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Management of Severe Accidents

Extending Plant Operating Procedures Into the Severe Accident Regime

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Battelle's Columbus Laboratories

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ABSTRACT

This study examines the feasibility and value/impact of extending emergency operating procedures into the severe accident regime. It reviews the types of knowledge needed to develop such procedures and the applicability of existing regulatory review criteria. A method is developed and illustrated in two test cases.

This study concludes that it is feasible to develop procedures for operators to mitigate the consequences of accidents progressing past the onset of core damage. A preliminary value/impact assessment indicates a significant likelihood of there being an overall net positive benefit of developing mitigative procedures. A phased program has been proposed. First a pilot study should develop the application of the methods used in this feasibility study and provide more precise information for a detailed value-impact assessment. Based on the results of the pilot study, extension to a greater population of plants may be justified.

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1. INTRODUCTION

Following the occurrence of the events at Three Mile Island, Unit 2, (TMI) in March, 1979, the U.S. Nuclear Regulatory Commission (NRC) and the nuclear industry embarked on a major program to develop requirements, and responses to those requirements, for a new generation of emergency procedures.* These new emergency procedures are based on a different philosophy concerning the way in which operators should respond to abnormal events.

Previously, operators were required to identify the nature of the event's initiation, such as a loss of coolant accident (LOCA) or a turbine trip, and then select a procedure appropriate to that particular circumstance. The events at TMI showed that diagnosis by operators may not always be correct, and as a result the new emergency procedures place much less reliance on the operators to identify the specific initiating events correctly. Rather, they identify operator actions to be taken in response to specific plant states, such as excess power generation and low primary-system coolant inventory. In most cases, guidance is provided for instances in which one or more of the various safety functions, such as power control, primary-system inventory control, and decay heat removal are threatened. This philosophy has been termed "function-based" in contrast to the earlier "event-based" approach.

The NRC's requirements for the revised emergency procedures have been identified in several documents, principally NUREG-0660[2], -0737[3], and -0899[4], which are summarized in Section 2 of this report. Although these NRC requirements address the need to establish operator responses for preventing or limiting core damage, the requirements are less clear on the extent to which procedures should address operator actions to be taken once significant core damage or containment failure has occurred. In some cases, particularly for boiling-water reactors (BWRs), actions to maintain the integrity of the containment and to contain a severe accident are identified. In general, however, utilities have not vigorously identified or evaluated actions that would mitigate the consequences of severe accidents for the purpose of incorporating the resulting insights into emergency procedures.

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* Throughout this report, we refer to "emergency procedures," the written instructions provided to plant operators to help bring the plant under control during an abnormal event. A companion report, "Accident Management: Perspectives on Managing Severe Accidents in Commercial Nuclear Power Plants,"[1] distinguishes among risk, accident and emergency management. Using the definitions in that report, the procedures would be termed accident response procedures. However, we retain the more familiar terminology here for the reader's benefit, though occasionally we use the terms interchangeably.

NRC has requested Battelle's Columbus Laboratories to evaluate the feasibility and value/impact of extending emergency procedures further into the severe accident regime. Our findings are reported here. This report consists of seven sections and an appendix. Section 2 discusses the knowledge required for developing emergency procedures; it also summarizes the current NRC criteria for function-based procedures and their relevance to mitigative-action procedures. Section 3 summarizes our views on the degrees of confidence in the current knowledge about the phenomena associated with severe accidents. Section 4 describes a method for developing strategies to mitigate the consequences of severe accidents taking into account the current uncertainties and applies that method to a BWR and a PWR. Section 5 assesses the value/impact of a program for extending and incorporating insights about severe-accident management into plant-specific preparations such as training. Section 6 summarizes the findings of this work and makes recommendations for future work. Appendix A describes the "Ultimate" (U) procedures developed by the French utility, Electricité de France; these U-procedures exemplify procedures directed primarily at mitigating the consequences of severe accidents.

2. REQUIREMENTS FOR DEVELOPING EMERGENCY PROCEDURES

Severe accidents are currently defined by the NRC as "those in which substantial damage is done to the reactor core whether or not there are serious off-site consequences."(5) Further, accident management is defined as "that set of actions taken by the plant operating crew to gain control of the outcome of an abnormal event at the earliest possible time and with the minimum adverse consequences."(1)

The revisions being implemented in emergency procedures continue to emphasize preventing core damage. All indications are that the outcome will be successful. Nevertheless, the defense-in-depth philosophy suggests that some attention be paid to those sequences of events in which, for whatever reason, the onset of substantial core damage manages to occur.

In this analysis, we emphasize those alternatives available to the operating crew for terminating core degradation, for maintaining containment integrity, or for otherwise reducing the radiological source term, given that substantial core damage has begun.

This emphasis is believed to be appropriate for the following reasons:

- It is a major area of accident preparedness in which a systematic effort is not reflected in function-based procedures.

- Several programs by NRC and others (such as IDCOR) have examined ways of reducing the frequencies of severe accidents. Few have examined ways of minimizing the consequences of these accidents, should they occur, by making the best possible use of current designs.

- Several programs are indicating potential changes in assumptions about source terms (the amount and types of radioactive materials that could be released following a severe accident). This program, by concentrating on possible operator actions to manage severe accidents, provides an opportunity to evaluate the significance of the changes in source terms. Specifically, the substantial reductions in source terms could indicate that the relatively modest design and operational changes investigated here show a much more favorable value/impact relationship than do some proposed large-scale changes.

2.1 The Knowledge Base

All the wisdom of the plant's designers, owners, operators, and regulators theoretically is reflected in the plant's operating procedures. Given the plant's physical construction and a threat to its

normal operation, the emergency (i.e., accident response) operating procedures define the relationship between the plant and the people on the spot--the operating crew.

Several bases of knowledge are required to develop emergency procedures. The three principal bases of knowledge are phenomenological knowledge, plant knowledge, and human-factors knowledge as shown in Figure 2.1. Of particular importance when considering emergency procedures to help manage severe accidents is an understanding of degraded core behavior, containment phenomena, and fission product behavior. Such knowledge is distilled from studies and evaluations of severe accident sequences, such as those performed in the NRC's Severe Accident Research Program, the Industry Degraded Core Rulemaking (IDCOR) Program, and specific plant studies, such as the Level 2 and 3* probabilistic risk assessments (PRAs). This knowledge indicates how the plant would respond physically to accident initiating events and to subsequent automatic and manual actions. Section 3 examines the existing phenomenological knowledge base in more detail.

Plant knowledge represents the existence and status of plant systems (such as valve alignments), their associated controls, and their accessibility and operability under severe accident conditions. In other words, these represent the means by which operators may respond to severe accidents. This information may be in the form of a systems analysis, a set of plant drawings and descriptions, or it may exist in the minds of plant personnel.

As shown in Figure 2.1, phenomenological knowledge and plant knowledge can be combined to form accident response strategies. These strategies are descriptions, expressed in engineering terms, of what can be done to manage severe accidents successfully. For example, during a severe accident a strategy might be to vent the containment wetwell of a boiling-water reactor (BWR) at the design pressure, thereby avoiding uncontrolled containment rupture. This particular case involves assessing trade-offs among intentionally relieving pressure build-up, possibly releasing small amounts of radioactivity to prevent potential uncontrolled release later on, and assuring the ability to reisolate containment. This example requires knowledge of the plant (that specific vent lines exist and their valve controls will survive and are accessible) and of the containment phenomena and source terms (the effects of venting on pressure build-up and fission product scrubbing).

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* Level 2 PRAs are those that evaluate in-plant consequences such as containment performance and source-term magnitudes. Level 3 PRAs include the scope of Level 2 PRAs but also estimate casualty and property-damage levels. The terminology is taken from the PRA Procedures Guide, NUREG/CR-2300.

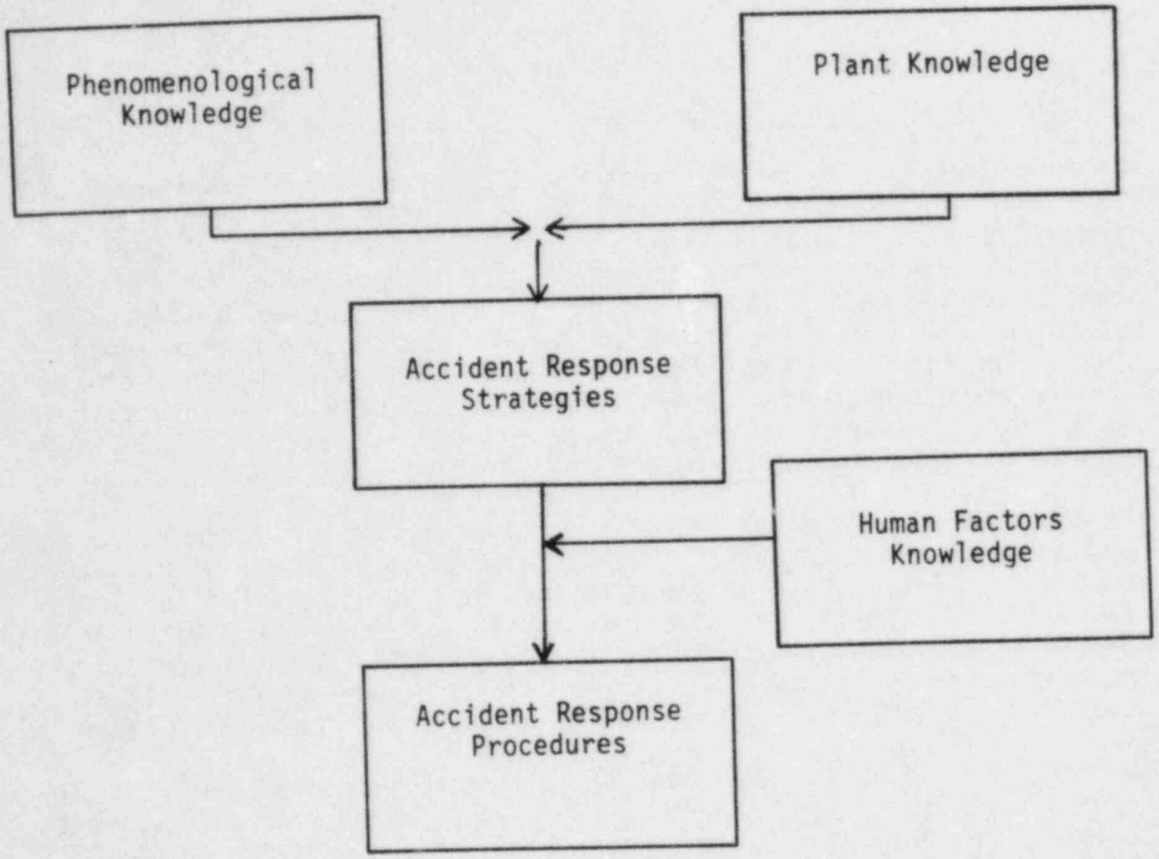

Figure 2.1 Process for developing accident response procedures.

The translation of such a strategy into procedures requires the addition of a third body of knowledge, that of human-factors knowledge. This knowledge provides ways for identifying the steps that have to be performed and the decisions that have to be made by the operating crew (such as by performing task analyses), their probability of success, and how those steps can best be put in the form of written procedures, or other appropriate guidance.

Bodies of knowledge exist in each of these three areas. However, the depth of knowledge varies, and this will determine how specifically strategies and procedures can be written. For example, if considerable uncertainty exists as to the containment loads and performance during core melt accidents, only broad guidelines may be practical. More specifically, if the rate of pressure buildup is unpredictable, then venting via a specific vent line may not provide assurance of mitigation, and developing very detailed procedures for its use may be inefficient and ineffective.

2.2 Current NRC Requirements

The NRC identified three phases of development for function-based procedures in its TMI Action Plan (NUREG-0660), which was subsequently enlarged upon in the Clarification of the TMI Action Plan (NUREG-0737). The first phase was the development of interim procedures for small-break LOCAs and inadequate core cooling such as occurred at TMI-2; the second phase was the development of generic emergency procedure guidelines. The final phase was the implementation of the emergency procedure guidelines on a plant-specific basis. At present, few plants have implemented their revised emergency procedures, with most of the others committed to implementation by the end of 1985.

The NRC review to date has emphasized the emergency procedure guidelines, together with writers' guides that describe how to produce plant-specific procedures. The results of these reviews have been reported in Safety Evaluation Reports (SERs). NRC intends not to review the procedures for all individual plants, but rather to rely on the proper application of the guidelines by licensees. The viability of this approach will be assessed eventually by NRC site audits at a limited number of plants.

The requirements of the NRC cover three areas: technical basis, human-factors aspects, and the process by which the procedures will be implemented. The documents that describe these requirements are NUREGs-0578[7], -0660, -0737 (plus its supplement Generic Letter 82-33[8]), and -0899. The following are the requirements and the specific documents containing them.

1. Transient Analyses as Basis

   Licensees are required to perform realistic plant-transient analyses to identify important operator safety considerations, such as the prevention of core uncovery and the prevention of more serious accidents (NUREGs-0578, -0660, and -0737). These analyses are to be performed for a range of initiating events, including those presented in the final safety analysis report (FSAR) and for natural phenomena such as earthquakes and floods; multiple equipment failures and operator errors should be considered in the analyses (NUREGs-0578 and -0737). The analyses are to include the times up to and beyond the onset of inadequate core cooling and continue into the event to ensure that all relevant phenomena are identified, including those associated with failures of long-term decay heat removal (NUREGs-0578 and -0737).

2. Human-Factors Aspects

   The procedures are to be function-oriented and not to require operators to diagnose specific events (Supplement 1 of

NUREG-0737, and NUREG-0899). Operators should be directed to monitor safety functions continuously (NUREG-0899). The specific operator actions are to be identified by means of task analyses, which will help in confirming the adequacy of information displays (Supplement 1 of NUREG-0737, NUREG-0899). NUREG-0899 also contains many guidelines for style, layout and other similar features of procedures documents.

3. Development Process

The steps required in developing the revised emergency procedures are the analysis of transients, preparation of generic emergency procedure guidelines, conversion of guidelines into plant-specific procedures, and the retraining of operators (NUREG-0578). Specifically, the generic guidelines are to be translated into emergency procedures using a writers' guide (NUREG-0899). The generic guidelines require validation by the owners' groups (the groups responsible for their creation), and are to be reviewed by the NRC. The resulting procedures must undergo a process of validation and verification (NUREG-0899). The process used to develop the guidelines is to be documented in sufficient detail to show the flow of information from its technical base to its use in the guidelines; this information is to be available for the translation of the generic guidelines into plant emergency procedures (NUREG-0899).

2.3 Relevance of NRC Requirements to Severe Accident Procedures

Many of the current NRC requirements can be considered directly relevant to procedures for mitigative actions. One principal reconsideration is in the definition of the analyses to be performed since the function-based procedures are primarily focused on preventing or delaying core damage.

Substantial additional analyses of plant and operator performance during severe accidents would be needed to indicate the opportunities for and effects of intervention by the operating crew. These analyses, for a representative range of severe accident sequences, would concentrate on the behavior of damaged cores, containment performance, and source terms.

Also it is not obvious at this point that severe accident response strategies need be or should be translated into detailed written procedures directed at the plant's operators. Rather, as we will discuss in Section 6, it is possible that the severe accident procedures should be written for a different group of people, e.g., the shift safety engineer, technical advisor, or engineering support staff, though some preplanned guidance to operators is thought to be helpful.

Other criteria, such as the need for verification, validation, and documentation, are appropriate. The use of realistic calculations is also appropriate insofar as we have confidence in their validity, a matter of considerable concern.

3. SEVERE ACCIDENT PHENOMENOLOGY

A starting element in developing severe accident strategies is the gathering of phenomenological information associated with severe accident progression (as discussed in Section 2.1). This knowledge has advanced significantly over the past five years but is still subject to large uncertainties. With this in mind, the technical feasibility of developing severe accident management strategies was evaluated in terms of whether our understanding of phenomena and our confidence in existing analytical tools are sufficient to develop such strategies with confidence.

The purpose of this section is to summarize the current state-of-the-art in knowledge of severe accident phenomena from the perspective of how confidently we can predict the effects of mitigative actions on core-melt accident consequences. The source of knowledge used in this report is primarily the NRC-sponsored Severe Accident Research Program (SARP). Insights and alternate views published by the Industry Degraded Core Rulemaking (IDCOR) program are also noted. The emphasis in these discussions is on those aspects of severe accident phenomena that are directly relevant to accident management, where the bottom line is how this information helps in developing effective strategies which minimize the consequences of a degraded-core condition. Reference 1 provides a more detailed discussion of this topic.

Though residual uncertainty will always exist in our understanding of severe accident phenomena, this uncertainty need not necessarily preclude the development of effective accident management strategies. This is because the process of developing strategies to mitigate the consequences of a severe accident does not require the same degree of detail for resolution as is required for a "best-estimate" calculation of, for example, a fission-product source term. The uncertainties may be incorporated in the overall strategy by considering the <u>probability</u> that a particular action will result in a <u>desired</u> plant state. This philosophy underlies the methodology developed in this study and is described in more detail in Section 4.

