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Accident Management Information Needs

Methodology Development and Application to a Pressurized Water Reactor (PWR) with a Large, Dry Containment

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ABSTRACT

In support of the U.S. Nuclear Regulatory Commission (NRC) Accident Management Research Program, a methodology has been developed for identifying the plant information needs necessary for personnel involved in the management of an accident to diagnose that an accident is in progress, select and implement strategies to prevent or mitigate the accident, and monitor the effectiveness of these strategies. This report describes the methodology and presents an application of this methodology to a Pressurized Water Reactor (PWR) with a large dry containment. A risk-important severe accident sequence for a PWR is used to examine the capability of the existing measurements to supply the necessary information. The method includes an assessment of the effects of the sequence on the measurement availability including the effects of environmental conditions. The information needs and capabilities identified using this approach are also intended to form the basis for more comprehensive information needs assessment performed during the analyses and development of specific strategies for use in accident management prevention and mitigation.

SUMMARY

Accident management is an essential element of the Nuclear Regulatory Commission (NRC) Integration Plan for the closure of severe accident issues. This element will be used to consolidate the results from other key activities under this plan to enhance the safety programs for nuclear power plants. Implementation of accident management will ensure that planned actions and preparatory measures are developed which will enhance the capability of nuclear power plant personnel to effectively manage severe accidents.

Instrumentation was identified by the NRC as one of the five areas in which the risks associated with severe accidents can be further reduced. This identification was based on the importance of reliable plant status information, which is needed by personnel to successfully manage severe accidents. Without adequate plant status information and guidance to ensure its proper use, plant operating personnel cannot reliably diagnose the occurrence of an accident, determine the extent and nature of the challenge to plant safety, monitor the performance of automatic systems, select and implement corrective strategies to prevent or mitigate the safety challenges, and monitor their effectiveness.

To support the NRC accident management work relating to instrumentation, a program is being conducted with the objectives of:

1. Developing a methodology that will identify (a) the information needed to understand the status of the plant during a broad range of severe accident conditions including corrective actions, (b) the existing plant measurements which could be used to directly or indirectly supply these information needs, (c) the potential limitations on the capability of these measurements to function properly under the conditions that will be present during a wide range of postulated severe accidents, and (d) the conditions in which information from the measurement systems could mislead plant personnel.

2. Applying the developed methodology to a pressurized water reactor (PWR) with a large, dry containment to assess the capabilities and limitations of the methodology and to identify, on a generic basis, potential limitations in the instrumentation.

3. Evaluating a typical severe accident sequence for a PWR with a large, dry containment to test the methodology's capability to assess the availability of information for sequences identified in probabilistic risk assessments.

To satisfy the first objective, a four-step methodology was developed to identify nuclear power plant information needs during severe accidents and to determine the extent to which these needs could be met by instrumentation currently in use at the plants. These steps are listed as follows:

1. A hierarchical tree structure is used to identify the relationships between plant safety objectives, challenges to the safety objectives, mechanisms that cause the challenges, and strategies that would mitigate or prevent the mechanisms.

2. Each branch point in the safety objective trees, developed in Step 1, is reviewed to determine what information would be needed to decide whether the plant is either approaching or at a state that would correspond to that branch point. Possible sources of this information (feedwater flow, reactor coolant temperature, containment pressure, etc.) is then identified and assessed to see how well the information at the plant represents that which is needed.

3. The instrumentation that exists at the plant that will supply the needed information is identified.

4. The information which may mislead personnel involved in accident management is identified based on the lack of information that is needed to clearly distinguish individual branch points on the safety objective trees.

The information-needs methodology was applied to a PWR with a large, dry containment to accomplish the objective. This application was based on typical plant features and instrumentation but would not be totally representative of a particular plant. The results from this application can be summarized as follows:

- Safety objective trees were developed. Although the severe accident information presented on the trees is not new or unique, the structure of the trees allows easy visualization of what is important for a broad spectrum of severe accidents. This broad perspective also

provides insights for identifying the situations in which strategies may be effective in preventing or mitigating a severe accident.

- The assessment of information needs for a PWR with a large, dry containment indicate there is not sufficient instrumentation to understand the status of the core during heatup and relocation. There is also insufficient information to determine whether there is excessive energy transfer to the vessel lower head and whether failure should be expected. The lack of information in these two areas increases the difficulty in making a decision of whether to use plant resources for strategies relating to preserving vessel integrity or to use strategies aimed at preserving containment integrity.

- About twenty information needs exist for which personnel involved in accident management have the potential to be mislead because they must rely on interpretation of instruments that do not directly supply the needed information. Five of these needs were judged to be important and could be grouped into two types of information needs: (a) those related to the inability to understand core status following the initiation of fuel melting, discussed above, and (b) those related to the inability to determine the location of flow bypass or leaks from the containment, such as the location of an interfacing system loss–of–coolant accident.

- There may be instrumentation or other means that can be used to provide information that is not currently available. For example, source range instruments and self–powered neutron detectors may supply information on core status during relocation, but this information would require the capability for special interpretation, which does not currently exist. Computational aids that interpret or extend the use of existing instrumentation have the potential to provide additional needed information.

To meet the third objective, a typical severe accident sequence for a PWR with a large, dry containment was used to (a) demonstrate the use of information–needs methodology along with the results of its application, and (b) provide detailed information on possible limitations for the instrumentation during a specific, severe accident sequence. An evaluation process was developed to fulfill the purposes described for this assessment. This process used the following steps:

1. A detailed definition of the conditions and timing for the specific sequence for evaluation was developed, and the accident sequence was divided into phases based on the timing of key events and the phenomena that occur.

2. The conditions under which the instrumentation can reasonably be expected to fail or provide faulty information were identified. This information was obtained from the qualification reviews for the plant instruments as they pertain to Regulatory Guide 1.97.[1]

3. The availability of the instruments during each phase of the sequence was determined using the instrument failure conditions developed in the previous step. The types of failures considered were: support system failures, failures resulting from exceeding environmental qualification conditions or conditions determined by testing or analysis, and failures resulting from exceeding the instrument range. Also, needed instrumentation that was not installed was identified.

4. The safety objective trees and information needs developed for a PWR with a large, dry containment were used along with the sequence definition to identify, on the trees, which safety functions were challenged, the mechanisms that caused the challenges, the strategies used for prevention or mitigation, and the information and instrumentation needed to follow the severe accident sequence as it progressed through each of the four phases.

5. The information needs developed in Step 4 were compared to the information available for each phase of the severe accident sequence, as identified in Step 3. This comparison determined the availability and limitations of plant instrumentation for accident management during each phase of the accident sequence.

The process above was applied to a risk–important sequence for a PWR. This sequence was a pump seal rupture loss–of–coolant accident with the loss of injection and eventual loss of containment spray. In the evaluation of the sequence, it was conservatively assumed that all instruments failed when their qualification conditions were exceeded. Important plant-specific findings were:

- Sufficient information exists to manage the accident up to the time prior to the onset of fuel and cladding melt.

- During the time period between the initiation of core melt and core relocation into the reactor vessel lower head, there may be faulty indications or no information available on system parameters within the reactor vessel for accident management since the instrument qualification limits are exceeded.

- There is no information available to accident management personnel to indicate an approach to vessel lower head failure.

- Following lower head failure, it is possible that faulty indications will exist or that no information will be available to monitor containment response, since all of the containment instrumentation qualification limits would be exceeded.

- The range of several instruments important to accident management were exceeded during the sequence, including the hot and cold leg resistance temperature detectors (RTDs) and core exit thermocouples. Therefore, these instruments would not provide reliable information for the latter stages of the sequence even if they continued to operate after their qualification conditions were exceeded. For some PWRs, the containment pressure range may also be exceeded, although this was not the case for the plant used in this study.

- To circumvent the faulty instrument indications or their unavailability due to support system failures and severe environmental conditions, computational aids could be used to supply the missing information.

This application did not identify any significant shortcomings in the use of information-needs methodology. However, the assumption that all instruments failed when their qualification conditions were exceeded may be too simplistic and conservative. If a more realistic assessment of this situation is desired, a detailed evaluation of measurement survivability would be needed for a range of possible severe accidents. This evaluation would need to consider the entire instrument system that is exposed to harsh conditions, including the transducer, cabling, electronics, etc. It is recommended that an environmental envelope encompassing risk-important severe accident conditions be developed to provide insight for the range of conditions that would be needed for an evaluation of instruments. The development of this environmental envelope is planned for a PWR and boiling water reactor (BWR) in FY-1990.

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ACCIDENT MANAGEMENT INFORMATION NEEDS VOLUME 1 – METHODOLOGY DEVELOPMENT AND APPLICATION TO A PRESSURIZED WATER REACTOR (PWR) WITH A LARGE, DRY CONTAINMENT

1. INTRODUCTION

Accident management is an essential element of the NRC Integration Plan for the closure of severe accident issues. This element will consolidate the results from other key elements such as Individual Plant Examination (IPE), Containment Performance Improvement (CPI), and Severe Accident Research Programs (SARP) in a form that can be used to enhance the safety programs for nuclear power plants. Accident management will ensure that planned actions and preparatory measures are developed which will enhance the capability of nuclear power plant personnel to effectively manage severe accidents. The NRC has identified five general areas in which the risks associated with severe accidents can be further reduced through accident management.[1] These five areas are: (1) accident management strategies, (2) training, (3) guidance, (4) instrumentation, and (5) decision-making responsibility.

