Nuclear Safety

CSNI Technical Opinion Papers

No. 1:
Fire Probabilistic Safety Assessment
for Nuclear Power Plants

No. 2:

Seismic Probabilistic Safety Assessment
for Nuclear Facilities

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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

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

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

NUCLEAR ENERGY AGENCY

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

The mission of the NEA is:

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

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

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

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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

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

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

In implementing its programme, the CSNI establishes co-operative mechanisms with the NEA 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 the NEA Committee on Radiation Protection and Public Health and the NEA Radioactive Waste Management Committee on matters of common interest.

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FOREWORD

The mission of the NEA Working Group on Risk Assessment (WGRisk) is to advance the understanding and utilisation of probabilistic safety assessment (PSA) in ensuring the continued safety of nuclear installations in Member countries. In pursuing this goal, the Working Group takes various methodologies into account for identifying contributors to risk and assessing their importance. While the Working Group focuses on the more mature PSA methodologies (Level 1, Level 2, internal, external, shutdown), it also considers the applicability and maturity of PSA methods for investigating evolving issues such as human reliability, software reliability and ageing issues.

In the late 1980s and early 1990s, PWG5 (renamed WGRisk in 2000) issued several important statements on risk assessment. The paper on "The Role of Quantitative PSA Results in NPP Safety Decision Making" represented a major international consensus in this area and is still seen as an essential document. While developing and obtaining consensus on technical opinion papers is considered a very difficult task, the product is viewed as one of the most important aspects of the WGRisk programme of work.

The technical opinions contained herein have resulted from the work of two separate tasks groups including experts in each subject. Both groups developed state-of-the-art reports and held workshops to further the discussions and derive these papers. The Seismic Paper received additional support and input from the Working Group on the Integrity of Components and Structures (IAGE). The final products were reviewed and accepted by the members of WGRisk and the NEA Committee on the Safety of Nuclear Installations (CSNI).

The NEA Secretariat wishes to express particular thanks to Mr. Reino Virolainen, task leader for Fire PSA, and Dr. Charles Shepherd, task leader for Seismic PSA, who provided clear insights on the objectives and overall co-ordination in completing the tasks. Dr. Nathan Siu for Fire PSA and Dr. Robert Budnitz for Seismic PSA, who as experts in their respective fields, provided much of the basis for the in-depth technical analysis provided in the papers as well as many hours in editing and finalising them.

The NEA Secretariat also wishes to thank the experts listed below, who provided valuable time and considerable knowledge towards the development of these papers:

Fire PSA Technical Opinion Paper

Dr. Jeanne-Marie Lanore and Mr. Remy Bertrand, IRSN, France; Mr. Reino Virolainen, STUK, Finland; Mr. Magiel F. Versteeg, VROM, the Netherlands; Dr. Charles Shepherd, NII, United Kingdom; and Dr. Pieter De Gelder, AVN, Belgium.

Seismic Technical Opinion Paper

Mr. Pierre Sollogoub, CEA, France; Dr. Alex Miller, NII, United Kingdom; Dr. Masyasandra Ravindra, EQE, United States; Dr. Nilesh Chokshi, NRC, United States; Mr. Joseph Murphy, NRC, United States; and the members of the Working Group on the Integrity of Components and Structures (IAGE).

TABLE OF CONTENTS

Technical Opinion Paper No. 1:

Fire Probabilistic Safety Assessment for Nuclear Power Plants

Technical Opinion Paper No. 2:

Seismic Probabilistic Safety Assessment for Nuclear Facilities

TECHNICAL OPINION PAPER No. 1

Fire Probabilistic Safety Assessment For Nuclear Power Plants

TECHNICAL OPINION PAPER ON FIRE PROBABILISTIC SAFETY ASSESSMENT FOR NUCLEAR POWER PLANTS

Introduction

This technical opinion paper represents the consensus of risk analysts and experts in the NEA Member countries on the current state of the art in fire probabilistic safety assessment (PSA) for nuclear power plant design and operation. The objective is to present a clear technical opinion on the current state of fire PSA to decision makers in the nuclear community. As such, the intended audience is primarily nuclear safety regulators, senior researchers and industry leaders. Government authorities, nuclear power plant operators and the general public may also be interested.