A cornerstone of this approach is a quantitative estimate of the uncertainties in the relevant phenomena. Phenomena believed to be relevant to evaluating severe accident management strategies are discussed below. The discussions are presented by topic: degraded-core behavior and containment loads; containment performance; and fission product behavior. Unresolved issues of significance are described in terms of how they affect our confidence in predicting the systems behavior under degraded-core conditions or how operator actions alter the accident sequence.

3.1 Degraded-Core Behavior and Containment Loads

The primary objective of accident management in the area of degraded-core behavior (both in- and ex-vessel) is regaining the capability of cooling the fuel without imposing an unacceptable or immediate threat to containment integrity. Whether or not a degraded core is coolable depends upon many factors (geometric configuration, composition, mass, and temperature distribution of the material). Current methods (such as MARCH 2 as used in the recent evaluation of source terms reported in BMI-2104(9)) apply a lumped parameter modeling approach, which determines integral effects, masking many details. More sophisticated, mechanistic models (for example, SCDAP and MELPROG) are under development. Both types of codes have a very limited data base for validation so for all practical purposes their credibility is comparable.

Some pragmatic conclusions may be drawn, however, from existing experimental data and from experience with a wide variety of modeling assumptions. If the core has not slumped into the lower head of the reactor vessel, the likelihood of arresting the melt progression and cooling the core is reasonably high, provided that sufficient coolant injection is available. Witness the TMI-2 core, which was recovered from what was clearly an advanced stage of degradation. The function-based emergency procedure guidelines facilitate the operating crew's ability to recognize this condition and to bring all available sources of water to bear on restoring core cooling.

If the core has slumped into the lower head, introducing water is unlikely to cool the molten mass, and melt-through is probably inevitable. Also, the time window for the successful introduction of coolant at this stage would seem to be very narrow.

If the lower head fails, the coolability of the molten mass and debris following ejection from the vessel depends upon the energetics of the ejection and the plant-specific containment floor configuration. With the reactor vessel pressurized (as in a TMLB' sequence in a PWR), an energetic melt ejection is likely, which would disperse the debris across the containment floor. The debris should be coolable in this configuration, provided water can be supplied to the various containment floor elevations (cavity, sump, and floor). The possibility of debris dispersal, however, raises the likelihood of two otherwise low-probability containment failure modes: (i) direct heating of the containment atmosphere, and (ii) for a BWR with Mark I containment, melt-through of the steel wall of the drywell at the floor elevation.

If the reactor vessel is not under high pressure when the lower head fails, the molten debris is likely to be retained within the reactor cavity. In this confined configuration, similar to that in the lower head, the ability to cool the debris is less likely. Other factors must

also be considered, such as whether the cavity was dry or filled with water when the debris dropped, and the plant-specific geometry of the cavity.

Given the location of the core debris at the time when coolant becomes available for flooding the core, it is possible to specify the minimum requirements to restore coolability. Whether or not the core would actually be cooled depends upon fuel-coolant interaction phenomena that are uncertain.

Although a definitive means is not available to determine whether a coolable debris-bed configuration will be formed, it is possible to identify alternative phenomenological scenarios and to evaluate whether a proposed accident management strategy is affected by the phenomenological uncertainty. If an accident management strategy could be detrimental under some phenomenological assumptions or if one wants to determine the optimal strategy, subjective assignment of probabilities can be made for the different phenomenological scenarios. The assignment of such subjective probabilities is commonly performed in probabilistic risk assessments, for example.

The details of core-melt progression can strongly influence the timing and magnitude of loads on the containment by providing the driving force or material source for interactions. Three specific examples are core/coolant interactions (steam explosions), the production of aerosols and non-condensable gases by core/concrete interaction, and hydrogen generation by Zircaloy oxidation. There are some potentially important differences of opinion within the technical community in these areas.

While most experts agree that small-scale steam explosions are possible, IDCOR maintains that high-energy, containment-threatening explosions are impossible. The current NRC position is that they are unlikely but have not been shown to be impossible. These positions are philosophically quite different. In practice they could affect accident response decisions only if one believed there is a significant probability of attaining the requisite conditions needed to produce containment-threatening steam explosions.

The mass of hydrogen generated during meltdown predicted by the IDCOR models in MAAP has differed from that predicted by MARCH by as much as a factor of four. This difference can have a major impact on the probability of early containment failure by overpressure or hydrogen combustion for specific sequences in particular plant designs. Uncertainty also exists in the rate of containment pressurization that arises from gas production during core-concrete ablation. Although the timing of containment failure can be affected by this uncertainty, the phenomenon occurs rather late in many sequences, by which time accident consequences are relatively insensitive to this uncertainty. Overall, there is a growing consensus among the NRC contractors comprising the Containment Loads Working Group on the conditions that can lead to containment failure for different plant designs.

One of the areas of containment loads that has received a great deal of attention is hydrogen behavior and control. Research has significantly reduced the uncertainties in understanding concentration limits for combustion, the pressure loads due to burns, and equipment survivability. Uncertainties remain, however, in combustion control. Although devices to ignite hydrogen deliberately have been installed in ice-condenser (PWR) and Mark III (BWR) plants, it is not clear how well they would work in certain environments (e.g., with containment sprays or fans operating); they do not work at all under station-blackout conditions. Since hydrogen combustion can be a significant contributor to early containment failure in Mark III and ice condenser containment designs, resolving questions pertaining to the hydrogen generation rates, combustion limits and resulting containment loads is essential for assessing accident management strategies.

The importance of these phenomena to accident response should not be underestimated. Each creates the potential for added risk from some operator actions under degraded core conditions that might otherwise not exist. For example, restoring coolant injection to a degraded core introduces large quantities of steam which can feed the Zircaloy-oxidation process and can add to containment pressure loads. In determining the best accident response strategy, the potential value of arresting core degradation must, therefore, be weighed against the probability of early containment failure.

3.2 Containment Performance

Whether, and how, the containment will fail is a function of both the loads produced in an accident and the performance of the containment. Over the past few years our improved understanding tends to suggest decreasing estimated loads, increasing estimated strengths of containments, and greater confidence in our ability to predict the conditions under which failure would occur.

In WASH-1400[10] there were six principal mechanisms of containment failure postulated for a PWR. These were

- by an energetic missile generated by an in-vessel steam explosion (α)
- inadequate isolation of openings and penetrations (β)
- hydrogen burning (γ)
- overpressurization by steam and noncondensable gases (δ)
- melt-through of the containment floor (ε)
- containment bypass through a low-pressure interfacing system LOCA (V).

For the BWR studied in WASH-1400, several variations on these basic mechanisms were postulated. WASH-1400 quantified the probabilities of these failure mechanisms given the progression of system failures leading to core melt.

Research since then has helped improve our ability to understand and analyze containment performance during severe accidents. Several additional containment failure mechanisms have been identified and the importance of containment performance on radiological source terms has been reemphasized.

The NRC has established the Containment Performance Working Group to evaluate the most likely modes of failure for the reference plants being analyzed in the Source Term Reassessment Study (NUREG-0956[11]). Although this Working Group has concluded that enhanced leakage of containments, particularly at gasketed penetrations, is more likely than an energetic structural failure, the data needed to support a leakage model are sparse. Experimental programs now underway should enable these current uncertainties to be reduced over the next several years.

3.3 Fission-Product Behavior

In recent years substantial knowledge has been gained about fission product release rates, chemical and physical forms, retention mechanisms, and the like. Although an objective of accident management is to reduce the risks from severe accidents, knowledge of the specific mechanisms dominating fission-product behavior during an accident is of little direct value to the operating crew during the accident. It is known that sprays, suppression pools, and ice condensers are important decontamination devices. If they function in an accident, they will have a major effect on accident consequences. Otherwise time, which allows natural processes to reduce airborne radioactivity, is the operators' best friend. In developing accident management strategies, decontamination factors need be known only with enough certainty to evaluate whether the improvement in the fission-product source term is worth any potential increased risk from the action taken.

The predominance of recent evidence suggests there is adequate justification for lowering estimates of radioactivity released during severe accidents. Substantial disagreements exist over the degree to which these estimates should be lowered for specific designs and sequences. Also there are some phenomena (e.g., revaporization of fission products deposited on surfaces), the understanding of which is still considered by some[12] to be inadequate to support general reductions of source terms.

The current degree of confidence in the analytical models applied to determine radionuclide release under specific LWR accident conditions has been estimated quantitatively in the QUEST program (Quantification of Uncertainties in Estimating Source Terms) at Sandia National Laboratories. The uncertainties in fundamental phenomena can be quite

large; for example, present estimates of uncertainty in the suspended aerosol activity in the containment are on the order of 100*. Such large uncertainties may not be important, however, if the absolute value of the source term is small. Several experimental programs are underway to reduce the magnitude of the uncertainties by providing data to validate existing models or form a base for developing new ones.

3.4 Summary

To summarize on the matter of severe accident phenomenology, our understanding of these phenomena has improved markedly in recent years but important uncertainties persist. These uncertainties must be accounted for, but they need not preclude the development of successful accident management strategies.

The principal uncertainties affecting severe accident management appear to be related to those phenomena exhibiting a threshold type behavior (e.g., hydrogen burn or detonation, steam explosion, containment loss of isolation, molten core dispersal) which can yield quantum changes in the direction a sequence takes. In those cases where important phenomena are uncertain, alternative phenomenological scenarios can be treated probabilistically. Given that we can define those directions, our recently improved understanding of core degradation, containment performance, and fission product behavior allows us to predict the subsequent course of events with relative confidence.

Controlling core degradation and the associated containment loads is of utmost importance. If it becomes impossible or impractical to do so, then the operating crew's accident response resources should still allow them to manage containment performance and perhaps fission product transport effectively provided the knowledge, information, and equipment needed to do so is available.

The key then to developing successful accident management strategies is to develop strategies which, if implemented, are "forgiving" toward those threshold type phenomena or alternatively, to accept the residual risk presented by those phenomena. Note also that we must still translate these strategies into forms that the operating crew can apply in real time, as discussed in Section 2.1.

A method of developing mitigation strategies for severe accident conditions that uses this approach is described in the next section.

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* For the sequences studies in the Surry PWR involving core melt caused by failure to maintain core cooling (e.g., TMLB' using the terminology of the Reactor Safety Study, WASH-1400).

4. A METHOD FOR DEVELOPING SEVERE ACCIDENT MANAGEMENT STRATEGIES

The principal goal for any severe accident management strategy is to minimize the radiological consequences of degraded-core conditions. Attaining this goal requires thought quite different from that involved with traditional design-basis analyses; that is, it must be assumed that an extremely low probability series of events has occurred, which have crippled at least some of the engineered safety features, resulting in total loss of one or more safety functions. The operating crew may be unable to return the plant readily to a safe, stable state. The challenge then becomes: if core melt is inevitable, how can the consequences be minimized?

A pragmatic approach adopted here is to ask the more tractable question: "For an accident sequence leading to core melt, what actions can a reactor operator take to reduce the consequences (i.e., fission-product source term) to the environment?" In most cases this implies actions that will prevent early containment failure. As a corollary we would like to know if there are relatively modest changes in procedures, training, instrumentation, or control which can yield significant improvements in the expected outcome of a severe accident.

A method of developing severe accident management strategies that takes this approach is outlined below. This method has been applied to two reference plants to demonstrate its feasibility. The results of this trial application are described in Sections 4.2 and 4.3.

4.1 Description of the Method

A core-melt challenges the final barrier to release of radioactivity to the environment, the containment structure. The nature of an environmental release depends upon the mode and timing of the containment breach, both of which may be influenced by operator actions.

The uncertainties associated with core-melt progression, containment loads and performance, and equipment survivability propagate uncertainty in the mode and time of containment failure for any accident sequence. Left unmitigated, an accident sequence may result in any one of a set of likely failure modes whose relative probabilities may be estimated. These probabilities are based on the confidence in predictions of the phenomena governing the progression of events and the containment failure criteria.*

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* Such an assessment is currently underway at Sandia National Laboratories in the Severe Accident Risk Rebaselining and Risk Reduction Program (SARRP).

With operator intervention, the relative probability of a more favorable containment failure mode (one resulting in a lower consequence) may be increased over less favorable modes. In essence, the initial accident sequence, with its associated set of probable containment failure modes, becomes a different sequence, with a different distribution of probable failure modes, upon an operator action. This principle underlies our method for developing accident management strategies.

An illustration may best describe how this method works. For a given nuclear plant, a containment performance tree may be constructed; this tree identifies the various modes and times of containment failure caused by all known potential mechanisms. The endpoint of each branch represents some containment failure mode and, for the accident sequence being analyzed, an associated level of consequence (fission-product source term). The basic structure of the tree, however, is not a function of the accident sequence being analyzed. Only the probabilities of each branch and the associated level of consequence will vary with the accident sequence and operator actions being considered.

The source term associated with each failure mode need not be known quantitatively but can be estimated as being high, medium, or low, based on our phenomenological understanding. The relative probabilities and consequences associated with each failure mode are then labeled adjacent to the corresponding branch on the containment performance tree.

An example is given in Figure 4.1, which depicts a BWR/MARK-I containment performance tree for sequences initiated by a large LOCA with subsequent failure to cool the core (sequence AE in the nomenclature of the Reactor Safety Study). In this example, containment venting is examined. Prior to implementation of the revised emergency procedure guidelines, operators would not have been expected to vent the containment in the AE sequence. In this case the most likely modes of containment failure would involve late leakage (possibly due to high temperature in the drywell) or late gross shell failure caused by gradual overpressurization. In either case the release of fission products would be at least partially scrubbed by the suppression pool, and the consequences would be characterized as medium. An important possible outcome, however, involves overpressure failure occurring during the core meltdown period (a case analyzed in BMI-2104). Although this is not believed to be a likely outcome, the potential consequences could be high. If, however, as recommended in the revised emergency procedure guidelines, the operators are instructed to vent the containment atmosphere through the wetwell ventilation lines prior to containment failure, the pattern of containment failure modes would be altered, and the overall accident consequences reduced. The potential results of this action are shown in Figure 4.1 by arrows identifying the column of initially likely failure modes and associated consequences to the new probable failure modes and consequences. By weighing the potential improvement in the source term against the probability that an

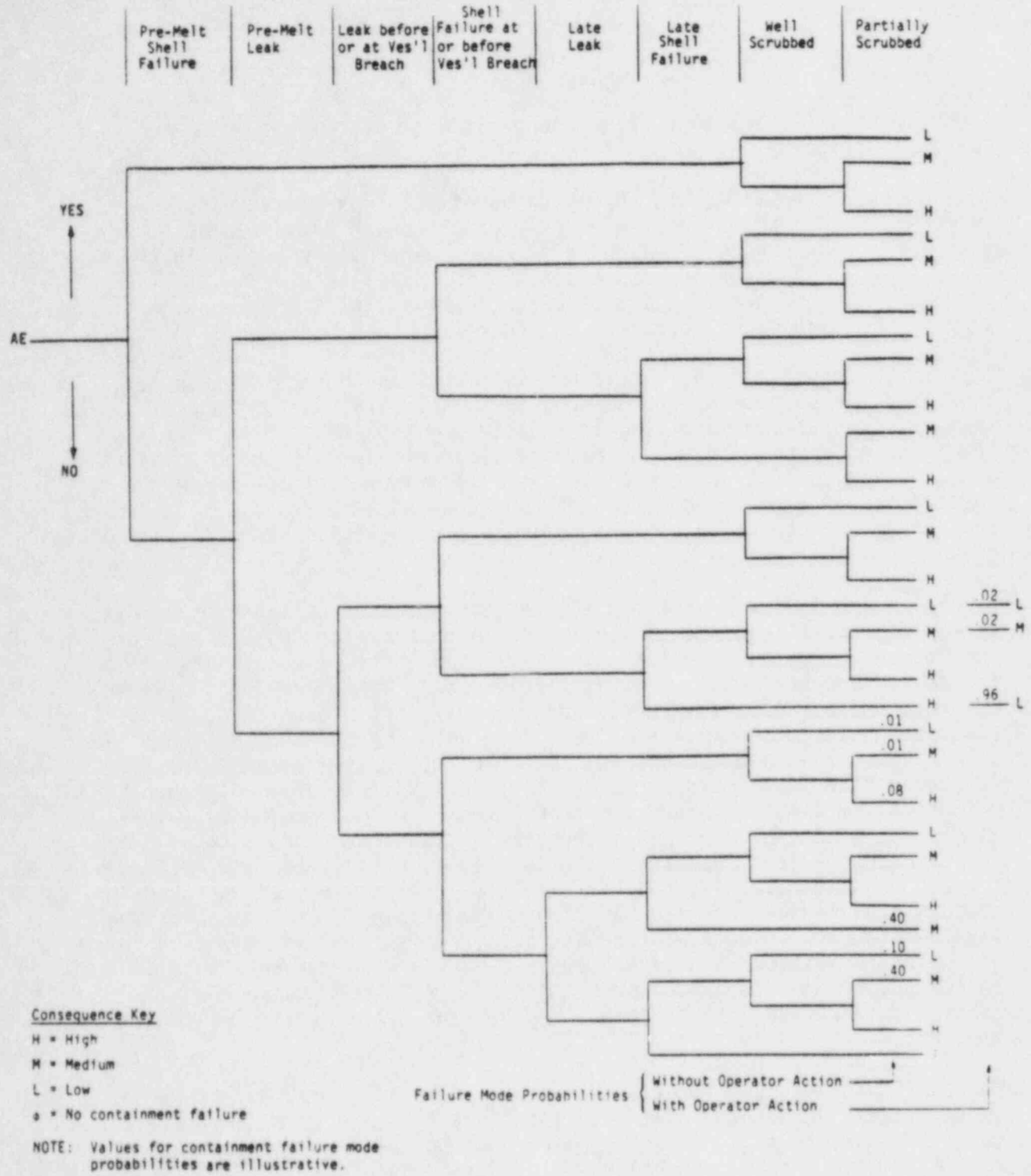

Figure 4.1. Containment performance tree for a BWR/Mark I plant.