Instrumentation was included as one of the five areas because of its importance to the success of personnel involved in severe accident management. Plant personnel (reactor operators, shift technical advisors, technical support center personnel, etc.) are responsible for diagnosing the occurrence of an accident, determining the extent of the challenge to plant safety, monitoring the performance of automatic systems, selecting strategies to prevent or mitigate the safety challenge, and implementing the strategies and monitoring their effectiveness. Without adequate plant status information and guidance to ensure its proper use, these operating personnel cannot reliably identify and accomplish the actions necessary for accident management.

The safety-related instrumentation installed in nuclear power plants is primarily designed and qualified for preventing and mitigating accidents that are less than or equal to the severity of a design basis accident. The ability of the instrumentation to supply the information needed for management of severe accidents has not been investigated in a comprehensive manner for conditions typical of a broad range of severe accidents. Therefore, the objective of the work presented in this report is to aid in determining the extent to which existing plant instrumentation is capable of supplying severe accident information by:

1. Developing a methodology that will identify: (a) the information needed to understand the status of the plant during a broad range of severe accident conditions including corrective actions, (b) the existing plant measurements which could be used to directly or indirectly supply these information needs, (c) the conditions in which information from the measurement systems could mislead plant personnel, and (d) the potential limitations on the capability of these measurements to function properly under the conditions that will be present during a wide range of postulated severe accidents.

2. Applying the developed methodology to a pressurized water reactor (PWR) with a large, dry containment to assess the capabilities and limitations of the methodology and to identify, on a generic basis, potential limitations in the instrumentation.

3. Evaluating a typical severe accident sequence for a PWR with a large, dry containment to test the capability of the methodology to assess the availability of information for sequences identified in probabilistic risk assessments.

The remainder of this report describes how the three elements of the objective for this work were accomplished, and it also shows the results. Section 2 describes the methodology developed for identifying the information needs for management of severe accidents (element 1 above). Section 3 describes the results from the application of the methodology to a PWR with a large, dry containment (element 2 above). Section 4 presents results from the application of the methodology to a specific severe accident sequence for a PWR (element 3 above). The summary and conclusions are presented in Section 5, and references are listed in Section 6. Appendices are used for documenting the information developed during the application of the methodology to both a PWR and a specific severe accident sequence. Volume 2 of this document contains the appendices.

2. METHODOLOGY AND APPROACH

The methodology developed to identify the information needed to manage a severe accident and to determine the ability of existing instrumentation systems to supply these needs is described in this section. This section also provides a description of the evaluation process used to assess the results from the application of the methodology for their applicability to a specific severe accident sequence. An example application of this methodology to a PWR with a large, dry containment is presented in Section 3, while use of the evaluation process for a PWR with a large, dry containment is given in Section 4.

2.1 Information–Needs Methodology

A four–step approach was developed for identifying nuclear power plant information needs during severe accidents and for determining the extent to which these needs will be met by information currently in use at the plants. These steps and their relationship are illustrated in Figure 1. A brief description of the purpose and products for each step is presented, followed by a more detailed description of the methodology for the individual steps.

The purpose of the first step in this approach is to identify the high level safety objectives for the plant and to provide a means to relate these safety objectives to accident management strategies that have been identified for accomplishing these safety objectives. The relationships identified in this step can be displayed in the form of a hierarchical tree that provides insights on the types of information that would be necessary to ensure that the plant safety objectives for severe accidents are met. The product of the first step would be a set of safety objective trees that identify the relationships between safety objectives, challenges to these safety objectives, and strategies that would mitigate or prevent these challenges.

The purpose of the second step is to consider each branch point in the trees developed in Step 1 and determine what information would be needed to decide whether the plant is at a state that would correspond to each branch point. Once the information needed to identify the positions on the tree have been determined, the possible sources of this information (feedwater flow, reactor coolant temperature, containment pressure, etc.) would be identified and assessed to see how well the information at the plant represents the information needed. The product of this step is the identified information needs and an assessment of the availability of this information at the plant.

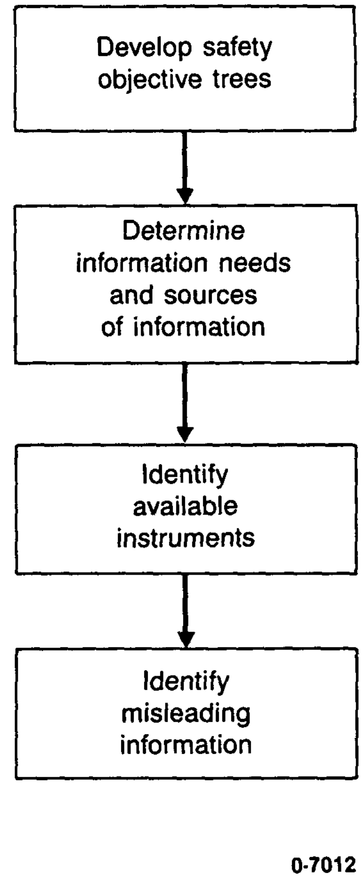

Figure 1. Steps in methodology development.

The purpose of the third step is to identify whether the instrumentation that exists at the plant will supply the needed information identified in Step 2. The product of this step would be a means of relating existing plant instrumentation to information needs and an identification of information needs that are not supplied by existing information.

The purpose of the final step is to identify situations in which the operator may be misled based on the lack of information needed to clearly distinguish the difference between individual branch points on the tree or the types of information that are available. The discussion below provides a more complete description of each of these steps.

2.1.1 Develop Safety Objective Trees. The first step in the development of the methodology utilizes a "top down" evaluation that requires an identification of the top-level objectives of severe accident management. These objectives were based on the NRC definition of accident management (see Reference 1):

> Accident Management encompasses those actions taken during the course of an accident by the plant operating and technical staff to: (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) maintain containment integrity as long as possible, and (4) minimize offsite releases.

The four items listed in this definition are appropriate as statements of the safety objectives for accident management. Use of the first objective in the development of a methodology for *severe* accidents was not considered to be appropriate since core damage would have already occurred in order for the accident to progress to the stage where it would be considered to be severe. The remaining items were selected as the safety objectives for severe accident management and were restated as: (1) prevent core dispersal from the vessel, (2) prevent containment failure, and (3) mitigate fission product release.

These three top-level objectives for severe accident management can be related to actions, generally called strategies, that can be used to ensure that the objectives are met if an accident occurs. In order to ensure that these safety objectives are met, certain critical plant conditions, or *safety functions*, must be maintained within acceptable limits. An accident will present *challenges* to the safety functions which have the potential to cause the safety functions to exceed the acceptable limits. These challenges are caused by different *mechanisms* that occur in the plant. Finally, various *strategies* can be identified and implemented for preventing or mitigating the mechanisms that cause the safety function challenges.

The categories described above—safety objectives, safety functions, challenges, mechanisms, and strategies—form a natural hierarchy that defines the roles of personnel and equipment involved in accident management. Identification of the various levels in the trees is a logic-driven iterative process that requires input from experts in severe accident behavior and personnel with plant operations experience. Figure 2 presents an example that shows one branch of a safety objective tree for the second safety objective, Prevent Containment Failure, for a PWR with a large, dry containment.

The completed safety objective trees are used in the second step of the methodology as a tool to systematically determine the operating staff's information and measurement needs. It is also possible to evaluate the tree structure for specific severe accident scenarios to determine the effects of the scenario on the safety objectives, to identify challenges to the safety functions, to assess those strategies that are disabled by the event, and to choose those remaining strategies that are appropriate to mitigate safety function challenges. The use of the trees for evaluation of a specific severe accident sequence is discussed in detail in Section 4.

2.1.2 Determine Information Needs. The types of information needed for severe accident management can be identified by considering the tasks that must be accomplished to support the severe accident management safety objectives. These tasks or activities include:

1. Monitoring the status of the safety functions
2. Detecting challenges to the safety functions
3. Identifying, if possible, the specific mechanisms that could be causing the safety function challenges
4. Selecting and implementing strategies for maintaining or restoring challenged safety functions
5. Monitoring the performance of the strategies to determine their effectiveness.

Each of these activities can be related to a branch point on the safety objective trees discussed in Section 2.1.1 above.

To identify information needs, the branch points in these trees are examined to decide what information is necessary to (a) determine the status of the safety functions in the plant, i.e., whether the safety functions are being adequately maintained within predetermined limits, (b) identify plant behavior (mechanisms) or precursors to this behavior which indicate that a challenge to plant safety is occurring or is imminent, and (c) select strategies that will prevent or mitigate this plant behavior and monitor the implementation and effectiveness of these strategies. The information needs for the challenges to the safety functions are not examined since the summation of the information needs for all mechanisms associated with a challenge comprise the information needs for the challenge itself.

