HSK Review Completion Date	Scope of PSA	Objectives/ Applications
Mühleberg GE BWR/4, Mark-I	PLG/1990	PLG+RMA 1990	1991/final review report 1992	Full-scale study including external events	Needed for final operating permit. Used as Living-PSA
Beznau Westinghouse 2 loop PWR	PLG/1990	Westinghouse 1991	1992/final review report 1993	Full-scale study including external events	Needed for final operating permit. Used as Living-PSA
Gösgen Siemens-KWU 3 loops PWR	PLG/1992	PLG+Stone & Webster 1993	1995	Full-scale study including external events	Used as Living-PSA
Leibstadt GE BWR/6 Mark-III	EWE*/ RELCON 1993	Not known	1995	Full-scale study including external events	Used as Living-PSA

* EWE: Electrowatt Engineering Services, UK

UNITED KINGDOM

Abstract

Reliability and risk analysis in nuclear safety have a long history in the United Kingdom. Probabilistic targets for radioactive releases were used in the sixties in the design of the UKAEA prototype reactors at Winfrith and Dounreay, and in the seventies, the CEGB adopted probabilistic design safety guidelines for its civil nuclear power stations. PSA, therefore, plays an important role in the design, safety analysis, and licensing of U.K. nuclear power reactors. The Sizewell B PWR development, recently approved for start of construction, involved two PSA studies in the pre-construction design and licensing process and a further, more detailed, study is being performed for the final safety report prior to operation. The pre-construction studies were subjected to detailed scrutiny and public debate at the Sizewell Public Inquiry over the period 1983 - 1985.

Programme Development

Advanced Gas Cooled Reactors

Preparation of safety cases for refuelling is well underway.

The design aim was to refuel the reactors during operation at high power. To date, NII has only accepted the case for on-load refuelling at 30% power at 4 of the 14 reactors. The others are refuelled off-load with the reactor un-pressurized.

* The safety cases under preparation include a PSA and are being done in a stage-wise manner; that is the licencee intends to make a case for off-load pressurized refuelling to be followed up by submissions for part load operation etc.

* Safety cases for the irradiated fuel dismantling facilities for some AGRs have been submitted to NII for fuel up to 4 KW decay heat. These cases are to a large extent dependent on PSA.

* PSAs in support of the flask handling facilities at Heysham 2 and Torness have been submitted and are currently being assessed by NII.

Magnox Reactors

The Long Term Safety Review (LTSR) process is still in progress. The LTSR includes a limited scope PSA which, together with engineering analysis has identified the need to install additional protection against frequent faults:

* diverse feed systems are being installed at a number of plants. The new systems are being located remote from the turbine hall to provide protection for hazards such as fire or steam release. The design of the systems is accompanied by level 0 PSAs with reliability targets derived from the original LTSR PSA.

* diverse, secondary guardlines are to be installed at a number of plants. These new guardlines are to provide additional protection against frequent faults and will automatically trip the Boron Ball Shutdown Devices (BBSDs), should the control rods

fail to enter the core. As with the additional diverse feed systems, the design of new guardlines is accompanied (and influenced) by a level 0 PSA which will ultimately be incorporated into the overall level 1 PSA.

* The LTSR indicated the value of PSA and consequently they are being revised and extended as part of the life-extension programme for these reactors. Enhanced level 1 PSAs for Calder Hall and Chapelcross have been submitted by BNFL and these are currently being assessed by NII.

Sizewell B

The construction of the Sizewell "B" PWR is proceeding subject to regulatory control at key points in the programme. Installation of the RPV, which required NII consent, has just taken place and the containment is now closed.

In parallel with this, the Pre-Operational Safety Report (POSR) is being prepared by the licencee, Nuclear Electric and this includes a detailed PSA of the as-built plant.

The detailed work on the PSA is underway and the topics which have been under consideration are as follows:

* continued development of common cause failure (CCF) methodology to estimate probabilities to be used in the fault tree analysis. This is an important, although difficult, topic since the PSA carried out at the design stage indicated that CCFs could make a significant contribution to the calculated risk.

* the level 1 PSA is nearing completion and progression to the level 2 PSA is underway. For Sizewell B the approach being adopted is to use small event trees to model the containment behaviour. These small event trees have fewer nodes than those in NUREG 1150.

* the PSA is intended to include contributions to the risk from a wide range of internal and external hazards. Currently NE are providing descriptions of the methodologies used.

* the next stage of the PSA is to include the contribution to the risk from all shutdown states.

* a peer review, funded by NII, on the PSA methodology is to be carried out by an independent consultant. A further contract on the application of parts of the PSA (fault & tree analysis) is currently underway.

* a project to determine the effectiveness of those protection systems to be installed Sizewell B which were not present on the SNUPPs design is also part way through. This study employs sensitivity analysis removing "Sizewell only" systems and is making use of the Pre-Construction Safety Report (PCSR) PSA.

United Kingdom Atomic Energy Authority (UKAEA).