The tree is structured to delineate the mode and time of containment failure (branch points). Consequences for each branch are labeled in the right margin.

An operator action of venting the containment from the wetwell, prior to containment failure, is depicted. It is assumed that containment venting is initiated prior to vessel breach, but not necessarily prior to core damage. Venting the containment is considered a "leak".

improvement will take place, the change in risk associated with an operator action may be judged.

In the case of venting, the potential benefit is associated with switching from a branch of the event tree with high consequences to one with lower consequences. Although the high consequence branch in the event tree without venting has a small probability (estimated as 8 percent), concern about this type of accident could be great enough to warrant the associated change in procedures.

It must be recognized that it is possible that an operator action may aggravate the situation, depending upon the specific circumstances. Instead of an action resulting in a desired plant state which leads to lower consequences, the action could yield a higher consequence failure mode. Examples of this were cited in the previous section--for example, restoration of coolant injection may cause enhanced Zircaloy oxidation or a steam explosion. Estimates of these probabilities are essential in evaluating alternatives for action.

By performing this exercise for alternative operator actions at various times in a given sequence, a list of preferable mitigative actions for that sequence may be compiled. The next step is to repeat the exercise for other sequences which present significantly different challenges to the reactor/containment system. For each case the plant state and its observable symptoms* should be identified when the operator action is taken. When the sequence-specific lists of mitigative actions are completed, the final step is to develop generic strategies which would effectively mitigate a broad array of severe accident sequences. This is done by comparing preferable operator actions in response to similar plant states derived from the different accident sequences and noting those common to several sequences. In some cases, the actions taken in response to one sequence may be illogical during another because of the way the sequence is defined (for instance, actions to initiate some form of coolant injection when the sequence is defined by complete loss of injection). These anomalies should not eliminate the action from consideration since lost functions may be regained at some time in the accident.

The final product is a function-based set of actions that may be taken during severe accident conditions, independent of the sequence itself, to reduce the off-site release of hazardous material. These actions may not be as well defined as actions in existing emergency operating procedures, but may be presented on an "engineering level"; that is,

* State variables may include, but are not limited to, whether or not core melting has occurred and, if so, the location of the molten material (e.g., within the core region, vessel lower head, or on containment floor), the containment thermodynamic conditions, and atmosphere composition. The associated symptoms include temperatures, pressures, flows, and radiation levels.

specific components or even systems may not be identified. The reasons for this are discussed further in Section 6.

For a given plant state, several possible operator actions may exist that, collectively, do not openly suggest an overall strategy. The above methodology, however, provides some supporting data to develop a mitigation strategy. Each action has an estimated measure of improvement (e.g., decreased source term) and a probability of success. The optimal strategy employs those actions that produce the greatest reduction in source terms with the highest probability of success. Because of the large uncertainties and degree of subjective judgment associated with the assignment of branching probabilities and source terms, it is essential that sensitivity studies be performed to test the robustness of the preferred strategies. In particular, the possibility of synergistic effects of multiple operator actions must be considered before judging the merit of the strategy.

Developing a coherent and technically sound severe accident management strategy by the method outlined above requires a great deal of information, which in some cases is not yet generated. The method requires estimates for the probabilities of potential containment failure modes for the set of sequences to be analyzed, including those with operator intervention. A technique for estimating these values has been developed (the Sandia study in conjunction with SARRP). However, it is based upon a set of source-term calculations (i.e., BMI-2104) in which phenomena associated with operator actions were not considered. This technique would, therefore, need to be expanded to include these effects. This expansion would require quantifying the uncertainties associated with such phenomena as degraded core coolability, hydrogen generation during core quenching, and fission product retention (that are described in Section 3). This task requires much subjective judgment. However, analyses may be performed to determine the range of potential plant states likely to result from an operator action. Combined with an estimate of the probability of certain phenomenological scenarios, the confidence in achieving a desired plant state can be estimated.

Order of magnitude estimates of fission product source terms are also necessary for sequences following operator interventions, to determine the beneficial reduction in consequences resulting from operator actions. In principle, these estimates are necessary for each of the probable failure modes.

Applications of this method to two types of nuclear plant designs are described in the following sections. The applications are intended only to illustrate the utility of modifying containment performance trees to evaluate the effectiveness of and the risk associated with a single operator action for a specified initiating sequence. The containment performance trees presented are one of many possible ways of delineating the mechanisms that can significantly influence the consequences associated with various containment failure modes. A more detailed

study of this type may identify criteria quite different from the examples shown here.

The applications identify probable containment failure modes before and after an operator action. These identified modes were not determined by combining the probabilities associated with each branch point and, in fact, are not given quantitative probabilities. For the purpose of this illustration, the probable failure modes and associated consequences are based entirely upon subjective judgment.

4.2 Application of Method to Browns Ferry

Browns Ferry Unit One (BF-1), is a BWR-4 reactor system with a MARK-I containment. The containment performance tree was constructed to delineate the potential containment failure modes for any severe-accident sequence one might wish to consider. This is the basis for Figure 4.1. Branch points are based upon the time and size of failure of the containment boundary. The significant effects of the suppression pool or containment sprays on fission-product retention were considered by including branch points that distinguish how well the release is scrubbed. It was assumed that a containment leak (as opposed to gross shell failure) may or may not preclude gross shell failure and could be, at best, only partially scrubbed (by sprays). Containment failure modes are then defined at the end-points of the final branch.

The sequences considered for BF-1 were a transient with loss of decay heat removal (TW*), transient with loss of injection (TQUV*), and a large break LOCA with failure of coolant injection (AE*). These sequences were selected because SASA analyses or source term calculations existed to provide some analytical support for assessing plant response and consequences for (at least) some of the probable failure modes. In particular, the Oak Ridge SASA analyses[13-15] included a great deal of useful information on plant configuration, system operability limits, system response prior to core damage, and suggested actions to prevent or delay core melt. The results of BMI-2104 source-term calculations for the Peach Bottom plant[16] (also a BWR-4 with a Mark I containment) were used as a basis for correlating containment failure mode with consequences. The sequences were also selected to represent a broad range in accident progression, containment loads, available safety equipment and probable containment failure modes.

For each sequence, the expected results of specific operator actions are illustrated on the BF-1 containment performance tree in the same fashion as shown in Figure 4.1. The actions taken to mitigate a TW sequence are illustrated (beginning on page 4-8) in Figures 4.2 and 4.3, TQUV in 4.4 and 4.5, and AE in 4.1, 4.6, and 4.7. Each figure is self-contained in

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* Sequence identifiers are expressed in the Reactor Safety Study nomenclature.

that the state of the plant at the time the action was taken, the action itself, the anticipated plant response to the action, and issues requiring further analysis for resolution are summarized below the containment event tree. These descriptions are not based on specific calculations to determine the plant response under alternative phenomenological scenarios, as would be done for a rigorous analysis. Instead, the likely results of the action are based upon our current understanding of the relevant phenomena.

Summary of Results for Brown's Ferry

In some cases, we have evaluated actions that are clearly beyond the scope of current emergency operating procedures and the design intent of systems. In all cases, however, actions were considered if they would be physically possible with only minor hardware modifications (e.g., defeating interlocks, and racking out breakers). Evaluations of actions were considered over a wide range of plant states from before core-damage to after containment-failure with release. Many of the actions illustrated may actually be taken at other times in the accident than we have described. The effects of an operator taking a given action at different times in the accident must be studied parametrically and has not been considered here.

Some general observations may be made from these cursory studies, however. It appears that actions may be taken even after significant core damage that can either arrest the accident or reduce the consequences of it. This conclusion must be backed by further detailed analysis, but it is significant since the risk associated with severe accidents can be reduced if effective accident management strategies can be developed. Although some of the operator actions illustrated relied upon regaining lost safety functions at some time into the sequence, they should not be dismissed as unreliable for accident management. Actual accident scenarios will in all likelihood entail only partial, intermittent or temporary loss of function. A comprehensive accident management strategy should be capable of dealing with any combination of these.

Some operator actions were found to be very effective in one sequence but not in others under similar conditions, as illustrated in Figure 4.7. In this case containment sprays are initiated to reduce the pressure load on the drywell. The efficacy of containment sprays to reduce containment pressure depends upon the partial pressure of steam in the containment atmosphere. During a sequence such as AE, in which the drywell loads are primarily due to the buildup of steam (non-condensables being transported to the torus), sprays can be effective in reducing the drywell pressure. In other sequences, the drywell loads may be due to the accumulation of non-condensable gases or the spray temperature may not differ significantly from the steam temperature. In either case, initiating sprays may have an insignificant effect on pressure. (The summary continues on page 4-20.)

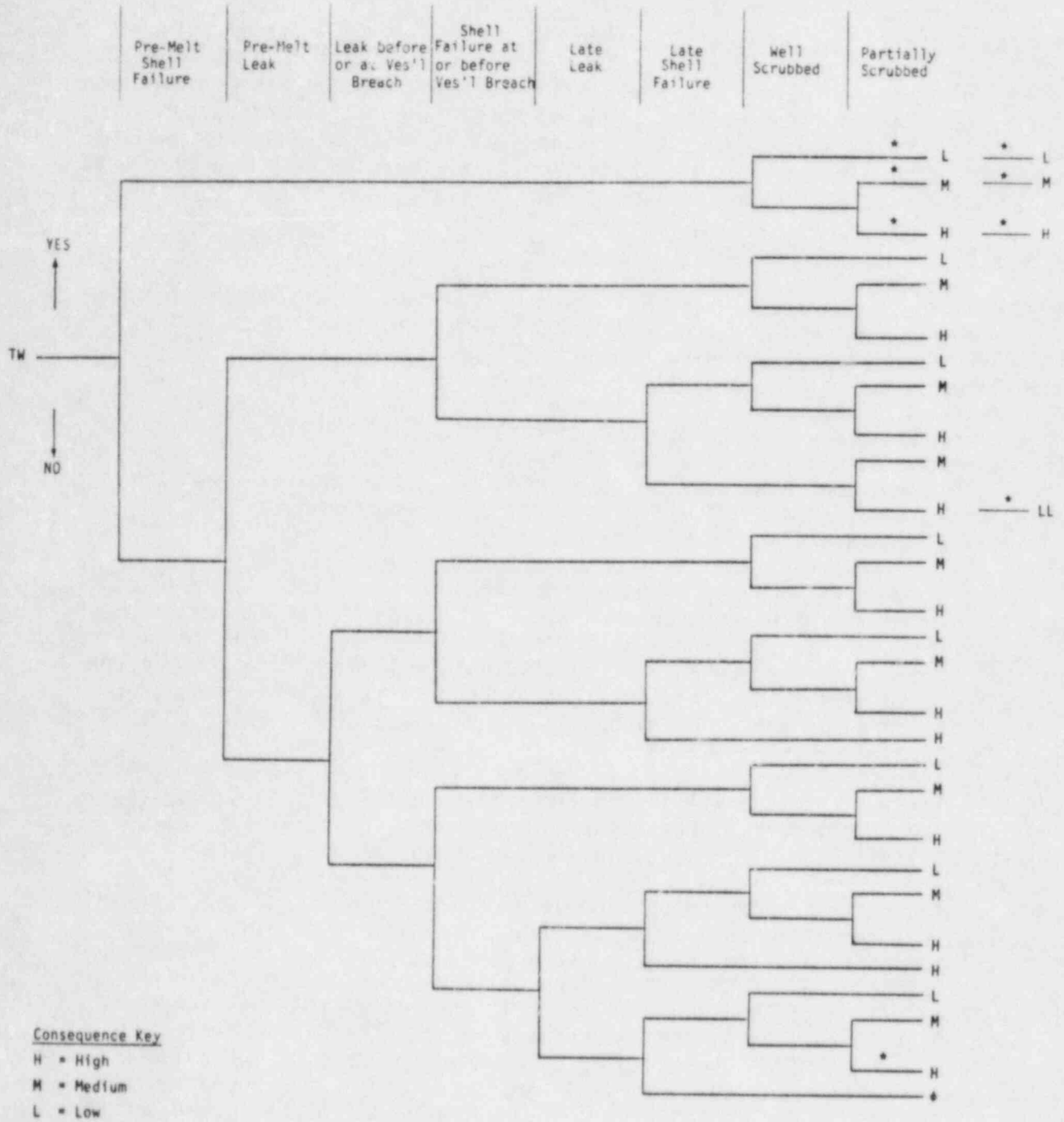

Figure 4.2. Operator action to mitigate Browns Ferry sequence TW (example 1).

Notes for Figure 4.2.

PLANT STATE: Vessel inventory and makeup supply normal. Reactor shutdown. Containment pressure and temperature high and rising.

ACTION TAKEN: Feed and bleed torus water. Pump hot torus water to condenser wetwell and replace with cold condensate water.

RESULTS OF ACTION: The intent is to stop or reduce the pressurization rate of containment by extending the period that the suppression pool acts as an effective heat sink. Minimizing the heatup of the suppression pool also reduces the probability of causing the failure of ECC systems due to high temperature conditions, in particular HPCI and RCIC. The change in failure mode probabilities is characterized by a reduced probability of pre-melt overpressure at the expense of an increased probability (and decrease in consequence) of a pre-melt leak (the feed and bleed operation itself).

If the feed and bleed operation is completely successful, coolant injection failure and, hence, core degradation is averted. This action, therefore, also has the effect of reducing the probability of the TW sequence leading to core damage.

UNCERTAINTIES: This action only applies to a TW sequence that does not preclude operation of the main condenser, such as an MSIV closure. The flow rates of coolant required to maintain a cool suppression pool need to be evaluated.

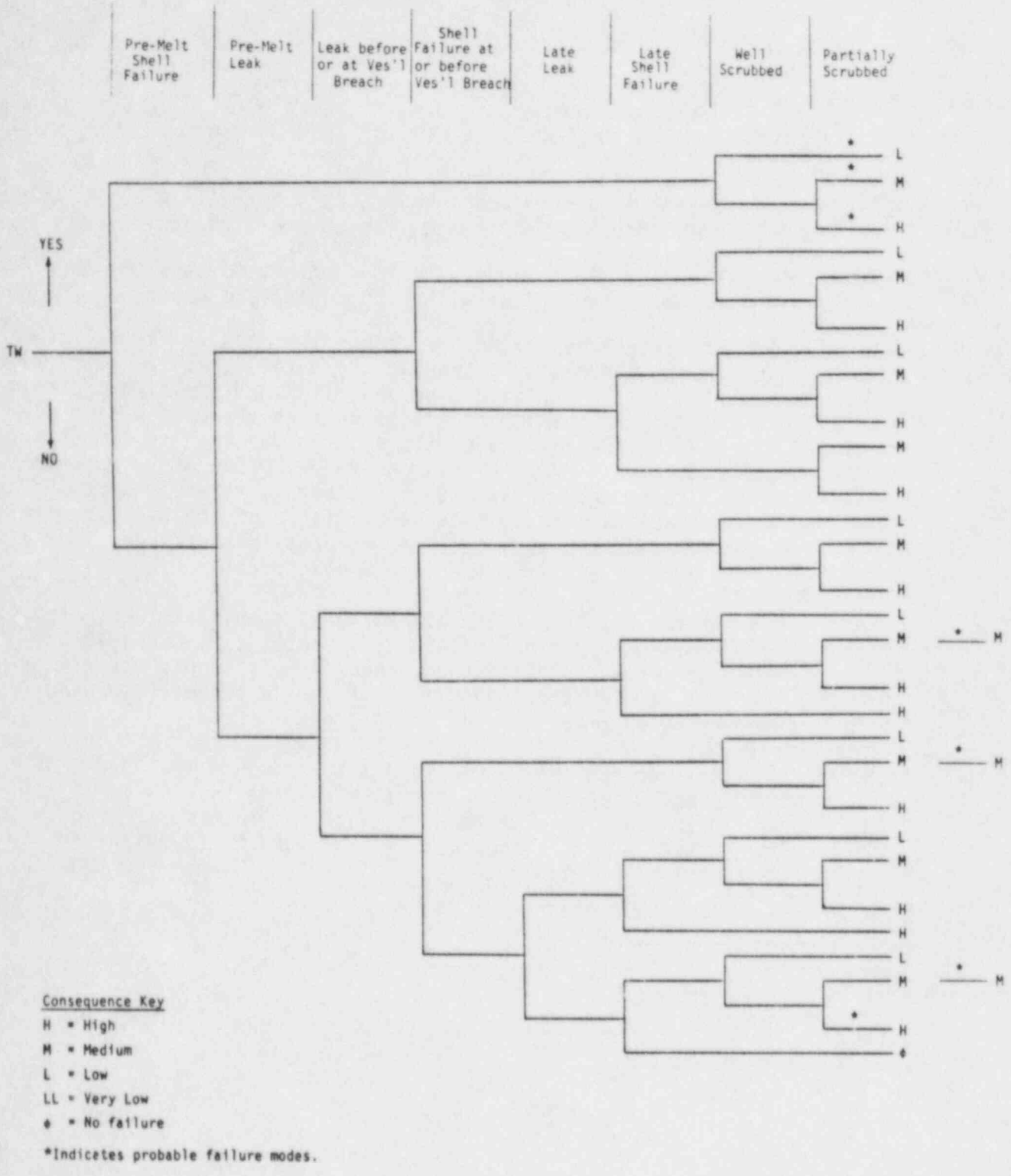

Figure 4.3. Operator action to mitigate Browns Ferry sequence TW (example 2).