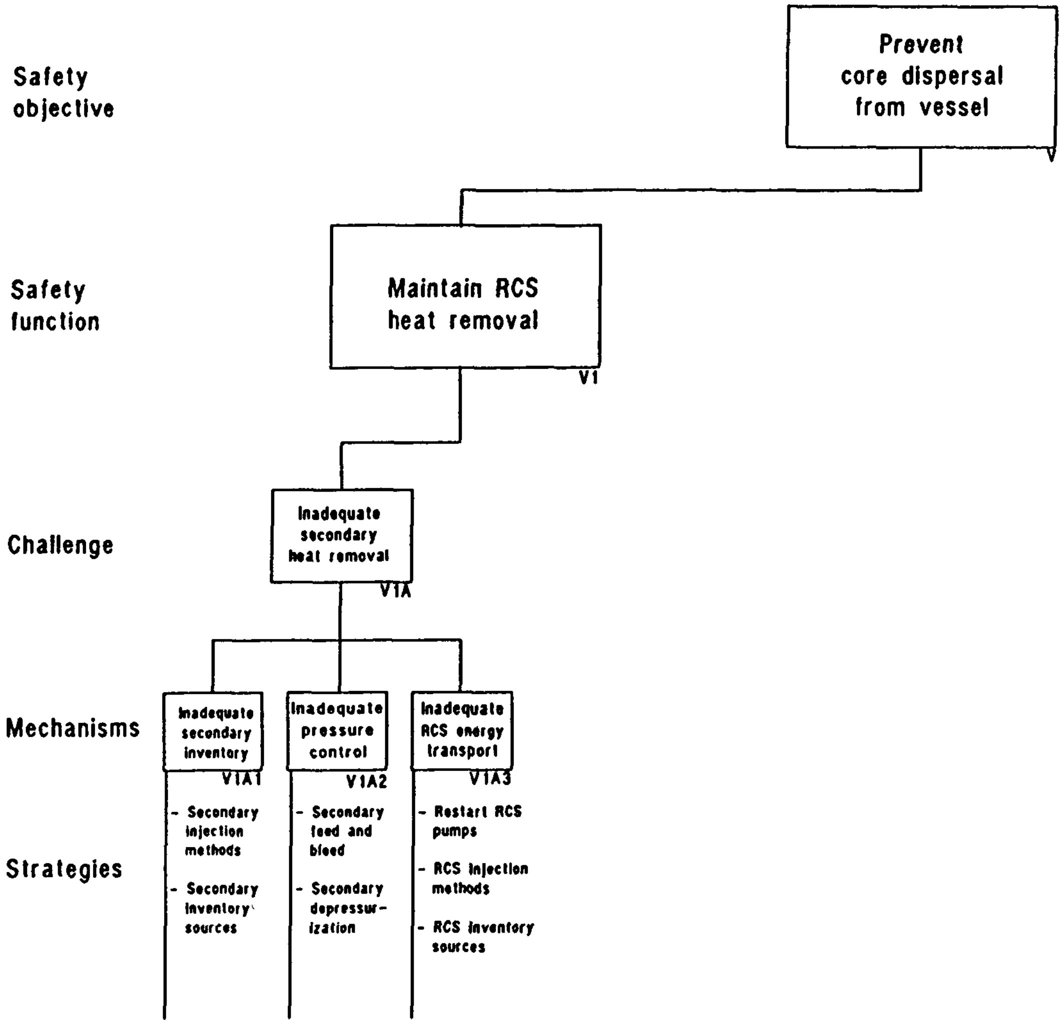

Figure 2. Example of one branch of a safety objective tree.

To aid in the systematic identification and display of the accident management information needs, a table-based format was developed. Table 1 shows an example of the structure of this table. The rows on the table correspond to the five levels of information listed previously which were derived from the levels of the safety objective trees. The first section (row) of the table contains the information needs that relate to the safety function. This section is used to describe the information needed to determine whether the safety function is being maintained within the accepted safety limits. The second section (row) of the table displays information to identify a specific mechanism that may be a challenge to a safety function. Two different categories of information are important for identifying mechanisms: indicators and precursors. The indicators include information that identifies when a mechanism is actually occurring and challenging a safety function. The precursor information identifies whether a mechanism would be expected to occur in the future based on currently available information.

The final three categories (rows) relate to strategy selection and evaluation. The Selection Criteria category identifies the information needed to determine which strategies should be selected for a given situation, including consideration of the plant conditions under which the strategy can operate and be effective. The Strategy Initiation category gives the information needed for the operating staff to determine whether a strategy has been implemented as intended. The Strategy Effectiveness row describes the information needed to determine whether the strategy is having its intended effect; that is, whether implementation of the strategy is having a beneficial effect on the status of the safety function that is being challenged.

Table 1. Example structure of the information needs table

	Information Needs	Direct Information Source	Indirect Information Source	Available Instruments	Potential Instruments
Safety Function					
Mechanism	Indicator				
	Precursor				
Strategy	Selection Criteria				
	Strategy Initiation				
	Strategy Effectiveness				

The respective columns in the table format include the identified information needs, the sources of the information categorized as to how well they represent the information needs, and the existing measurements that could supply the needed information. The information sources are subdivided into those that are considered to be either direct or indirect. A direct information source is one that can be used to provide information that will positively determine the presence or absence of a specific condition on the safety objective tree. For example, for the safety function addressing pressure control, a pressure measurement is a direct information source for understanding challenges to the safety function. An indirect information source can be used to infer the needed information, but there may be conditions where the information source may provide ambiguous results. For example, core exit temperature may provide reasonable information for fuel cladding temperatures for some system conditions but would not provide accurate indication for other combinations of system flow and fluid conditions.

Development of the input to the rows and columns requires the expertise of personnel with diverse backgrounds. A team of personnel with operations, instrumentation, and severe accident experience are needed to produce the needed information.

2.1.3 Identify Available Information. Instruments that have the potential to supply information needs can be identified from the many instruments available at the plant by using various specific sources, such as piping and instrumentation diagrams, system instrument lists, and documentation showing compliance with Regulatory Guide 1.97.[2]

For the severe accident conditions represented by the information needs, there may be some information needs that existing instrumentation will not have the capability to supply. In addition, there may be existing instrumentation that does not have the needed range or is not qualified for conditions typical of those that will occur for some severe accidents. To assess these plant instrument limitations for severe accidents, a compilation of plant conditions that correspond to the identified information needs for a wide range of severe accidents would be required. This compilation would have to rely on the results of analyses performed with one or more severe accident computer codes. Ranges for existing measurements and results from their environmental qualifications could then be compared to parameters calculated during the analysis of severe accidents. Then, judgements could be made regarding the capability of existing measurement equipment to survive harsh environments and supply accurate, unambiguous information.

The selection of representative plant conditions to judge the capabilities of existing instruments would be difficult because there are several different severe accident computer codes that provide results that extend from conservative to what is considered "best estimate." Since the process of selecting representative conditions would be complex and time consuming, assessment of generic range and qualification conditions for instruments is outside the scope of this study. However, a strategy–specific evaluation of ranges and qualification conditions was performed for a single severe accident sequence, which is reported in Section 4.

2.1.4 Identify Misleading Information. There are several ways in which the information supplied by the instrumentation could mislead personnel involved in accident management. Examples include:

1. Using information from instruments that include large error components because they are operating outside of their specified operating conditions

2. Using information that is in error because the instruments are either damaged or failed

3. Inferring information from an indirect source without consideration of implicit limitations.

Alerting the personnel involved in accident management that instrumentation is outside its specified range of qualification conditions could be easily accomplished if the environmental conditions for the instruments were measured or could be estimated based on the characteristics of the accident. A determination of the amount of error in the information would be much more difficult, but could be based on the type of instrument and the known or expected conditions. For the second example, identification of measurements that are failed or damaged could be determined through cross comparisons with similar or supporting instruments, which is a practice that could be used during the response to an accident.

The use of information from indirect sources could be misinterpreted in a way that would mislead accident management personnel. Misinterpretation of information may occur due to the lack of understanding of the limitations of the instruments. For example, use of the core exit temperatures to infer core cooling could mislead the operators if an injection path is restored to the hot leg. The injection into the hot leg may significantly reduce the core exit temperature; however, for low injection rates, portions of the bottom of the core may continue to heat up and melt. In this situation, the information supplied by an instrument appears to fulfill the information need for verification of core cooling; however, core degradation continues due to a misinterpretation of an indirect source of information.

Results from the table-based format on information needs described in Section 2.1.2 can be used to identify the situations in which the greatest potential exists for misleading the operator. This identification is accomplished by determining which information needs do not have direct information sources and must rely only on indirect sources. Direct information sources would be difficult to misinterpret, if properly displayed, since they correspond one-for-one with the information need. As an example, if the safety function of interest is containment pressure, then the information need is the containment pressure history and the direct information source is the containment pressure measurement. However, if the information need does not have a direct information source, then assessment of the situation and determination of what action should be taken must rely on indirect information sources. The potential for misinterpreting the accident conditions and misleading the operator increases with the use of indirect sources of information since there can be ambiguities in the information they provide. For example, interpretation of the core exit thermocouple readings as indicative of the fuel rod cladding temperature has the potential to mislead the personnel involved in accident management for a wide range of potential severe accident conditions unless the limitations typical of those discussed above and the possibility for large uncertainties is included in the interpretation. Therefore, if the number of information needs that are being supplied by indirect sources is large, the potential to mislead the operator is even greater.

