SAND90-1844C

APPLICATION OF NUREG-1150 METHODS AND RESULTS TO ACCIDENT MANAGEMENT*

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ABSTRACT

The use of NUREG-1150 and similar Probabilistic Risk Assessments in NRC and industry risk management programs is discussed. "Risk management" is more comprehensive than the commonly used term "accident management." Accident management includes strategies to prevent vessel breach, mitigate radionuclide releases from the reactor coolant system, and mitigate radionuclide releases to the environment. Risk management also addresses prevention of accident initiators, prevention of core damage, and implementation of effective emergency response procedures. The methods and results produced in NUREG-1150 provide a framework within which current risk management strategies can be evaluated, and future risk management programs can be developed and assessed. Examples of the use of the NUREG-1150 framework for identifying and evaluating risk management options are presented. All phases of risk management are discussed, with particular attention given to the early phases of accidents. Plans and methods for evaluating accident management strategies that have been identified in the NRC accident management program are discussed.

INTRODUCTION

The risk from five nuclear power plants was examined during the NUREG-1150 program.[1] When the analyses of the plants were complete, an effort was undertaken to examine the implications of NUREG-1150 for risk management initiatives.[2] The term "risk management" was used in place of "accident management" because a more comprehensive evaluation was performed. Strategies for preventing accident initiators, preventing core damage and providing more effective emergency response were examined, in addition to the strategies normally evaluated for accident management. The framework provided through the

*This work supported by the U.S. Nuclear Regulatory Commission and performed at Sandia National Laboratories, which is operated for the U.S. Department of Energy under contract DE-AC04-76DP00789.

NUREG-1150 analysis presented a means within which current risk management strategies could be evaluated and future risk management strategies could be developed and assessed.

This initial risk management work provides the base for an expanded role of probabilistic risk assessment (PRA) in risk management and demonstrates its usefulness. Using the integrated framework of a PRA, operator actions that could mitigate or terminate a severe accident can be identified, and the impact of the action on risk can be quantified. In addition, such studies can be used to assist the Nuclear Regulatory Commission (NRC) in prioritizing severe accident research such that detailed evaluations are performed for the operator actions that have the greatest potential for risk reduction.

The development of system and phenomenological models to support risk management is relatively straightforward. However, implementation of risk management strategies in the form of additional emergency procedures is more complex. Of particular concern is the development of procedures that are sufficiently flexible to deal with the range of possible accident progression outcomes. The uncertainties in accident progression clearly indicate the need for symptom and function oriented procedures, rather than event based procedures. The feasibility of actually performing the strategies must also be considered. Factors that must be considered include environment, equipment availability, operator training, and regulatory restrictions.

A key aspect of managing severe accidents is the availability of reliable monitoring instruments and displays. In developing current risk management plans, it should be recognized that much of the available instrumentation is not designed to operate in the severe pressure, temperature, and radiation environments that may occur in the risk-dominant accident sequences.

PRA USE IN RISK MANAGEMENT

Risk management programs at nuclear power plants have two basic objectives:

1. Minimize the public health risk from nuclear power plants, and

2. Provide the capability for operators and decision-makers to effectively respond to and thereby reduce the probability and consequences of severe accidents.

Severe reactor accidents involve extremely complex system and phenomenological responses that are often nonintuitive. When developing and evaluating risk management options it is important to understand how a particular action may affect other portions of the accident progression. For example, intentional depressurization in a PWR might mitigate the threat from direct containment heating (DCH), but it also affects the amount of hydrogen generated during core degradation which could affect the containment failure probability from hydrogen combustion. The phenomena in severe accidents are also highly uncertain. The current uncertainty in the phenomenology yields a wide range of potential outcomes which must be considered when developing risk management strategies.