Notes for Figure 4.3.

PLANT STATE: Due to high environmental temperatures, HPCI and RCIC no longer operate. Containment pressure is high enough to cause loss of SRV control air function, preventing the operator from maintaining a depressurized vessel. With no coolant injection, reactor coolant system boils off through the SRVs to the suppression pool.

ACTION TAKEN: Use low-pressure injection systems to flood drywell floor.

RESULTS OF ACTION: If injection is through the torus, the addition of cool water improves vapor suppression. Flooding the drywell floor is in anticipation of core meltdown. Core debris may be cooled following vessel melt-through and pool provides some fission product attenuation.

UNCERTAINTIES: The containment pressure response associated with the reduction in free volume, as the torus and lower drywell fill with water, is uncertain. There may also be hydrodynamic instability problems in the torus as it fills with water and drywell gases continue to enter through the vent pipes.

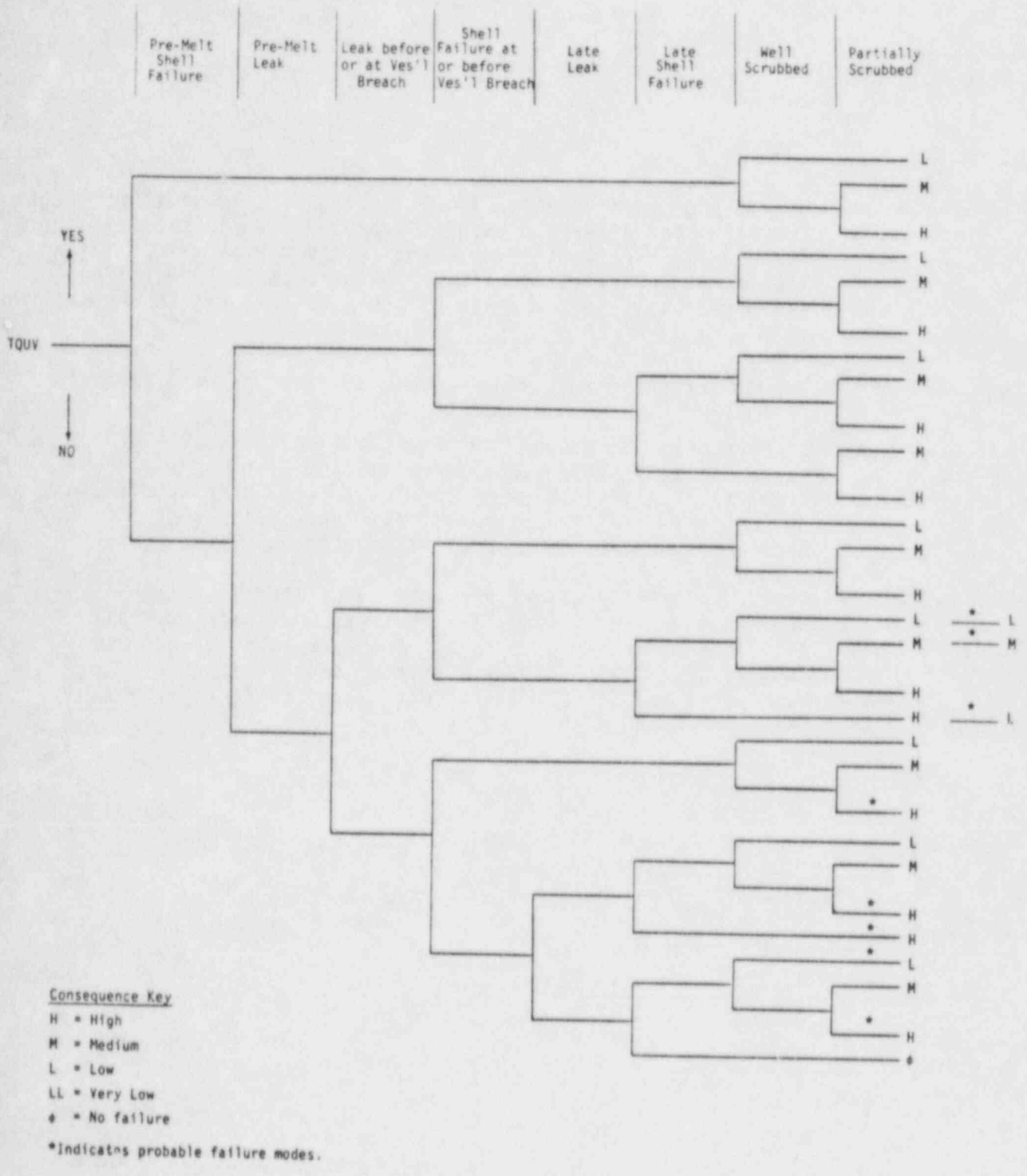

Figure 4.4. Operator action to mitigate Browns Ferry sequence TQUV (example 1).

Notes for Figure 4.4.

PLANT STATE: Vessel at high pressure (with no low pressure ECC operable, the reactor vessel was not depressurized), and relieving to suppression pool. Significant core damage. Containment is intact but pressure is high enough to render SRV control air system inoperable, precluding vessel depressurization.

ACTION TAKEN: Containment is vented from wetwell atmosphere. Vented gases are passed through off-gas filter systems.

RESULTS OF ACTION: The intent is to allow a scrubbed, filtered, and elevated release of low consequence to prevent an overpressure failure of the drywell which would cause an unscrubbed, unfiltered, and high consequence release. Even if the standby gas treatment system subsequently fails, the fission product attenuation afforded by the suppression pool and ventilation system is significant.

UNCERTAINTIES: The principal uncertainties lie in the flow capacity of the wetwell ventilation systems at elevated pressures in Browns Ferry.

NOTE: Discussions with TVA and ORNL personnel indicate that containment venting procedures are included in the new Browns Ferry emergency procedures based upon the General Electric EPGs. The criteria for their execution were not known for this study.

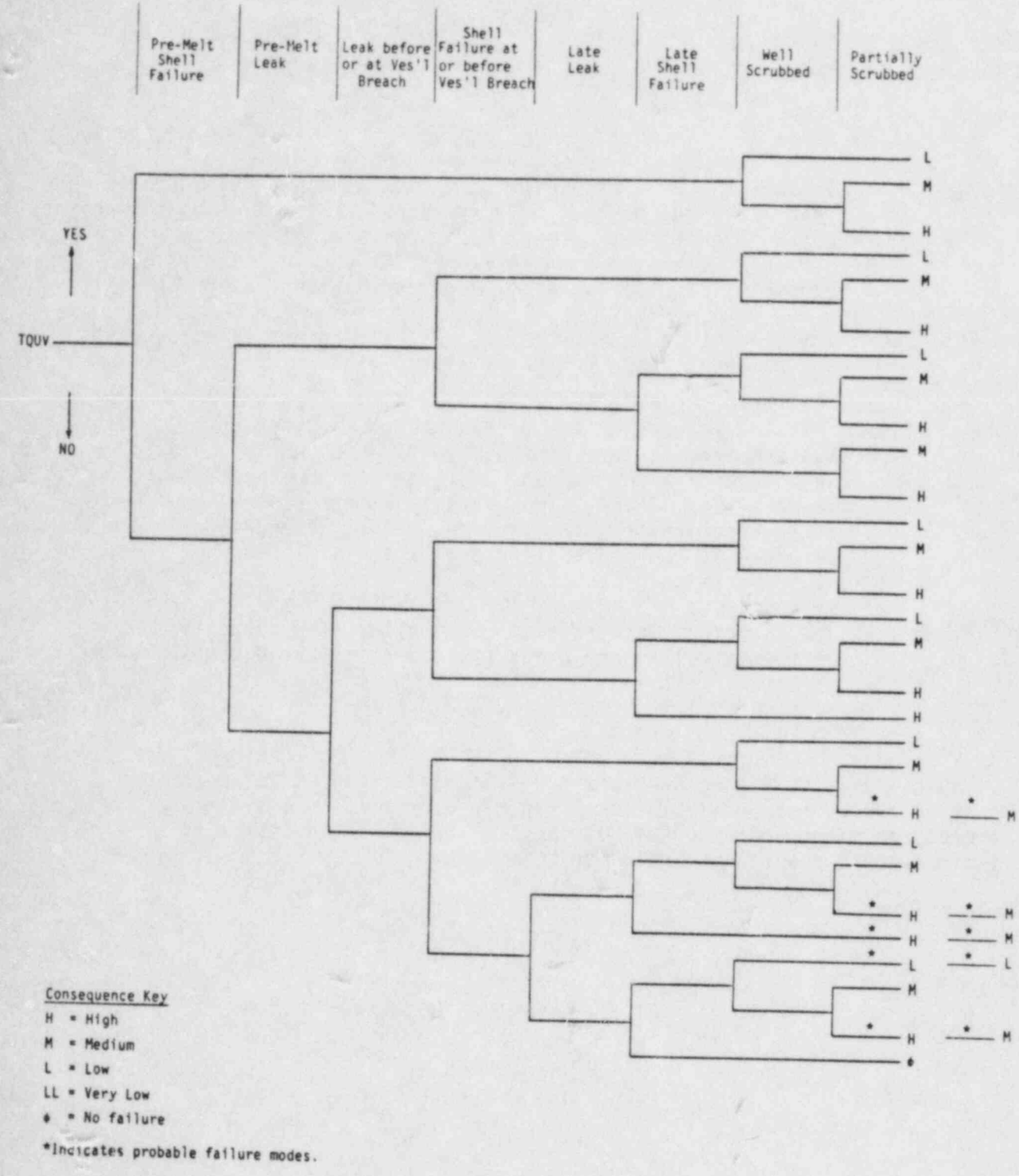

Figure 4.5. Operator action to mitigate Browns Ferry sequence TQUV (example 2).

Notes for Figure 4.5.

PLANT STATE: Core material is dispersed across the containment floor. Containment has been breached by over-pressure failure or drywell penetration failure. Reactor building radiation monitors may indicate the general vicinity of containment breach.

ACTION TAKEN: If reactor building smoke detectors have not initiated the "pre-action" function of the fire sprinkler system, manually initiate.

RESULTS OF ACTION: High temperature of containment release gases should melt fusible links in sprinkler heads. Pre-action function is required to initiate the water supply. Sprinklers act to remove aerosols and airborne fission products, condense steam and lower the source term.

UNCERTAINTIES: The effectiveness of the fire sprinklers in attenuating the source term is questionable in light of the uncertainties in sprinkler location and flow distribution. The system is primarily designed to protect cable trays, not general building areas. The uncertainties in how the drywell shell itself would fail also complicate the unlikely picture of a simple hole in the wall. It is, therefore, not clear that the sprinkler spray would pass through a significant amount of release gas.

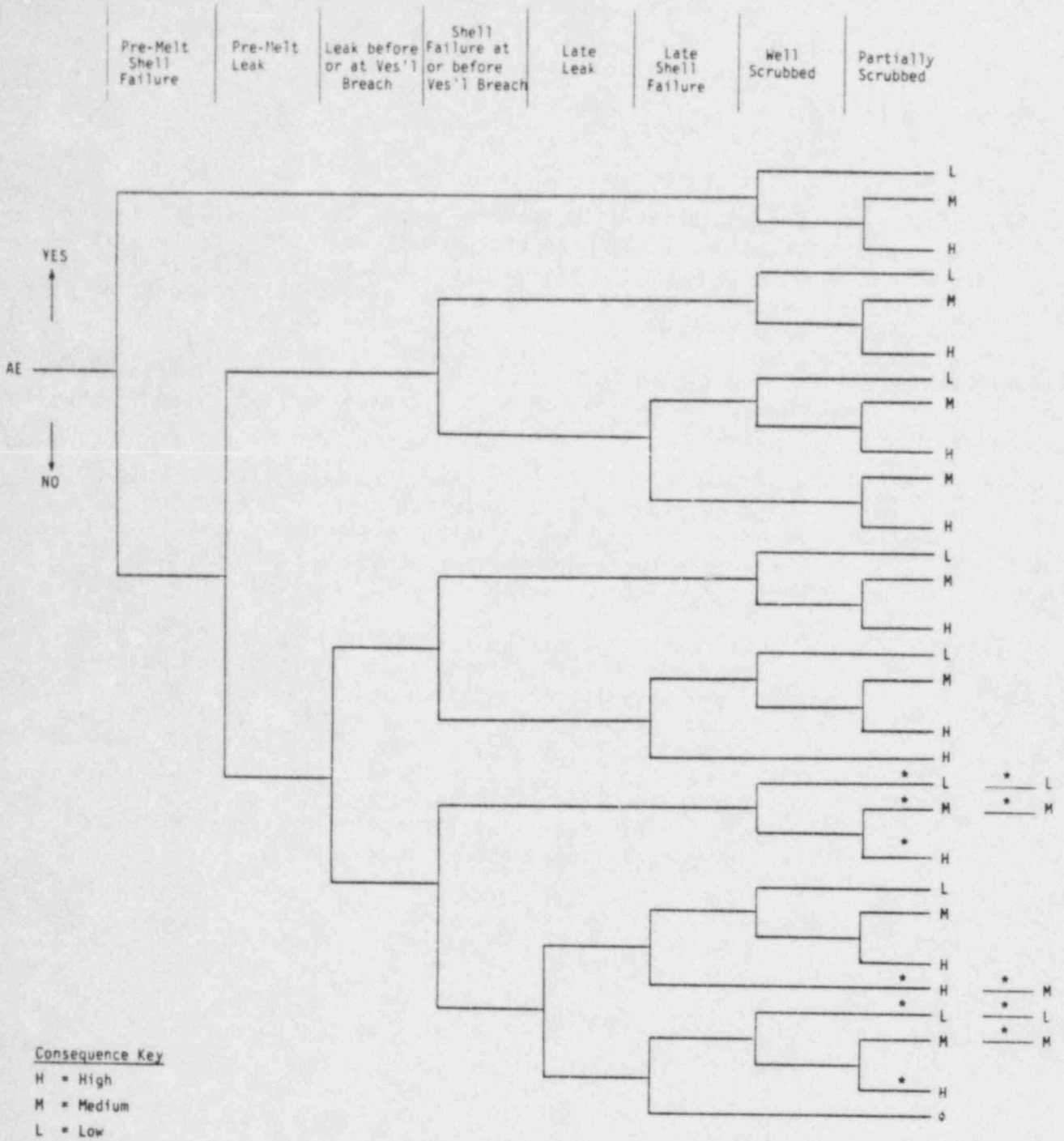

Figure 4.6. Operator action to mitigate Browns Ferry sequence AE (example 1).

Notes for Figure 4.6.

PLANT STATE: Core melt has progressed through vessel lower head melt-through. Core material is dispersed across containment floor. Containment integrity is lost due to overpressure failure in drywell.

ACTION TAKEN: Regain partial injection and deliver as containment sprays.

RESULTS OF ACTION: This action does not actually influence containment failure mode since it is taken after containment failure; it can, however, influence the consequences.

Containment sprays reduce the release driving force by cooling the containment atmosphere. It also is a removal mechanism for airborne activity. Both results serve to lower the source term.

UNCERTAINTIES: It may be possible for the sprays to cause the development of a combustible mixture in the containment. If spray actuation causes a great enough depressurization of the containment atmosphere by condensing large amounts of steam, reactor building air may be drawn into the containment through the drywell breach. Since most of the pre-inerting gas would have flowed out of the containment at drywell failure, a combustible mixture might result. This scenario also requires the air in the reactor building to not have been replaced by steam at containment failure.

Hydrogen combustion in the reactor building is probably more likely.

