

## **NUCLEAR ACCIDENTS AND ASSOCIATED ENVIRONMENTAL RISK**

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The paper presents a critical review on the recent view of some safety related issues concerning nuclear accident analysis and its environmental impacts. The philosophy of defence in depth, nuclear accident classification, and quantitative evaluation of environmental risk are among the issues being discussed. The problems of nuclear data harmonization and/or models, building trust and transparency between regulatory guides and public, and alternatives for relocation pathways are also addressed and evaluated.

### **INTRODUCTION**

The present paper addresses some safety related topics in their updated understanding and shows how they are linked towards an efficient quantitative risk assessment.

Any design change in nuclear power plants should be subjected to safety oriented evaluation to make sure that the change is not jeopardous

In each of the topics addressed in this paper, there are extensive research programs and studies. Within each topic, consistent matching between the subtopics could be preserved but the main objective of this review paper is to highlight the link between the topics and to assure that they are all targeting to minimization of radiological risk. By the end of the last section if you have the impression that all topics are related, then my goal will have been successfully achieved.

### **THE CONCEPT OF MULTIPLE BARRIERS**

To minimize the escape of radioactive nuclides released from nuclear reactors during normal operation and accidental conditions, the concept of multiple barriers is introduced in the design of these facilities. These barriers represent a sequence of obstacles to block the passage of radioactive nuclides from the fuel, or where they may originate, to the surrounding population. Seven barriers (all physical except the last one) normally exist [1] and designated as; fuel, cladding material, closed coolant system, reactor vessel, containment, site location, and evacuation plan. The concept of multiple barriers is still in force and a great effort through many R&D programs has been directed towards strengthening its effectiveness. Many computer codes are available to calculate the source term ( $Q$ , Ci /sec), the actual activity in the reactor coolant and steam generator [2], and finally the amount dispersed ( $\chi$ , Ci / m<sup>3</sup>) for routine and accidental releases. Most of these codes are expanded to cope with the effectiveness of the last barrier. Relocation is one of the protective actions that have been introduced to protect the public following an accidental release of radioactive material into the environment [3].

## DEFENCE IN DEPTH

The Atomic Energy Commission (AEC) has adopted, as a safety philosophy, the concept of defence in depth with three levels of safety [1]. These are intended to provide a kind of defence of depth to prevent or minimize radioactive releases to the environment. The three levels precept as follows:

- Level-1: Design for maximum safety in normal operation and maximum tolerance for system malfunction. Use design features inherently favorable to safe operation, emphasize quality, redundancy, and inspect ability and testability prior to acceptance for sustained commercial operation and over the plant lifetime.
- Level-2: Assume that incidents will occur in spite of care design, construction, and operation; provide safety systems to protect operators and the public; to prevent or minimize damage when such incidents occur.
- Level-3: Provide additional and appropriate safety systems, based on the evaluation of effects of hypothetical accidents, where some protective systems are assumed to fail simultaneously with the accident they are intended to control.

The concept of defence in depth as postulated by the AEC was accepted by the time it was written, however with the experience gained from the operation of NPPs, it was found to revise the concept and rewrite it in a more specific way [4]. The defense in depth can be considered as consisting of a set of actions, items of equipment or procedures, classified in five levels. The prime aim of each of which is to prevent degradation liable to lead to the next level and to mitigate the consequences of failure of the previous level.

The five levels are designated as follows:

- Level-1: Prevention of abnormal operation and failures via *conservative design and high quality in construction and operation*.
- Level-2: Control of abnormal operation and detection of failures via *control, limiting and protective systems and other surveillance features*.
- Level-3: Control of accidents within the design basis via *engineered safety features and accident procedures*.
- Level-4: Control of severe plant conditions including prevention of accident progression and mitigation of severe accident consequences via *complementary measures and accident management*.
- Level-5: Mitigation of radiological consequences of significant off-site releases of radioactive material via *off-site emergency procedures*.

