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PRA: A POWERFUL ENGINEERING DECISION TOOL

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1. INTRODUCTION

There are at present two engineering approaches to deal with nuclear safety: the classical deterministic and a new probabilistic approach. The latter approach is complementary to the former. Conventional deterministic engineering is the result of 6,000 or more years of human engineering experience. An empirical collection of data on "good engineering practices" improved design by feedback of experience. Public risk was minimized by the design objective of avoiding failure, but risk was never explicitely estimated.

Conventional nuclear engineering is relatively new (about 50 years), however when the WASH-1400 was published with the results of the PRA of two power plants the records of the nuclear plants was already equal to 2000 reactor year. Such a consideralbe record demonstrates that safety was fair when the power plants were built within the standards of good conventional engineering, but this now must be complemented by new approaches.

The PRA approach is an effective way to estimate the probabilities of rare events in complex technical systems in the absence of an adequate empirical record. When the PRA is applyed to nuclear power plants, a number of systems improvement have been identified. In same cases a significant reduction of core melt probabilities could be obtained by minor modifications of the plant design. The PRA nature allows the verification of the coherence of the safety plant concept. Therefore the PRA results are carefully analysed by industry, regulatory authorities and by legal bodies. Changes in design or operation were and will be stipulated. The PRA is a powerful decision tool.

The IAEA has defined PRA as "the appropriate application of probabilistic risk assessment methods to nuclear safety decisons". In this context, PRA is the probabilistic analysis of reactor accident risk (risk probabilities x consequences). To achieve such an analysis, the PRA must treat all processes involved, from the initiating event to the health effects subsequent to a realease. It is general practice now to describe the levels of scope in a PRA by using the terminology of the PRA Procedures Guide (NUREG/CR- 2300). A Level-1 PRA covers the analysis of initiating events and system response to determine the frequency with which severe core damage or core melt occurs. A Level-2 PRA continues the analysis from core melt on through containment failure to calculate the source terms on radioactive releases. Finally, a Level 3 PRA uses the source terms, with their associated probabilities, to calculate off-site consequences.

It is important to note that each of these three analytic regimes is quite different in its degree of probabilistic character. Level 1 is heavily probabilistic, since it involves predicting the success or failure of many reactor components such as valves, pumps, and switches.

The second level, progressing from the failure to provide adequate core cooling through core melt and on to radioactive release, is essentially nonprobabilistic in character. Cooling system success analysis does enter the assessment of containment failure, but the bulk of Level 2 analysis is in the modeling of the complex physical phenomena associated with core melt. Although risk or PRA codes are used, it is really not probabilistic analysis.

The third level is somewhat like Level-2, a matter of modeling plume dispersion, health effects, etc. However, the results are quite dependent on weather conditions at the time of an accident, and that is treated probabilistically. Major work in such areas as the phenomena of core melt and fission product transport are not ordinarily considered PRA.

PRA techniques have gained broad recognition because they provide engineering and safety insights which are not attainable through any other means, and because they can shed light on the technical issues in situations where normal deterministic thought processes break down. Most PRA applications to date have been estimates of public risk from severe accidents. In the future, the emphasis will probably be on accidents that are less severe in terms of public risk, but which could have substantial economic consequences for public utilities.

A plant specific PRA is the best and most comprehensive analytical tool available today for analyzing the overall safety of a nuclear power plant. It helps improve operator training programs and plant procedures for operations, surveillance and maintenance, including emergency procedures for operators. In addition, the PRA process provides an unique opportunity for engineers on the project team to become intimately familiar with plant operations and procedures, since the PRA requires very specialized skills, as well as a complete spectrum of knowledge of plant systems and operations. Plant-specific PRAs can be used to propose changes in technical specifications aiming optimizing plant availability.

