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THE PHILOSOPHY OF SEVERE ACCIDENT MANAGEMENT IN THE US

A J BARATTA

NUCLEAR ENGINEERING DEPARTMENT, PENN STATE UNIVERSITY

231 SACKETT, UNIVERSITY PARK, PA 16802

USA

Summary

The US NRC has put forth the initial steps in what is viewed as the resolution of the severe accident issue. Underlying this process is a fundamental philosophy that if followed will likely lead to an order of magnitude reduction in the risk of severe accidents. Thus far, this philosophy has proven cost effective through improved performance. This paper briefly examines this philosophy and the next step in closure of the severe accident issue, the IPE. An example of the authors experience with deterministic calculations as applied to severe accident analysis is examined.

In the wake of the TMI accident, growing attention has been paid to the management of severe accidents. This paper reviews the current US Nuclear Regulatory Commission's philosophy on severe accident management and presents a brief summary of Penn State's experience with the initial stages of severe accident management. Although the study examines a BWR scenario, the lessons learned are of value to any plant, PWR or BWR.

The NRC considers that accident management is a key element to the closure of the severe accident issue. The term accident management includes both preventative and mitigative elements. It is viewed by many as a cost-beneficial method to achieve substantial reduction in the risk to the public and the utility from severe accidents.

Pre-TMI, the emphasis on safety relied heavily on the engineered safeguard features. In the post-TMI era, the approach has become a multifaceted one emphasizing improved operational performance, human factors, realistic performance of systems and operators, and the use of probalistic studies. There is little doubt that plants today are safer than they were prior to TMI. One only needs to review the US capacity factors to see the significant improvements achieved. A less obvious and harder to quantify improvement is the introduction of emergency operating procedures (EOP's) which are symptom based and which include events that go beyond the traditional design basis accident.

Accident management is a broader application of the existing EOP philosophy. As Victor Stello recently noted,[1] it uses the existing data base of knowledge gained from plant specific severe accident analysis, the Individual Plant Examination or IPE to obtain insights and understanding of how to:

a. Prevent core damage,
b. Terminate the progress of core damage if it begins and retain the core within the reactor vessel,
c. Maintain containment integrity as long as possible,
d. Minimize off-site release.

Just as analyses with reactor transient analysis codes such as RETRAN and RELAP have lead to a better understanding of operational transients and hence improved operations, the process is expected to improve accident response and meet the above goals.

Accident management includes much the same philosophy as transient analysis does, in that it involves measures taken before an accident occurs. Improved training, hardware, and procedures are just some of these aspects expected to result from an effective management program.

The NRC's Accident Management Program outlined in Figure 1[2] has four fundamental objectives.

a. The development of technically sound strategies for maximizing the effectiveness of personnel, equipment and procedures in precluding and mitigating severe accidents,
b. The availability of instrumentation and equipment adequate to diagnose and control severe accidents,
c. Ensuring that operators, engineers and management are well trained to follow an event beyond design basis accidents,
d. And ensuring the availability of a sound technical basis for assessing the effectiveness of such strategies and capabilities.

In the NRC's view, an adequate severe accident management program includes a mechanism for implementing accident management procedures, training operators, engineers, and management, ensuring the availability of adequate instrumentation, a clear chain of command for decision making, and finally the availability of computational aids to assess plant status and guide in the response to such accidents.

NUREG 1150[3] is the basis sited as an example of the type of analysis required to identify those areas which need to be addressed to reduce the risks. It is felt that the NUREG studies have already identified a number of simple measures capable of significantly reducing the risk. As Stello pointed out, the modification of diesel-driven firewater systems to provide a source of cooling water for the core contributes to the prevention of core damage and reduction in the probability of containment failure.

One of the first steps in the process is to obtain an in-depth understanding of those features unique to a specific plant that both contribute to the probability of core damage as well as reduce its likelihood. This understanding is achieved in part through the IPE process.

Using both probalistic and deterministic techniques, the IPE attempts to identify the major contributors to core damage and radionuclide release. As outlined in Generic Letter No 88-20[4] and NUREG-1335[5], the process objectives are to have each utility, "(1) develop an overall appreciation of severe accident behaviour; (2) understand the most likely severe accident sequences that could occur at is its plant; (3) gain more quantitive understanding of the overall probability of core damage and radioactive material release; and (4) if necessary, reduce the overall probability of core damage and radioactive material release by appropriate modifications to procedures and hardware that would help prevent or mitigate severe accidents". The objective being to ensure compliance with the NRC Safety Goal Policy Statement.[6]

At Penn State, we have concentrated on the application of deterministic techniques to aid in understanding the most likely severe accident sequences. This process involves the modelling of the plant using one or more codes such as MAAP, BWR SAR, RELAP/SCDAP, and TRAC.

