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STATISTICAL SAFETY EVALUATION OF POWER REACTORS

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Building 4 R. E. Alexander		STATEMENT OF PROBLEM Evaluate the possibility of improving safety evaluations of power reactors by applying the statistical technique developed for the SNAP 10A flight test.			
Building 2 W. W. Davis C. W. Griffin R. S. Hart A. A. Jarrett W. B. Sayer					
Building 5 D. J. Cockeram B. W. Colston R. L. Detterman R. J. Gimera J. Hale L. I. Moss A. R. Piccot W. J. Roberts R. J. Rohlfis C. V. Hendricks		ABSTRACT The standard maximum credible accident approach to safety evaluation is reviewed briefly and its shortcomings are noted. The statistical approach and its advantages are outlined. An example is presented in which the MCA-type accident does not determine the reactor hazard. The development of the statistical approach is discussed and it is concluded that conditions are propitious for general application of stochastic methods. Appropriate advances are suggested.			

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I. INTRODUCTION

The standard safety analysis for nuclear power stations is based on the maximum credible accident (MCA).^{1,2,3} By this approach, various accidents are analyzed to determine the resulting activity release. An accident, the MCA, is then postulated which releases more activity and presents greater potential hazards than any accident considered credible for the plant. The combination of site and plant safety features must reduce the off-site doses to criteria levels. Thus, it is concluded that the plant presents no undue hazard to the public. Actually, it has been demonstrated that the worst accident considered credible, occurring once, will not produce injurious radiation doses off site.

The shortcomings of the MCA safety evaluation are well known.^{4,5} In general, it does not determine the magnitude of the hazard. The dividing line between credible and incredible is purely subjective. Focusing attention on the most spectacular accident may obscure subtle possibilities and lead to neglect of less severe accidents which may be more important because they occur more frequently. Conversely, actual hazards might be dominated by an "incredible" accident because the resulting damage would be great, even though the likelihood of its occurrence is small. The MCA analysis generally does not focus attention on the problem areas nor does it lead to appropriate system safety improvements. The MCA result is not useful in comparing nuclear to conventional hazards. The consequences of normal operations, which produce air and water pollution, are ignored. While engineering safeguards are considered vital by the Advisory Committee on Reactor Safeguards,⁶ present accident analysis techniques discourage their development.

The standard (MCA) approach to nuclear safety analysis does not determine the hazard magnitude nor indicate where improvements should be made.

II. THE PROBABILISTIC APPROACH

The hazard (R) from a reactor (or any other device) is the product of the damage from a particular accident (D_i) times the probability of occurrence for that accident (P_i) summed over all accidents.⁴

$$R = \sum_{i=1}^n D_i P_i$$

R might be expressed in overexposures/yr or \$/yr. Normally, no attempt is made to determine R, ostensibly because appropriate data are unavailable, and because no such evaluation is required. The maximum value of D_i for which the corresponding value of P_i appears to be "significant" is obtained but this is only a part of the problem. In the probabilistic approach, each accident is analyzed. A value of each D_i and each P_i is obtained and the total hazard is determined.

In the intercontinental ballistic missile field, very high reliabilities have been obtained using fault tree analysis techniques with modern computers. The fault tree analysis is closely related to the statistical safety analysis in that both require and permit consideration of small details while maintaining cognizance of the overall problem. Actually, fault tree analysis is an important aspect of the probabilistic approach to safety studies. Statistical safety evaluations utilize the fault tree analysis system which has proven so effective in improving ICBM reliability and can lead to similar improvements in nuclear reactor safety.

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The following example is presented to illustrate the general probabilistic approach. Each of the blocks in the simplified diagram (Figure 1) represents the result of a detailed fault tree analysis. Consider accident I, loss of coolant flow. The probability of occurrence value (P) in the first block indicates that the analysis predicted that flow loss would occur 10^{-3} times per year. Flow loss does not necessarily produce an off-site hazard. It may be detected by the low flow detector which has been determined to have a reliability of 0.98*. If the low flow detector fails (solid line), fission products will be released to the containment vessel. A hazard results if the containment vessel, with a reliability of $(1-10^{-9})$, also fails. Success of the low flow detector (broken line) can cause the emergency coolant system to be actuated by the automatic system (reliability $1-10^{-4}$) or the operator (reliability 0.98). The hazardous condition is then averted unless the emergency cooling (reliability 0.98) and the containment fail. Considering all the possible ways it could develop, the probability of a hazardous release due to loss of coolant flow (P_I) is 2×10^{-14} per year. Considering the various possible meteorological conditions, the possible degrees of containment failures, etc., the average number of overexposures per release (D_I) is 500. Thus the risk from a loss of coolant flow accident is 10^{-11} overexposures/yr. Similarly, the risk associated with each accident is obtained; the total risk is the sum of all of these individual risks.

*The probability of failure, P, is one minus the reliability.