Those involved in the performance of a PRA study gain valuable engineering and safety insights. Conceptual insights are the most important benefit and the most general of these is the intirely new way of thinking about reactor safety, by forming a logical structure that transcends normal design practices and regulatory processes. PRA thought processes introduce a much-needed realism into safety evaluation, in contrast to the more conventional deterministic thinking that often masks important matters, owing to its conservative qualitative approach and its rigid adherence to design basis accident. PRA eliminates large numbers of events that do not contribute to core melt probability and risk estimates. Its concepts can help improve understanding in many applications across a broad spectrum of activities, including engineering, licensing and operations.

The PRA nature is dynamic and is meant to be continuously updated with inputs from new and better data, modelling and plant information. The long involvement of a utility with PRA builds an in-house capability to deal with many special areas of PRA utilization. The Philadelphia Electric Company, for instance, has been involved since the begining of the WASH-1400 program and now is able to use PRA for many purposes. Applications within the company were not limited to nuclear power plants, but also include studies related to electric power transmission and fossil power plants. In order to develop PRA usefulness to the maximum in the nuclear industry, definite commitment is necessary on the part of utility and plant management for development and retaining utility personnel trained in PRA techniques and commiting these people to perform PRA studies and keeping them current. PRA is a time consumming and costly process, requiring very specialized skills and dedication of the senior operation staff and engineering personnel. Among the several fields of PRA utilization it is worthwhile to point out the following: identification of systems and components important to safety: evaluation of technical specifications, including limiting conditions of operation (LCO); backfitting, including cost/benefit analysis; training of operators, plant staff, and regulators; design evaluation, including common-cause failures and human errors; identification, evaluation, and ranking of safety issues; emergency preparedness planning; allocation of inspection activities; accident management; simulation of accident scenarios; test, maintenance and repair policy; compliance with target values; and risk management. Recently the PRA has been used in the Chemical Industry and in Civil Engineering (i.e. Railways and Dan Construction).

Additionally, government and industrial organizations today are conducting various research programs to resolve problems encountered in the utilization of PRA. Among areas being addressed are the identification and treatment of uncertainties; the considerable degree of judgement needed in almost all aspects of human reliability evaluation; the disagreement about frequency of occurrence of common cause failures and the accepted means of analysis; the methodology developments needed for analysis of systems interactions; the large uncertainties associated with calculated risks from external initiatiors and the characterization of source term uncertainties".

2. HISTORICAL DEVELOPMENT OF PRA

Probabilistic methodology in reliability and safety evaluation can be traced back to the early 1940s at least, when quantitative probabilistic safety requirements were first proposed for the aeronautics industry. The probabilistic term then was that an airplane accident would not happen more than once per 100 000 hours flight time.

In 1942, when the high failure record (about 100%) of the German V-2 rocket program was studied, the concept of dependency between parts of a system was introduced as a step ahead of the prevailing belief that a system was only as good as its weakest part. One of the members of V2 project team, a mathematician named Robert Lusser, suggested a theory to explain the failures. He believed that the reliabilities of the system was equal to the product of the reliability of the parts. Thus, while previous thinking said that the system was only as good as its weakest part, according to Lusser's theory the rocket parts were dependent on each other. Of course, Lusser was not totally correct, but he hit on something important: He started a logical system analysis technique that looked not only at individual parts, but at how the parts worked together as a whole.

During the two following decades, initial concepts were refined and expanded to include the development of statistical models for analysing component failures and reliability theory.

In 1960 further development was achieved with the use of logical analysis in connection with the Apollo program US National Aeronautics and Space Agency (NASA). An important tool called "failure mode effect and criticality analysis" was then introduced. In 1962 the Bell Telephone Laboratories in the USA used the fault-tree technique in a study for the US Air Force on the reliability of launching and controlling Minuteman missiles. In the nuclear power field, in 1967 FR Farmer of the United Kingdom proposed a limit line for accidental iodine releases in terms of probability of occurance.