Background and General Methodology

Depending on the design and operational characteristics of a given nuclear power plant, fire can be a significant or even dominant contributor to the overall risk for that plant. From an analytical standpoint, a significant number of fire probabilistic safety assessment (PSA) studies have resulted in estimates of mean fire-induced core damage frequencies (CDF) of 10^{-4}/yr or greater, predicted contributions to total CDF (i.e., CDF from all contributors) of 20% or greater, or both. From an empirical standpoint, many fire events have been of some safety significance. Of these, a few have been serious core damage precursor events. Key examples include the fires at: Browns Ferry (U.S., 1975), Armenia (Armenia, 1982), Vandellos (Spain, 1989), and Narora (India, 1993).

Not all large nuclear power plant fires are significant from a public safety point of view, nor are all safety significant fires large. Differences in such details as the routing of key electrical cables, the separation and orientation of important cable trays, the fire protection scheme used for a particular compartment, and the procedures employed by plant operators in response to a fire can dramatically alter the risk significance of real and hypothesised fires.

Fire PSA is a quantitative tool for addressing these details and showing how they relate to risk.

The general approach for performing a fire PSA is little changed from that used in the earliest commercial nuclear power plant fire PSA studies (performed in the early 1980s). As part of this approach, potentially important scenarios are identified through a consideration of the fire hazards (ignition sources and fuels) in each plant fire area and of the plant equipment (including electrical cables) that may be damaged by a fire. Of particular interest are fire-initiated scenarios involving the triggering of a plant transient (an "initiating event") and the degraded response of plant systems and operators. The scenario frequencies are quantified by estimating the frequency of fire initiation, the conditional probability of fire-induced damage to critical equipment given the fire, and the conditional probability of core damage given the specified equipment damage. (Post-core damage issues, e.g., containment failure and radioactive material release, can also be treated, but this has only been done in a limited number of studies.) Typically, the fire initiation frequency is estimated using a simple statistical model for fire occurrences; the likelihood of fire damage is estimated using combinations of deterministic and probabilistic models for the physical processes of fire growth, detection, suppression, and equipment damage; and the likelihood of core damage is estimated using conventional PSA systems and accident progression models. Because of the large number of scenarios that need to be considered, the assessment process is iterative and involves the extensive use of screening techniques to focus analysis resources on the most important scenarios. More detailed discussions on the overall fire PSA approach and specific application issues can be found in the references cited at the end of this paper.

This assessment process provides a rational framework for addressing the complex phenomenology underlying a fire event in an explicit manner. It uses a wide variety of information sources (e.g., model predictions as well as event data), and provides a risk context for discussions of areas where there are significant controversies and uncertainties. Also, because the fire PSA analysis elements correspond directly with the traditional fire protection definition of defence-in-depth (whose elements are fire prevention, fire detection and suppression, and fire mitigation), the fire PSA can be a useful communications tool in determining the relative importance of fires in different areas of the plant. This information cannot be obtained from the design basis deterministic analysis.

Fire PSA Applications

Over the years, fire PSA has proven to be a useful tool for nuclear power plant designers, operators, and regulators. Many of the fire PSA studies have been performed as part of larger PSA studies. The earliest such analysis was performed in 1975 as a supplement to WASH-1400 (the U.S. Nuclear Regulatory Commission's Reactor Safety Study); this limited study was aimed at providing a quick estimate of the risk implications of the Browns Ferry fire. The analysis indicated that the CDF associated with that fire was around 10^{-5}/yr, or about 20% of the CDF due to causes addressed in the main body of the study. It also noted the usefulness of developing a more detailed fire PSA methodology (including improved models and data). Another early study was performed in 1979 as part of a PSA for a proposed high-temperature, gas-cooled reactor design. The analysis focused on the risk contribution of cable spreading room fires; it concluded that the core heat-up frequency due to such fires was also around 10^{-5}/yr, or about 25% of the total core heat-up frequency due to all causes.

The first comprehensive, detailed fire PSAs for commercial nuclear power plants were performed in 1981 and 1982 as part of the Zion and Indian Point PSAs, respectively. A key question addressed by both PSA efforts was if additional accident mitigation systems (e.g., a filtered vented containment) were needed for the two plants. The study results indicated that the fire risk for Zion Units 1 and 2 was relatively small (the mean CDF was about 5 x 10^{-6}/yr for each unit; this was about 10% of the total mean CDF) and that the fire risk at the Indian Point plants was relatively large. (For example, the mean fire-induced CDF for Unit 2 was about 2 x 10^{-4}/yr, or about 40% of the total mean CDF.). As the Zion and Indian Point fire PSAs were performed by the same analysis team using the same analysis methodology and tools, these studies demonstrated how plant-specific features could greatly affect fire risk. More importantly, the studies also identified plant design changes for reducing risk (e.g., fire barriers, a self-contained charging pump, provisions for an alternate power source in the event of damaging fires) that were far more cost-effective than the proposed accident mitigation systems prompting the studies.