The PRA methods developed for NUREG-1150 provide an integrated analysis framework that can be used to evaluate the potential ramifications of a certain action over a wide range of possible outcomes. The framework provides the capability to compare various strategies based on selected risk measures, such as health and economic risk. It should be recognized that reducing total core damage frequency does not always reduce total risk for a plant. The consequences for each sequence must also be considered. For example, replacing pump seals in a PWR might yield a lower core damage frequency, yet higher overall risk, if the strategy increases the containment failure probability from DCH.

The core damage fault trees and event trees, as well as the accident progression event trees, provide a logical framework for identifying potential risk management strategies. The framework assists in identifying potential options at each stage of the accident.

A key area where the NUREG-1150 methods can contribute to risk management is in the treatment of uncertainties in accident progressions. PRA results can supplement detailed deterministic calculations by identifying alternative outcomes for the important accident sequences. By identifying these alternatives, along with their frequency of occurrence, the operators are made aware of the uncertainty in severe accident progression and the need for sufficient flexibility to deal with a spectrum of potential outcomes.

INITIAL RISK MANAGEMENT WORK

Late in the NUREG-1150 program, an effort was undertaken to examine the implications of NUREG-1150 and similar PRAs for risk management initiatives.

Risk management can be divided into five separate, but interrelated phases:

1. Prevention of accident initiators (reliability management),

2. Prevention of core damage (accident management),

3. Implementation of an effective emergency response (emergency response management),

4. Prevention of vessel breach and mitigation of radionuclide releases from the reactor coolant system (accident management), and

5. Retention of fission products in the containment and other surrounding buildings (accident management).

"Accident Management" is a term that is often used in place of "risk management"; however, the former is usually applied to the late stages of phase 2 and phases 4 and 5. Thus, risk management is a more comprehensive approach.

The current test, maintenance, and operating procedures in place at most plants are generally good with respect to the first two phases of risk management. While additional work is still needed in these areas, it is the late phases of a severe accident that need the most attention and development.

A risk-based methodology for identifying and evaluating risk management options for each of the five phases listed above was developed. The general approach consists of determining the important contributors to risk, identifying options to reduce the impact of the contributors, and evaluating the risk impact of the options. More discussion of the approach for each of the five phases of risk management is provided in the following sections, and then quantitative examples are provided for Phase 1 and 2 strategies.

Phase 1 - Prevention of Accident Initiators

The first step in reducing the impact of accident initiators is to identify the initiators that are important to risk, which is a straightforward process with the availability of a PRA. It is extremely important to recognize that the initiators most significant to risk are not usually those that are the most frequent, so merely reducing the total number of trips may not significantly reduce plant risk. After the important initiators are identified, the next step is to ascertain the root causes of those failures. Before the frequency of these events can be reduced, the reasons for their occurrence must be understood.

The next step is to identify options for reducing the frequency of important initiators, which may include:

1. Improvements to operating procedures,

2. Improvements to test and maintenance procedures,

3. Changes to technical specifications and limiting conditions for operation,

4. Changes to hardware and system configurations, and

5. Adding or revising automatic "early time" responses.

The final step in this process is to evaluate the potential risk reduction of each option using the PRA framework. Options can be evaluated in terms of their impact on the core damage frequency (CDF) and/or overall risk.

Phase 2 - Prevention of Core Damage

The occurrence of an initiating event leads to challenges to the plant safety systems. Operators must bring the plant to a subcritical condition with adequate water inventory and decay heat removal.

The evaluation of phase 2 options is similar to the evaluation of phase 1 options. This process includes the identification of important accident sequences, hardware failures and human errors within these sequences that contribute most to the CDF, and determining the root causes of these failures. Enhanced risk management options can then be proposed, which may include:

1. Improvements to operating procedures,

2. Improved operator training and staffing,

3. Improved test and maintenance procedures for safety-related systems,

4. Hardware modifications.

Similar to the evaluation of the phase 1 options, the final step for phase 2 options is to evaluate the potential risk reduction of each option using the PRA framework. Options can be evaluated in terms of their impact on the CDF and/or overall risk.