In a recent IAEA document, the defence in depth is defined as: "hierarchical deployment of different levels of equipment and procedures in order to maintain the effectiveness of physical barriers, placed between radioactive material and workers, the public or the environment, in normal operation, anticipated operational occurrences and, for some barriers, in accidental conditions at the plant". This definition assigns the equipment to be used and procedures to be followed to maintain the effectiveness of the five barriers explained before and pertain to certain accident classification [5].

## **ACCIDENT CLASSIFICATION**

All reviewed accident classification schemes follow the inverse relationship between accident frequency and accident severity.

The first attempt to classify nuclear accidents was performed by the American Nuclear Society (ANS) [1] in accordance with anticipated frequency of occurrence (from high to low) and potential radiological consequences (from low to high). The scheme recognizes four conditions as:

- Condition I: Normal operation and operational transients.
- Condition II: Faults of moderate frequency.
- Condition III: Infrequent faults.
- Condition IV: Limiting faults.

It should be noted that Loss of Coolant Accident (LOCA) and design basis fuel handling accidents belong to condition IV.

The second classification was proposed by the Nuclear Regulatory Commission (NRC) [6] where the classification scheme was expanded to fall in nine classes. The classifications are considered as guidelines for the applicant for a reactor license to analyze a set of postulated, severe accidents to show that the facility can be operated without undue risk to the health and safety of the public. The scheme starts with trivial incidents (class-1) and ends with Design Basis Accident (DBA) considered in the Preliminary Safety Analysis Report (PSAR) and Final Safety Analysis Report (FSAR) (class-8). Class-9 is devoted to accidents that are more serious than class-8 such as station blackout during the course of LOCA.

The expansion proposed by NRC classification allows the safety analyst to be more specific in defining the accident and thus helps in the quantitative evaluation of accident consequences.

Recently [5] the operational status of the NPP is linked to the level of defence in depth (five levels) as: normal operation, anticipated operation occurrences, design basis accidents (DBA), severe accidents beyond the DBA, and post severe accidents.

## **MONITORED PARAMETERS AND DATA ACQUISITION**

Accurate measurements of neutronic and thermal-hydraulic parameters via reliable and diverse instrumentation system have an important role in timely and effective functioning of the reactor protection system [7].

In general, the monitored parameters are fed to operation system instrumentation (then to operation system control circuits) and protection system instrumentation (then to protection system logic circuits). Besides, monitored parameter signals can be displayed directly for manual operation system actuation.

The new generations of LWRs (ABWR/APWR/ESBWR) utilize advanced instrumentation systems with sophisticated transducers and data processing techniques suitable for nuclear applications [8-11]. ESBWR has taken an innovative approach to reduce systems and components and introduce passive systems to simplify the overall plant and to minimize the financial risk to the constructing utility.

Many developments had been achieved in data processing, storage, and retrieving. One of good examples is the development of a plant navigation system [12] to assist power plant

operators by automatically displaying the plant situation and plant operation procedures on a CRT screen when abnormalities occur.

In the same context, the study of neutronic/thermal hydraulic coupling through transient simulation is an important issue in improving performance of instrumentation in NPPs. The CRISSUE-S [13] project re-evaluates fundamental technical issues in the technology of LWRs related to the interaction between thermal-hydraulics and neutron kinetics. This project was undertaken in light of the advanced computational tools that are currently available to the nuclear energy community.

## **HUMAN FACTORS AND MAN-MACHINE INTERFACE**

Analysis of experience gained in the operation of NPPs pointed out the need of giving continuous and serious attention to human reliability research. The contribution of human reliability to overall NPP performance is evident from many serious events (including the Three Mile Island and Chernobyl accidents) in which human error was one of the major contributing causes. The human component has in fact been identified as both the initiating as well as the secondary event in abnormal occurrences and accidents. "How to improve the human reliability and man-machine interface" is a topic that has been discussed extensively [14, 15, and 16]. Multi attribute utility theory can be utilized to assign an optimum set of human performance shaping factors that yield minimum human error probability [17].