In 1975, the Reactor Safety Study WASH-1400 applied the techniques of probabilistic risk assessment and logical systems analysis to two "typical" nuclear power plants. Unfortunately, some people in the nuclear industry chose to concentrate on parts of this study's shortcomings and assumptions, rather than looking on it as a first step in the implementation of a powerful tool new to nuclear safety analysis. The consequence was that probabilistic techniques became subject to much controversy.

The Three-Mile Island (TMI) accident in 1979 changed the picture dramatically. Post-accident study groups, notably the Kemeny Commission, urged greater emphasis on probabilistic techniques. In its report, the Kemeny Commission recommended that "continuing in-depth studies should be initiated on the probabilities and consequences (on-site and off-site) of nuclear power plant accidents, including the consequences of meltdown, as part of the formal safety assurance program.

Moreover, when it was found that WASH-1400 - contained an accident sequence in which the relief valve on the primary coolant system opened under high system pressure, but failed to close when pressure was reduced (exactly the same component failure that occurred in the Three-Mile Island accident), the use of probabilistic methodology in nuclear safety gained additional momentum.

Thirteen years after the publications of WASH-1400, eight year after TMI and two year after Chernobyl, the probabilistic technique today is a maturing and useful tool for evaluation of reactor safety. Many PSA studies have been made and more are in progress in several countries.

Some studies such as WASH-1400 in the US, have been sponsored by governmental organizations. The NRC has a very large program expanding the use of PRA in regulatory activities and has in recent years turned to PRA to provide insights and make NRC regulatory requirements more coherent and rational. For this the NRC used resources of MIT and the National Laboratories; Sandia, Oak Ridge, Lawrence Livermore, Idaho, Brookhaven, etc. Another example is the Federal Republic of Germany's Risk Study (GRS). Many others studies were conducted entirely by the industry, or as joint ventures. Today there are a half-dozen firms with know-how in PRA, offering widely differing strategies on how utilitites can satisfy NRC PRA requeriments or the need for a state-of-art full-scope PRA for internal purposes. Recently, a NSSS vendor submitted to US NRC a PRA as part of the General Electric Standard Safety Analysis Report (GESSAR-II).

The IAEA has programs concerning such aspects of the field as: the trend leading from estimates of overall risk through identification of dominant accident sequences and on to reliability analyses of systems important to safety; the initiation of PRA programs in many Member States; the PRA effort on decision making; preparation of technical reports in the areas de PRA and operating experience, and training and support by means of short courses to meet the needs of utility managers who plan to incorporate a PRA group into their activities.

Up to now,in the Soviet Union, PRAs have only been perfomed on a qualitative basis. But the Soviet nuclear organizations need to have in-depth and plant-specific PRAs in order to determine whether possible safety - related backfits, including containments, pressure suppression pools, and filtered containment vents - will really yield the desired level of safety. In - depth PRAs are also needed for quantitative assements of the safety levels of new reactor tecnologies now under development in the USSR, in order to meet new social and economic criteria for acceptable nuclear power plant risks. Because Soviet organizations do not have all the methodology needed to do in - depth plant - specific PRAs, they are seeking to aquire US PRA methods and codes. The Soviets have involuntary acquired a vast and unique body of knowledge on what happens when the reactor containment is breached and radiactive products are dispersed into the atmosphere. The Soviet experience includes decontamination and dosimetry systems and many other findings invaluable to the nuclear community. The Soviets can exchange these important data for U.S. PRA expertise. Their authorities are convinced that with "radical improvement" of design and new equipment reability such as is forseen for the new advanced LWR reactors now under design-under the guidance of PRA results - the risk can be lowered at last by two or three orders of magnitude.

Sometime in 1988 the NRC will formally request an Individual Plant Examination (IPE) for every reactor not recently analyzed in a probabilistic risk assessment (PRA). The scope of the IPE is the identification of weakness or unusual conditions or configurations that could give an unit a higher severe accident risk with the objective of correcting the vulnerability or compensating for it. The estimed NRC deadline time for compliance with the IPE is 30 months. The IPE must follow a special model, the IPEM method.