2.2 Process for Evaluation of Information–Needs Methodology

This section describes an evaluation process that can be used to assess results obtained from the application of the methodology described in Section 2.1 for a specific severe accident sequence. The purpose of this assessment is to identify potential problems with the methodology and the results of its application to a PWR with a large, dry containment. The assessment will also provide specific information on possible limitations for the instrumentation during a specific severe accident sequence.

The evaluation process that has been developed to fulfill the purposes described for this assessment is illustrated in Figure 3. A brief description of each individual step in this process follows.

2.2.1 Define Sequence Conditions and Timing. The purpose of the first step, Define Sequence Conditions and Timing, is to select an accident sequence, or collection of sequences, for which the information-needs evaluation is to be performed and provide a detailed definition of the conditions and timing for the sequence. In developing this definition, the following actions should be taken to obtain the information:

1. List the initial conditions for the sequence.
2. Identify the equipment and support system failures.
3. Obtain key parameter plots of the pertinent reactor coolant system (RCS) and containment, thermal-hydraulic responses. These plots should include RCS pressure, temperature, and liquid level along with core exit temperatures and containment pressure and temperature.
4. Tabulate the key sequence of events for the severe accident sequence.
5. Divide the accident sequence into phases based on the timing of key events and the phenomena that are occurring.

The information developed during the definition of sequence conditions and timing will serve as the basis for determining the instrumentation performance when exposed to differing levels of severe accident conditions. Examples of sequence phases would be (a) the time prior to core uncovery or (b) the time between vessel failure and containment failure.

2.2.2 Identify Instrument Failure Conditions. The purpose of the second step is to identify the conditions under which the instrumentation can reasonably be expected to fail. This information can be obtained from the qualification reviews for the plant instruments as they pertain to Regulatory Guide 1.97 (see Reference 2). To accomplish this, it is necessary to:

1. Identify the instrument qualification ranges. In other words, determine the maximum environmental temperature, pressure, humidity, radiation, and seismic levels that can exist without the instruments showing inaccurate information to the operators during an event.

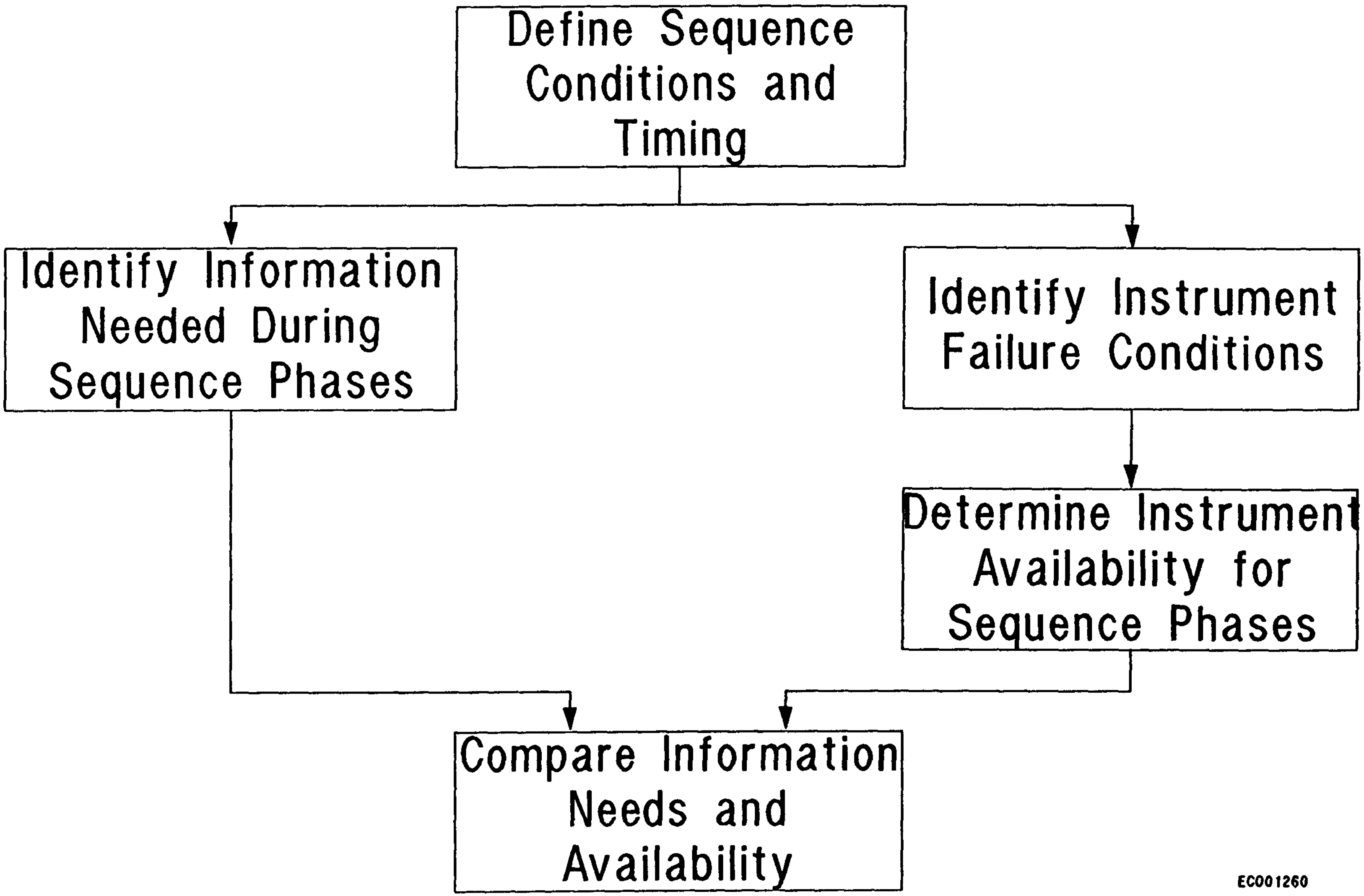

Figure 3. Process for evaluating a severe accident sequence.

2. Identify the key system ranges (pressure, temperature, radiation, etc.) experienced during each phase of the accident using the information developed in Step 1.

3. Identify the failure conditions for the instruments. A bounding case would be to assume the instrumentation fails when its qualification conditions are exceeded. This case is known to be conservative based on the experience gained during the Three Mile Island accident where some instrumentation continued to operate under severe conditions. A more realistic identification of failure conditions could be obtained through testing or through analysis of individual instrument types. However, this assessment is beyond the scope of this study.

   Categorization of the instruments, based on their qualification conditions, could be used to readily identify the equipment available during the accident phases. Since instruments have different qualification conditions imposed for both design basis accidents (DBA) and non–DBA–type events, with both harsh and non–harsh environmental conditions, these categories could be used.

4. Identify other equipment failures that would occur as a consequence of the accident conditions.

The results from identifying instrument failure conditions will be used in future steps to assist in determining what instrumentation is available during the identified phases of the accident sequence.

2.2.3 Determine Instrument Availability for Sequence Phases. The purpose of the third step is to determine the availability of the information needed from the instruments during each phase of the sequence. This determination is accomplished using the instrument failure conditions developed in Step 2. The types of failures that would restrict the information available during each phase, and should be considered in this determination include:

- Failures due to support system failure (electrical power, instrument air, service water, etc.)
- Failures resulting from exceeding environmental qualification conditions or conditions determined by testing or analysis
- Failures resulting from exceeding the instrument range
- Failures resulting from the lack of instrumentation available to provide the needed information.

The matrix of sequence phases and failed instruments will be used for developing comparisons with the information needs for the sequence to determine what necessary information is not available as the sequence progresses.

2.2.4 Identify Information Needed During Sequence Phases. The purpose of the fourth step is to identify the information needed to maintain the plant safety functions and to select and monitor the effectiveness of strategies. The information needs could be identified by applying the methodology described in Section 2.1 to the plant as a whole, or they could be identified for a specific sequence by slightly modifying the methodology to consider challenges resulting from only that sequence. To track the flow of information needs as they develop throughout the accident sequence, information needs are identified by noting on the safety objective trees and the tables described in Section 2.1.2 the status of the safety functions, the mechanisms causing the challenges, and the relevant strategies for prevention or mitigation. This notation would be used for each phase of the sequence.

For each safety function, the following classification system should be used to denote the status for each phase of the sequence:

- The safety function is not challenged, so the status is okay
- The safety function is challenged
- The safety function has been lost, i.e., the core is not coolable inside the vessel
- Information deficiencies exist.

For each mechanism, the following classification system should be used to denote the state for each phase of the sequence:

- The mechanism is present and the safety function is being challenged

- The mechanism is not present, but the potential exists for this mechanism to occur in the future

- The mechanism is not present nor is it expected to occur

- Information deficiencies exist.