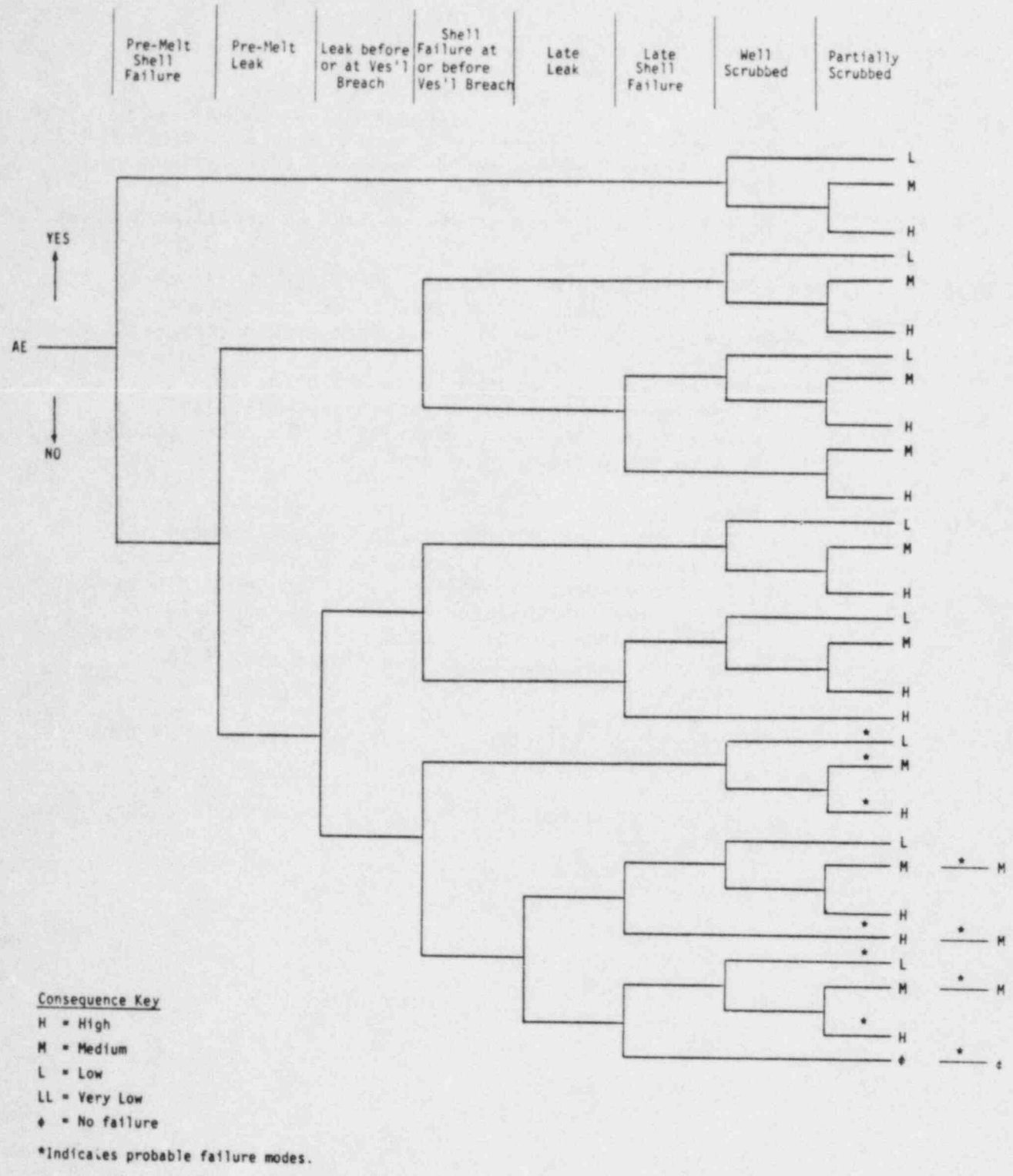

Figure 4.7. Operator action to mitigate Browns Ferry sequence AE (example 2).

Notes for Figure 4.7.

PLANT STATE: Significant core damage. Primary containment intact but pressure and temperature are high and rising rapidly. Partial injection capability is restored such that a competition for resources allows either ECC or containment sprays to function, but not both simultaneously.

ACTION TAKEN: First line-up coolant to containment sprays. Operate sprays until containment pressure is below a value p. Then switch over to ECC injection mode.

RESULTS OF ACTION: The value p is such that when ECC is initiated, the anticipated pressure spike associated with core quenching will not fail containment. If core quenching is successful, a longer term alternating scheme between ECC and containment spray operation may be necessary to maintain containment integrity as well as core cooling.

UNCERTAINTIES: The pressure rise associated with quenching the core in various degrees of degradation needs to be assessed and compared to the corresponding containment base pressures.

NOTE: This action is unique to the AE sequence since it is characterized by a high steam content in the drywell atmosphere. Delaying ECC injection by first initiating containment sprays may not be a beneficial action for other sequences in which the drywell atmosphere is composed of mostly noncondensable gases.

After performing the evaluation exercise on the containment performance tree for several plausible operator actions during the reference accident sequences, the actions were observed to fall into two categories defined by their immediate objective. In most cases the actions are intended to prevent or at least delay containment failure. However, other actions are directly aimed at reducing radionuclide release, given a failed containment. Some individual actions may fulfill either objective (e.g., turning on containment sprays), but others are distinct to one (e.g., ensuring activation of reactor building fire sprinklers). This might suggest a natural means of structuring the transitions required among the relevant types of emergency "procedures"* that would correspond to dramatically changing plant states.

Current emergency operating procedures, based upon the BWR Owners Group Emergency Procedure Guidelines (EPGs), become irrelevant after fuel integrity is lost and gross core damage begins. There is also only limited guidance for primary containment control when entire safety functions are lost. Under these conditions a transition to some form of guidance for maintaining containment integrity is required. Example specific actions have been considered here for specific sequences. Should the available actions not be capable of preventing ultimate containment failure, operator guidance should be augmented with "procedures" designed to minimize the environmental release.

In summary, the method outlined in Section 4.1 appears to provide a useful means of evaluating the effectiveness of and risk associated with specific operator actions taken to mitigate severe accidents in a BWR-4 with a Mark-I containment. The uncertainties in plant response noted for the actions evaluated can be addressed in a more extensive analysis using existing analytical tools. It is necessary to perform such an analysis to quantify the degree of confidence in a specific operator action yielding a desired plant state. Later, in Section 5.2, the potential value of such actions is assessed.

4.3 Application of Method to Zion

The containment performance tree for the Zion PWR, a Westinghouse design with a large dry containment, was constructed in a manner similar to that for BF-1. Branch points are based upon the time and size of containment failure. The washout of airborne radionuclides by spray operation is explicitly considered as the final branch point. Again it was assumed that a leak may or may not preclude a later catastrophic failure of the containment structure by overpressure or hydrogen burn.

The expected consequences associated with the containment performance tree for Zion rely on different factors than those for Browns Ferry. This is reflected in the manner in which the performance tree was constructed. The consequences

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* The word "procedure" is placed in quotes here to caution against interpreting it to strictly mean written instructions.

associated with the various containment failure modes in Zion are primarily time-dependent. That is, in general, fission product source terms tend to decrease as time to containment failure increases. In contrast, for the BWR/MarkI system, the element of time is overshadowed by the effects of sprays and fission product scrubbing in the pressure suppression pool on source terms. Effective operator actions for Browns Ferry were those that took advantage of the fission product retention afforded by these components, whereas effective actions in Zion were those that delayed containment failure.

The sequences considered for Zion are a small break LOCA with a loss of coolant injection (S_2D) and a transient with loss of core cooling and containment cooling (TMLB'). Fewer extensive SASA-type analyses were available for Zion than for Browns Ferry to support the analysis of operator actions; this limited the scope of the application. The two sequences were analyzed by Sandia[17], including some evaluation of mitigative actions. Fission product source term estimates for specific failure modes were also estimated as part of the BMI-2104 study,[18] but without considering mitigative actions.

For both sequences, the effect of three operator actions are illustrated on the Zion containment performance tree. The actions taken to mitigate a TMLB' sequence are illustrated in Figures 4.8 through 4.10 and S_2D in Figures 4.11 through 4.13. As in the Browns Ferry application, the likely results of the actions illustrated are speculative based upon our current understanding of the relevant phenomena, with support from the limited analyses available.

Summary of Results for Zion

Similar conclusions were drawn in this exercise as for Browns Ferry. Using a containment performance tree to visualize the effects of operator actions is both useful and instructive. If reasonable estimates of probable containment failure modes can be determined over the range of phenomenological scenarios, the method provides some quantitative estimate of the value of operator intervention in terms of the magnitude and likelihood of achieving a reduced radiological source term.

An additional observation was made during the Zion exercise, however. It appeared that fewer options were available for operator actions during the sequences considered for this reactor/containment design than for the BWR/MarkI. In particular, the TMLB' sequence in Zion renders most equipment unavailable. Station blackout at Browns Ferry does not remove all emergency safety features from service since backup DC power is available for a limited but significant period following a loss of all AC power. Operator actions are available during this period to extend the duration of DC power to maintain control of high pressure coolant injection systems. A key reason for the difference in available

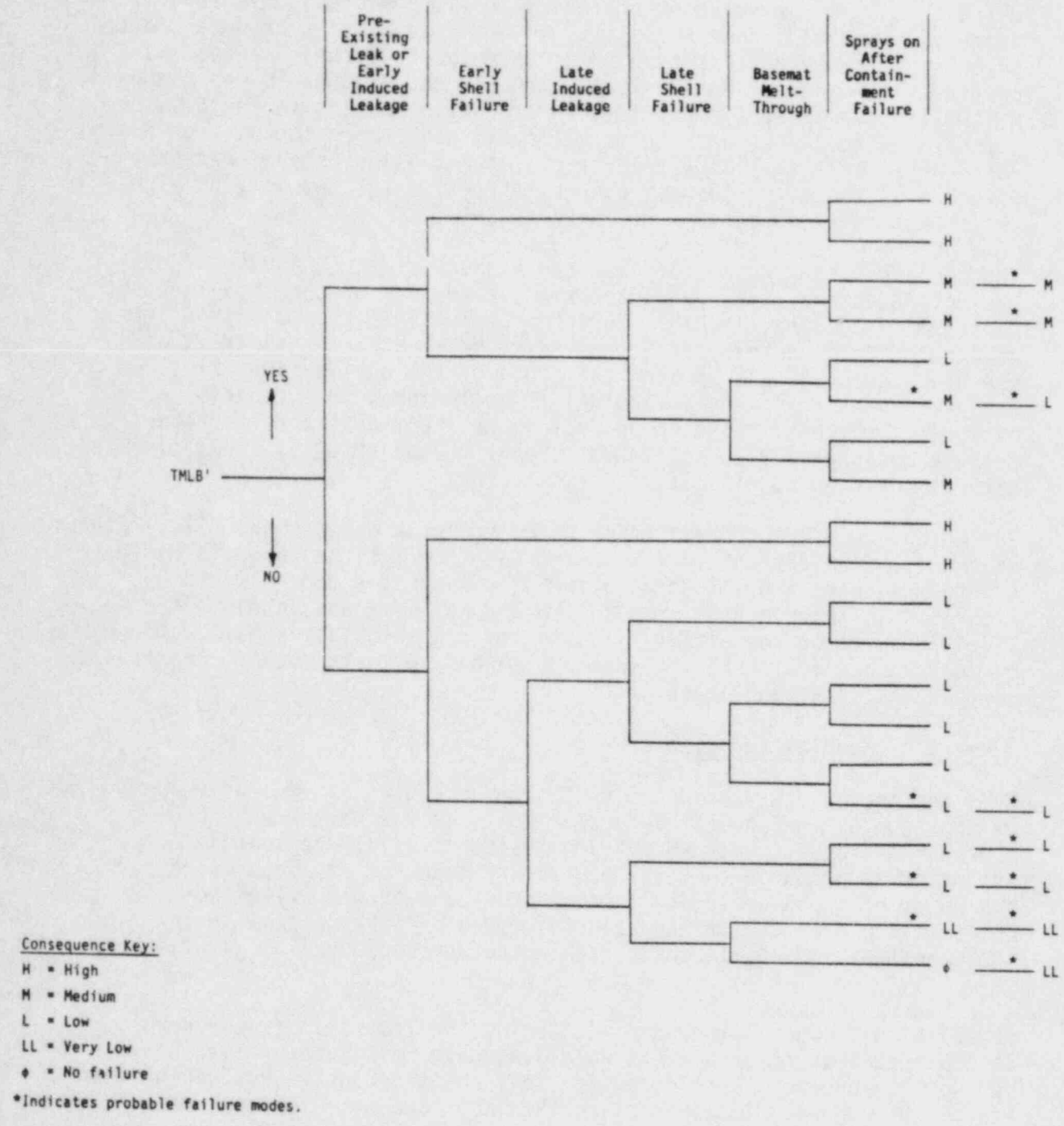
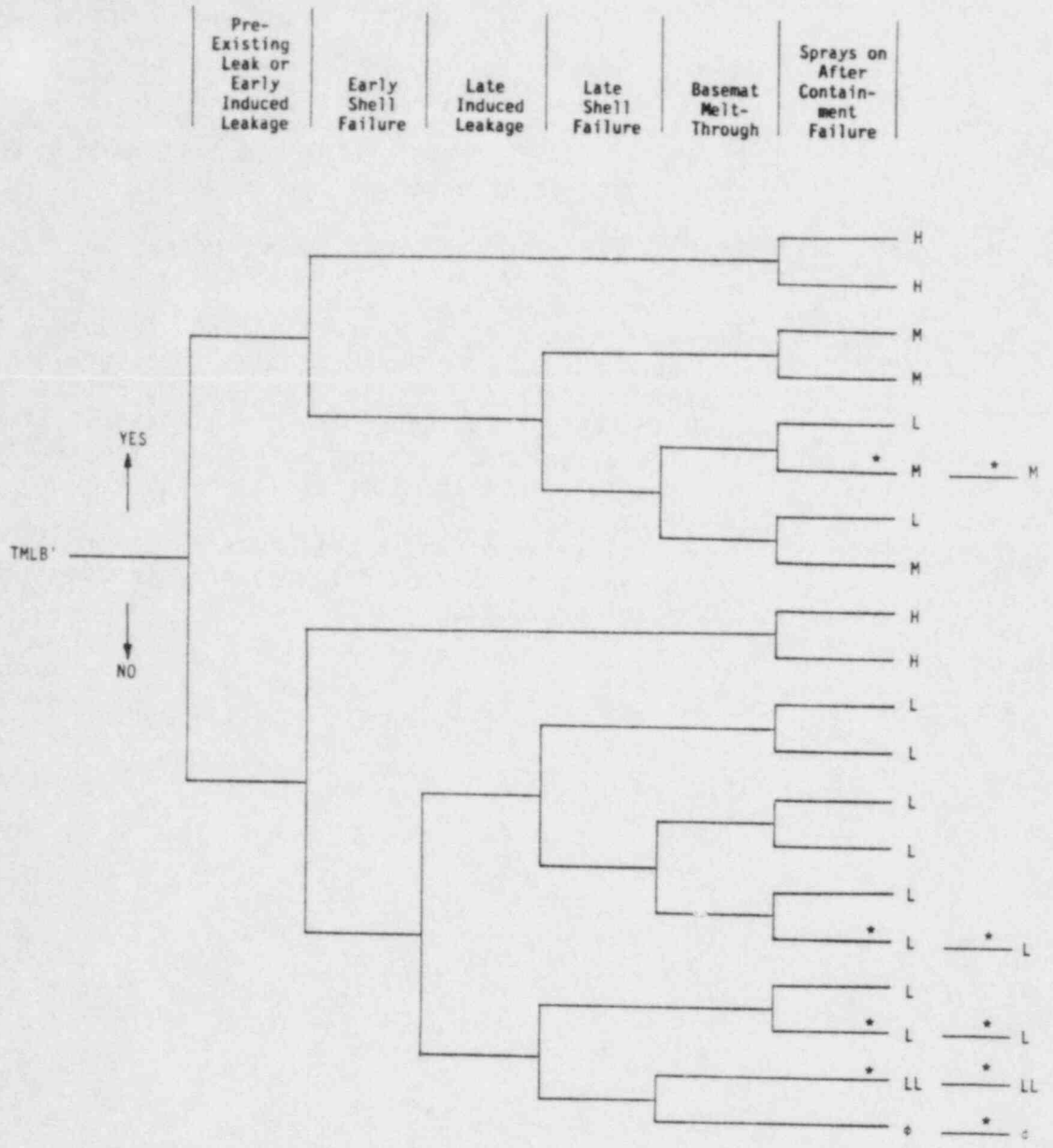


Figure 4.8. Operator action to mitigate Zion sequence TMLB' (example 1).

Notes for Figure 4.8.

PLANT STATE:	Core uncovery has been of long enough duration to expect core damage to have begun.
ACTION TAKEN:	Actuate diesel driven containment sprays.
RESULTS OF ACTION:	Sprays condense steam in containment atmosphere allowing hydrogen and other noncondensables to accumulate, provided system is operating in the injection mode (drawing suction from the RWST). Although the containment is likely to survive global hydrogen deflagrations, the development of locally detonable mixtures is possible.
UNCERTAINTIES:	Potential range of loads associated with probable distributions and ignition criteria for hydrogen combustion needs to be examined.



Consequence Key:

H = High

M = Medium

L = Low

LL = Very Low

ϕ = No failure

*Indicates probable failure modes.