The sites operated by the UKAEA, originally exempt from licencing, have now been licenced. These sites include Dounreay, site of the Prototype Fast Reactor (PFR) and Winfrith, where the Steam Generating Heavy Water Reactor (SGHWR) is located.

PSAs were prepared in support of UKAEA's applications for licences. The PSAs addressed UKAEA's own criteria, which include targets for individual risk of premature death to a member of the public of 10^{-6}/yr and to workers of 10^{-5}/yr. The PSAs for PFR and SGHWR have been considered by NII. In the case of SGHWR, UKAEA's decision to bring forward the closure of the reactor to 1990 reduced the need for follow up of the PSA by NII. For PFR, assessment of the PSA has led to a dialogue with UKAEA on the potential for plant improvements, taking into account the remaining lifetime, and further probabilistic analysis of the plant. This dialogue is on-going.

Nuclear Chemical Plant

Like Sizewell "B", the Thermal Oxide Reprocessing Plant (THORP) at Sellafield is being constructed under the regulatory system of consents to proceed at key points in the programme. The licencee, British Nuclear Fuels (BNFL) is currently completing a full PSA as part of the Design Safety Report. By the end of 1992, this will be supplemented by a Plant Safety Case in which the PSA will reflect the development of operating procedures during inactive commissioning.

The PSA is based on fault identification through HAZOP (Hazard and Operability) studies. These include both a top-down approach looking at potential hazards such as criticality, fire, explosion or loss of containment and finding ways in which they might be caused and a bottom-up approach which systematically considers whether each possible failure of an item of equipment or variation of process conditions could lead to a hazard.

Analysis of the consequences of faults is based on a database of release fractions and decontamination factors based on an extensive study of the literature and on specific experimental studies. Off-site consequences are evaluated for the critical groups using site specific dose-release ratios (ie dose to a member of the critical group in Sv per TBq of a specific radionuclide released - these vary with release height).

The results of the PSA will be compared with the licencee's accident risk criteria which include a target of 10^{-6} per year for the risk of premature death of a critical group member from the whole Sellafield site. Subsidiary criteria include target frequencies for aerial releases over 100 mSv, for criticalities and for leakage to ground.

Similar PSAs are being carried out for other major new plants including the Enhanced Actinide Removal Plant (EARP) for reducing marine discharges, and the New Oxide Fuels complex at Stringfields.

For operational plant at Sellafield, the PSA also includes a form of importance analysis for identifying key Safety Mechanisms and Operating Rules under the licence conditions.

PSA has also been used in the review of the at existing nuclear chemical plant at other UK sites. At the Capenhurst enrichment plant, individual and societal risks from releases of uranium hexafluoride have been evaluated. Reprocessing plant at Dounreay has also been analysed.

Status and Outlook

Both the CEGB and UKAEA use PSA-based criteria and guidelines in plant design and licencing. These are based on frequency/release (i.e. level 2), although in the case of the CEGB guidelines the emphasis is on the prevention of releases by protection against fuel damage, with claims for containment "in reserve". The principal numerical guideline for severe accidents (uncontrolled releases of radioactivity) is set at 10^{-6} per year in the CEGB design safety guidelines.

The CEGB design safety guidelines were created in response to, and in accordance with, the safety principles of the U.K. Nuclear Installations Inspectorate (UKNII). In the U.K., the licensee/operator has the responsibility for plant safety, and the inspectorate monitors the fulfilment of that responsibility. A parallel scheme applied in the UKAEA for the operation of research and prototype reactors, where the operating site has primary responsibility and the Safety and Reliability Directorate has an inspectorial/monitoring role.

The use of the probabilistic design safety guidelines by the U.K. generating boards ensures that at least detailed level 1 studies will be done for all future U.K. nuclear power plants. The large research efforts in levels 2 and 3 analysis in the UKAEA and CEGB have created a capability for levels 2 and 3 analysis which is likely to be applied to future plants, even though it may have a different status in licensing from the level 1 analysis which is now definitely required.

UNITED KINGDOM

Plant	Analysing Team/ Date of Starting/ Completion	Methods Used/ Procedure Guide	Goal of the Analysis/ Insights or Results Application
Sizewell B	NNC/CEGB/1+	Large FT	Pre-construction safety report and licensing
Sizewell B	CEGB/W/NRPB/3	Large ET & multiple support states	Risk assessment
CDFR	NNC/1+	Large FT	Design concept safety report
CDFR	UKAEA/3	Containment event tree	Assessment of WCA consequence mitigation
AGR (Heysham II) (Torness)	NNC/CEGB/1+ NNC/SSEB/1+	Large FT	Pre-operational safety report and licensing
Magnox	CEGB/1-	System FT	Long term safety reviews and life extension

Notes

Level 1+ indicates some assessment of source terms for design basis accidents.
Level 1- indicates a restricted level-1 analysis
ET = event tree
FT = fault tree
WCA = whole core accident
NNC = National Nuclear Corporation
CEGB = Central Electricity Generating Board
SSEB = South of Scotland Electricity Board
NRPB = National Radiological Protection Board
UKAEA = United Kingdom Atomic Energy Authority
W = Westinghouse

UNITED STATES

Abstract

The conduct of PRAs by the NRC and the nuclear industry has grown steadily since the publication of the Reactor Safety Study (WASH-1400). The range of applications for PRA results and insights in achieving safe plant design and operation has also grown. In the next few years, all U.S. plants will be subjected to probabilistic analysis to identify and eliminate risk vulnerabilities.