The IPEM assesses risks and identifies any plant-specific weakness by comparing an unit, system by system, to a generic model developed by Idcor (Industry Degraded Core Rulemaking Program), which has been reviewed and santioned by NRC. The IPE provides a quicker and cheaper way to meet NRC requirements because it is a risk assessment specifically tailored to meet the objective of identifying severe accident vulnerabilities in a much more highly focused review. Eventually the IPE can be used as a "bridge" to a Level-1 PRA, when the utility is left with an in-house plant-specific model that can be used to address other regulatory needs and could also be enhanced for use in operations, maintenance, and training.

3. HUMAN FACTORS

Consideration of the effects of human error on operation, control, maintenance and equipment-testing is very important in any industrial activity, and actuarial records of industrial experience show that human interatctions, either with a system or other humans which can impact the frequency of an accident sequence, are a vital consideration in regard to plant safety. It is estimated that 80 to 90% of accidents in the Chemical Industry involve the human element and data from the airline industry during the past 15 years indicate that 60 to 80% of airplane crashes were attributable to human errors. Studies such the GRS report human error as contributing up to 63% to core melt frequency, which makes it obvious that analysis of human interactions with plant equipment merits the same degree of attention as analysis of the hardware systems.

Although human interactions can mitigate an accident sequence as well as exacerbate it, virtually all reactor accidents are traceble to one or more human errors in design, fabrication, construction, maintenance or operation and half of public risk derives from human errors in the area of operation, maintenance, testing, cognitive errors and manipulation of controls.

In spite of the lack of a large data base on human behaviour, quantitative estimates of the contribuitions to core melt probability or risk from human actions can be derived from PRA results. A step-by-step review of procedures, taking in account stress, number of operators, availability of information to the operator and the frequency of such things as switching errors made before and during the accident would yield the errors which should be included in the fault tree, in order to obtain a quantitative estimate.

The review of PRA studies indicates that it is necessary to account for five types of human interactions which may mitigate or exarcebate an accident. Before an initiating event, plant personal can affect availability and safety either by inadvertently disabling equipment in testing or maintance, or to the contrary they can improve the availability of the systems by restoring failed equipment through correct testing and maintenance. By committing an error, plant personel can initiate an accident. By following procedures during the course of an accident, plant personel can operate properly stand-by equipment and correctly terminate the accident. Contrarily plant personel attempting to follow procedures, can make mistakes that aggravate the situation or fail to terminate an accident.

Modeling human errors caused by misperception about the state of the plant made an important advance in PRA. For example, use was made of the local simulator in the Seabrook power plant study. Seabrook specific risk scenarios were used as input to the simulator and the operators response was observed. The results were documented and utilized as operator action input to the plant event sequence model and the result is a new mechanism for proposing and quantifying corrective actions relating to risk due to operational procedures, by means of specialized operator training. Therefore PRA results point out operations staff actions, thus providing possible focal points for training programs and improving existing procedures for operations, surveillance and maintenance.

Another very important field is the Licencee Event Reports (LERs). The NRC - Contractor (Oak Ridge National Laboratory, and the industry's Institute of Nuclear Power Operation) are both analyzing LERs with similar methods. The LERS are used to determine whether any precursors to core melt or severe core damage accident are to be found in these reports. A valuable result of the study was the development of a method to direct attention to the more serious or threatening events reported.

Humam reliability analysis is a complex subject that has not lent itself to relatively straightfoward models like those for component and system reliability assessments. During the past six years EPRI/NUS has conducted a human reliability program using simulator experiments to analyse human reliability for PRA studies, with the participation of seven utilities and their full-scope nuclear power simulators. A large number of experiments has been conducted since 1983 employing simulators at the Bugey, Paluel and Caen EDF training centers. EPRI and EDF have signed a collaborative agreement. The EDF and EPRI programs are fully combined and constitute the largest source of simulator data in the world, from which numerous safety insights could be derived.