Phase 3 - Implementation of Effective Emergency Response

Emergency response involves actions outside the plant before and after an accident to reduce public exposure to radiation. A specific emergency response will be comprised of some combination of evacuation, sheltering, decontamination, and interdiction strategies. Emergency response can be very site-specific, and is strongly influenced by population density, road systems, weather conditions, and interactions with and between local and state governments. Some existing emergency response strategies consider alternatives such as graded response or sheltering. There is very little guidance concerning correlation of the emergency response with the anticipated progression of the accident. For example, the relationship between containment failure or venting and evacuation strategies should be considered.

PRA information can assist the utilities and NRC in several parts of the emergency response decision-making process. The pre-accident evaluation process should determine the important contributors to risk by characterizing possible source terms, then determining the risk importance of factors such as site conditions, including population characteristics and road conditions, and the operator's predictive capability. Next, both short-term and long-term emergency response actions can be identified to reduce the risk. Finally, integral risk evaluations of alternative strategies can be performed to support the development of site and accident-specific response strategies.

Phase 4 - Prevention of Vessel Breach

If core damage is inevitable or has already occurred, then the goal of risk management is to arrest the degradation process and retain the fission products and core materials within the vessel and reactor coolant system. Recovery may be attempted at any time, from when the fuel rods are intact to when the corium is lying on the bottom of the reactor vessel. Currently, there are no detailed procedures related to the timing and injection of water into an overheated core. There is usually little or no guidance beyond instructions to flood the core, if at all possible.

It is probably best to deal with situations in this phase of risk management in terms of plant states (collections of symptoms defining the plant status, e.g., pressure, temperature, and radiation levels) and functional responses. In evaluating various options using the NUREG-1150 methods, the following steps would be included:

1. Identify the risk important plant states,

2. Identify the possible plant state variables that could identify these states,

3. Determine the ability of the operators to use available instrumentation to identify existing plant states,

4. Identify possible functional responses, and

5. Evaluate the probability and consequences of potential outcomes for each functional response.

Once the evaluation is complete, appropriate strategies can be selected and implemented. This implementation could involve procedures, guidance and hardware modifications along with modifications to training and plant practices.

The major goal of this phase is to obtain a coolable core and minimize radionuclide releases. A number of risk management strategies that could be proposed to achieve this goal include:

1. Addition of improved instrumentation,

2. Use of non-safety systems to provide makeup water,

3. Varying the rate and location of injection, depending on the particular plant state, and

4. Increasing or decreasing the primary system pressure, as appropriate for the scenario.

Analyzing the possible outcomes of various actions is a complex process. The NUREG-1150 methods provide a framework for evaluating each possible recovery scenario from a probabilistic standpoint to identify potential outcomes and assess their influence on overall risk.

Phase 5 - Retention of Fission Products

If the primary system boundary is breached, fuel and radionuclides will be released to the containment, and risk management will be oriented toward preserving containment integrity and/or strategies to reduce off-site radioactive releases. At this point, the risk management environment is changed in a number of important ways. First, the plant state characterization will rely more heavily on containment parameters, and the key diagnostic data are provided via different pathways. Second, different time scales may now govern the accident. Third, the systems and actions available for responding to the accident are largely different. Finally, the interface with off-site emergency response decisions is at its most critical stage.

The approach to this phase of risk management is similar to that for Phase 4 in that plant states and functional responses can form the basis for selecting risk management strategies. The five steps previously identified for Phase 4 are also utilized to develop risk management strategies for Phase 5. A number of strategies may be considered, including:

1. Addition of improved instrumentation,

2. Management of combustible gases,

3. Injection of water into containment,

4. Venting strategies,

5. Additional methods for containment heat removal,

6. Additional methods for reducing suspended aerosols, and

7. Strategies for controlling high pressure melt ejection.

Examples

The methodology described in the previous sections was demonstrated by evaluating the risk impact of potential risk management strategies, current risk management practices, and recent plant changes. Because this was a brief effort, the strategies were limited to phases 1 and 2, and the strategies were evaluated by determining the impact on CDF, rather than the impact on total risk.