Programme Development

The mid-1970s in the U.S.A. was marked by the first large-scale application of Probabilistic Risk Assessment (PRA) techniques to nuclear power plants, the Reactor Safety Study (WASH-1400). This study demonstrated that a nuclear power plant could be analyzed in an integrated and systematic fashion, and served to develop risk curves for a BWR and PWR for comparison with other sources of societal risks.

The pace of PRA development was further enhanced by the Lewis Committee evaluation and critique of the Reactor Safety Study. Among this Committee's recommendations was re-examination of the fabric of the regulatory process to explicitly incorporate more rational and cohesive methods for decision making. The nuclear community reacted by exploring ways of systematically applying probabilistic technique NPPs in Europe and the United States.

The Reactor Safety Study Methodology Applications Program (RSSMAP) was initiated by the USNRC following completion of the Reactor Safety Study. The objectives of the RSSMAP were to identify risk-dominant accident sequences for a broader group of reactor design (one BWR, and three PWR plant designs), to compare these sequences with those identified in the Reactor Safety Study, and to identify risk-significant plant design differences. The RSSMAP programme was the first significant attempt to explicitly use PRA to identify specific aspects of plant design and operation which impact safety.

The Lewis Committee report also proposed the initiation of a programme to review Licencee Event Reports (LERs) of operational events that have occurred at LWRs, to identify and categorize precursors to potential severe core damage accidents. This programme, named the Accident Sequence Precursor Programme, was initiated in 1979. The thrust of the programme is to use LERs and other plant data, estimated system unavailabilities, and the expected frequency of initiating events to evaluate the potential impact of (1) safety system unavailability, and (2) initiating event occurrences. The first major report of the programme covered events occurring between 1969 and 1979. Additional reports have been issued covering events through 1985.

The utilization of PRA was further stimulated by the accident at Three Mile Island (TMI). The TMI accident resulted in a realization that the potential for accidents other than design-basis accidents needed to be addressed more thoroughly in the U.S. regulatory process. Since transients, small LOCAs, and human errors were identified in the RSS as major contributors to risk and were determined to be contributors to the TMI accident, there was a renewed and increased focus on the use of PRA. Furthermore, the Kemeny Commission and the Rogovin investigation strongly encouraged the use of PRA techniques in the regulation of nuclear power.

The Interim Reliability Evaluation Programme (IREP) was initiated subsequent to TMI to address the concern that differences in the design and operation of a plant may have a significant impact on core melt frequency. Two BWR and three PWR plants were analyzed under IREP. Principal objectives of IREP were to identify those accident sequences which dominate risk and to develop state-of-the-art plant system models which could be used as a foundation for subsequent more intensive application of PRA. The importance of uncertainties in component failure and human error data was also evaluated under the IREP effort.

Proceeding in parellel with RSSMAP and IREP were utility-sponsored PRAs for four plants judged by the USNRC to pose potentially large risks due to the high population densities near their site, and for a fifth plant to provide risk insights to assist in evaluating proposed regulatory requirements. These studies represented and important breakthrough in that they were the first to be sponsored, managed and directed by utilities. Also, these studies were the first to treat external events.

Following issuance of the Safety Goal Policy Statement in 1983, an effort was undertaken by the USNRC to collect available information of PRA studies concerning the risks of plants licensed in the U.S., and to prepare a reference document containg a common base of information on such matters as the dominant contributors to core melt and public risk, the strengths and weaknesses of current plant designs, and the usefulness of PRA and the safety goals in assessing such strengths and weaknesses. This effort culminated in issuance of NUREG-1050.

The most recent and advanced PRA methodology development program sponsored by the USNRC is the Risk Methods Integration and Evaluation Programme (RMIEP) initiated in 1982 and scheduled to be completed in 1988. The objectives of RMIEP are (1) to integrate internal, external, and common cause risk methods, (2) to evaluate PRA technology developments and provide the basis for improved PRA procedures, and (3) to identify, evaluate and display the uncertainties in PRA risk predictions which stem from limitations in plant modelling, PRA methods, and data. The LaSalle plant, a BWR 5 Mark II containment, will be used as the RMIEP reference plant.

Status and Outlook

The USNRC developed and issued its Severe Accident Policy Statement in 1985 and followed with its policy implementation plant (Implementation Plant for Severe Accidents and Regulatory Use of New Source Term Information) in 1986. This plan provided for the resolution of severe accident issues through (1) a systematic examination by industry of plants for risk contributors and (2) regulatory use of improved source term information.