For many years the OECD/Halden Reactor Project has been involved in the development and evaluation of advanced man - machine systems for nuclear plant control using a full scope simulator. Moreover, the laboratory has recently been enhanced by the installation of dedicated artificial inteligence machines for expert system development.The ultimate design objective will be an integrated information display and control system. The Toshiba Corporation in Japan has also developed computer based technologies and their application to the man - machine interface in nuclear power plant control rooms. An advanced control pannel system is now in use at one operating power plant in Japan. The control panels are of three kinds: main reactor, cooling system and balance of plant.The process computer incorporates elements of artificial intelligence and performs the following functions: high speed color graphic display, plant operation monitoring, core performance monitoring and management. The new control equipment reduces operator burden and failures.

As has been pointed out, it is very important to understand human interaction with complex processes. But emphasis on human "error" is a negative approach and can be non-productive, particulary when the "correct" human action has not been clearly defined prior to the incident. In this regard, some utilities and other organizations, as was shown, have expended great resources in a positive attempt to maximize human perfomance in such critical roles as the man-in-space program, the commercial airlines, airforce, deep-sea exploration etc. Central to this issue of the human factor in those activities is the role 'hat human play in reducing risk. Therefore, in order to make suostantial improvement in human performance, one needs to know to what extent the risk is actually reduced in order to give optimum attention to operator and personel training in surveillance and maintenance.

4. PRA SOFTWARE AND GUIDES

Since PRA methodology was published in the WASH-1400 much has been accomplished. In these 13 years, identified weak areas have been pursued, especially in modeling, data collection and in treatment necessary for support of PRA studies. In the area of computer codes, large varieties of codes are available for quantitative evaluation of large fault trees and event trees, thus aiding identification of common-cause failures and analysing propagation of uncertainties.

Apart from the detailed PRA reports which have been published describing methodology and results, there are many reports issued by the US National Laboratories and by the NRC. The availability of PRA literature is now quite general. The Electric Power Research Institute (EPRI) has a large research program related to various aspects of PRA. (EPRI computer codes are available under licence from the Electric Power Software Center, Dallas, Texas). Various guide books have documented procedures for conducting a PRA. In the US the NRC supported development of the PRA Procedure Guide (NUREG/CR-23000) in a joint effort with the of Electric Power Research Institute (EPRI), the Institute of Electrical and Electronic Engineers (IEEE), and the American Nuclear Society (ANS). This is a compendium of methods for conducting a PRA. For internal use the NRC also developed a first draft of PRA Review Material (NUREG/CR-3485). In January 1983 the IREP procedure guide for conducting a PRA was published .

The future of risk management lies in the computerization of the plant specfic PRA. It is envisioned that a user at a computer key will be able to trace though the entire risk assembly/ dissambley process, to identify the various sources and aspects of risk and to explore the efficiency of the corrective actions or modifications.

4.1 Data Base

The situation regarding data base in the US is very encouraging at present due to the results of comprehensive research sponsored by the NRC for improvement of data used in PRA with scope to monitor system performance and develop inputs to probabilistic risk assements already avaliable. In addition there is also availiable the data base (Std 500984) of the Institute of Electrical Engineering (IEEE), a standard for nuclear plant risk and reliability analysis. The Nuclear Plant Reliability Data System (NPRDS), published by the Institute of Nuclear Power Operations (INPO), and updated periodicaly, contains data from all operating nuclear power plants in a consistent and comprehensive manner and provids an on-line access for data entry and retrivel. In Europe there is a centralized system of four data bank, collecting and organizing information related to operation of light-water reactors (The European Data Bank). Since 1982 there is also a joint undertaking by Electricite de France and INPO, for a data collection and analysis program on human failures; times at which failures occur; place of failures; time between failure occurrence and detection and characteristics of tasks leading to failures.