Figure 4.9. Operator action to mitigate Zion sequence TMLB' (example 2).

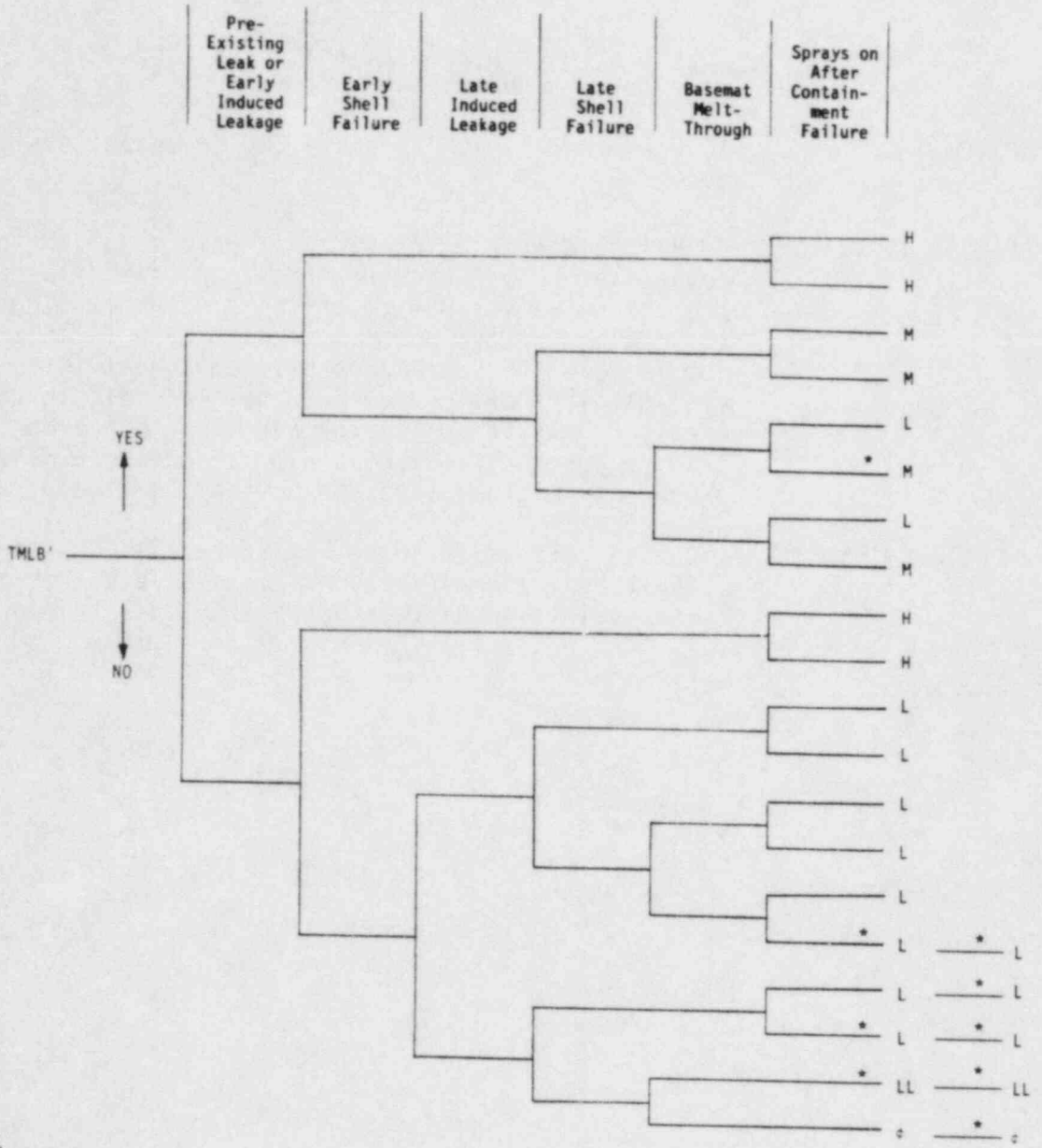
Notes for Figure 4.9.

PLANT STATE: Vessel at high pressure, relieving through PORV. Significant core melt. Containment intact.

ACTION TAKEN: AC power regained. Initiate ECC to vessel via injection mode.

RESULTS OF ACTION: Core should quench, provided it hasn't collapsed into the lower head. Quenching results in a brief, large hydrogen projection rate. Due to previous steam loading of containment, hydrogen burns are unlikely. If this action is not followed by some means of containment integrity control, late overpressure or basemat melt-through are likely. Only if the energy addition to the containment from the primary coolant can be stopped via makeup cooling or the steam generators, is no failure likely.

UNCERTAINTIES: Amount of time allowed to remove hydrogen burden must be estimated. Long-term condensation of steam on containment surfaces introduces the possibility of late hydrogen burns.



Consequence Key:

H = High

M = Medium

L = Low

LL = Very Low

ϕ = No Failure

*Indicates probable failure modes.

Figure 4.10. Operator action to mitigate Zion sequence TMLB' (example 3).

Notes for Figure 4.10.

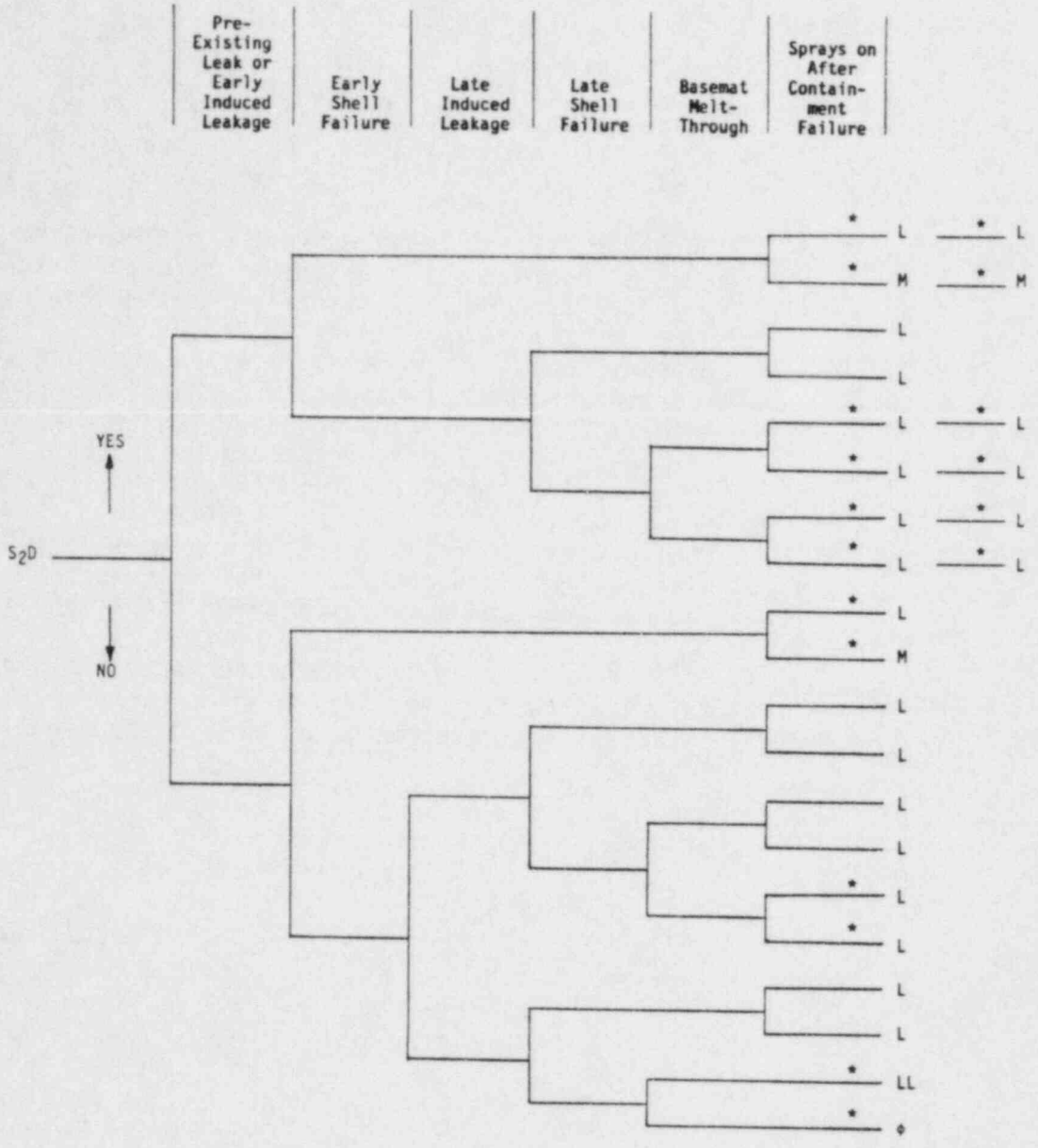
PLANT STATE: Vessel breached and core material dispersed across containment floor. Containment intact but pressure and temperature are rising.

ACTION TAKEN: AC power is restored. Prevent containment spray actuation, control containment pressure at high but below failure levels with one fan cooler. Start hydrogen recombiners.

RESULTS OF ACTION: Overpressure failure is averted and provided steam inerting can be maintained long enough to remove a significant amount of hydrogen, large scale burning may be avoided. Local burning may increase the likelihood of high temperatures failing penetration seals.

UNCERTAINTIES: Onsite hydrogen recombiners are not designed to operate with environments in excess of 5 mole-percent hydrogen. Need to consider arrangements for high-capacity recombiners.

It is not clear what temperature the containment gas may attain while pressure is held constant. High temperatures increase the likelihood of penetration seal failure.



Consequence Key:

H = High

M = Medium

L = Low

LL = Very Low

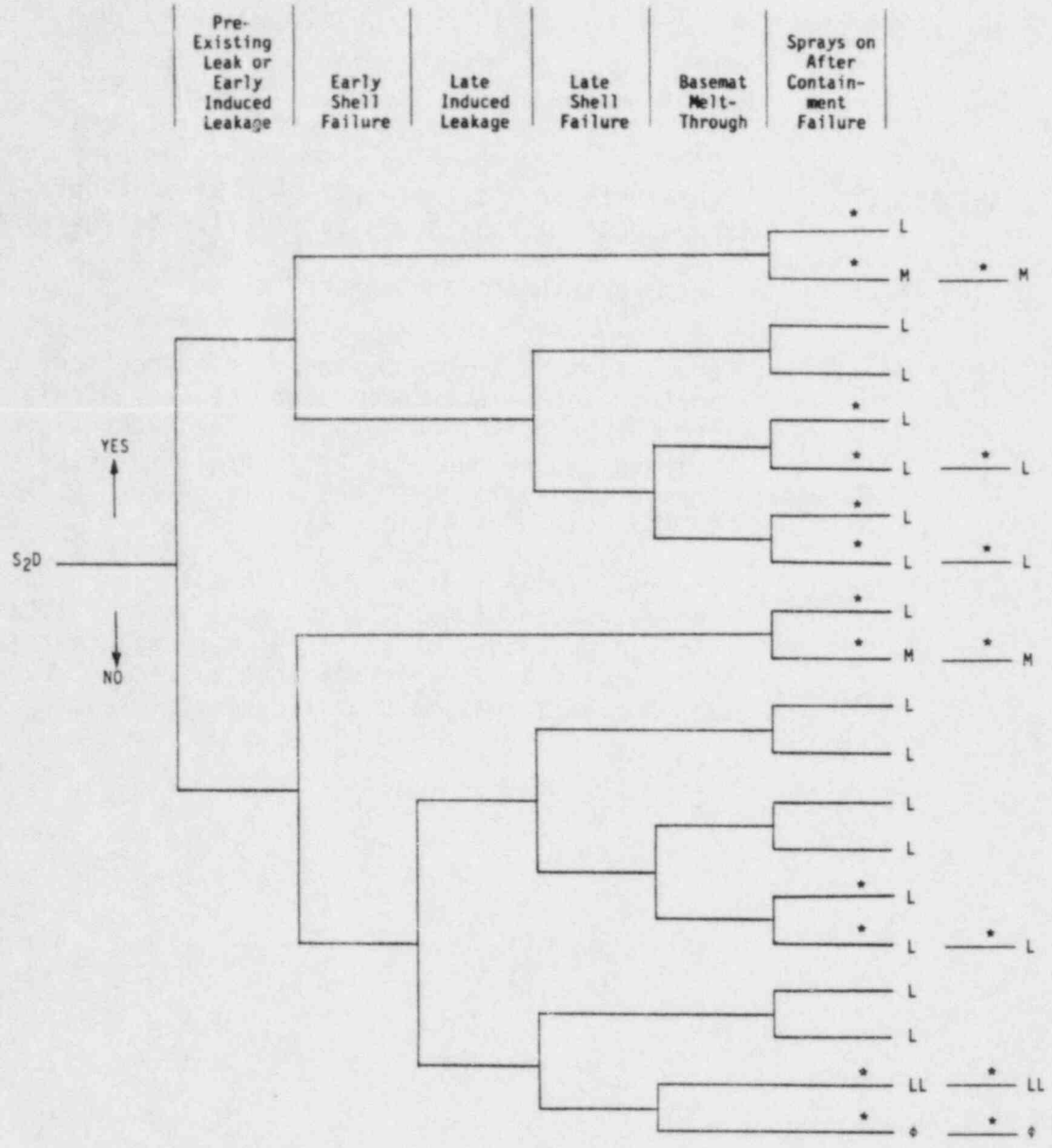
ϕ = No failure

*Indicates probable failure modes.

Figure 4.11. Operator action to mitigate Zion sequence S_2D (example 1).

Notes for Figure 4.11.

PLANT STATE:	Core uncovery is imminent. Containment sprays or fan coolers have been maintaining a relatively low containment pressure.
ACTION TAKEN:	Vent containment atmosphere.
RESULTS OF ACTION:	The intent is to reduce the free oxygen content of the containment before core damage occurs, significantly increasing radiation levels. If enough oxygen can be removed, subsequent hydrogen burns may be oxygen limited. However, an early burn may still be more probable due to the loss of inerting gases.
UNCERTAINTIES:	The relative humidity of the containment atmosphere increase with time. The amount of air that can be removed per pound of vented atmosphere, therefore, decreases. It is unlikely that sufficient oxygen can be removed to make a significant difference.



Consequence Key:

H = High

M = Medium

L = Low

LL = Very Low

ϕ = No failure

*Indicates probable failure modes.

Figure 4.12. Operator action to mitigate Zion sequence S_2D (example 2).

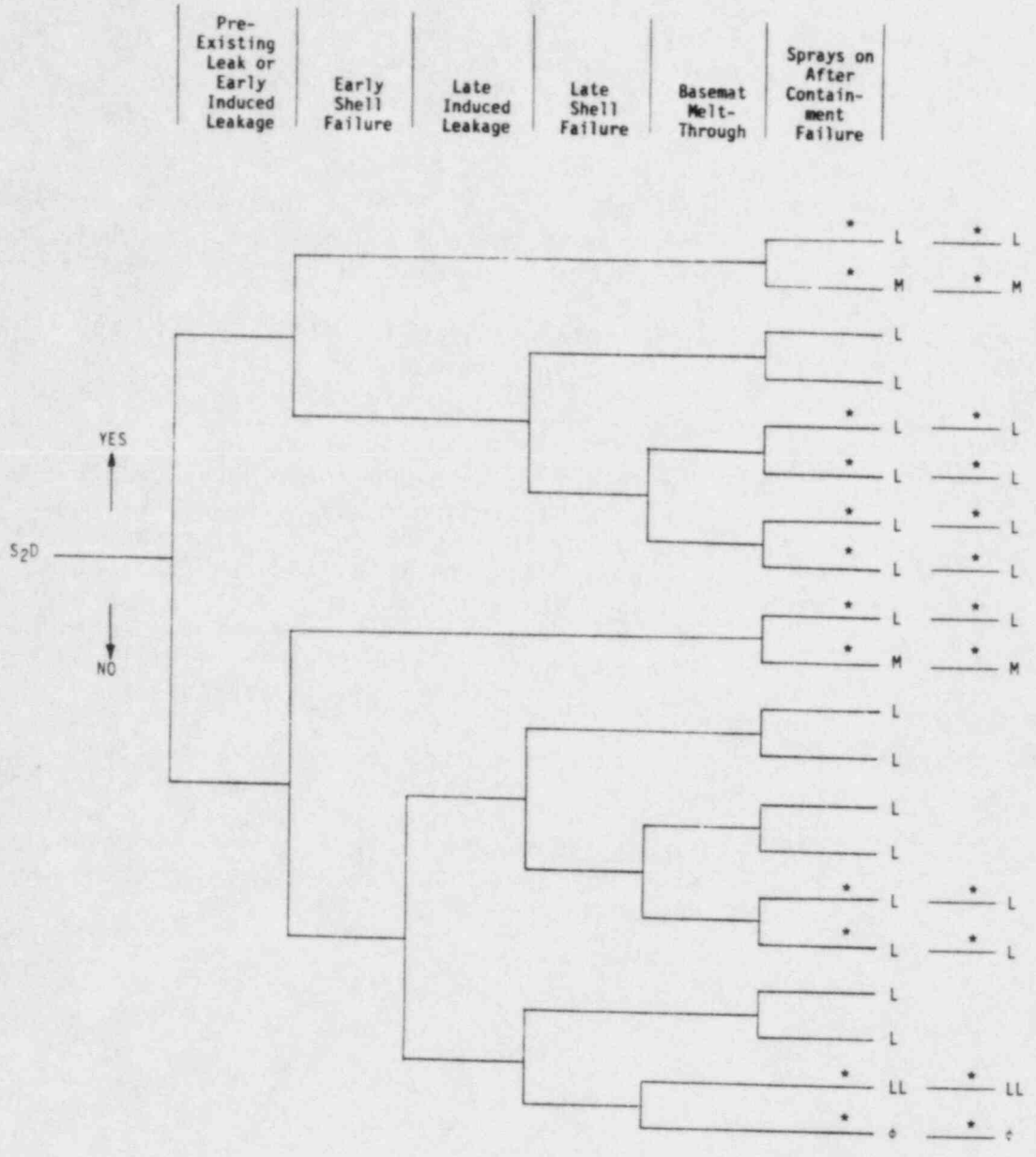
Notes for Figure 4.12.