In our most recent case, a study of the containment response was performed for a large BWR during a steam line break.[7] The analysis developed a more realistic temperature time profile for the containment (See Figure 2). The more realistic criteria in turn allows qualification of equipment to be extended for a much longer time frame altering the response of the operators to such a scenario. The study also showed the desirability of early initiation of containment spray. Other studies by a US utility were used as a basis for blackout event tree timing. These studies showed that one operating fan cooler will subcool sump water sufficiently to provide successful cold leg recirculation without component cooling to the RHR heat exchangers.

Such experiences show the value of a thorough engineering analysis of one's own plant. These analysis must, however, be viewed with some uncertainty as are any complex thermal hydraulic calculations. To overcome this, sensitivity studies using several codes are performed for the same events and results compared. This approach allows one to gain increased confidence in the methodology and to identify any short comings.

References

1. V Stello, "NRC Perspectives and Plans on Accident Management," Proceedings-PSA'89, Pittsburgh, PA, ANS (April, 1989).

2. "Revised Severe Accident Research Program Plan," FY1990-1992," NUREG-1365, USNRC, Washington, DC (1989).

3. "Severe Accident Risks: An Assessment of Five US Nuclear Power Plants," NUREG-1150, USNRC, Washington, DC (1989).

4. D Crutchfield, "Individual Plant Examination for Severe Accident Vulnerabilities - 10CFR 40.54(f)," USNRC Generic Letter No 88-20, (November 23, 1988).

5. "Individual Plant Examination: Submittal Guidance," NUREG-1335, USNRC, Washington, DC, (1989).

6. "Policy Statement on Severe Reactor Accidents Regarding Future Design and Existing Plants," USNRC, Federal Register, Vol 50, p 32138, (August 8, 1985).

7. K Smith and A J Baretta, "Containment Response to a Small Main Steam Line Break," ANS Trans, (to be published, June 1990).

Captions

Figure 1. USNRC's Severe Accident Program leading to closure of the severe accident issue.

Figure 2. Comparison of original GE developed containment atmosphere temperature profile during a small break steam line LOCA with that calculated using BWR LTAS and CONTAIN.