In more recent years, fire PSAs, have been performed for many types of reactors. The increasing capacity of the PSA hardware platform has enabled more detailed modelling to be carried out for all aspects of the fire PSA, including fire growth, suppression and the impact on systems. The most frequent application of fire PSA (or related analytical tools based on fire PSA, e.g., the US Electric Power Research Institute's FIVE methodology) involves the search for plant-specific vulnerabilities/weaknesses and associated potential improvements. This has been done in a number of Member countries for their operating plants, often as a supplement for deterministic safety analyses. Many of the studies have shown that fire can be an important or even dominant contributor to risk and have identified design or operational improvements that would reduce this risk. Many of these improvements have been implemented by the plants. Other recent or current fire PSA applications include the support of: risk-based configuration control for operating plants; plant staff training; advanced reactor design development; generic (i.e., non-plant specific) fire protection issue identification and prioritisation (e.g., the effect of smoke on manual fire fighting effectiveness); inspection activities (e.g., by identifying areas of focus); regulatory effectiveness analyses (e.g., evaluation of the risk significance of exemptions to specific fire protection regulations) and research and development prioritisation. An example of a systematic fire risk reduction policy is from the Loviisa 1 and 2 units. The living fire risk assessment associated with risk reduction measures has been used in a systematic fashion for improving the plant's safety. The additional sprinkler and physical protections of several safety significant cables, pressure air pipes and hydraulic oil stations of turbine by-pass valves and others have lowered the risk contribution of fires significantly since the early 1990s.

In general, the performance of the fire PSA will enhance the understanding of the fire risk compared with the consideration of the deterministic analysis. The increased understanding is summarised as follows:

Comparison of Deterministic and PSA Fire Analysis		
Issue	***Deterministic Approach***	***Fire PSA Approach***
Extent of equipment/ cable damage	Generally assumes all equipment in fire compartment will be damaged or fire spread will be limited by physical separation and /or installed suppression systems	Uses fire modelling to determine extent of damage from specific sources
Likelihood of fire	Assumes fire may occur regardless of sources present	Evaluates component/activity-based fire frequencies for specific plant locations based on generic and/or plant-specific experience
Coincident equipment failures	Assumes equipment unaffected by the fire will be available for plant shutdown	Considers random failures of unaffected equipment coincident with fire damage

Comparison of Deterministic and PSA Fire Analysis (cont.)		
Issue	***Deterministic Approach***	***Fire PSA Approach***
Operator reliability	Assumes operators will take actions directed by procedures having demonstrated minimum time, instrumentation and access is available	Considers likelihood of potential operator error given stress factors imposed by particular fire, spurious alarms or lack of alarms, existing operating procedures not adapted to the situation induced by fire and activity-related conditions (e.g. degraded instrumentation, smoke, time limitations and training
Offsite power and other non-safety systems	Often assumes non-safety related components and services (e.g. offsite power and main feedwater) are unavailable	Only assumes components and services are unavailable if shown to be damaged by the fire; otherwise considers random failure probability coincident with the fire
Fire protection systems	Has specific requirements regarding installation and operability depending upon fire hazard; if these are met analyses assume system is effective	Accounts for random failure of automatic and manual fire detection and suppression systems depending upon specific types of system installed and fire fighter response during drills; also addresses random barrier element failure probabilities

It is anticipated that in the near future, as regulatory decision making increasingly addresses risk considerations, the use of fire PSAs will naturally increase as well. Fire PSA applications currently undergoing development include: the evaluation of the risk significance of fire protection inspection findings; the development of risk information to support the resolution of generic fire protection issues (e.g., fire-induced circuit failures); and the development of fire PSA methods and data to support proposed changes to a plant's current licensing basis. Regarding this last application, it is anticipated that, in some of the Member countries, there will be increasing demand to relax regulatory requirements in situations where it can be demonstrated that such relaxations involve either no or negligible increases in risk. The appropriate application of fire PSA to support fire protection relaxations is an ongoing challenge.