5. IMPROVEMENT OF UTILITY MANAGEMENT OF NUCLEAR POWER OPERATIONS

In the endless debate about nuclear power, one of the aspects in greatest need of enhancement is that of utility management. Many utilities in the US and elsewere were unprepared for the complexities of nuclear power and the dedication it requires. This was inevitable, given the overenthusiansm prevailing when the electric energy demand was growing at a rate of 7% per year and nuclear energy seemed to be an ideal solution. Now there is quite a surplus, which apparently will prevail until the year 2004. At present, in a period of operation, the US is concentrating on perfecting operation rather than expansion. There are at present in the world 416 reactors producing electricity in 26 countries and some type of effort to imp,ove operation will be necesssary. The regulatory authorities are shifting part of their scrutiny from the plants themselves to the organizations running them. Specific areas, in need of attention, are training and organizational structures. Formerly, in the US, both areas were left to the discretion of the utility, but are now being addressed by the NRC and INPO, which are now taking drastic measures to improve operations.

Requirements for training show remakable variation. Good training programs are expensive and qualified instructors are scarce. It is important to set standards for training and establish reasonable programs to achieving them. The INFO is beginning to do this by establishing an accredited training program in order to achieve optimum progress. It is also necessary for the regulatory authority to impose these standards. Criteria for organizational structures will be harder to define. One obvious factor however is that the highest level of the utility management must be involved with their particular plants and fully commited to their better operation in order to achieve the acceptable level.

In contrast with Finland's four plants which averaged a phenomenal 92.28%, the US units in 1987 continued to average only less than 60% capacity factor. Other countries also had high capacity factors: Hungary averaged 88,79; Switzerland 84.66%; and Belgium 83.25%. As far as performance is under consideration nationality seems more influential than the type of reactor involved, according to data. Finish, Hungarian, Swiss, Belgian and German plants have high performance. Japan has an average of 77%, but performance is hampered by long statutory outages. Japanese units, once on line, are highly reliable and very seldon have forced outages. Why do those countries have such high performance? If you make inquires you soon descover that all utilities are making full use of all engineering tools used to improve plant management, including the use of PRA. The economic incentive for good operation of the 109 plants in the US is very great. The plants produce 40 billion dollars in eletric power per year. An improvement of the capacity factor from 60% to 75% means a saving of 5 billion dollars per year. Therefore there is an important incentive for the improvement of plant management and market for PRA is very good..

6. THE DEVELOPMENT OF A STANDARDIZED LWR DESIGN,OPTIMIZED FOR SAFETY, RELIABILITY AND ECONOMY

The complexity of nuclear plants is partially the result of extensive safety measures not taken in other energy technologies. The reactor plant designs currently available could be significantly improved. An effort to rethink the concepts by which reactors have been designed could result in LWRs that are cheappear, safer, more operable, and perhaps smaller than the type presently in use. Safety was generaly engineered as a superimposed addition to a basic design primarily optimized for power criteria, with a substancial increment in economic cost as a result. To improve safety significantly, the effort must draw on all that has been learned about the characteristics of safe reactors and integrate the best features into a package that would enphasize resiliency and passive safety features as well affordability and economy. The Westinghouse Eletric Corp. in a joint venture with Mitsubishi Heavy Industries and General Eletric Corp. with its join programs with others partners are efforts with these objectives. If the technical scope of the design is to maximize safety objectives as a preliminary element of design, this could be done using PRA as a guide. The EPRI is conducting such an Advance Light Water Program with several other cooperating utilities and manufacturing organizations. A complete reactor and plant design is extremely expensive and no corporation is likely to be able to finance alone, it unless it sees a major market. A good solution would be a Government-initiated nonproprietary design that could more easily draw on the work of a few vendors and archtect-engineers. A national or even an international design could have a better chance of sucess. The safety analyses also might be done on international base, making the design licencable thoughout the world as an standard reactor type. With international direct feedback the safety will be optimized to the maximum extent. There are several advantages to a standardized design. The cost would be predictable, since regulatory and construction uncertainties could be cleared before construction started. Costs also could be lowerd by incorporating improved construction techniques. It should be cheaper to operate because it would be designed to operate at a high capacity factor, with lower fuel cycle costs and lower operator exposures. The Advanced LWR design might represent an important step toward acceptance of nuclear power, since it would be safer, at least by a factor of ten or more when compared with current LWRs, as was shown by early calculations using PRA. It is important that the high safety standard of such and advanced LWR would be perceived by the public.

