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The evolution of the U.S. nuclear regulatory process

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ABSTRACT

This paper provides historical perspectives and insights on the early development of the U.S. nuclear regulatory process and its subsequent evolution towards risk-informed processes. After the landmark Reactor Safety Study (WASH-1400) and the TMI-2 accident, the U.S. Nuclear Regulatory Commission (NRC) began to use probabilistic risk assessment (PRA) methods and insights in regulatory applications as deemed necessary or useful. In 1995, the NRC adopted a policy that promotes increasing the use of probabilistic risk analysis in all regulatory matters to the extent supported by the state of the art to complement the deterministic approach. The NRC then started moving toward a much expanded use of PRAs in what is termed risk-informed regulatory approach. This paper discusses the challenges and the success stories of the use of probabilistic assessment of the risk to support and inform regulatory decisions.

1. Introduction

The passage of the 1954 Atomic Energy Act provided an opportunity for private companies to build and operate nuclear reactors. The Act also gave the Atomic Energy Commission (AEC) broad authority to establish regulations “necessary in order to enable it to find that the utilization or production of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public” (Atomic Energy Act, 1954). The Atomic Energy Act did not provide a formal definition of “adequate protection.” Rather, Congress left it up to the AEC to give a practical meaning to this term based on its technical expertise and on all the relevant information. Today, the U.S. Nuclear Regulatory Commission (NRC), created in 1974, operates under the same Congressional authority.

In the early years of development of nuclear power plants, both the technology and its governing regulations were in the formative stages. The AEC safety philosophy, as summarized in a March 14, 1956 AEC letter to the Congress of the United States, was based on the proposition that the ultimate safety of the public depends on three factors: (1) Recognizing all possible accidents that could release unsafe amounts of radioactive materials; (2) Designing and operating the reactor in such a way that the probability of such accidents is reduced to an acceptable minimum; (3) Protecting the public from the consequences of such accidents, should they occur, by the appropriate combination of containment and isolation (U.S. Atomic Energy Commission (AEC), 1956). However, at the time, the operating experience with power reactors and the state of knowledge of safety analysis had not progressed to the point where it was possible to use quantitative techniques to estimate the probabilities and consequences of accidents. Instead, conservative assumptions were used to bound “real” accidents and to provide upper bounds of the potential public consequences resulting from certain hypothetical accidents (the so-called “deterministic”[2] approach). The fundamental concept of defense in depth was invoked at the time to ensure that the unquantified probabilities of accidents were small.

The NRC and its predecessor, AEC, led the development of quantitative risk analysis for nuclear power plants. Nevertheless, the use of probabilistic risk assessment (PRA) methods and insights were usually limited to a variety of applications on a case-by-case basis as deemed necessary or useful. In 1995, the NRC adopted a policy that promoted increasing the use of PRA in all regulatory matters to the extent supported by the state of the art to complement the deterministic approach. The NRC started moving toward a

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[1] The views expressed in this paper are solely those of the authors and do not necessary represent those of either the ACRS or NRC.

[2] The term “deterministic” is not quite accurate. A better term is “conservative.”

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much expanded use of PRAs in what is termed risk-informed regulatory approach. This paper provides a historical perspective on the evolution towards a risk-informed regulatory process. The challenges and the success stories of the use of probabilistic assessment of the risk to support and inform regulatory decisions are also discussed.

2. Early years of nuclear power plant licensing

Beginning in 1961, the AEC began defining a standard regulatory prescription to licensing of nuclear reactors. Reactor siting was the first issue addressed with the new approach. Regulations for site selection were developed as 10 CFR Part 100, “Reactor Site Criteria,” in 1962. Part 100 was developed, in part, based on the assumptions that an upper limit of fission product release could be estimated and the containment building, as a final element of defense against the release of radiation, would hold even if a severe accident were to occur. In conjunction with Part 100, the concept of a maximum credible accident (later designated as a design basis accident) was developed to evaluate the acceptability of a potential site (siting limits) and containment design requirements (Nourbakhsh, 2012).

By the mid 1960s, as proposed plants increased significantly in size, the Advisory Committee on Reactor Safeguards (ACRS) became concerned that a core meltdown accident, particularly one in which the plant's emergency core cooling system (ECCS) failed to operate as designed, could lead to a breach of the containment. At the “prodding” of ACRS, the AEC established a special task force to look into the problem of core meltdown in 1966 (Walker and Wellock, 2010). The task force, chaired by William K. Ergen, a former ACRS member, issued its report in October 1967 (Ergen, 1967). The report offered assurances about the reliability of the ECCS designs and the improbability of a core meltdown, but it also acknowledged that a loss-of-coolant accident (LOCA) could cause a breach of containment, if the ECCS failed to perform. Therefore, containment could no longer be regarded as an unchallengeable barrier to the release of radioactivity. This finding represented a “milestone in the evolution of reactor regulation” (Walker and Wellock, 2010). In an ACRS letter on the task force report dated February 26, 1968, the Committee recommended, as it did in its 1966 report on safety research, that a “vigorous program be aimed at gaining better understanding of the phenomena and mechanisms important to the course of large-scale core meltdown” (Advisory Committee on Reactor Safeguards (ACRS), 1968).