In all new generations of NPPs and nuclear system designs, the target of availing improved operating and maintenance performance is weighed equally with other targets such as improved power generation and greater safety margins [11].

At this point, one should point out to an important document [18] produced by the U.S.-NRC on human factors engineering program review model. The document is prepared to be used by the NRC staff to review the Human Factors Engineering (HFE) programs of applicants for construction permits, operating licenses, standard design certifications, combined operating licenses, and for license amendments. The purpose of these reviews is to verify that accepted HFE practices and guidelines are incorporated into the applicant's HFE program. The review methodology provides a basis for performing reviews that address twelve elements of the HFE program.

Finally and for the sake of completeness, one should draw the attention that continuous effort should be directed towards more and more improvement in human reliability (HR). Improvement should involve the three components of HR i.e., stimulus (S), operator (O), and response (R) [7].

An operator's diagnosis and interpretation of what is going on is a crucial matter that may result in either positive or negative interference.

Many studies address different ideas to optimize the manipulation of displays and controls and hence allowing the operator to execute the timely and proper corrective action. Based on risk evaluation, some studies group all possible sequences and design a corrective action panel (colors can be utilized to express the severity) to enable the operator to correctly understand what is going on and to take the appropriate corrective action(s) during the course of an accident [19].

## **ENGINEERED SAFETY FEATURE (ESF)**

ESF is one of three integral subsystems constituting the reactor protection system (the other two are the reactor shutdown system and the reactor protection logic). It includes containment isolation, emergency core cooling, pressure reduction, emergency or auxiliary power, and air filtration. The ESF response time is that time interval from when the monitored parameters exceed its actuation set-point at the channel sensors until the ESF equipment is capable of performing its safety function. An extremely challenging point is when the designer finds himself comparing this time with the rate of accident propagation.

There is no doubt that introducing passive components (e.g. accumulator in PWR (Emergency Cooling Injection System) in the design of ESF will positively enhance its effectiveness.

## **SAFETY QUANTITATIVE TOOLS AND RISK ASSESSMENT**

Now, it the time to incorporate all topics discussed in the previous sections towards minimizing the radiological risk from accident consequences in NPPs.

The identification of accident sequences, finally leading to radioactive releases to the environment, is greatly facilitated by Event Tree Analysis (ETA) [20]. By identifying the initiating event (e.g. pipe break) and the situation of reactor trip and ESF (branching stages), all possible sequences could be easily outlined. If no logical skipping is allowed, the number of sequences is  $2^n$ , where n represents the number of branching stages. Once the event tree is constructed, Fault Tree Analysis (FTA) for each system involved can be utilized to compute its failure probability (Failure mode and effect analysis (FMAE) is also useful). These probability values (and their complements) are then used to calculate the probability of occurrence of each sequence in the event tree.

The probability values for components involved are derived from failure rate data (including the human component) [21]. Failure rate data in reference [19] are collected from different industries since not enough data from actual operation experience from nuclear industry have been collected by that time. Revised data bank that includes the experience gained from NPP operation, recent types of components/transducers, and man machine interfaces, are extremely needed.

The probability value computed from ETA constitutes the first term in the product formulae for risk evaluation. The second term (consequences) can be obtained from expert judgments.

## **FUTURE NEEDS**

Harmonization in generating and using knowledge on energy risks is a big challenge towards nuclear risk prevention and mitigation [22]. In this context, the possible future needs could be figured out as:

- Further harmonization of risk assessment regarding methods & data .and quality assurance (presenting uncertainties /limitations)
- “Risk scale” as common reference to classify, compares and communicates different energy hazards.

- User-friendly international knowledge repositories on energy risks (e.g. “risk maps”, web-based information systems, etc).
- Universal data bank for component and human failure rates based on actual operational experience in NPPs.

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