7. IMPACT OF RISK ASSESSMENTS IN PUBLIC ACCEPTANCE OF NUCLEAR POWER

The Rasmussen Report (WASH-1400) instead of fulfilling one of its purposes; to give safety assurance to the public, on contrary had a particulary negative effect on public acceptance, as a consequence of the public debates and the disagreements among the "experts". The "dispute among experts arguments" were more like monologues, with the partes arguing at one another instead of responding to what the opposing expert has actually stated, rejecting data which might develop the opponent's case; interpreting ambiguous data differently; and consequently, increasing polarization. The debate over the study continues until today, with critics arguing methodology and data quality. The result of the debate, widely publicized, was negative because the public, rather than attempting to follow the debate and sort out the facts for themselves, simply concluded that nuclear tecnology has not yet been perfected. If the "experts" cannot agree on whether or not nuclear power is safe, the average citizen is likely to assume it is unsafe.

The anti-nuclear community focused on the health consequences of the extremes of analytical probability distributions as a means of stimulating fear. The PRA does not reasure the public, because "de minimis" probability does not mean "de minimis" consequence. However, in this regard the Chernobyl accident had very important consequences. The scale of events was the theoretical "maximum credible accident" which opponents of nuclear power held was too high a price for mankind to pay for that source of energy. Even so, Chernobyl caused far fewer deaths than the tens of thousends forecasted for a "maximum credible accident". In fact, the accident has not undermined the nuclear industry's claim to be the safest of large-scale source of energy. Most of the 31 fatalities were among the firefighters exposed to radiation and to fires started by the accident. However (and this must be strongly emphasized), no fatality has been reported among the public. Therefore the accident blew away the major fear full argument used by nuclear opponents. What people saw inside the blazing reactor was certainly a horror, but one with magnitude to which they are well accostumed to see in toll of human lives in many accidents common to everyday life. The lack of "early victims" made opponents of nuclear energy concentrate the attack on the longtime effect of radiactive fall-out; they omitted however, important information, to such as that Chernobyl released less radiation than the athmospheric weapon testing between 1945-62; and that the over all biologic effect is only a small fraction of the natural radiactive background effect.

Some of the opponents of nuclear energy did good service to PRA improvement when they correctly pointed out weaknesses in some cenarios, data and uncertainties. The people developing PRA took note and corrections were made using these observations by competent scientific critics. Therefore part of the progress on PRA is owed to a positive action of opponents.

Public perception of risk and public nuclear acceptance differs greatly from one country to another. If Public perception were, in a country, was based on misinformation, improved public education programs might, in principle, be an appropriate response. Analysis of recent survey data in France indicates a current public acceptance of the same level as before the Chernobyl accident. France's fossil fuel and hydropower resources are poor. At present 50% of the primary energy comes from nuclear power and more than 70% of electric energy is of nuclear origin. Consequently for France, nuclear power is a must and the public belief is that nuclear energy is the most important factor in reducing France's reliance on foreign oil and other economic vulnerabilities. The same happens in same other industrial countries.

It is necessary to approach post-accident management with a more comprehensive mix of man-machine systems, including the specialist teams and organized services now part of the emergency plans. This implies extending the PRA analysis to a post accident risk management system (Level 3 PRA).