Several potential risk management strategies were examined for the Surry plant. These strategies were aimed at lowering the frequency of station blackout sequences, which dominate the CDF for this plant. For each strategy, the ratio of the CDF calculated when the strategy was included to the base NUREG-1150 CDF was determined. This ratio is called the CDF reduction factor. One of the potential phase 1 strategies examined was adding an additional diesel generator. As shown in Figure 1, this strategy reduced the CDF by a factor of 1.9. Three of the phase 2 strategies were extending the battery life, using an onsite gas turbine generator, and adding improved reactor coolant pump seal material (to reduce the frequency of seal LOCA events). The impact of these strategies on the CDF is also shown in Figure 1. The CDF reduction factor ranged from 1.1 to 1.6. The combined effect of implementing all four of these strategies was also evaluated. The PRA framework accounts for the interrelations among the options, which for this case results in a lower CDF reduction factor for the integral evaluation than would be calculated by summing the individual reduction factors.

The effect of some of the risk management procedures currently in place at Surry and Peach Bottom were also evaluated. When evaluating these current procedures, the CDF reduction factor was calculated as the ratio of the base NUREG-1150 CDF to the CDF calculated without credit for the strategy The reductions in CDF that have occurred from four of these practices are shown in Figure 2. For Surry, the procedures for feed and bleed cooling (to provide heat removal when steam generator feedwater is not available) have reduced the CDF by a factor of 1.5. Cross connects to the auxiliary feedwater and high pressure injection systems at Surry Unit 2 have reduced the CDF for Surry Unit 1 by a factor of 3. For Peach