A technical data base to support implementation of the USNRC severe accident and safety goal policies was developed through completion of a series of plant-specific USNRC-sponsored risk studies for five plants. These studies provided the technical information source from which the USNRC was able to extract the relevant data and insights that then developed into NUREG-1150 analyses for severe accident frequency estimation did not differ from those characteristically found in other probabilistic risk assessments. However, the NUREG-1150 analyses contain detailed containment event tree and uncertainty analyses heretofore not attempted in other probabilistic risk assessments on such a scale.

NUREG-1150 was issued as a second draft for peer review in June 1989, and a peer review panel, chaired by Dr. Herbert J.C. Kouts of Brookhaven National Laboratory has been formed. All related contractor documents should be available by December 31, 1989. An Assessment for Five U.S. Nuclear Power Plants was published in final form in December 1990. A new Volume 3 was added which responds to the peer review comments received.

As part of the NUREG-1150 documentation, procedure manuals for conducting an analysis of a similar scope have been prepared and are being published as NUREG/CR-4550, Volume 1, "Analysis of Core Damage Frequency: Methodology Guidelines," and NUREG/CR-4551, Volume 1, "Evaluation of Severe Accident Risks: Methodology for the Accident Progression, Source Term, Consequence, Risk Integration, and Uncertainty Analyses."

A generic letter relative to individual plant examinations was issued by the U.S. Nuclear Regulatory Commission in November 1988 which requested each plant to conduct a search for severe accident vulnerabilities. Submittal guidance for these studies was issued as NUREG-1335 in August 1989.

Based of recent surveys, 32 probabilistic risk analyses are currently in progress in the United States; 13 are scheduled to be completed by the end of 1989. In addition, 4 studies using the IDCOR individual plant examination methodology are also in progress. Apploximately 38 U.S. plants have completed PRAs as of this date.

New research programmes have been instituted by the USNRC to reexamine the risk significance of interfacing system LOCAs, with primary emphasis on the human interface with the system, and to evaluate risk associated with low power and shutdown conditions.

In accordance with the implementation plant for the severe accident policy statement, a systematic examination of all plants for severe accident vulnerabilities will soon be requested by the USNRC. The purpose of this defined as the Individual Plant Examination (IPE), is for each utility to understand what could possibly go wrong in its plant so that it would be prepared to handle such events. The IPE will involve a thorough examination of the plant design and operation to identify dominant severe accident sequences and their contributors. Licensees will then assess areas of potential improvement and will implement justifiable corrective actions. Besides the use of a level-1 PRA to examine plant design and operation, the Industry Degraded Core Rulemaking group (IDCOR) has developed a methodology, termed the IPEM, which is a second option available to industry to perform the IPE, subject to the conditions identified by the staff in its review of the IPEM.

The USNRC has developed a pilot programme, called the Integrated Safety Assessment Program (ISAP), which used plant-specific PRAs and engineering judgement to rank all pending regulatory and licensing issues at two volunteer plants. The end result of the programme will be a living integrated schedule dealt with first. Issues of lowest priority are evaluated to check if they still warrant continued NRC and licensee resource expenditures. Furthermore, the USNRC is considering adopting a voluntary programme (ISAP II) which would allow licensees to access the benefits demonstrated in the pilot ISAP, along with new benefits (possible such as plant-specific resolution of generic and unresolved safety issues) based on the experience gained under the pilot program.

Status of Individual Plant Examinations

Under the provisions of the United States government regulations I0CFR 50.54f, the USNRC has requested that each licencee of a commercial nuclear power plant in the United States perform an "individual plant examination" (IPE) to identify potential severe accident vulnerabilities in that plant. The scope of such IPEs is to include the equivalent of a Level I and II analysis for events initiated during full power operation by internal and external initiators. In response to the USNRC request, almost all licencees indicated that a PRA would be performed and used to identify any potential vulnerabilities.

In fiscal year 1991 (which ended September 30, 1991), eight IPEs were submitted to the USNRC for review. By present schedules, 56 IPEs will be submitted in FY 1992, 9 in FY 1993, and 5 in FY 1994. At the present time, the staff has completed its review of the first IPE submittal, with a safety evaluation report to be issued by September 30, 1991. The final staff report is presently scheduled to be completed by the end of FY 1994.

The primary objective of the staff's IPE review is to determine whether licencees met the intent of Generic Letter 88-20, i.e., that each licencee (1) develop an overall appreciation of severe accident behavior through their involvement in the IPE process; (2) understand the most likely severe accident sequences that could occur at their plant; (3) gain a quantitative understanding of the overall probability of core damage and radioactive material releases; and, (4) reduced that overall probability of core damage and radioactive release by modifying procedures and hardware to prevent or mitigate severe accidents. The IPE reviews are being performed by NRC teams consisting of four members: (1) a team leader and coordinator with a general PRA background, (2) front-end systems analyst, (3) a back-end containment performance analyst, and (4) the equivalent of an additional person in specialized areas such as human factors or structural engineering. The IPE review basically involves a two step process. All IPE submittals are expected to undergo a first step (Step 1) review which will check the IPE for completeness and consistency with previous IPE findings and conclusions. The review will also check that the licencee considered Containment Performance Improvement (CPI) recommendations in their IPE and that they explicitly addressed USI A-45 decay heat removal concerns. The staff also expects that licensees performed an in-house peer review of their IPE as part of the process.