The AEC sponsored research programs to more fully understand the effectiveness and capability of the ECCS. The results of some of the semi-scale tests performed in early 1970s raised concerns about the adequacy of ECCS. The AEC's attempt to keep the information regarding these tests away from the public and congressional oversight led to congressional hearings before the Joint Committee on Atomic Energy. During the ECCS controversy, Senator John Pastore, then the Chairman of the Joint Committee on Atomic Energy, wrote a letter to the Chairman of AEC, James Schlesinger, requesting a comprehensive assessment of reactor safety. This letter seems to have been an impetus for the WASH-1250 study, “The Safety of Nuclear Power Reactors and Related Facilities,” (U.S. Atomic Energy Commission, 1973) which was circulated as draft for comments in 1972. However this study did not provide a quantitative assessment of the risk in a probabilistic fashion as discussed in Senator Pastore's letter[3] (Okrent, 1981). The AEC then initiated a major study to estimate the probability of a severe accident which resulted in the publication of the landmark Reactor Safety Study (WASH-1400) (U.S. Nuclear Regulatory Commission, 1975) in 1975 and the beginning of the science of probabilistic risk assessment as applied to nuclear power plant safety.

3. Reactor Safety Study (WASH-1400)

The Reactor Safety Study (WASH-1400) (U.S. Nuclear Regulatory Commission, 1975) was the first systematic attempt to provide realistic estimates of risk to the public from potential accidents in commercial nuclear power plants. Two specific reactor designs were analyzed in WASH-1400: the Peach Bottom Atomic Power Station, a boiling water reactor (BWR) with a Mark I containment, and Surry, a 3-loop pressurized water reactor (PWR) with a subatmospheric containment. A major conclusion of the Reactor Safety Study was that the low probability-high consequence accidents involving core meltdown, containment failure, and failure of engineered safety features dominated the risk to the public. The study also pointed out the significance of human errors and support systems.

The WASH-1400 report stimulated a great deal of debate after its release. In June 1976, the Committee on Interior and Insular Affairs of the U.S. House of Representatives, chaired by Representative Morris Udall, held hearings on the findings of the study. These hearings found that the study seemed to be misleading in the certainty and comprehensiveness of its conclusions (Keller and Modarres, 2005). Representative Udall suggested that an outside review panel be formed to take a closer look at how the study arrived at its conclusions (Keller and Modarres, 2005). The NRC then asked Professor Harold Lewis of the University of California at Santa Barbara to chair an independent review group, which produced what is now known as the “Lewis report” (Lewis et al., 1978). The Lewis report concluded that the WASH-1400 study was overall a “conscientious and honest effort”, an “important advance” over earlier quantitative analyses of reactor safety, and employed a “sound methodology” that should be used more widely by the NRC. Among the shortcomings that the Lewis Committee identified in WASH-1400 was the difficulty to follow the detailed thread of any calculation through the report. The Lewis report was particularly critical of the Executive Summary for being “a poor description of the contents of the report” and for not adequately indicating the full extent of the consequences of reactor accidents and the uncertainties in their probabilities. For this reason, the NRC withdrew its endorsement of the Executive Summary although it did not repudiate the study itself.

The WASH-1400 report identified significant weaknesses in the traditional “deterministic” regulations. The risk significance of human errors, common-cause failures, and support systems had not been appreciated in the traditional system. Later PRAs, such as those for the Zion and Indian Point plants, continued to identify major contributions to risk (e.g., earthquakes and internal fires) that had not been attracted much attention in the traditional regulatory system.

4. Lessons learned from the TMI-2 accident

The March 28, 1979 accident at Three Mile Island Unit 2 (TMI-2) led to the reexamination of the design basis and the consideration of regulations for protection against severe accidents. The reexamination of the design basis was prompted by the fact that the TMI-2 accident involved a small-break LOCA, whose consequences should have been bounded by those of a large-break LOCA, but became much more severe due to misunderstanding of the event

[3] A statement in the senator's letter that essentially defines a PRA is the following: “… prepare a report which, by addressing the probability of occurrence and consequences of the spectrum of accidents which could befall a nuclear power plant, would represent an assessment of the risks involved in the use of nuclear plants.”

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