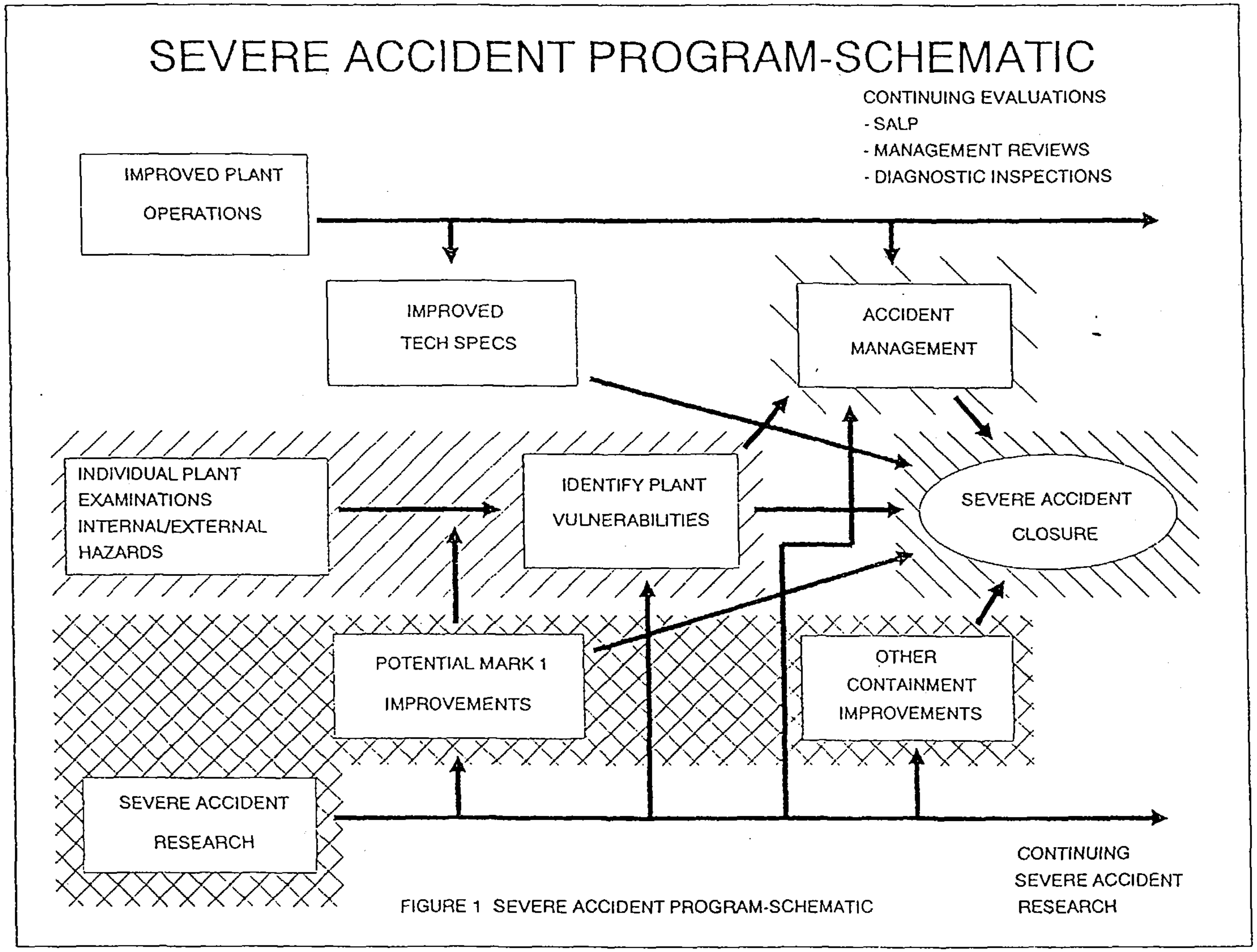

FIGURE 1 SEVERE ACCIDENT PROGRAM-SCHEMATIC

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COMPARISON OF CONTAINMENT TEMPERATURE PROFILES

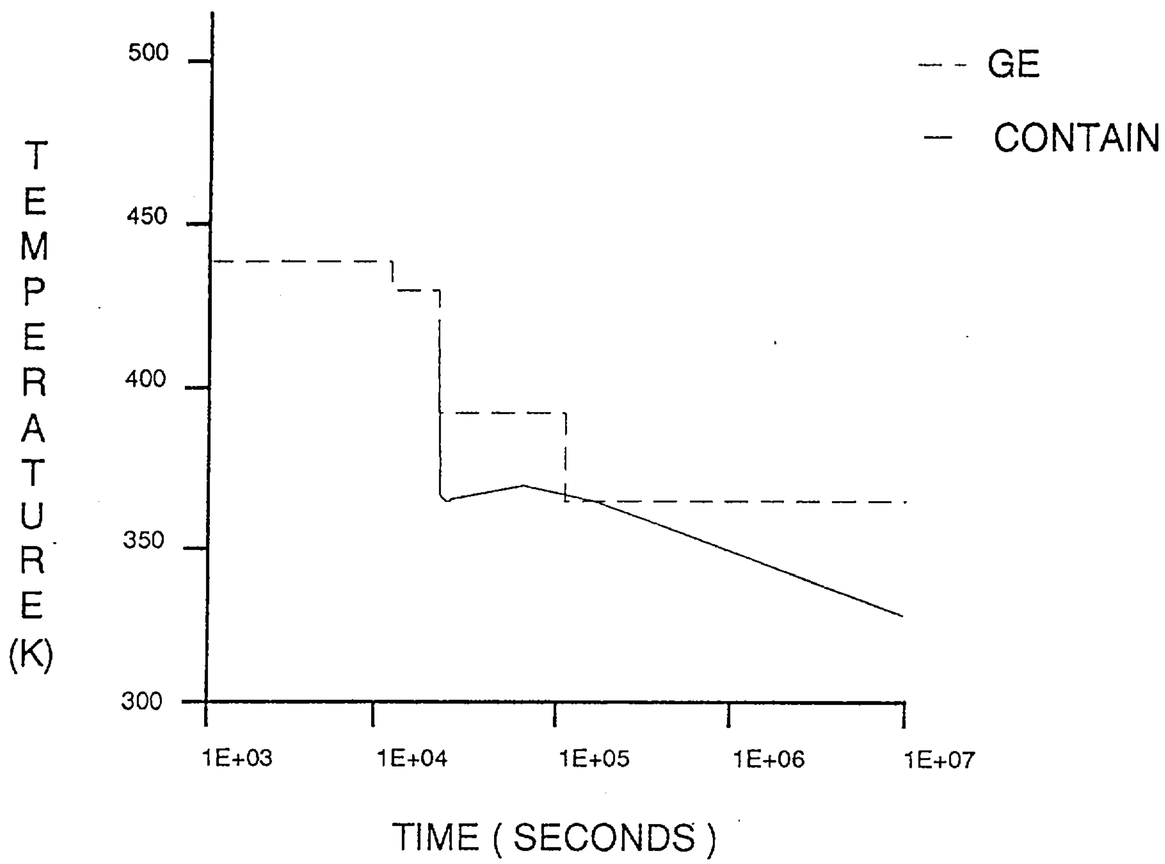

FIGURE 2

A J BARATTA