Interaction between the NRC staff and the licencee is an important part of the Step 1 review. NRC review teams are expected to formulate and transmit both general and specific questions to the licencee in areas identified as being important. Once a licencee's response is received and reviewed, a meeting is held to discuss any differences that may exist between the review team findings and the licencee. An attempt is made at that time to resolve any outstanding issues by contacting technical experts familiar with the area in question. All review findins and conclusions that result form Step 1 review will be documented in a letter report which becomes the final report (Safety Evaluation Report) is the IPE is found acceptable.

Upon completion of the Step 1 review, the team decides whether a more detailed "Step 2" audit is required. A Step 2 audit could be initited if (1) a submittal has findings which appear to be inconsistent with past PRA experiences or expectations and suggest weaknesses in either the applied methodology or plant's operation characteristics, or (2) if a specific plant has unique characteristics that are not well understood. The Step 2 audit will enhance the NRC's understanding of licencee's IPE and provide a better perspective from which to evaluate the licencee's IPE process.

The second phase in the review process involves a more indepth examination of many aspects of the IPE technical analysis, e.g., examination of analytic models, pertinent input data, or the quantification process. Under Step 2, NRC employed contractors perform site visits, plant walkdowns, and audit tier 2 supporting information. The Step 2 also includes an assessment of an limitations or weaknesses in the licencee's IPE methodology previously identified by the NRC team members under Step 1. Although Step 2 is performed in more depth, neither the Step 1 nor the Step 2 review will be at the level required for validation of the IPE's detailed findings.

The following additional points are also worth noting:

1. The U.S., in particular the U.S. Nuclear Regulatory Commission, does not have a "PSA programme" per se. There exists a number of policy and regulatory documents which discuss or require PRA studies of varying depths and which differ significantly in prescriptiveness. The current reference documents are:

(a) "Safety Goals for the Operation of Nuclear Power Plants," Federal Register, Vol. 51, No. 149, August 4, 1986 (attached).

(b) Generic Letter No. 88-20 "Individual Plant Examinations for Severe Accident Vulnerabilities," November 23, 1988 (Specifies a PRA of level 1).

(c) 10.CFR 50.34(f) (Specifies systems & sequences to be analyzed).

(d) "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," Federal Register, Vol. 50, No. 153, August 8, 1985.

(e) Standard Review Plan, NUREG-0800, Section 10.4.9, Auxiliary Feedwater System (PWR), Rev. 2, July 1981. (Limited to analysis of PWR auxiliary feedwater system).

(f) 10 CFR 52.47

The performance of a PRA, with regards to these references, is the responsibility of the individual applicant. Furthermore, studies done in accordance with References b,c,e & f are reviewed by the NRC staff; there is no published agency policy with regard to independent peer reviews.

2. Reference (a) above is purely a policy statement which contains qualitative and quantitative goals. Reference (b) does not require PRAs of existing plants, but states that PRAs are acceptable as individual plant examinations. Reference (c) requires that PRAs be performed, on specific plants whose license application was pending as of February 16, 1982, within two years of issuance of a construction permit. Reference (d) contains a policy statement that PRAs can be used, in part, to demonstrate acceptability of new designs. Reference (e) prescribes a limit on unreliability of auxiliary feedwater systems and prescribes PRA methods and component failure data for use in the analysis. Reference (f) contains a requirement that an application for design certification must contain a design-specific probabilistic risk assessment.

3. Regulatory actions to deal with problems, whether revealed by PRAs or not, are usually on a case by case basis and depend upon many factors. Nine of these factors appear in 10 CFR 50.109, otherwise known as the "Backfit Rule," number of plant-specific problems which required no agency action because licensees voluntarily corrected the problems. Other PRA-revealed-problems, such as the V sequence revealed by WASH-1400, have been generic in nature. These have often been proposed as Generic Safety Issues (GSIs), and hence, are put through a formal GSI procedure involving prioritization, assignment of a task manager (if the priority warrants), forming a task action plan, designing a technical assistance or research program to provide a technical basis for resolution, and finally subjecting the proposed resolution to many reviews by individuals, committees, and even public comment should the resolution involve rulemaking or a regulatory guide.