For each strategy, the following classification system should be used to denote the status for each phase of the sequence:

- The strategy is available

- The strategy is unavailable as a result of failed equipment

- The strategy is active, i.e., it is being utilized during the sequence phase

- There may be strategies identified on a set of generic safety objective trees that are not installed at the particular plant being evaluated

- There are information deficiencies for the strategy.

This step will produce a set of safety objective trees and a set of information–need tables that will indicate which safety functions are being challenged, the mechanisms causing the challenges and what strategies are available for use. Also, the information needs and instrumentation required to follow the severe accident sequence as it progresses through each phase will be shown. This status information will be used in the next step for comparison to the instrumentation that is available.

2.2.5 Compare Information Needs and Availability. In the final step, a table is developed to compare the information needs to the information available for each phase of the severe accident sequence using the information developed in the preceding steps. The purpose of the table is to present the final product, which is an assessment of the availability of plant instrumentation for accident management during each phase of the particular sequence in question. This table is referred to as an Accident Management Information Assessment table. Information regarding the potential to mislead the operator or to add confusion of the accident to management can also be obtained from this table by examining where the information needs do not exist or are supplied by indirect information sources.

An application of the assessment process described above is presented in the application to a specific severe accident sequence in Section 4. However, before the application to the sequence is presented, the methodology is applied to a PWR with a large, dry containment, as explained in Section 3. Section 3 also identifies the needed instrumentation for accident management as established by the safety objective trees using the table–based format discussed earlier for this plant type. This tabular information is then used to perform the assessment presented in Section 4.

3. APPLICATION OF INFORMATION–NEEDS METHODOLOGY TO A PWR WITH A LARGE, DRY CONTAINMENT

The methodology described in Section 2.1 has been applied to a PWR with a large, dry containment. This application was based on a knowledge of the plant features and instruments that are typical of some PWR plants but may not be totally representative of a particular plant. The objective of this application is to assess the methodology, demonstrate application to a plant, and provide an important evaluation of the status of information needs for this general class of plants. The results of this evaluation can also form the basis for additional studies or plant–specific studies to determine the status of plant information needs. Section 3.1 describes the development of safety objective trees for a PWR with a large, dry containment. Section 3.2 provides the information needs that are based on this safety objective tree. Section 3.3 provides an identification of existing measurements from a specific plant to fulfill the information needs at a representative plant, and Section 3.4 presents results which assess the potential to mislead the operator based on the results from the previous three sections.

3.1 Safety Objective Tree Development

Three safety objectives were identified for a PWR with a large, dry containment in Section 2.1 based on NRC information concerning accident management. Since this assessment and evaluation is being performed for severe accidents, inclusion of the Prevent Core Damage[a] safety objective is not appropriate. The current emergency operating procedures are intended to address this objective. The remaining safety objectives which are used in this development are: (a) Prevent Core Dispersal from Vessel, (b) Prevent Containment Failure, and (c) Mitigate Fission Product Section 2.1, has been applied to a PWR with a large, Release from Containment. The methodology for developing safety objective trees, described in dry containment. The development was not based on a specific plant, but on information that is generally typical of some Combustion Engineering and Westinghouse PWRs. Personnel with expertise in severe accidents and PWR operations were used to develop and review the trees.

a. Core damage is considered to have occurred when the fuel rod cladding has ruptured and fission products have been released into the reactor coolant system (RCS).

It should be noted that the strategies shown on each safety objective tree are only examples and are not considered to be a complete set for the mechanisms and safety functions. Some strategies may not be practical under certain circumstances but are included to illustrate that there may be conflicting requirements for some plant safety functions. Most strategies presented are general in nature and would require further evaluation to determine whether they would adequately maintain the appropriate safety functions for a specific plant configuration. A brief description of the safety objective trees developed for the three safety objectives is discussed below.

3.1.1 Safety Objective Tree: Prevent Core Dispersal from Vessel. The safety objective for preventing the core from being dispersed from the vessel into the containment is important for both short–term and long–term accident management because the strategies and actions associated with mitigating the effects of a degraded core are less complicated when the core material is retained within the boundary of the reactor vessel. In addition, there are relatively large uncertainties in the response of the containment parameters when molten core material exits the vessel and interacts with the containment structures. These uncertainties can be avoided if the core remains within the vessel.

The structure of the Prevent Core Dispersal from Vessel safety objective tree is shown in Figure 4. Three safety functions were identified that would support this safety objective. These safety functions were selected based on an understanding of the types of safety functions that are important for the previous phase of accident management (prevent core damage), together with the recognition that the complexity of system behavior and the extent of system failures during a severe accident limits the range of available actions. For the Prevent Core Damage phase, the safety functions traditionally used for RCS–related accidents are (a) Maintain Core Heat Removal, (b) Maintain RCS Heat Removal, (c) Maintain Reactivity Control, (d) Maintain RCS Inventory Control, and (e) Maintain RCS Pressure Control. Once core damage has occurred, the focus of accident management shifts to emphasize the prevention of further core degradation and to ultimately maintain the core within the vessel. In this situation, maintaining core heat removal would be the highest priority. RCS heat removal would also be a high priority since it would be necessary to support

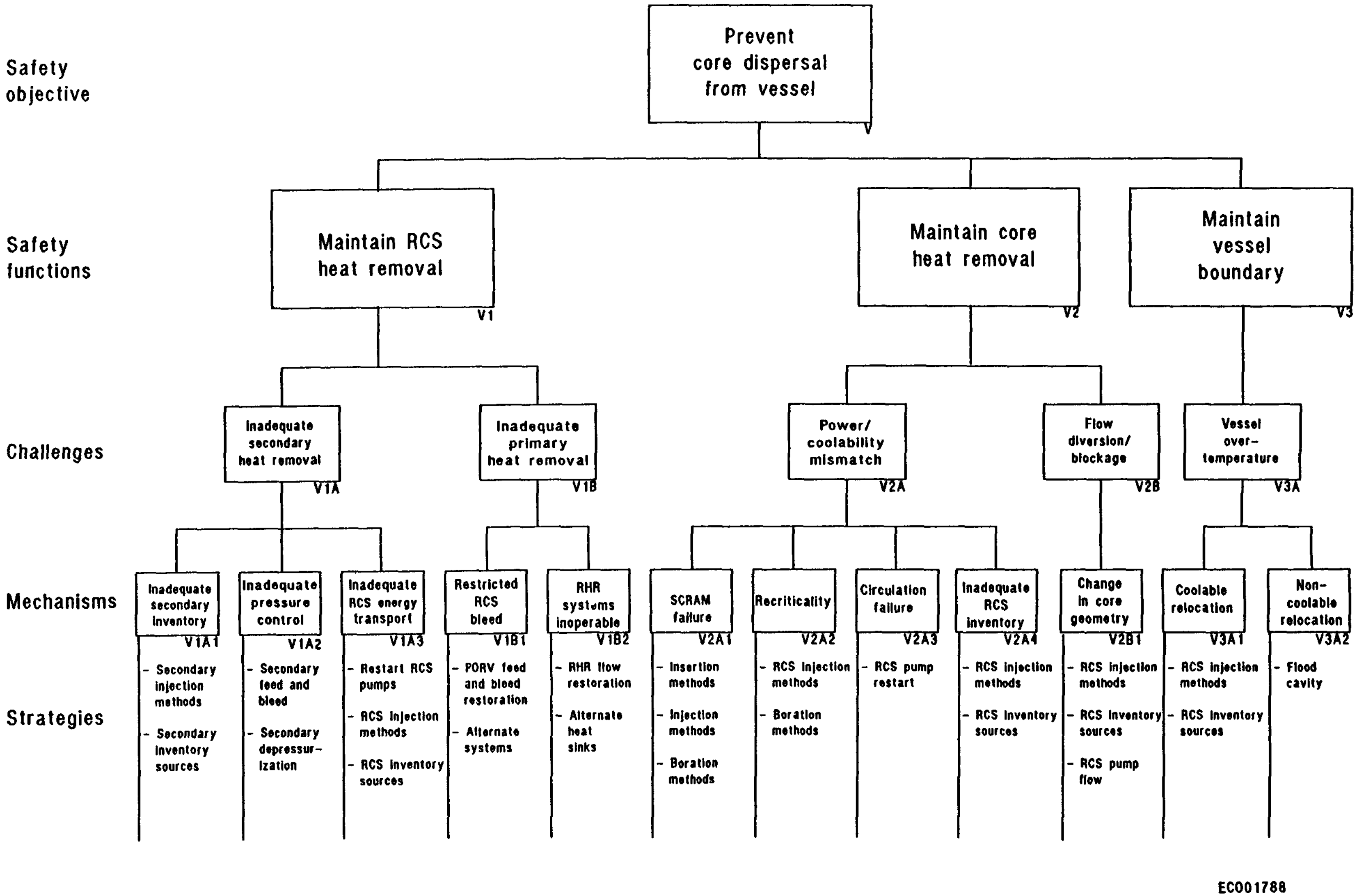

Figure 4. Safety objective tree: Prevent Core Dispersal from Vessel.

long-term core cooling. These two safety functions are designated in Figure 4 as Maintain RCS Heat Removal and Maintain Core Heat Removal. Reactivity control and Inventory control then become mechanisms which would lead to challenges related to core cooling. Pressure control would be included in portions of certain strategies for preventing or mitigating those mechanisms associated with an RCS energy imbalance.