PLANT STATE: Core uncovery has been of long enough duration to expect core damage to have begun. Containment is intact and sprays have been on to maintain low containment pressure.

ACTION TAKEN: Turn containment sprays off. Operate coolers to maintain pressure at some elevated level.

RESULTS OF ACTION: The intent is to prevent the development of a combustible containment atmosphere by allowing the steam concentration to build up as hydrogen is released. If successful, the probability of late failures (basemat melt-through or induced leakage) increase and early hydrogen burn failure reduced. However, if hydrogen burns do occur, with the base pressure now elevated, the likelihood of a burn causing containment failure increases.

UNCERTAINTIES: The range of steam and hydrogen relative concentrations need to be evaluated for the case in which the early steam release is condensed by sprays. If combustible mixtures are possible, the associated pressure rises need to be evaluated as to whether they may threaten containment integrity.



Consequence Key:

H = High

M = Medium

L = Low

LL = Very Low

ϕ = No failure

*Indicates probable failure modes.

Figure 4.13. Operator action to mitigate Zion sequence S_2D (example 3).

Notes for Figure 4.13.

PLANT STATE: Significant core melt. Containment intact and sprays have been on to maintain low containment pressure.

ACTION TAKEN: Switch sprays over to recirculation mode.

RESULTS OF ACTION: Fan coolers alone can maintain containment cooling. Aerosol attenuation can still be accomplished with higher temperature sprays and RWST is not depleted by continuous operation of sprays in the injection mode. The action preserves RWST inventory in the event that vessel injection is restored, at no expense to maintaining containment integrity.

This action does not lend itself to visualization on the containment event tree since failure mode probabilities and consequences are unaffected. Instead, this action reduces the probability of the initiating sequence by preventing the RWST from being depleted should ECC injection capability be restored.

UNCERTAINTIES: The ability of the plant specific containment spray pumps to operate (without cavitation damage) while taking suction from a water source at or near the saturation temperature must be evaluated.

operator actions in Zion and Browns Ferry is the difference in the relationship between the reactor coolant system and containment. One encloses the other in the Zion design, with the containment relying upon mechanical equipment for reducing loads due to energy addition from the primary coolant system. The reactor coolant system and containment are more closely coupled in the BWR/MarkI design, with the suppression pool in the torus providing a very large passive reservoir to dissipate energy. The Zion containment structure is much larger and stronger, which compensates for this difference.

Finally, our study has shown that the quantitative estimation of containment failure mode probabilities is a key element in the utility of the modified containment performance tree method of investigating the effects of operator actions on severe accident consequences. The benefits gained or the dangers associated with operator actions cannot be evaluated by simply indicating changes in probable containment failure modes, as done in our two plant exercises. It is also possible that the probable containment failure modes are not changed by an operator action; the effect often lies in the change in their relative probabilities.

A good example of this latter point is given in Figure 4.12, which illustrates the effects of turning off the sprays in the Zion containment after the core uncovers in a S_2D sequence. The probable containment failure modes are essentially unchanged; however, a significant change in their relative probability is anticipated. The action may significantly reduce the likelihood of an early failure due a hydrogen burn and increase the likelihood of a late failure. This potential benefit is not apparent in Figure 4.12 since failure mode probabilities are not indicated.

5. VALUE/IMPACT ASSESSMENT

This value/impact assessment follows the guidelines presented in NUREG/CR-3568, A Handbook for Value/Impact Assessment[19]. This analysis provides preliminary estimates of the net benefit of extending emergency procedures into the severe accident regime. The proposed action whose value and impact we are estimating is summarized in Section 5.1: the pilot study and its phased extension to all plants. As is appropriate for a scoping analysis, we have used several of the approximate values for public health effects and core-melt frequencies, contained in Appendix B of NUREG/CR-3568, supplemented by plant-specific parameters where necessary.

Many simplifications and generalizations have been made from the limited data available to illustrate the possible outcomes of subsequent phases of study. This assessment should be considered as an illustration of what may be possible.

5.1 A Proposed Program for Extending Emergency Procedures into the Severe Accident Regime

As discussed in Section 2, emergency procedures are undergoing a major evolution, presumably for the better. They are being rewritten to improve simultaneously their scope of coverage, their relevance to the plant's expected physical behavior, and their ease of interpretability. For the most part, the new procedures continue to emphasize the prevention of core damage, with little real guidance being given to managing containment integrity. This is not surprising considering that prevention traditionally takes precedence over mitigation.

Our investigations to date indicate that extending procedures into the severe accident regime is technically feasible. What appear to be at issue are the appropriateness of the guidance they provide and the value/impact balance associated with their implementation. These are also highly technical issues whose resolution depends to some extent upon our understanding of plant phenomenology and operating crew behavior during severe accidents. It appears that extended procedural guidelines written at an engineering or safety function level provide comparable value at substantially less impact than do very specific procedures written at the system or component level.

Developing improved methods of analysis and better strategies for managing severe accidents is of little value unless such efforts are transposed from research-space into regulatory positions and plant operations. A program to do that is proposed below. It would take a phased approach toward disseminating the insights of previous work, analogous in some ways to that used in developing symptom-oriented procedures. It builds upon the results of work reported here and in the companion report on accident management perspectives[1].

In summary, the proposed program is comprised of three phases. The first phase is a plant-specific pilot study sponsored by NRC; a principal goal of this pilot study would be to develop and test methods for identifying explicit actions that licensees could take to improve their capability to manage severe accidents. The pilot study would serve to identify specific methods best suited for the analysis, and to confirm or deny the value of extending the program to a greater population of plants.

In the second phase, similar plants would be grouped into classes, and for a plant representative of each class, analyses using the methods developed in Phase 1 would be performed to generate severe-accident management strategies. These results would then be made available to individual licensees and the NRC for their consideration on a plant-specific basis. For the third phase, licensees would apply the results of Phase 2 to their own facilities. The information on strategies would be most useful in training the senior engineering support staff, shift technical advisors, or others likely to be involved in decision-making during severe-accident conditions. This work could be performed primarily by licensee independent safety evaluation groups, engineering staff or other in-house analytical groups.

5.2 Value Assessment

Using the definition in the handbook for value/impact assessment, values are measures of the public benefits that the NRC is required to pursue as its statutory mission. For this exercise we focus on two attributes: public health and off-site property interdiction. Attributes such as occupational exposure, on-site property loss, regulatory efficiency, and improvements in knowledge are excluded at this stage of the analysis. The omission of these attributes tends to underestimate value, but we believe not significantly so, considering the purpose of the exercise.

5.2.1 Public Health

The evaluation of accident mitigation strategies in Section 4 used a simple classification of "high-medium-low" for the nature of releases from endpoints of the containment performance trees. For this analysis these were assumed to be approximately order-of-magnitude reductions from high to medium to low. "High" roughly corresponds to releases in the range of 10 percent to 100 percent of the volatile fission products (most significantly, iodine and cesium). "Medium" corresponds to releases of 1 percent to 10 percent of the volatile fission products, with "low" being less than 1 percent. As a first approximation, we have assumed that the public health effects (specifically total man-rems) are linearly proportional to the size of the releases.

Two sources of health effect estimates have been used. The first is NUREG/CR-1989[17], which reports radiological consequences for postulated severe accidents for the Zion PWR plant. The second is the Severe Accident Risk Assessment (SARA) for Limerick Generating

Station,[20] a BWR plant. These estimates are used only to provide preliminary indications of potential value. Predicted health effects could vary significantly from one plant to another because of differences in assumptions about population distribution, the expected radiological releases from the plant, weather patterns, evacuation speeds, and release timing.

Figure 2-3 of NUREG/CR-1989 indicates a mean range of 10^7 to 10^8 man-rem per accident for the highest outcome sequences involving bypass of the containment (sequences resulting from loss of coolant accidents in pipework outside containment--the interfacing systems LOCA).* The frequency of such sequences is on the order of 10^{-7}/year, giving an expected value of 1 to 10 man-rem/year of reactor operation.

For accidents not involving bypass of the containment, the next highest outcome results from containment failure above ground due to over-pressure or hydrogen burning; this has a mean man-rem value on the order of 10^7. Based on a typical core-melt frequency of 5×10^{-5} (Appendix B of NUREG/CR-3568) and an assumed conditional probability of containment failure due to overpressure or hydrogen burn of 0.1, the expected value is

$$10^7 \times 5 \times 10^{-5} \times 0.1, \text{ or}$$

50 man-rem per year of reactor operation.

It should be noted that containment failure-mode probabilities are highly plant- and sequence-specific, as discussed in Section 3.2. The assumed failure-mode probability (0.1) is within the range of values for many risk-dominant sequences and plants. It is recognized that for some types of containments (e.g., BWR Mk III, PWR large dry), the probability will probably be less.

As a result, the "high" outcomes for sequences in Section 4 correspond as a first approximation to 50 man-rem per year of reactor operation. "Medium" would then correspond to about 5 man-rem, and "Low" to less than 0.5 man-rem, using the order-of-magnitude relationships.

For comparison, the BWR calculations in the Limerick Severe Accident Risk Assessment also report point-estimate man-rem contributions per year of reactor operation (Table 10-18 or SARA). That report identifies the largest value as 77 man-rem per reactor-year for source term "OPREL" corresponding to containment overpressure failures above ground. The

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* For perspective, recall that 10^4 man-rem is usually equated with one statistical latent cancer fatality. Also, the estimates of the integral off-site dose received as a result of the TMI-2 accident appear to be converging around 3000 man-rem.

next largest is 6.8 man-rem per reactor-year ("C_{123}"-suppression-pool draining due to containment cracking). Other leakage failures are in the range 4.8 to 1.2 man-rem per year of reactor operation.*

These values are roughly comparable as a first approximation to those for Zion (77 vs. 50 man-rem for "High", 6.8 vs. 5 for "medium"). We will use the Zion values as the basis for the subsequent analyses.

If, as a result of implementing the program described in Section 5, the expected outcomes of all sequences originally classified as "high" (50 man-rem) consequence are now reduced to "medium" (5 man-rem) consequence, and if there are no associated changes in core-melt frequency, then there would be an effective reduction of 45 man-rem (50-5) per reactor-year expected for each plant. Table B.1 of NUREG/CR-3568 projected (as of September 1982) a total of approximately 3800 reactor years of operation to come.** If any regulatory actions requiring mitigation procedures were in place by September 1986, there would be about 3500 reactor years remaining. If a regulatory program were to be fully implemented by September 1988, there would be about 3300 reactor years of operation remaining.

Based on an expected reduction of 45 man rem per reactor year, a program in place by September 1986 would result in an overall reduction of about 157,500 man-rem (45 x 3800) and for a program by September 1988, a reduction of 148,500 man-rem (45 x 3300). Using the value of $1000 per man-rem identified in Table B.3 of NUREG/CR-3568, this reduction corresponds to approximately 160×10^6 and 150×10^6.

The above analysis assumes that all high-consequence sequences can be transformed to medium-consequence sequences with a probability of 1.0. If instead we assume a 0.5 probability of mitigation action being taken and being effective, this would result in a reduction of 25 man-rem per year of reactor operation in the expected health effects. For a program in place by September 1986, the overall value would be about 90×10^6, and for one in place by September 1988, the value would be approximately 80×10^6.

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* The Limerick health effects are for the population up to 50 miles from the plant; extending the analysis to 500 miles approximately doubles the health effects. However, Limerick has a surrounding population density which is considerably higher than that of most other nuclear power plants in the United States. Using the lower values is considered to be more representative and results in a more conservative (i.e., lesser) estimate of the values.

** For those plants now in operation or possessing construction permits.

5.2.2 Off-site Land Interdiction

The analysis of the value of avoiding the interdiction of land outside the plant boundaries is based on values provided in the Limerick Severe Accident Risk Assessment. The expected annual contribution to economic risks for the set of consequences resulting in source-term "OPREL" is $5,500* per year of reactor operation (Table 10-20 of SARA). The contribution from the next significant set of source-term sequences is $520 per year of operation--again one order of magnitude less. A first approximation of the value of avoiding interdiction is therefore to assume an annual expected value of $5,000 for complete assurance of source term reduction from "high" to "medium". Based on a regulatory program fully implemented by September 1986, the value is $(3500 \times 5000) = \$18 \times 10^6$, and for a program implemented by September 1988, the value is $\$17 \times 10^6$. If a probability of 0.5 is assumed for the overall effectiveness of reducing the source-term size, the value for the September 1986 program is $\$9 \times 10^6$ and for the September 1988, program is $\$8 \times 10^6$.

5.2.3 Overall Value

Table 5.1 summarizes the estimated overall expected values for programs fully implemented by September 1986, and September 1988, for completely effective (1.0) source-term reduction and for 0.5 effective reductions.

Table 5.1. Summary of expected values of a program to extend emergency procedures into the severe accident regime.

	September 1986		September 1988	
	Completely Effective	0.5 Effective	Completely Effective	0.5 Effective
Public Health	160×10^6	90×10^6	150×10^6	80×10^6
Land Interdiction	18×10^6	9×10^6	17×10^6	8×10^6
Total $	178×10^6	99×10^6	167×10^6	88×10^6

* Based on 1980 dollar values.

5.3 Impact Assessment

NUREG/CR-3568 defines the impacts of a program as "the other consequences resulting from the proposed action." In the context of this study, we measure the impact as the monetary costs of implementing the program to extend emergency procedures into the severe accident regime. These include the costs to NRC and to the industry.

Section 5.1 of this report describes a phased program of development. The first phase is an NRC-sponsored pilot study of one U.S. power plant to develop methods, perform analyses, develop strategies and procedures and provide more details of the values and impacts of developing plant-specific mitigation procedures. The second phase extends the analysis methods to six or so more plants representative of the entire population of plants. The third and final phase extends the methods and insights to all remaining plants.

The estimated costs for these phases are:

Phase 1, pilot study--2×10^6 total

Phase 2, plant-specific analyses (6 plants)--1×10^6/plant

Phase 3, plant-specific analyses (128 plants)--
$0 - 1×10^6/plant.

At the time of this analysis, there are about 135 plants in operation or active construction, many of which are near duplicates. If we assume that only one set of analyses is required for duplicate plants and we take credit for those analyzed in Phases 1 and 2 as duplicates, then approximately 80 additional sets of analyses would be needed. On this basis, the overall impact could be as large as

$$(2 \times 10^6) + (6 \times 1 \times 10^6) + (80 \times 1 \times 10^6) = \$88 \times 10^6.$$

5.4 Comparison of Values and Impacts

NUREG/CR-3568 provides two ways of comparing values and impacts: the ratio method and the net-benefit method. The net-benefit method is used here because we are considering more than one kind of value for only one impact.

The various values and impacts, while expressed in common monetary units, use different year-bases for the estimates. For example, the values for land interdiction are expressed in 1980 dollar terms. The costs of the plant-specific analyses are based on experience in the early 1980's. Both of these estimates are probably somewhat low

compared with the costs, say for 1986 to 1988 (the expected timescale for any resulting program). However, at this stage, we believe it unnecessary to adjust for such differences, especially since there are many approximations and assumptions already in the analyses.

Based on the results of Section 5.3, the expected value of a program is expected to lie in the region of $\$175 \times 10^6$ to $\$88 \times 10^6$, depending on the likelihood assumed for the effectiveness of mitigative procedures on reducing the source terms, and on the date for implementing a program. The estimated impact is estimated to be $\$88 \times 10^6$. As a result, the net benefit is expected to lie in the range of \$0 to 87×10^6. This analysis suggests that the net benefit of such a program is positive.

6. SUMMARY AND CONCLUSIONS

We conclude that it is feasible to develop emergency procedures to provide guidance for operators to mitigate the consequences of severe accidents. Section 3 indicates that sufficient knowledge exists to formulate strategies to mitigate severe accidents. In addition, we have developed a process, based on the containment performance trees for creating and illustrating the effectiveness of such strategies.