UNITED STATES

Plant	PRA Program Sponsor	Analyzing Team	Scope	Methods Used Procedure Guide	Insights or Results
ARKANSAS NUCLEAR ONE-1					
B&W PWR 850 MWe	IREP	SNL	Level-1 Internal Events	IREP Procedure Guide (NUREG/CR-2728)	NUREG/CR -2737
Dry containment Steel-lined concrete Started operation 1974	TAP-A45*	SNL	Level-3 Internal & External Events	PRA Procedures Guide (NUREG/CR-2300)	NUREG/CR -4713
BIG ROCK POINT GE BWR/1 72 MWe Dry containment Steel Sphere Started operation 1962	Utility	Delian	Level-3 Internal Events	Patterned after WASH-1400 with refinement	PRA Big Rock Point Plant Vol. I
BROWNS FERRY-1					
GE BWR/4 1065 MWe	IREP	EG&G	Level-1 Internal Events	NUREG/CR-2728	NUREG/CR -2802
Mark I Containment, Steel Drywell and Wetwell Started operation 1974	Utility	PL&G	Level-3 Internal & External Events	PLG-0209 NUREG/CR-2300	

* PRA Completed on Decay Heat Removal System Only

UNITED STATES (continued)

Plant	PRA Program Sponsor	Analyzing Team	Scope	Methods Used Procedure Guide	Insights or Results
BRUNSWICK 1&2 GE BWR/4 821 MWe Mark I Containment, Steel-lined Concrete Drywell and Wetwell Started operation 1977 Unit 1 1975 Unit 2	Utility	EI	Level-2 Internal & External Events	NUREG/CR-2300	
CALVERT CLIFFS-1 CE PWR 845 MWe Dry Containment, Steel-lined concrete Started operation 1975	RSSMAP IREP IDCOR	SNL BCL SNL Utility	Level-1 Internal Events Level-1 Internal Events Level-2 Internal Events	Patterned after WASH-1400 NUREG/CR-2728 IDCOR Individual Plant Evaluation Methodology (IPEM)	NUREG/CR -1659 Vol. 3 NUREG/CR -3511
CATAWBA W 4-loop PWR 1145 MWe Ice condenser Containment, Steel-lined concrete Started operation 1985 Unit 1 1986 Unit 2	Utility	Utility		NUREG/CR-2300	

Plant	PRA Program Sponsor	Analyzing Team	Scope	Methods Used Procedure Guide	Insights or Results
COOPER GE BWR/4 778 MWe Mark I Containment, Steel Drywell and Wetwell Started operation 1974	TAP-A45*	SNL	Level-3 Internal & External Events	NUREG/CR-2300	NUREG/CR -4767
CRYSTAL RIVER-3 B&W PWR 825 MWe Dry Containment, Steel-lined concrete Started operation 1977	IREP Utility	SAIC Utility SAI	Level-1 Internal Events Level-2 Internal Events	NUREG/CR-2728 NUREG/CR-2300	NUREG/CR -2515
DIABLO CANYON 1&2 W PWR 1086 MWe Unit 1 1119 MWe Unit 2 Dry containment, Steel-lined concrete Started operation 1985 Unit 1 1986 Unit 2	Utility	PL&G	Level-3 Internal & External Events	PLG-0209 NUREG/CR-2300	

* PRA Completed on Decay Heat Removal System Only

UNITED STATES (continued)

Plant	PRA Program Sponsor	Analyzing Team	Scope	Methods Used Procedure Guide	Insights or Results
FITZPATRICK BWR/4 821 MWe Mark I Containment Started operation 1975	Utility				
GESSAR GE BWR/6 1269 MWe Mark 3 Containment	GE	GE	Level-3 Internal & External Events	NUREG/CR-2300	NUREG-0979 Supplements 2,3 and 5
GRAND GULF-1 GE BWR/6 1250 MWe Mark 3 Containment Started operation 1985	RSSMAP	SNL BCL	Level-1 Internal Events	Patterned after WASH-1400	NUREG/CR -1659 Vol. 4
	IDCOR	EI	Level-3 Internal Events	Rebaselining of Previous Results	IDCOR Task 21.1 Report
	IDCOR	Utility	Level-2 Internal Events	IPEM	
	NUREG-1150	SNL	Level-3 Internal Events	NUREG/CR-4550 Vol. 1	NUREG-1150 NUREG/CR -4550 Vol. 6 NUREG/CR -4551 Vol. 4 NUREG/CR -4700 Vol. 4

UNITED STATES (continued)

Plant	PRA Program Sponsor	Analyzing Team	Scope	Methods Used Procedure Guide	Insights or Results
HADDAM NECK W PWR 575 MWe Dry Containment, Steel-lined concrete Started operation 1968	Utility	Utility	Level-1 Internal Events	NUREG/CR-2300	NUSCO-149 CYPSS, Feb. 86*
INDIAN POINT 2&3 W 4-loop PWR 873 MWe Unit 2 965 MWe Unit 3 Dry containment, Steel-lined Concrete Started operation 1974 Unit 2 1976 Unit 3	Utility	PL&G	Level-3 Internal & External Events	PLG-0209 NUREG/CR-2300	I.P. Prob. Saf. Study Vol. 12
LASALLE 2 GE BWR/5 1078 MWe Mark 2 Containment, Steel Drywell and Wetwell Started operation 1984	RMIEP	SNL	Level-3 Internal & External Events	NUREG/CR-2300 Refined Methodology	