The third safety function, Maintain Vessel Boundary, is intended to maintain a relocated core within the vessel lower plenum.

For ease in relating the discussion to the various levels of the tree, each level has been assigned a unique identifier that is descriptive of its position on the tree. Thus, the Maintain RCS Heat Removal safety function has an identifier called "V1." One of the mechanisms causing a challenge to this safety function is identified as "V1A1." The letter "V" identifies this safety function as relating to the Prevent Core Dispersal from Vessel safety objective tree. Each of the safety functions, challenges, mechanisms, and strategies in this tree are explained below.

3.1.1.1 Safety Function: Maintain RCS Heat Removal. The capability to remove energy from the RCS must be maintained in order to maintain long-term cooling of the core. There are two challenges that influence the capability to maintain RCS cooling. The first challenge, Inadequate Secondary Heat Removal (V1A), would occur if the steam generators could not remove all of the energy being produced in the core. The second challenge, Inadequate Primary Heat Removal (V1B), would be expected if the steam generators were not available to remove heat directly from the RCS.

Challenge: Inadequate Secondary Heat Removal. The mechanisms that would contribute to this challenge are Inadequate Secondary Inventory (V1A1), Inadequate Pressure Control (V1A2), and Inadequate RCS Energy Transport (V1A3). Inadequate Secondary Inventory would be expected to occur when the methods of delivering water to the secondary side of the steam generator had failed or when there was not sufficient water available for delivery. The potential strategies that have been identified to control or prevent this mechanism are Secondary Injection Methods and Secondary Inventory Sources.

The second mechanism, Inadequate Pressure Control (V1A2), would be required if the steam generator secondary side pressure was sufficiently high that the corresponding temperature exceeded the primary side temperature or if the secondary-side pressure needed to be reduced so alternate inventory addition systems could be used. A potential strategy for this mechanism includes opening valves that could relieve steam, but which require manual initiation either locally or from the control room.

The final mechanism for this challenge, Inadequate RCS Energy Transport (V1A3), would occur when the circulation of coolant between the reactor core and the steam generator was not sufficient to transport the necessary amounts of energy. Circulation failure could occur when there is a relatively high energy generation rate in the core and when the reactor coolant pumps are not operating or when natural circulation is insufficient or has been interrupted. The buildup of noncondensable gases could influence this interruption through degradation of heat transfer or reduction of heat transfer area. Strategies for this mechanism are Restart Reactor Coolant (RC) Pumps, RCS Injection Methods, and RCS Inventory Sources. The last two strategies would attempt to increase the inventory of the RCS to enhance natural or forced circulation. The effectiveness of any of these strategies would be strongly dependant on the conditions in the RCS and would have to be examined carefully prior to implementation.

Challenge: Inadequate Primary Heat Removal. The mechanisms that would contribute to this challenge are Restricted RCS Bleed (V1B1) and Residual Heat Removal (RHR) Systems Inoperable (V1B2). Feed and bleed of the RCS has been shown to be an effective method of removing energy if the steam generators are unable to remove sufficient energy. Therefore, the Maintain RCS Heat Removal safety function would not be challenged if the power-operated relief valve (PORV) was fully operable. However, long-term RCS bleed may be restricted during a severe accident as a result of failures in the hardware or operation of the Pressurizer PORVs. The feed portion of the feed and bleed approach is considered part of inventory control for removal of heat from the core and is discussed as one of the mechanisms influencing the Maintain Core Heat Removal safety function.

Potential strategies for Restricted RCS Bleed are (a) the management of battery capabilities and recharging to extend the time of availability of electrical power, (b) the use of gas cylinders or reservoirs to store sufficient gas supplies to allow long-term manipulation of pneumatic PORVs, and (c) protection of PORV controls (solenoid valves, etc.) against over-temperature failure. Use of alternate systems, such as the low-temperature, over-pressure protection system, may also be possible under special circumstances, but would require detailed evaluation.

The RHR Systems Inoperable mechanism would likely result from the failure of pumps or the capability to remove energy through the residual heat removal system heat exchangers. This mechanism would become a concern only when plant conditions were appropriate for the use of the RHR system. The strategies would be very plant-specific and could include finding alternate methods of pumping water from the RCS to the heat exchangers or alternate means of supplying cooling water to the secondary sides of heat exchangers by alternate pumping systems or by alternate water sources.

3.1.1.2 Safety Function: Maintain Core Heat Removal. Energy removal from the core must be restored and maintained to halt the progression of core damage. There are two challenges that influence the capability to maintain adequate core heat removal. The first challenge, Power/Coolability Mismatch (V2A), would occur when there is excess energy being generated in the core compared to the capability of the injected coolant to remove this energy. The second challenge, Flow Diversion/Blockage (V2B), is a special case in which the coolant is restricted from entering the core, or portions of the core, as a result of changes in the geometry of the core material. Examples would be the formation of rubble beds upon collapse of core material or the formation of subchannel blockages resulting from a melt relocation process.

Challenge: Power/Coolability Mismatch. The four mechanisms that would contribute to the Power/Coolability challenge are: SCRAM Failure (V2A1), Recriticality (V2A2), Circulation Failure (V2A3), and Inadequate RCS Inventory (V2A4). The SCRAM Failure mechanism would occur if the control rods did not insert sufficient negative reactivity into the core to enable shutdown to decay heat levels. The strategies for this mechanism would be similar to those for an anticipated transient without scram (ATWS). For some plants, alternate methods of inserting the control rods or alternate methods of injecting and developing additional borated water sources may be needed. Consideration of the unavailability of electrical power or other plant resources such as plant air or service water would need to be considered.

The Recriticality (V2A2) mechanism may occur if the core temperatures are sufficiently high to allow the control rod material to melt and relocate followed by an addition of water that is insufficiently borated. Since water must be present to cause recriticality and increase the power levels, cooling would also take place, and this mechanism may not result in significant relocation of additional core material. However, recriticality is not considered to be an acceptable core condition since adequate cooling could be difficult to ensure for some configurations of a damaged core. The initial strategy would be to identify alternate means of inserting the control rods into the core. Additional strategies would include use of alternate injections methods and sources similar to those described for the Inadequate RCS Inventory (V2A4) mechanism. The presence of the unborated water necessary to cause recriticality would be possible only if normal sources of injection water had been exhausted and an alternate unborated source was being used. A potential strategy for this mechanism would involve providing a means of adding boron to water injected either at the source or directly into the injection lines. Strategies that would inject highly borated water for long periods of time world require an evaluation of the potential for boric acid precipitation to disrupt long-term cooling.

The Circulation Failure (V2A3) mechanism would result in a challenge if there was not sufficient flow to the core to remove the energy being generated. If power levels were above the decay heat levels, forced circulation would likely be necessary. At decay heat levels, natural circulation of either single or two-phase fluids may be sufficient to remove the lower amounts of energy if the steam generator is available for energy removal. The strategies would be similar to those described for the Inadequate RCS Energy Transport (V1A3) mechanism.

The Inadequate RCS Inventory (V2A4) mechanism would result in a challenge to core heat removal if there was not sufficient flow to keep the core very nearly immersed in water. If the core heat generation is at or near decay heat levels with low fuel and cladding temperatures and the geometry of the core has not been seriously distorted, maintaining the core covered with water may provide sufficient cooling to stabilize the core and prevent further degradation. Research is currently being sponsored by the NRC to define the limitations on coolability for a damaged core. Strategies for the Inadequate RCS Inventory mechanism that could provide the necessary inventory include alternate RCS Injection Methods and alternate RCS Inventory Sources.

Challenge: Flow Diversion/Blockage. If there is insufficient cooling and if extensive core degradation and relocation begins, blockages could occur either in subchannels or on a broader scale that would restrict cooling of some portions of the core and challenge the capability to maintain core heat removal. The mechanism causing this challenge has been designated Change in Core Geometry (V2B1). If a geometry change occurs, the core could transform into one or more configurations depending on the specific conditions of the material and the availability of cooling. The geometry could range from a rubble bed to a widespread crust of melted and refrozen core material that supports molten material. The capability to cool the various geometries could require different types of strategies. Unfortunately, there are not accurate means of determining the geometric configuration of the core as an accident progresses, so selection of geometry specific strategies would not be possible. Three general strategies that have the potential to provide cooling for different core geometries include alternate RCS Injection Methods, alternate RCS Inventory Sources, and RCS Pump Flow.

3.1.1.3 Safety Function: Maintain Vessel Boundary. If cooling of the core cannot be established sufficiently early in the accident, relocation of portions of the core to the vessel lower plenum may occur. At this stage of the accident, the safety function related to maintaining the core heat removal has been ineffective, and accident management efforts should be directed toward preserving the integrity of the vessel lower head, shown as the Maintain Vessel Boundary (V3) safety function in Figure 4.