The NRC requirements for the current generation of emergency procedures as summarized in Section 2 are applicable to extension into the severe accident regime with two exceptions. First, the specific technical analyses that are required to develop mitigation strategies are obviously different from those associated with prevention of core damage, though a similar program of analysis is required. Second, it is not obvious that detailed step-by-step procedures are appropriate for mitigative actions. For example, actions will not be required in short timescales, and specific plant systems may well have been degraded or failed by the events leading up to the core melting. In addition, any actions to deliberately release radioactive material to the environment will require concurrence at many levels of authorization, and hence senior plant-engineering staff will be available, for example in the technical-support center (TSC).

In comparison, the French 'U' procedures, as discussed in Appendix A, are written for engineering personnel and for the most part provide an engineering strategy, as opposed to detailed steps, which the plant engineering staff will enact at the time. Discussions with the French personnel indicate that, in their opinion, the most important feature of the 'U' procedures is that they are aimed at a different group of people (engineers versus operators), thereby providing redundancy and diversity in technical skills. The engineering staff are provided with training to help make decisions in using the 'U' procedures.

We believe that a similar approach is appropriate for the mitigation procedures discussed here, since this would result in a less expensive development program, and it is consistent with human behavior; that is, humans are very purposeful when they understand the goal that is important for them to achieve*.

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* Poor human behavior is often associated with either a mistakenly identified goal, a lack of understanding of how it can be achieved, or a misperception of its importance. See, for example, NUREG/CR-3010[21] for a discussion of the importance of goal-oriented behavior in responding to accidents.

In addition to demonstrating the feasibility of developing mitigation procedures and identifying relevant NRC requirements, the value/impact assessment in Section 5 suggests a positive net benefit, though there are many assumptions and approximations in these value/impact analyses.

As a result of this study, we propose performing a pilot study which would define and demonstrate more completely the kinds of analyses and knowledge required for plant-specific utility-performed studies; a handbook for such studies would be a logical and useful product. In addition, the pilot study should provide more detailed data for a value/impact assessment to supplement a decision on whether to motivate the industry to implement similar studies for the greater population of reactors.

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APPENDIX A

A REVIEW OF "U" PROCEDURES AND THEIR ROLE IN ACCIDENT MANAGEMENT

INTRODUCTION

The French national power utility, Electricite de France (EDF) is experimenting with the development of so-called "ultimate" (U) procedures to provide guidance to operators during severe accidents. This paper summarizes the nature of these procedures and the role they play in accident management. The principal purpose of these U procedures is to aid the operators in cooling the endangered core and maintaining the integrity of the containment during severe accident conditions.

Background to Development of U Procedures

The French PWR program is viewed as one of the world's more successful nuclear power programs, with continuing construction and operation of standardized PWRs of 900 and 1300 MWe rating. This program involves groups such as EDF (licensees and architect-engineers), Framatome (NSSS vendor), SCSIN (regulatory agency), and CEA/IPSN (government technical analysts).

The French approach to reactor safety* combines deterministic and probabilistic safety criteria to judge the acceptability of designs. Deterministic criteria are based on defense-in-depth, involving the demonstration of successive barriers to radiological releases (fuel clad, vessel and containment) and the adequacy of these barriers for a range of design-basis events. In order to define design-basis events, the French adopted a probabilistic criterion that the frequency of "unacceptable consequences" should be less than 10^{-6} per reactor year. "Unacceptable consequences" refers to the necessity for immediate evacuation around the site or roughly the equivalent of 5-50 rems projected whole body dose at the site boundary. The adoption of this criterion implies the need to consider failures of mitigating systems following initiating events as part of the design basis.

Based on this philosophy of considering design-basis events using probabilistic criteria, EDF has developed three types of procedures for accident conditions, "A", "H", and "U". "A" (for "Accidental") procedures relate to identified events requiring ECCS operation. "H" for "Hors-dimensionnement"--"beyond design", or, more appropriately, at the limit of design (see referenced paper by P. Tanguy)--procedures

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*P. Tanguy, "The French Approach to Nuclear Power Safety", Nuclear Safety, Vol. 24, No. 5, September-October, 1983).

provide guidance to operators for well characterized events in which important mitigating systems (e.g., scram, electrical supplies, main and auxiliary feedwater) have failed. An important feature of the A and H procedures is that they are based on the analysis of clearly identified families of accident sequences. Also it is clear that the goal of the A and H procedures is to satisfy the critical safety functions of criticality control and core cooling.

In order to provide a diverse means of preventing an unacceptable off-site release, EDF is developing a third type of procedure, called "U" (for "ultimate") procedures. The purpose of these U procedures is to provide guidance to operators for events where core cooling is not being satisfied, the goal being to minimize the likelihood of off-site releases. This type of procedure is intended to be applicable regardless of how the sequence previously developed, and is intended for use when use of the A and H procedures has been unsuccessful. Their primary goal is to guide the operator in managing the critical safety function of containment integrity.

"U" Procedures

Five U procedures have been or are being developed. These are:

U1 - last-resort measures to prevent core meltdown

U2 - actions in case of inadequate containment isolation

U3 - use of mobile units to supplement core-injection or containment spray systems

U4 - actions to flood the reactor cavity (certain plants only)

U5 - use of a controlled, filtered containment vent.

The initiation of the U procedures is the responsibility of the on-shift Safety and Protection Engineer, a member of the plant's engineering staff. For the A and H procedures, it is the shift leader and operators who are responsible for their implementation.

Two of the procedures, U1 and U3, are to prevent or delay the onset of coremelt. U2 through U5 are to reduce the likelihood of an uncontrolled failure of containment. The U3 procedure relates to the use of mobile systems for containment sprays as well as for injection into the vessel. The procedures concerned with maintaining containment integrity are developed for what the French consider the most probable containment failure modes*: U2-isolation failure (β) U4-base mat melt-through (ε);

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*Terminology used in the Reactor Safety Study (WASH-1400).

and U5-rupture from overpressurization (δ). The steam explosion (α) and hydrogen explosion (γ) causing direct failure of the containment were judged too improbable to merit specific consideration.

As of June, 1984, procedure U1 has been developed and is ready for use as a general operating guide. Linked to the French concept of the safety parameter display system, U1 provides guidance on characterizing the state of the plant in functional terms and on the need to implement subsequent U procedures. Table 1 illustrates the guidance provided in U1 to the operators on the use of secondary cooling or the use of ECCS depending on the state of the primary circuit. Figure 1 shows the decision logic that leads the operators into the use of this procedure.

Procedure U2 is reported to be available in part (Figure 2). The remaining U procedures are under study and are described as being an operating guide rather than an extensive list of specific instructions; the emphasis is on making evaluations before taking actions. However, they do provide a tentative listing of measures to be taken depending on the operators' knowledge of the plant state and the systems performance. Attention is focused on the last possible series of actions that are considered to have a beneficial effect.

EDF recognizes that the quality and validity of the guidance depends on the knowledge of plant and containment phenomena. They report that a major development effort is underway for the modeling of accidents involving two-phase flow in the primary circuit, though the primary intention of this work is to provide improved simulator models for operator training. In addition, work is progressing to improve the modeling of severe fuel damage and the behavior of radioactive material in the containment. These programs are expected to improve the knowledge base for U-type procedures.

In addition to the U procedure development program, the French research program related to accident management is developing an information-display system, to be watched by the on-shift safety engineer, to allow him to develop independent conclusions as to the state of the plant. It is expected that this information display will be related to the identified entry conditions for the U procedures.

Observations on the U-Procedure Concept

That the French are developing U procedures and incorporating them into their accident management program attests to the feasibility of extending procedures into the severe accident regime. The values and impacts associated with this action remain to be characterized.

The relevance of the U type of procedures to U.S. accident management practices is considered in three areas. These are:

Table A-1. U1 procedure: reactor cooling states and possible corrective actions.

State of primary circuit	Action* based on state of steam generators and ECCS**		
	At least one steam generator available	No steam generator available	
		ECCS	No ECCS
$\Delta T_{sat} \geq 10$ C	1', 2, 4'	1, 3, 4'	1, 4'
$\Delta T_{sat} < 10$ C and either core exit T < 350 C or core exit dT/dt ≤ 1 C/5 min	1, 2	1, 3	1
350 C ≤ core exit T < 650 C and core exit dT/dt < 1 C/5 min		1, 2, then 3 (if 2 for the past 5 min)	
Core exit T ≥ 650 C		1, 2, 3, 4	

*Definitions of number codes: 1, maximum ECCS; watch ECCS availability, take steps to restore ECCS if lost. 1', control ECCS flow in case of subsaturation. 2, speed up cooling process by available steam generators. 3, open the three pressurizer relief valves. 4, start primary pumps successively. 4', stop primary pumps.

**ECCS, emergency core-cooling system.

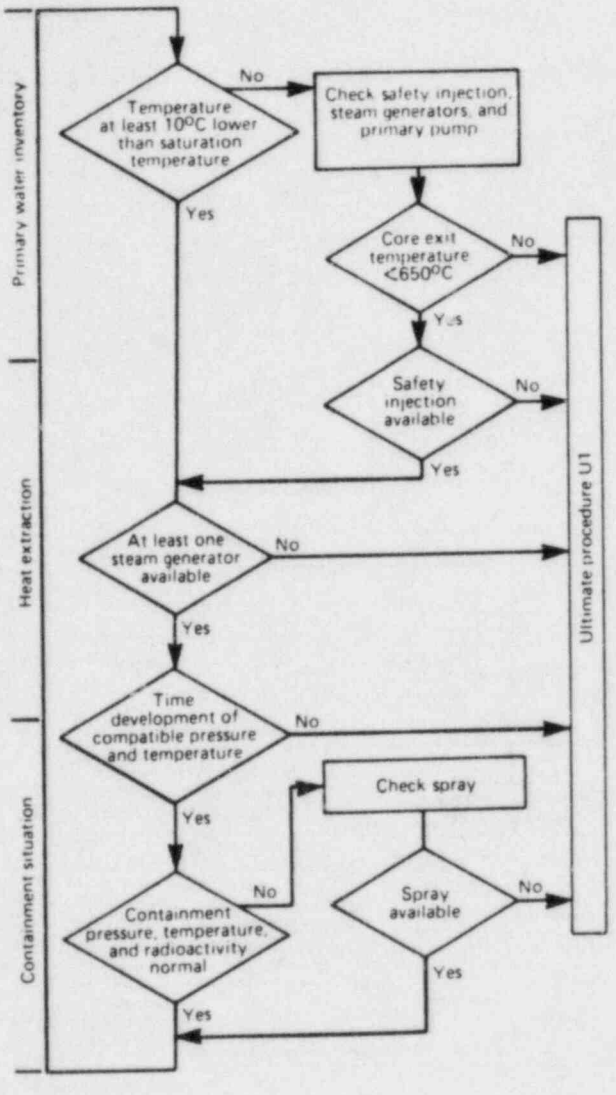

Figure A-1. Operator decision logic for use under PWR accident conditions.

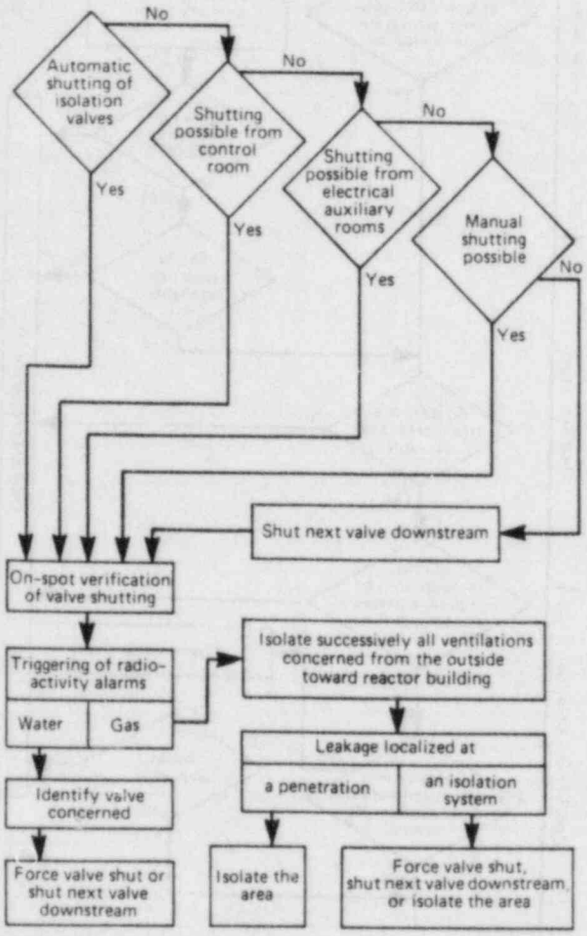

Figure A-2. Containment isolation U2 procedure flow chart.

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- The content and context of the procedures
- Their relationship with other elements of the severe accident-management process
- The level of information supporting the guidelines.

Apart from procedure U1, which provides a diagnostic process, the U procedures provide empirical guidance for a range of plant states that are judged possible though unlikely--the term used to describe these is "still conceivable". The guidance contained in these procedures emphasizes understanding rather than instructions, and is written for engineering interpretation. No attempt is made by EDF to provide guidelines for all imaginable states, and hence guidance for steam or hydrogen explosions is not provided. Such events are termed "hardly conceivable" and are assumed part of the so-called residual risks.

In some ways the emergency procedures guidelines (EPGs) developed by the various U.S. owners' groups in response to the NRC's post-TMI requirements (e.g., Supplement 1 to NUREG-0737), parallel the development of the U procedures. Specifically all the owner's groups' EPGs contain instructions operators can follow without needing to recognize the specific kind of initiating event (so-called function-based procedures as opposed to event-based procedures). This corresponds to the U procedures being used independently of the event-based A and H procedures. In addition, the EPGs provide guidance to varying degrees of detail on ways to protect the integrity of the containment--for example, the BWR EPGs include containment venting and use of the standby gas treatment system; the Westinghouse EPGs provide some guidance on hydrogen control.

Two important differences between the EPGs and the U procedures exist, however. First, the EPGs are intended to be used by the same operating staff that have previously been operating the plant, whereas the U procedures are written for the shift safety engineer (a role that has no direct equivalent in the U.S.--a shift technical advisor has no line authority to direct operations). Secondly, the emphasis in the EPGs is on actions to be taken, whereas the U procedures are more concerned with understanding the plant's condition. To a degree the second difference flows from the first; an engineer (by virtue of his professional expertise and training) should be more able to find impromptu solutions to a severe condition once he understands its nature.

Also keep in mind that the population of French reactors is highly standardized, consisting primarily of two types, 900 and 1300 MWe PWRs with large dry containments. The accident management strategies and implementing procedures are much more likely to be similar there than in the U.S. where the situation is quite the opposite. To assure that such procedures were positively affecting risk from U.S. plants would require a body of phenomenological and plant knowledge for each of the dozen or

so generic types of U.S. plant. Generating such knowledge requires substantial resources--the impact side of the value/impact equation.

The existence of U procedures is just one part of EDG's approach to planning for the management of severe accidents. For example, the initiation of the U procedures is controlled as part of the site emergency plan. Much of the phenomenological analysis that is being used to develop some of the U procedures is also being used to develop simulator models for operator training. In addition, the U procedures are deliberately designed to be somewhat orthogonal, or complementary, to the A and H procedures, in that they do not depend on the diagnosis of any particular initiating event as do the A and H procedures.

In the United States, there is no requirement for an integrated approach to the management of severe accidents, though in some cases there is a degree of connectivity between elements. For example, the Westinghouse Owner's Group (WOG)-sponsored safety parameter display system, called the Safety Assessment System (SAS), is designed in part to display data in a form directly related to the Westinghouse Emergency Reference Guidelines. Similarly, EPRI/NSAC sponsored the development of a disturbance analysis and surveillance system (DFASS) for BWRs which is directly related to the General Electric EPGs.

These individual cases are not necessarily representative of other NSSS systems or parts of the accident management process. For example, the latest ANSI requirements for operator training do not include severe-accident conditions; the EPGs have no intrinsic connections to the activation of the Emergency Plan by reference to the relevant emergency action levels.

Concerning the knowledge underlying the guidance provided in the U procedures, EDF acknowledges that many phenomena are not well understood even with the on-going research programs. As a result, the guidelines are based on current best-estimate understanding. The procedure developers recognize that this understanding is limited and, hence, provide a tentative list of actions with a greater emphasis on the portion related to explanation and evaluation. This would permit discussions of uncertainties, alternatives, and limitations of the explanations of the phenomena.

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This study examines the feasibility and value/impact of extending emergency operating procedures into the severe accident regime. It reviews the types of knowledge needed to develop such procedures and the applicability of existing regulatory review criteria. A method is developed and illustrated in two test cases. This study concludes that it is feasible to develop procedures for operators to mitigate the consequences of accidents progressing past the onset of core damage. A preliminary value/impact assessment indicates a significant likelihood of there being an overall net positive benefit of developing mitigative procedures. A phased program has been proposed. First a pilot study should develop the application of the methods used in this feasibility study and provide more precise information for a detailed value-impact assessment. Based on the results of the pilot study, extension to a greater population of plants may be justified.

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