It is important to note that, for all of the Member countries, when the results of PSAs are used to explicitly support decision making, they are not used as the sole basis for the decision making. Other sources of information, including other engineering analyses, are also used to support the decision. In other words, the decision-making process is "risk-informed", rather than "risk-based". Under this risk-informed approach, the decision maker can make use of information from imperfect or even flawed PSA models, as long as the use of the PSA results and insights improves on the non-risk-informed decision-making approach for the problem of interest. This is an important consideration for the use of fire PSAs because the fire PSA state of the art needs a number of improvements, as discussed below.

Fire PSA Methodology and Uncertainties

When used in a risk-informed decision-making framework, fire PSA is useful in that it provides a systematic, integrated method for evaluating the importance of fire protection issues. However, because of the lack of experimental data or validated models for key issues (e.g., fire growth, the impact of fire on equipment and operator performance), the results of current fire PSAs have significant uncertainties. A review of recent studies shows that, for a number of key situations where there currently is no consensus as to the "right" or "best" modelling approach, variations in modelling assumptions in a given situation can lead to large variations in estimates of fire-induced CDF and qualitatively different risk insights. Examples of issues lacking consensus include the treatment of the likelihood of challenging fires (through the use of so-called "severity factors") and the treatment of main control room evacuation in the event of a serious fire. The understanding of the impact of these issues on the core damage frequency can be as important as the numerical result. A lack of understanding such uncertainties can clearly affect a decision maker's use of the results of a fire PSA and lead to sub-optimal decisions.

The uncertainties in fire PSA results are not due to the general analytical approach described at the beginning of this paper. All current nuclear power plant fire PSAs use this approach or some slight variant on it. Rather, a good deal of the uncertainty is due to weaknesses and gaps in the current state of knowledge of certain aspects of fire scenarios. Significant uncertainties can arise in the estimation of the likelihood of important fire scenarios (e.g., when addressing the frequency of large, transient-fuelled fires or of self-ignited cable fires), the modelling of fire growth and suppression (e.g., when treating fire propagation through a stack of cable trays), the prediction of fire-induced loss of systems (e.g., when quantifying the likelihood of spurious actuations, when addressing the effect of smoke on equipment), and the analysis of plant and operator responses to the fire (e.g., when modelling operator actions during a severe control room fire).

Other uncertainties arise because not all fire PSAs follow the current state of the art. Many studies do not address uncertainties either formally (through quantification and propagation) or informally (e.g., through sensitivity studies). In many such cases, it is stated that conservative models are being used, but the degree of conservatism is not clear. Many other studies employ obsolete models for fire scenario development. Clearly, these implementation issues present a problem to a decision maker who wishes to use the results of the study.

In order to deal with the weaknesses and gaps in the fire PSA state of the art, research and development is needed. A number of countries have significant activities in this area; their results are expected to lead to fire PSA improvements in the near future. For instance, in France, the Institute for Radiological Protection and Nuclear Safety (IRSN) has performed fire tests related to fire-induced damage to electrical cables and is carrying out fire experiments related to electrical cabinets and to the propagation from one compartment to the adjacent ones. The objective of these fire tests is to reduce the uncertainties linked to the study of these fire scenarios that have an important contribution to the fire PSA results. It is recognised that improvements in the state of the art may very well lead to more complex and costly analyses (e.g., to address multiple ventilation scenarios when modelling fire behaviour in a room). The fire PSA research and development efforts need to address this concern in order to ensure that the improved fire PSAs can be performed efficiently. As the information base is built up, the models will become more robust and the uncertainties in the analysis will be reduced.

Recognising that there is a need to strengthen the information base, the OECD has established the OECD Fire Incident Records Exchange (OECD-FIRE) project. This project will provide a database repository for fire event data. The project activities will include the collection, categorisation and exchange of data related to the occurrence of fire events and the characteristics (e.g., location, fuel involved, severity, detection and suppression times, influence on plant operations).

In order to deal with problems in the implementation of the state of the art, a number of mechanisms might be useful. Within the U.S., an industry consensus standard which uses risk information in evaluating a plant's fire protection programme has been developed under the auspices of the National Fire Protection Association (NFPA). This standard includes a description of the technical issues that need to be addressed by a high-quality fire PSA.