The challenge that would influence the capability to maintain a relocated core within the vessel boundary is the Vessel Over-Temperature (V3A) challenge. Direct mechanical failure of the vessel at conditions other than high temperatures of the vessel wall has been examined. For example, pressurized thermal shock studies are not considered to have sufficient likelihood to be included in this tree. Mechanisms that would contribute to the Vessel Over-Temperature challenge are Coolable Relocation (V3A1) and Non-Coolable Relocation (V3A2). These mechanisms are intended to reflect the status of the core following relocation. If the core relocates in a coolable geometry, a challenge would occur if there was insufficient inventory and RCS heat removal to provide long-term cooling. If, upon relocation, the core material breaks into very fine particles and forms a mass that is relatively impermeable to water or forms very large pieces that are too large to transfer all the energy generated within, or if the material forms a pool in which molten material is in contact with the vessel head, the core material may not be easily cooled. Unfortunately, there are no existing measurements that have the capability to determine whether the core is, or is not, in a coolable geometry. Therefore, although different strategies are identified for these two mechanisms, it is doubtful that there is sufficient information available to determine which strategies are needed. Fortunately, many of the strategies are identical to those used in maintaining core heat removal, so they would likely be in the process of being implemented if the accident had progressed to the point of core relocation.

If the core relocates in a coolable geometry, the potential strategies should include a continuation of inventory addition to maintain water in the lower plenum and continuation of the Maintain RCS Heat Removal safety function. Since this mechanism would occur late in the course of a severe accident, the alternate RCS Injection Methods strategy and the alternate RCS Inventory Sources strategy would both be an extension of the Inadequate RCS Inventory (V2A4) strategies discussed previously. Since there would be a significant amount of time elapsed, recovery of the normal injection modes would be more likely. Strategies for the Maintain RCS Heat Removal safety function would require assessment of the specific equipment available and the plant conditions. Since the RCS would be highly voided and large quantities of noncondensables would be present, the steam generators may not be capable of providing adequate heat removal. Feed and bleed of the RCS appears to be the most successful means to remove RCS energy that is being generated by the relocated core.

If the core relocates in a non-coolable geometry, the only identified strategy that has the potential to prevent vessel failure would be to flood the cavity surrounding the vessel.

3.1.2 Safety Objective Tree: Prevent Containment Failure. The second accident management safety objective is designated as Prevent Containment Failure. This safety objective is important because the containment building is the final barrier that can prevent the release of fission products to the environment in the event of a severe accident. The tree for the Prevent Containment Failure safety objective is shown in Figure 5. Three safety functions were identified for this safety objective which would contribute to preventing containment failure and assuring containment integrity: (1) maintain over-pressure control to prevent structural damage and eventual rupture of the containment, (2) maintain over-temperature control to prevent failure of the containment structures from the effects of excessive temperature, and (3) maintain containment integrity from leakage, bypass, or penetration by internally generated missiles. These safety functions are respectively designated Maintain Pressure Control (C1), Maintain Temperature Control (C2), and

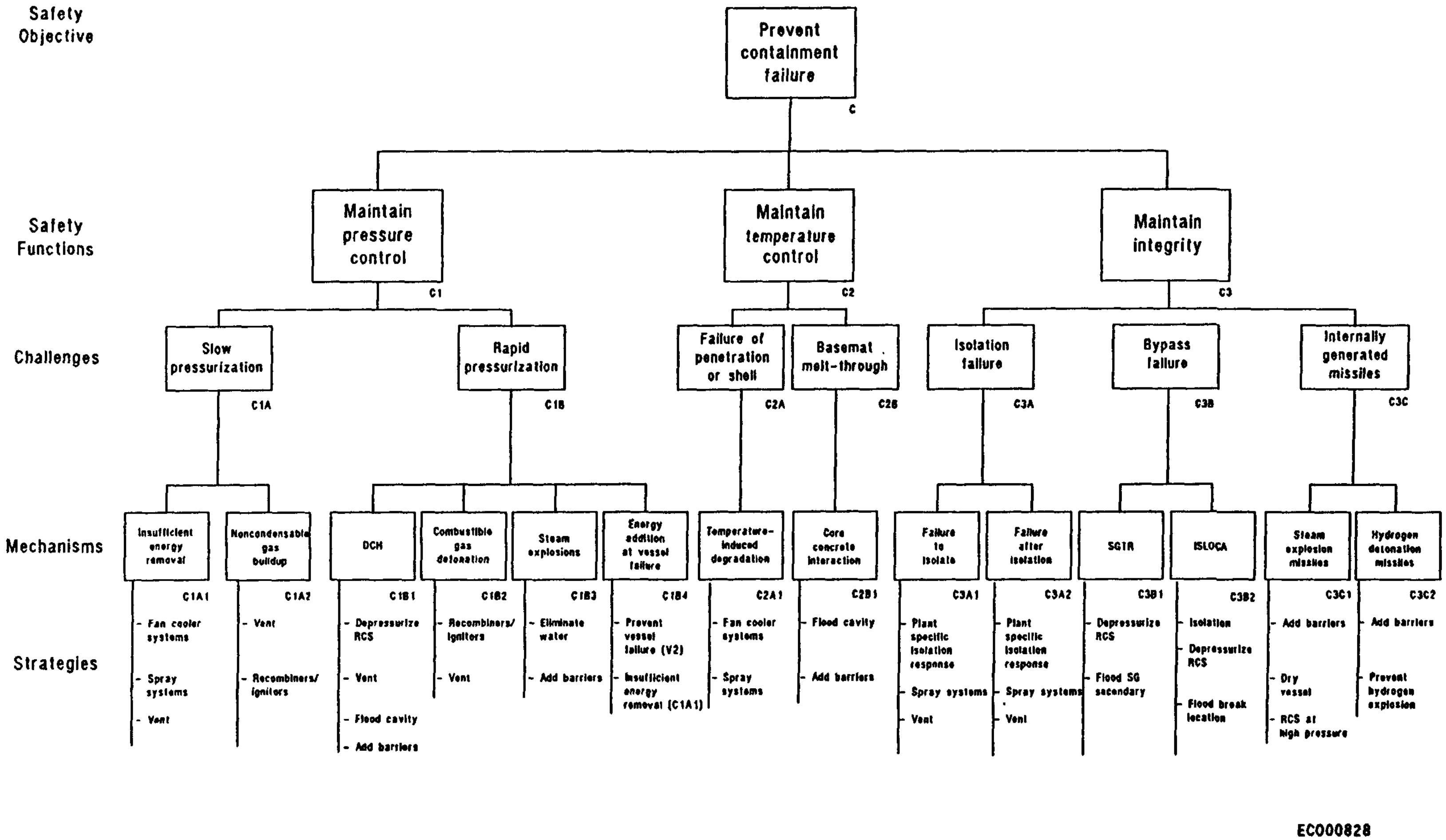

Figure 5. Safety objective tree: Prevent Containment Failure.

Maintain Integrity (C3). Each of these safety functions, along with their challenges, mechanisms, and strategies are explained below.

3.1.2.1 Safety Function: Maintain Pressure Control. There are two challenges that influence the capability to maintain control of the pressure in the containment. These two challenges were identified based on the types of mechanisms that influence pressurization, the potential information sources, and the effect of time available for implementation of potential strategies. In general, a Slow Pressurization (C1A) challenge would require little information to predict or confirm its occurrence, and there would be a significant amount of time to assess and implement strategies. A Rapid Pressurization (C1B) challenge would require a significant amount of information to diagnose the mechanisms and, in most cases, would need preventative strategies to be implemented prior to the occurrence of the challenge.

Challenge: Slow Pressurization. Two mechanisms were identified as contributing to the Slow Pressurization challenge: (1) Insufficient Energy Removal (C1A1), which would occur when heat removal systems fail to operate or could not operate at their required capacity, and (2) Noncondensable Gas Buildup (C1A2) from gases such as hydrogen, carbon monoxide and carbon dioxide. Under some accident conditions, these mechanisms would act together to over-pressurize the containment.

Three potential strategies have been identified for the Insufficient Energy Removal (C1A1) mechanism: Fan Cooler Systems, Spray Systems, and Vent. Venting has not been seriously considered for plants in the United States, but is being considered or used on PWRs with large, dry containments for plants in Europe.

Two potential strategies have been identified for the Slow Pressurization challenge from the Noncondensable Gas Buildup mechanism: Vent and use of Recombiners/Igniters. It is recognized that current recombiners and igniters do not have sufficient capacity to mitigate the effects of severe accidents where large quantities of noncondensable gasses are produced over relatively short periods of time.

Challenge: Rapid Pressurization. Prevention of containment failure during a Rapid Pressurization challenge (C1B) could, in most cases, be difficult since there may not be sufficient time for operations personnel to implement mitigating strategies. For these cases, identification of precursor conditions would be needed to successfully invoke preventative strategies that would be effective prior to the beginning of a rapid pressurization. Four potential mechanisms have been identified that would cause rapid containment pressurization: (1) high pressure melt ejection with sufficient force to disperse the molten core material and cause direct containment heating (DCH), (2) the buildup and detonation of combustible gases, (3) steam explosions that may occur in the containment, and (4) energy addition to the containment from steam and water that is expelled when the vessel boundary fails. These mechanisms are designated: DCH (C1B1), Combustible Gas Detonation (C1B2), Steam Explosions (C1B3), and Energy Addition at Vessel Failure (C1B4).