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Figure 1. Hazard Evaluation Diagram

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The total risk due to this reactor is .038 injuries/year; that is, injuries from the reactor will average 1 every 26 years. A 200 Mwe reactor can provide power for a community of about 200,000. Therefore, an individual's chances of being injured by the reactor in his community are about 2×10^{-7} per yr. He is 10 times as likely to be struck by lightning as to be injured by the reactor.²² His probability of accidental injury from conventional hazards is about .05 per yr.²² The average risk from a single automobile is about .02 injuries per year.²² Therefore, the reactor of this example is less dangerous than 2 automobiles. The community served by the reactor contains about 80,000 automobiles.

It is clear that for this reactor (1) the reactor does not significantly affect the safety of the people it serves and (2) the MCA type accidents do not contribute significantly to the hazard from the reactor.

A statistical evaluation approach has been used, particularly in Canada^{8,9} and Japan,¹⁰ for power reactors and it has been considered for nonnuclear facilities.¹¹ Gomberg applied the technique to site evaluation for the Enrico Fermi Atomic Power Plant¹² and a similar study was performed for the Ford reactor at the University of Michigan.¹³ Statistical hazard evaluation is in general use for space applications of nuclear power. Atomics International¹⁴⁻¹⁷ and Nuclear Utility Services^{18,19} have evaluated SNAP system hazards on a statistical basis and such an approach was recommended by the Atomic Industrial Forum.²⁰ Planning Research Corporation has completed a statistical evaluation of some U.S. power reactors.^{7,21} Holmes and Narver, Inc. is currently collecting reliability data on power reactors which will be valuable input for statistical safety studies. Thus, while this approach has met with limited success, it is neither new or untried.

Two problems have limited the use of the statistical approach to safety evaluations -- in addition to the AEC requirements.¹ First, the basic data for such calculations are quite limited. A statistical study must be more thorough, and therefore is more laborious than a MCA

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analysis. Neither problem is insurmountable and the possible advantages justify serious attention to statistical safety studies.

Probabilistic safety analyses would determine the total hazard and therefore permit useful comparisons between various reactors and various sites as well as comparisons between nuclear and conventional plants. A complete safety evaluation would improve understanding of safety measures and problems and call attention to areas where significant improvements are possible. Assurance of safety would be improved because (1) secondary hazards are not ignored in focusing attention on the spectacular accident and (2) the truly damaging accident is not ignored as "incredible". It appears that only through full, statistical safety analyses can nuclear safety research and development be appropriately directed and safety be ensured adequately to permit locating nuclear power stations near population centers.

The state-of-the-art is now appropriate for advancing statistical safety analysis. The art has been developing for a decade and reactors have been operated for some 23 years. The SNAP 10A flight test constitutes a true breakthrough in that the safeguards report¹⁴ was statistical and the operation was approved on this basis. The statistical approach to safety analysis has been used to determine appropriate areas for improvement.^{15,17} Studies evaluating nuclear power plant safety on a statistical basis^{7,21} have been conducted and these will be very helpful in adapting the SNAP 10A techniques to terrestrial plants. Data being collected by Holmes and Narver, Inc. will be valuable inputs to statistical evaluations and the analyses will point out areas where other data should be obtained. Nuclear safety technology has reached the stage where thorough statistical safety analyses are appropriate and needed.

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III. FURTHER DEVELOPMENT

The appropriate advances in the development of statistical evaluation techniques are clear. First, the SNAP 10A safety evaluation system should be modified for application to terrestrial nuclear power plants using the PRC results,^{7,21} and the experience of the Canadians and Japanese. Second, existing central nuclear power plants should be analyzed as built and with postulated changes in siting, containment, and other safety systems. This study should establish the value of various precautionary measures, indicate areas where improvements are needed or would be of significant value, show where data are needed, and develop the statistical evaluation technique to a generally useful state. Finally, the statistical evaluation technique should be detailed, perhaps in a handbook format, and made available for general use and a system should be established for collecting pertinent data.

IV. CAPABILITY

Atomics International possesses a unique combination of capabilities for this work based on:

1. More than 10 years experience in the nuclear field and reactor safety evaluation.
2. Pioneering the probabilistic approach to safety in the SNAP 10A flight test.¹⁴
3. Utilizing statistical safety studies to determine appropriate areas for improvement.¹⁵

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V. SUMMARY

An improved method for evaluation of nuclear power plant safety is needed because the present MCA approach neither determines nor leads to appropriate improvements in engineering safeguards. This need can be met by the statistical approach which has been developing for a decade and which recently reached maturity in providing the basis for approval of the SNAP 10A flight test. Other studies, related to central power stations, and data collecting activities have set the stage for developing the statistical analysis technique and using it to obtain direction for future development of engineered safeguards and for ensuring that future reactors present no undue hazard to the public, even when placed near population centers.

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