UNITED STATES (continued)

Plant	PRA Progam Sponsor	Analyzing Team	Scope	Methods Used Procedure Guide	Insights or Results
LIMERICK-1 GE BWR/4 1055 MWe Mark 2 Containment, Steel-lined Concrete Drywell and Wetwell Started operation 1986	Utility	Utility GE SAIC	Level-3 Internal Events	NUREG/CR-2300	NUREG/CR -3301
McGUIRE W 4-loop PWR 1180 MWe Ice condenser Containment, Steel-lined Concrete Started operation 1981 Unit 1 1983 Unit 2	Utility	Utility TEC	Level-1 Internal Events	NUREG/CR-2300	
MILLSTONE-1 GE BWR/3 660 MWe Mark 1 Containment, Steel Drywell and Wetwell Started operation 1970	IREP Utility	SAIC Utility	Level-1 Internal Events Level-1 Internal Events	NUREG/CR-2728	NUREG/CR -3301 NUREG/CR -3085

* PRA Completed on Decay Heat Removal System Only

UNITED STATES (continued)

Plant	PRA Program Sponsor	Analyzing Team	Scope	Methods Used Procedure Guide	Insights or Results
MILLSTONE-2 GE PWR 870 MWe Dry Containment Steel-lined Concrete with Enclosure Building Started operation 1975	Utility				10 CFR 50.59 Design Optimization
MILLSTONE-3 W 4-loop PWR 1154 MWe Subatmospheric Containment, Steel-lined Concrete with Enclosure Building Started operation 1986	Utility	Utility W	Level-3 Internal & External Events	PLG-0209 NUREG/CR-2300	NUREG-1152 NUREG/CR-4142 NUREG/CR-4143
OCONEE-3 B&W 2-loop PWR 887 MWe Dry Containment Steel-lined Concrete Started operation 1974	RSSMAP EPRI/NSAC IDCOR	SNL BCL Utility EPRI/NSAC Utility	Level-1 Internal Events Level-3 Internal Events Level-2 Internal Events	Patterned after WASH-1400 NUREG/CR-2300 IPEM	NUREG/CR-1659 Vol. 2 NUREG/CR-3301 NSAC-60 NUREG/CR-4374

UNITED STATES (continued)

Plant	PRA Program Sponsor	Analyzing Team	Scope	Methods Used Procedure Guide	Insights or Results
OYSTER CREEK GE BWR/2 620 MWe Mark I Containment, Steel Drywell and Wetwell Started operation 1969	Utility	PL&G		PL&G-0209 NUREG/CR-2300	
PALISADES GE 2-loop PWR 805 MWe Dry Containment Steel-lined Concrete Started operation 1971	Utility	Delian		NUREG/CR-2300	
PEACH BOTTOM 2&3 GE BWR/4 1065 MWe Mark I Containment Steel Drywell and Wetwell Started operation 1974	RSS IDCOR IDCOR NUREG-1150	SNL EI Utility SNL	Level-3 Internal Events Level-3 Internal Events Level-2 Internal Events Level-3 Internal Events	WASH-1400 Rebaselining of previous results IPEM NUREG/CR-4550 Vol. 1	WASH-1400 IDCOR Task 21.1 Report NUREG-1150 NUREG/CR-4550 Vol. 4 NUREG/CR-4551 Vol. 3 NUREG/CR-4700 Vol. 3

UNITED STATES (continued)

Plant	**PRA Program Sponsor**	**Analyzing Team**	**Scope**	**Methods Used Procedure Guide**	**Insights or Results**
POINT BEACH-1 W 2-loop PWR 497 MWe Dry Containment, Steel-lined Concrete Started operation 1970	TAP A-45*	SNL	Level-3 Internal & External Events	NUREG/CR-2300	NUREG/CR -4458
QUAD CITIES 1 GE BWR/3 789 MWe Mark I Containments, Steel Drywell and Wetwell Started operation 1972	TAP A-45*	SNL	Level-3 Internal & External Events	NUREG/CR-2300	NUREG/CR -4448
SEABROOK 1&2 W 4-loop PWR 1150 MWe Dry Containment, Steel-lined Concrete with Enclosure building	Utility	PL&G	Level-3 Internal & External Events	PL&G-0209 NUREG/CR-2300	Seabrook Station Probabilistic Safety Assessment, PLG-0300

* PRA Completed on Decay Heat Removal System Only

UNITED STATES (continued)