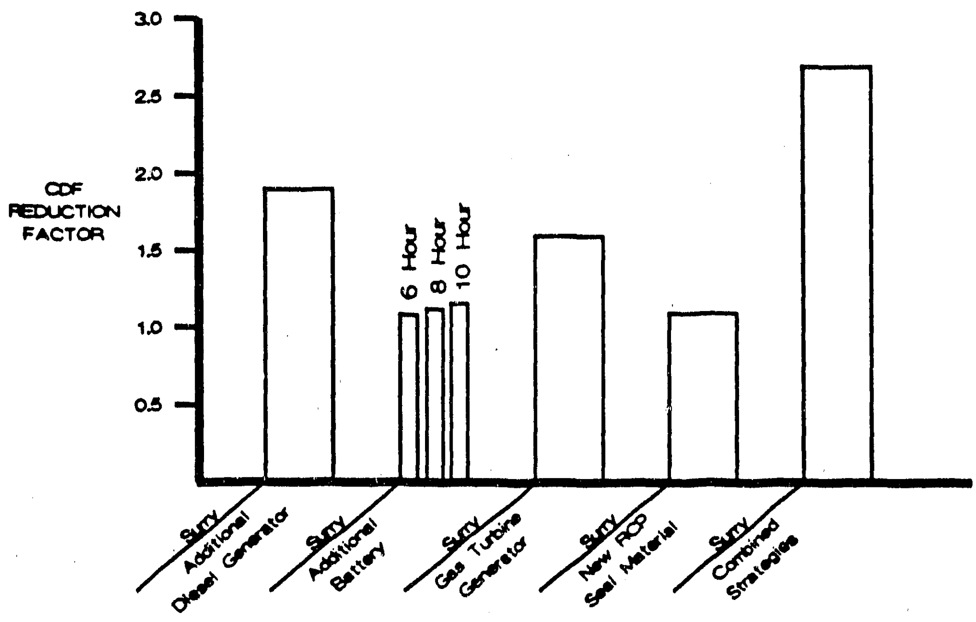

Figure 1. Effect of Future Strategies on Core Damage Frequency

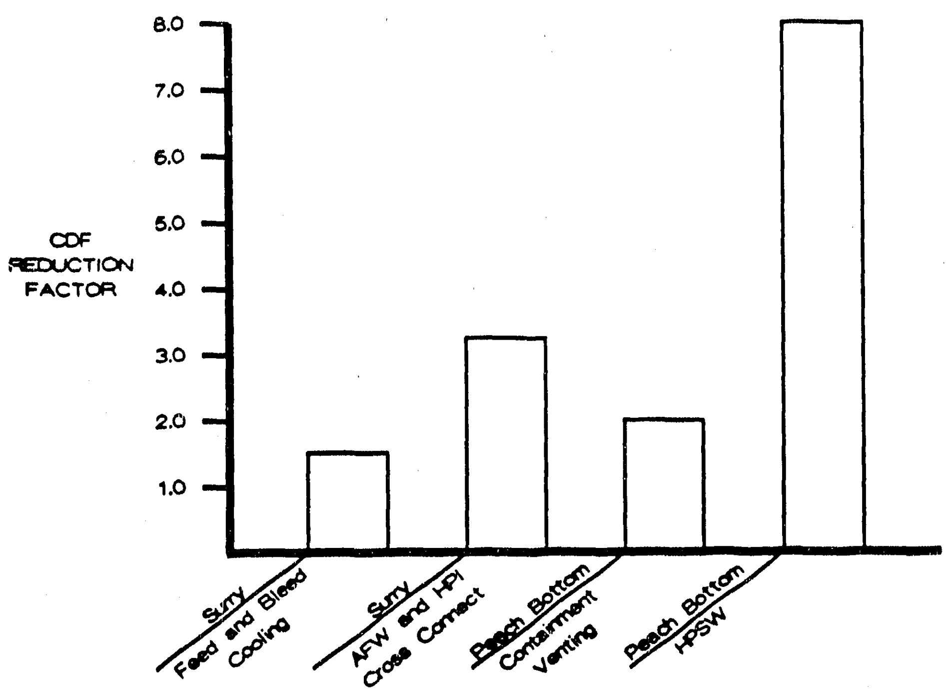

Figure 2. Effect of Current Strategies on Core Damage Frequency

Bottom, containment venting in sequences in which all containment cooling is lost (which prevents containment overpressurization and subsequent loss of core cooling) reduces the CDF by a factor of 1.9. Use of the high pressure service water (HPSW) system at Peach Bottom reduces the CDF by a factor of 8. This system draws water from a river and most of the components are located outside containment. Thus, the HPSW system is largely independent of other safety systems, and has a large impact on the CDF.

ACCIDENT MANAGEMENT STRATEGIES EVALUATION

Sandia National Laboratories is beginning an accident management program for the NRC that will further demonstrate the feasibility of using risk methods in accident management studies. While the examples discussed in the previous section were phase 1 and 2 strategies, the strategies evaluated in this accident management program will primarily be for phases 4 and 5. The objective of the program is to bring a risk perspective to accident management research by providing a systematic way to:

1. Evaluate the efficacy of proposed strategies,

2. Identify alternative strategies, and

3. Examine uncertainties.

The program is initially focusing on quantifying the risk reduction for operator actions that are identified in other portions of the NRC's accident management research. The impact of intentionally depressurizing the reactor pressure vessel during station blackout sequences at Surry is being quantified first. The next evaluation will consider a strategy of adding borated water in boiling water reactors (BWRs) to prevent recriticality following recovery of coolant injection. Thereafter, the impact of intentionally depressurizing the reactor pressure vessel will be reexamined for the Oconee plant.