Concluding Remarks

Fire PSA is a systematic tool for dealing with the complex issues that need to be addressed when ensuring fire safety at a nuclear power plant. It is a useful tool for supplementing deterministic analyses on which the reactor design and fire protection are based, as it highlights the strong and weak points of a plant's design and operation with respect to fire hazards. It has proven useful in supporting design, operational, and regulatory decision making, and this use is expected to increase in the coming years.

As is the case with the analysis of other internal plant initiating events, the results and insights from the fire PSA should be used as part of a risk-informed decision-making process (rather than the complete technical basis for decision making). The degree of support and scope of applications is again dependent on the accuracy and validity of the models used in the PSA. In the case of the fire PSA there are likely to be more limitations than for some other initiating events. The usefulness of fire PSA will increase as ongoing research and development efforts lead to improvements in the state of the art, and as improvements in the implementation of the state of the art lead to more consistent results.

Recommended for Further Reading

1. G. Apostolakis, M. Kazarians, and D.C. Bley, "Methodology for Assessing the Risk from Cable Fires," *Nuclear Safety*, 23, 391-407(1982). This pioneering paper describes key aspects of the technical approach used in the first comprehensive, detailed fire PSAs for NPPs.

2. "Fire Risk Analysis, Fire Simulation, Fire Spreading, and Impact of Smoke and Heat on Instrumentation Electronics," NEA/CSNI/R(99)27, February 2000. This report provides a perspective on the current fire PSA state of the art. It provides a general description of the fire PSA approach, identifies a number of important issues, discusses the state of the art with respect to these issues, and provides numerous references for additional reading.

3. "Proceedings from International Workshop on Fire Risk Assessment, Helsinki, Finland, 29 June-2 July, 1999," NEA/CSNI/R(99)26, June 2000. This report includes papers discussing current issues in fire risk assessment applications and issues, fire modelling, fire experiments, and fire research.

TECHNICAL OPINION PAPER No. 2

Seismic Probabilistic Safety Assessment for Nuclear Facilities

TECHNICAL OPINION PAPER ON SEISMIC PROBABILISTIC SAFETY ASSESSMENT FOR NUCLEAR FACILITIES

Introduction

This technical opinion paper represents the consensus of risk analysts and experts in the NEA Member countries on the current state of the art in seismic probabilistic safety assessment (SPSA) for nuclear facilities. The objective is to present a clear technical opinion on the current state of seismic PSA to decision makers in the nuclear community. As such, the intended audience is primarily nuclear safety regulators, senior researchers and industry leaders. Government authorities, nuclear power plant operators and the general public may also be interested.

Based on the large number of successful applications of SPSA around the world in recent years, it is clear that the methodology is now mature enough for routine use, to analyse the risk of core-damage accidents and radiological releases that might arise from earthquake-initiated events at nuclear power plants (NPPs), and to understand what is important from a risk perspective. The SPSA methodology can also be adapted for use at other nuclear facilities.

Like other PSA methodologies, SPSA represents a systematic method for examining and evaluating the risk significance of various structures, systems, components, and human actions as they could contribute to potential accidents – in this case, earthquake-initiated accidents. In addition, as with other PSA methodologies, the SPSA methodology explicitly handles the uncertainties that it deals with, and can therefore provide perspectives about the importance of its findings in light of those uncertainties.

Methodology

The aim of the SPSA methodology is to calculate the core-damage frequency, and the frequency of potential radioactive releases, from postulated NPP accidents. The methodology also provides insights into the relative importance of structures, systems, components and operator actions in contributing to the risk, and thus highlights the areas where changes to the design or operation of the plant would be most effective in reducing the risk. This would include upgrading the anchorages of equipment, improving the seismic capacities of structures, systems and components, or modifying operating procedures. SPSA results have enabled both plant managerial personnel and regulators to concentrate their efforts and resources where it matters most, and in the process to improve safety significantly, often at low cost. For example, Seismic PSA studies often identify anchorage problems with electrical equipment, pumps, and large tanks, and relay chatter problems, which can usually be fixed without large expense.

Because seismic PSA methods were developed several years after the PSA methods for analysing accident sequences arising from internal plant faults, there was a period when SPSA was not as mature as the internal-faults PSA methodology. That period has long since passed, so that today SPSA is widely used, often by itself but sometimes in conjunction with the long-established deterministic methods of seismic-safety analysis. Today, SPSA has shown its ability to evaluate a very wide range of NPP designs, ranging from the newest NPP designs now still on the drawing board to NPPs of the oldest vintage that sometimes were designed and built with little or no consideration of seismic issues. In these latter cases, SPSA has demonstrated that it can provide a rational basis for determining which are the most urgent and most cost-effective approaches to making such an NPP adequately resistant to earthquakes.