Plant	PRA Program Sponsor	Analyzing Team	Scope	Methods Used Procedure Guide	Insights or Results
SEQUOYAH 1&2 W 4-loop PWR	RSSMAP	SNL BCL	Level-1 Internal Events	Patterned after WASH-1400	NUREG/CR -1659 Vol. 1
1148 MWe	IDCOR	EI	Level-3 Internal Events	Rebaselining of previous results	IDCOR Task 21.1 Report
Ice Condenser Containment, Steel with concrete Shield Building	IDCOR	Utility W	Level-2 Internal Events	IPEM	
Started operation 1981 Unit 1 1982 Unit 2	NUREG-1150	SNL	Level-3 Internal Events	NUREG/CR-4550 Vol. 1	NUREG-1150 NUREG/CR -4550 Vol. 5 NUREG/CR -4551 Vol. 2 NUREG/CR -4700 Vol. 2
SHOREHAM GE BWR/4 819 MWe	Utility	SAIC	Level-3 Internal & External Events	NUREG/CR-2300	PRA/ Shoreham Nuclear Power Sta. Vol. 1 SAI-372-83-PA-01
Mark II Containment Steel-lined Concrete Drywell & Wetwell	IDCOR	Utility	Level-2 Internal Events	IPEM	NUREG/CR -4050
SOUTH TEXAS PROJECT 1&2 W PWR 1250 MWe Dry Containment Steel-lined Concrete Started operation 1987 (Unit 1)	Utility	PL&G		PLG-0209 (June 1981) NUREG/CR-2300	

UNITED STATES (continued)

Plant	**PRA Program Sponsor**	**Analyzing Team**	**Scope**	**Methods Used Procedure Guide**	**Insights or Results**
ST. LUCIE CE 2-loop PWR 830 MWe Dry Containment, Steel with Concrete Shield Building Started operation 1976	TAP A-45*	SNL	Level-3 Internal & External Events	NUREG/CR-2300	NUREG/CR -4710*
SURRY 1&2 W 3-loop PWR 788 MWe Subatmospheric Containment, Steel-lined Concrete Started operation 1972 Unit 1 1973 Unit 2	RSS NUREG-1150	SNL SNL	Level-3 Internal Events Level-3 Internal Events	WASH-1400 NUREG/CR-4550 Vol. 1	WASH-1400 NUREG-1150 NUREG/CR -4550 Vol. 3 NUREG/CR -4551 Vol. 1 NUREG/CR -4700 Vol. 1
SUSQUEHANNA 1&2 GE BWR/4 1050 MWe Mark II Containment, Steel-lined Concrete Drywell and Wetwell Started operation 1983 Unit 1 1985 Unit 2	IDCOR	Utility	Level-2 Internal Events	IPEM	

Plant	PRA Program Sponsor	Analyzing Team	Scope	Methods Used Procedure Guide	Insights or Results
THREE MILE ISLAND B&W PWR	Utility	PL&G	Level-1 Internal & External Events	PLG-2029 NUREG/CR-2300	
TROJAN W	TAP A-45*	SNL	Level-3 Internal & External Events	NUREG/CR-2300	
TURKEY POINT 3&4 W 3-loop PWR 693 MWe Dry Containment Steel-lined Concrete Started operation 1972 Unit 3 1973 Unit 4	TAP A-45*	SNL	Level-3 Internal & External Events	NUREG/CR-2300	NUREG/CR -4762
YANKEE ROWE W 4-loop PWR 175 MWe Dry Containment Steel sphere Started operation 1961	Utility	Utility EI	Level-3 Internal Events	NUREG/CR-2300	NUREG/CR -4589

* Completed on Decay Heat Removal System Only

UNITED STATES (continued)

Plant	PRA Program Sponsor	Analyzing Team	Scope	Methods Used Procedure Guide	Insights or Results
ZION					
W PWR 1040 MWe	Utility	PL&G	Level-3 Internal & External Events	PLG-0209 (June 1981) NUREG/CR-2300	Zion PSS Vol. 1
Dry Containment, Steel-lined Concrete	IDCOR	EI	Level-3 Internal Events	Rebaselining of previous results	IDCOR Task 21.1
	IDCOR	Utility	Level-2 Internal Events	IPEM	
Started operation 1973 Unit 1 1974 Unit 2	NUREG-1150	BNL	Level-2 Internal Events	NUREG/CR-4550 Vol. 1	NUREG-1150 NUREG/CR-4550 Vol. 7 NUREG/CR-4551 Vol. 5

W = Westinghouse
CE = Combustion Engineering
B&W = Babcock & Wilcox
BNL = Brookhaven National Lab
SNL = Sandia National Lab
SAIC = Science Applications Inc.
PL&G = Pickard, Lowe and Garrick Inc.
IREP = Intrim Reliability Evaluation Program
RSS = Reactor Safety Study (WASH-1400)
RSSMAP = Reactor Safety Study Methodology Applications Program
IDCOR = Industry Degraded Core Porgram
RMIEP = Risk Methods Integrated Evaluation Program
EPRI = Electric Power Research Institute
TEC = Technology for Energy Corporation

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STATUS OF PSA PROGRAMMES IN MEMBER COUNTRIES

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