The public must be assured that in the event of any foreseable accident, a risk management exists that can credibly provide protection from the point of view of health and safety. Therefore the end objective of the nuclear system design would be the assurance of a minimal amount of radioactive release to the public in the event of severe accident and that the potentially exposed public can be fully guarded from any significant exposure. There are several technical concepts for this post-accident management of radiactive release, including the containment building, its internal spray and cooling systems, filtered venting systems, external rain-out devices, and such similar approaches to reducing the "source term". According to Chauncey Starr a water spray, external to a containment building to remove the bulk of radioactive release in the event of a containment failure or a venting system failure can be very effective. It is known empirically that an external and easily testable "rain-out system" can decontaminate the gas passing through it by roughly a factor of 100. Thus the magnitude of the source term would be substancially reduced and consequently the "risk-radius" for an exposed population. In the FRG one nuclear power plant is already equipped with facilities of controlled containment filtered depressurization. Similarly France's EDF also is adopting filtered depressurization. More attention to the post-accident emergency measures should be given to assure those members of the public who might be exposed in event of radiactivity release that a credible system exists for guarding them from harm. The systems should involve medical and others civilian services, both for insuring that there is no health danger to any member of the public and for assuring the public that their health and safety has been provided for, similar to existing emergency fire protection and emergency medical services through out the world. It is important to remove the concept of the inevitability of harm in the event of a reactor accident which now pervades the general public.

8. CONCLUSION

Safety is never absolute in any human endeavour. It is impossible to build any energy source absolutely free of risk, but in nuclear energy when all safety principles are adequately implemented the plants should be very safe and effective in meeting society's needs for abundant energy.

The PRA is a sophisticated approach to estimate possibilities of rare events in a complex technical system, in the absence of an adequate empirical record. For the engineer it provides a powerful tool for studying system dynamics, for aggregating engineering judgements, for disclosing potential system failures, for communication among other engineers, and for establishing priorities in design and operation. The PRA produces only "best estimate" probability for various failure frequencies. The PRA is not a panacea and has limitations. Its usefulness, however, is limited only by the competence and intelligence of the engineers and of management involved.

As was emphasized previously by its nature, nuclear power plant PRAs require very specialized skills, as well as a complete spectrum of knowledge of plant systems and operations. The risk ~~is~~ assessment is a process requiring the dedication of senior operation staff and engineering personel. The analyses contain uncertainties well known to the industry and the results of PRA include estimates of these uncertainties. In spite of its shortcomings PRA is the best and most comprehensive analytical tool available for analyzing the overall safety of nuclear power plants. When PRA is done for the first time in a country, it requires careful preliminary work. It is necessary to select a highly qualified consultant (a major firm's expert in PRA), and to be sure that the consultant contract implies a full technology transfer of a full scope PRA meant for permament in-house use. A utility conducting a PRA study should dedicate several fulltime people to the project. These people should assume leading roles and maintain a total, start-to-finish involvement with the project. With methods available today it would employ about 30 man-years. A full-scope PRA reaquires the dedication of team of four to six utility engineers. The PRA is of highest interest for the regulatory authority and to architect engineers and all technical people involved with Nuclear Power, therefore for a country conducting a PRA for the first time, it is necessary to pool human resources in order to profit to the maximum from the lessons learned in managing risk assessment projects.

In order to develop the PRA to the point of maximum usefulness for the utility a definite commitment on the part of the management of the utility must be assumed regarding the manner of developing and retaining utility personnel trained in PRA studies, and of keeping them current. Finally, utility management must develop a willingness to face the problems identified by the PRA process and to resolve them.

PRA use for reducing accidents frequence is operationally important for engineers, but not sufficient to assure public acceptance. The nuclear community has an obligation to translate its quantitative outcomes into quantitative concepts of comparative risk, comprehensible to the public.

Two hundred years ago Thomas Jefferson said: "When the people are well informed, they can be trusted with their own government". Fifty three years ago Prof. H.D. Smyth pointed out: "The ultimate responsability for our nation's policy rests on its citizens and they can discharge such responsabilities wisely only if they are informed". The only way to acquire and preserve public confidence is by means of truth. By permanent, correct and complete flux of information to the public it is possible to build and preserve confidence Therefore it is up to the nuclear community to keep the public well informed and, in due time, inexorable, the public acceptance will groth up.

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