To evaluate the strategies, the impact of the operator actions on all portions of the PRA must be considered. An example of the impact of intentional depressurization is shown in Figure 3. Although intentional depressurization is being proposed to mitigate the threat from DCH, the action would also affect other portions of the accident progression. An integrated treatment is necessary to capture all such possibilities. Depressurization affects the accident timing, which affects the time available for recovery actions. This effect must be included in the core damage frequency analysis and the accident progression analysis, and will cause a change in the CDF and relative frequency of accident progression outcomes. By altering the accident progression, parameters such as hydrogen generation and steam explosion probability will be affected in addition to the likelihood of DCH. The effects of intentional depressurization on the source term analysis include changes in in-vessel fuel releases and deposition in addition to changes in plant releases because of the change in likelihood of containment failure. The consequence analysis is also affected because of changes in accident timing and plant releases.

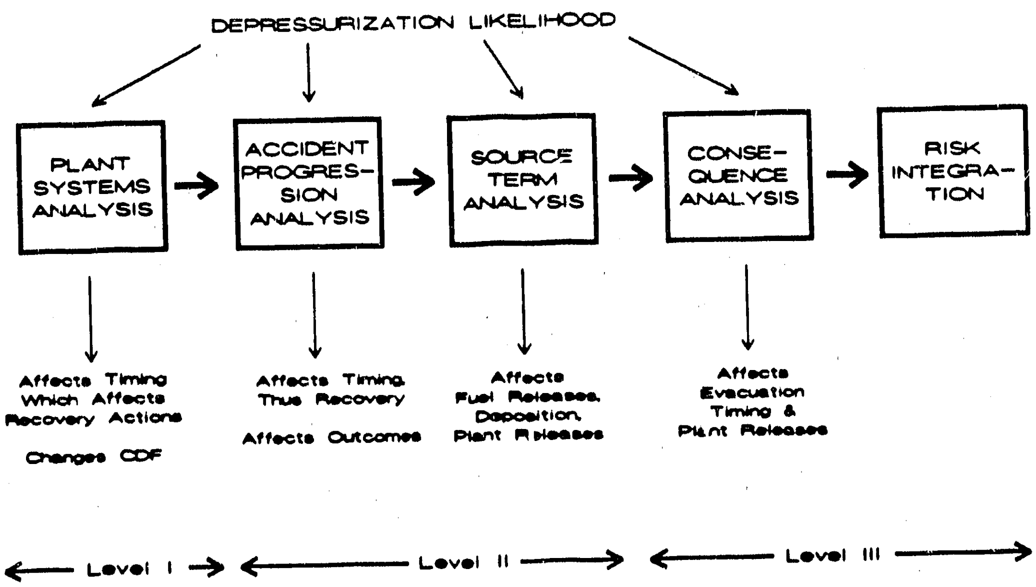

Figure 3. Impact of Intentional Depressurization on PRA

To perform the evaluation of specific accident management strategies, the parameters in the event trees that are affected by each strategy will first be identified. The quantitative effect of the strategy on these parameters will then be determined. A human factors analysis will be performed to determine the likelihood of the operator performing the action. Finally, the appropriate level of risk calculation will be performed and the results with and without the strategy included will be compared to evaluate the strategy.

SUMMARY

A general approach was developed for using PRA analyses to supplement risk management programs in all five of the identified risk management phases. This approach allows for the in-depth, integrated treatment of all phases of severe accidents. Further, alternative outcomes in the progression of severe accidents can be explicitly treated.

PRA techniques have been demonstrated to be effective in addressing risk in three different ways:

1. PRAs provide direct benefits by identifying plant vulnerabilities that can be corrected by the utilities,

2. Current risk management procedures and hardware can be examined to determine their efficacy and help assure correct implementation, and

3. Future risk management strategies can be developed and evaluated in an integrated fashion.

The nuclear industry has taken many positive steps to reduce risk since the accident at Three Mile Island. However, there are many improvements that are still possible. The capabilities identified and demonstrated in this paper can help to guide and evaluate future improvements in risk management programs.

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12/28/90