Although the SPSA methods are clearly mature enough for such uses, it needs to be recognised that, like the rest of PSA, SPSA produces estimates of the risk (core-damage frequency [CDF], etc.) that have large numerical uncertainties, which sometimes makes the results hard to interpret. In fact, the overall uncertainties in CDF are often larger than those in typical internal events PSA analyses. The leading contributor to the numerical uncertainty arises from the fact that the frequencies of potential large damaging earthquakes are not known very well in most places around the world – there simply have not been such earthquakes in historical times in most locations, and interpreting the geological record to produce estimates of the small likelihood of such events is controversial in the seismological community today and likely to remain so for a long time. There are also uncertainties arising from the other aspects of SPSA, consisting of engineering evaluations and so on. However, these are neither more nor less uncertain than the rest of PSA that evaluates accidents arising from internal-plant faults. The ability of the SPSA to analyse systematically and realistically the seismic capacities of structures and components, which is its strongest attribute, makes SPSA valuable quite aside from the fact that CDF uncertainties are large. In addition, the insights that can be gained regarding the relative strengths and weaknesses of the design and operation of the plant are still valid despite these uncertainties.

Research and Development

Areas where further research and development can help to enhance the SPSA methodology to extend the method's usefulness include reducing uncertainties, making the analysis methods easier to use and review, and extending the range of applications. Among the areas where the SPSA community is now working to improve the methods are the way correlations among failures are handled, the approach to quantifying human errors in the post-earthquake environment, the use of SPSAs for accident management, and the treatment of ageing effects in structural evaluations. Concerning seismic capacities of equipment, although generic data have sometimes been used, there are obvious limitations, and there is a clear need for seismic-capacity test data for some equipment that has never been tested. This is especially true in countries that did not take earthquake-safety issues as seriously as they should have when they were designing and constructing some of the earlier NPPs.

Another set of potentially important applications is in safety regulation. Specifically, SPSA insights can be used to alter the way current seismic regulations are framed and used, by allowing probabilistically-based methods to complement the existing deterministic approaches to regulation. Both for determining the seismic design basis needed at a given site, and for establishing the seismic design rules for structures and equipment, such approaches, using SPSA methods and insights, are already beginning to be developed, and in a few cases around the world they are now being embedded in regulations. They could become very widely adopted in the coming decade or so, because they offer the advantage over the traditional deterministic methods of an explicitly realistic analysis.

Concluding Remarks

A unique attribute of SPSA is its ability to integrate seismic-capacity and systems-engineering analyses and insights into a unified picture of the way the whole plant might respond to large earthquakes. This valuable attribute has often proven to be of importance in SPSA studies, which have sometimes identified potential vulnerabilities that involve both seismic-capacity weaknesses and systems-and-operational weaknesses whose interplay produces the safety concern. An example is a potential accident sequence resulting when an earthquake might damage certain equipment, which nevertheless would still leave the plant in a safe state unless a subsequent operator error were to occur. The ability to find such scenarios (which are often identified in SPSA studies), and the ability to evaluate various ways to eliminate them, is one of SPSA's greatest benefits, and a benefit that cannot be realised by any other analysis method.

Regarding the future, it is clear that SPSA is now mature and will continue to be applied more and more around the world, as an integral part of full-scope PSAs, for all types of NPPs. This will be true despite the fact that, like the rest of PSA, seismic PSAs produce results that are uncertain, often significantly uncertain, so that applications always require the use of judgement, validated by peer review. And in parallel, continuing methodological improvements around the world and continuing information exchanges among practitioners (especially through journals and international fora) will enhance SPSA's ability to support safe NPP operation.

References

1. "State-of-the-Art Report on the Current Status of Methodologies for Seismic PSA", NEA/CSNI/R(97)22, March 1998. This report provides an up-to-date review of the SOAR of the methodologies for conducting a seismic PSA at a nuclear power station, including six sub-methodologies that comprise the overall methodology.

2. "Proceedings of the OECD/NEA Workshop on Seismic Risk", NEA/CSNI/R(99)28, 10-12 August 1999. Provides a summary of conclusions and contents from 24 technical papers submitted on methodology and data for, uses of, and insights gained from seismic PSA/margin studies, and the development of risk-goal-oriented seismic design procedure.

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