The DCH mechanism has the potential to cause containment failure, particularly if the containment is at an elevated pressure when core melt ejection occurs. Strategies with the potential to mitigate the effects of DCH are: Depressurize RCS, Vent, Flood Cavity, and Add Barriers.

The Combustible Gas Detonation mechanism can result in large pressure increases in a short period of time. Although the size of a large, dry containment makes the probability of containment failure due to the burning of combustible gases relatively low, two strategies for mitigating this mechanism have been identified: Recombiners/Igniters and Vent.

The likelihood is very small that the Steam Explosions (C1B3) mechanism in the containment would cause sufficient pressurization to rupture a large, dry containment. Potential strategies that could be used to deal with this very low probability event are Eliminate Water from Cavity and Add Barriers in Cavity. It is recognized that the strategy to eliminate water from the cavity is not practical, but is included here to highlight the conflict between this strategy and strategies in which placement of water in the cavity would be desirable, such as those related to the Core Concrete Interaction mechanism, which is discussed on the following page.

The final mechanism for the Rapid Pressurization challenge is a rapid Energy Addition at Vessel Failure (C1B4). This energy addition would result from high-pressure and high-temperature water entering the containment from the RCS and would only cause failure if the large, dry containment was already near the failure limit. The most effective strategies would be those used to prevent failure of the vessel, shown on the Prevent Core Dispersal from Vessel safety objective tree and those associated with maintaining a low initial containment pressure for the Insufficient Energy Removal (C1A1) mechanism of the Prevent

Containment Failure tree. These strategies were discussed earlier.

3.1.2.2 Safety Function: Maintain Temperature Control. There are two challenges that have been identified as contributing to the Maintain Temperature Control (C2) safety function. These two challenges represent different ways in which a large, dry containment can fail from over–temperature. Over–temperature would be expected to cause Failure of Penetration or Shell (C2A) and Basemat Melt–Through (C2B). The possibility of contact of molten material with the containment wall during a high pressure melt ejection was considered, but rejected due to the very low likelihood of this occurrence in a large, dry containment.

Challenge: Failure of Penetration or Shell. Penetration, seal, or shell failure resulting from localized strains in the liner could be induced by the combination of high temperature conditions and containment pressure at levels higher than ambient. The mechanism of Temperature–Induced Degradation (C2A1) could affect a range of penetration and seal locations and types. The potential strategies would use alternate means for operating spray systems and fan cooler systems and would be identical to the strategies described previously for Insufficient Energy Removal (C1A1).

Challenge: Basemat Melt–Through. Basemat melt–through could occur if sufficient molten core material collected in the vessel cavity. The mechanism for melt–through is Core Concrete Interaction (C2B1). The strategies addressing the mitigation of the core concrete interaction are Flood Cavity and Add Barriers. The addition of barriers is a design change rather than a strategy; however, it is included as a strategy since strategies are to be envisioned to include all pre–event measures to prevent or mitigate accidents.

3.1.2.3 Safety Function: Maintain Integrity. The third safety function for the Prevent Containment Failure safety objective is Maintain Integrity (C3). This safety function would be challenged if piping, components, or equipment failed, which would prevent initiation or continuation of containment isolation. Three challenges have been identified: (1) isolation failures that result from the failure of systems to initially isolate the containment or failure of isolation systems after the initial isolation has been successful, (2) bypass of the containment resulting from a pipe or component failure outside the containment boundary, and (3) penetration or the containment boundary by internally generated missiles.

Challenge: Isolation Failure. Failure of the equipment in the containment isolation system (CIS) to initially isolate the containment or failure to maintain isolation over the full period of the accident would comprise the mechanisms for this challenge. Both of these mechanisms, Failure to Isolate (C3A1) and Failure after Isolation (C3A2), would initially utilize strategies to establish reisolation, which would rely on hardware specific to the containment penetration line where failure occurred. Availability of valves that could isolate, divert, or diminish the flow would depend on the configuration of the system and the capabilities of the valves and their actuators. If isolation fails, other strategies have been identified for mitigating the effect of fission product dispersal by reducing the driving force causing flow from the containment, or by flooding the leak location to reduce the inventory of fission products in the effluent leaving the containment.

Challenge: Bypass Failure. Two mechanisms have been identified that could lead to fission products bypassing the containment: Steam Generator Tube Rupture (SGTR) and an Interfacing System Loss–of–Coolant Accident (ISLOCA). Strategies identified to mitigate the SGTR (C3B1) mechanism are primarily focused on reducing fission product release based on the assumption that the RCS remains intact. These strategies are Depressurize RCS and Flood Steam Generator Secondary.

The second mechanism that would contribute to containment bypass would be an unisolable pipe break or component failure outside of the containment boundary, which is referred to by an ISLOCA (C3B2). The strategies identified to mitigate this mechanism are nearly identical to those identified for the steam generator tube rupture and include depressurization of the RCS and flooding of the break location. Fission product scrubbing could be accomplished for some ISLOCAs if the location of the break could be covered with one to two meters of water.

If either the SGTR or ISLOCA incidents progress to the point that the vessel is breached, previously discussed strategies involving containment depressurization may be effective in mitigating the release by reducing the driving potential for flow through the break and, in some cases, by reducing the fission products in the containment atmosphere. Additional strategies that will be discussed in the next section for reducing the containment fission product inventory (F1) could also be used.

Challenge: Internally Generated Missile. Two mechanisms identified for this challenge are Steam Explosion Missiles and Hydrogen Detonation Missiles. Both of these mechanisms are considered to have a very low probability of occurrence. Missiles from rotating machinery were not considered since they have been investigated extensively and their generation is not considered to be a severe accident issue.

The mechanism of in–vessel steam explosions (C3C1) could result in the generation of missiles that would penetrate the containment. This mechanism is generally referred to as an alpha–mode containment failure. Strategies to control this form of missile generation are Add Barriers, Dry Vessel, and RCS at High Pressure. Dry vessel is not considered to be an effective strategy, but is included to highlight the differences in strategies for other mechanisms such as Inadequate RCS Inventory.

Hydrogen detonation was identified as having the potential to generate missiles. This mechanism would have a low probability since there should be a limited source of missiles that could be generated through hydrogen detonation. The strategies include using barriers if potential missiles can be identified and eliminating the threat of combustible gas detonation using the same strategies discussed in the Combustible Gas Detonation (C1B2) mechanism for the Rapid Pressurization challenge.

3.1.3 Safety Objective Tree: Mitigate Fission Product Release from Containment. The third accident management safety objective is Mitigate Fission Product (FP) Release. This safety objective is important since it is intended to minimize the quantity of fission products released and to delay the release as long as possible if there is a failure of the containment boundary. The strategies associated with this safety objective would generally be implemented in conjunction with the strategies for the other two safety functions and, in most situations, would enhance the effectiveness of these other strategies. The tree for the Mitigate Fission Product Release from Containment safety objective, shown in Figure 6, details the safety functions that must be maintained, the challenges to the safety functions, the mechanisms causing these challenges, and the strategies that could potentially be employed to respond to these mechanisms.

Two safety functions were identified which would contribute to mitigating the release of fission products: (1) maintain control of the inventory of fission products suspended in the containment atmosphere, and (2) maintain control of the release of fission products that reside in the water inside of the containment. These safety functions are designated Maintain Control of FP Inventory in Containment Atmosphere (F1), and Maintain Control of FP Release From Containment Water (F2), respectively. A brief description of each of the safety functions follows.

3.1.3.1 Safety Function: Maintain Control of Fission Product Inventory in Containment Atmosphere. This safety function is concerned with controlling the concentration of fission products in the containment atmosphere. By reducing the quantity of fission products in the containment atmosphere, the amount available for release as a result of containment leakage or failure is also reduced. The challenge to this safety function is the presence of fission products in the atmosphere within the containment. Two mechanisms were identified that represent the types of fission products that could be dispersed within the containment during a severe accident. These mechanisms are Aerosol Dispersion (F1A1), and Gaseous Dispersion (F1A2).

The quantity of fission products suspended in the containment atmosphere in an aerosol form can be reduced through several potential strategies that take advantage of the physical structure and nature of aerosol particles. These strategies rely on both passive and active devices for the removal of aerosols and include Spray Additives, a Filter System, and Chemical Reactions.

The second mechanism that influences the availability of fission products for release is Gaseous Dispersion in the containment atmosphere. These gaseous fission products behave differently than aerosols and, therefore, require a different set of strategies. Potential strategies that could be used to reduce the concentration of gaseous fission products in the atmosphere are Chemical Reactions and Cryogenic System.

3.1.3.2 Safety Function: Maintain Control of Fission Products in Containment Water. This safety function deals with preventing the release of fission products present in the water inside the containment. If the fission products are held within the water, they are not available for release to the environment through the containment atmosphere and would be less of a threat if the water was inadvertently

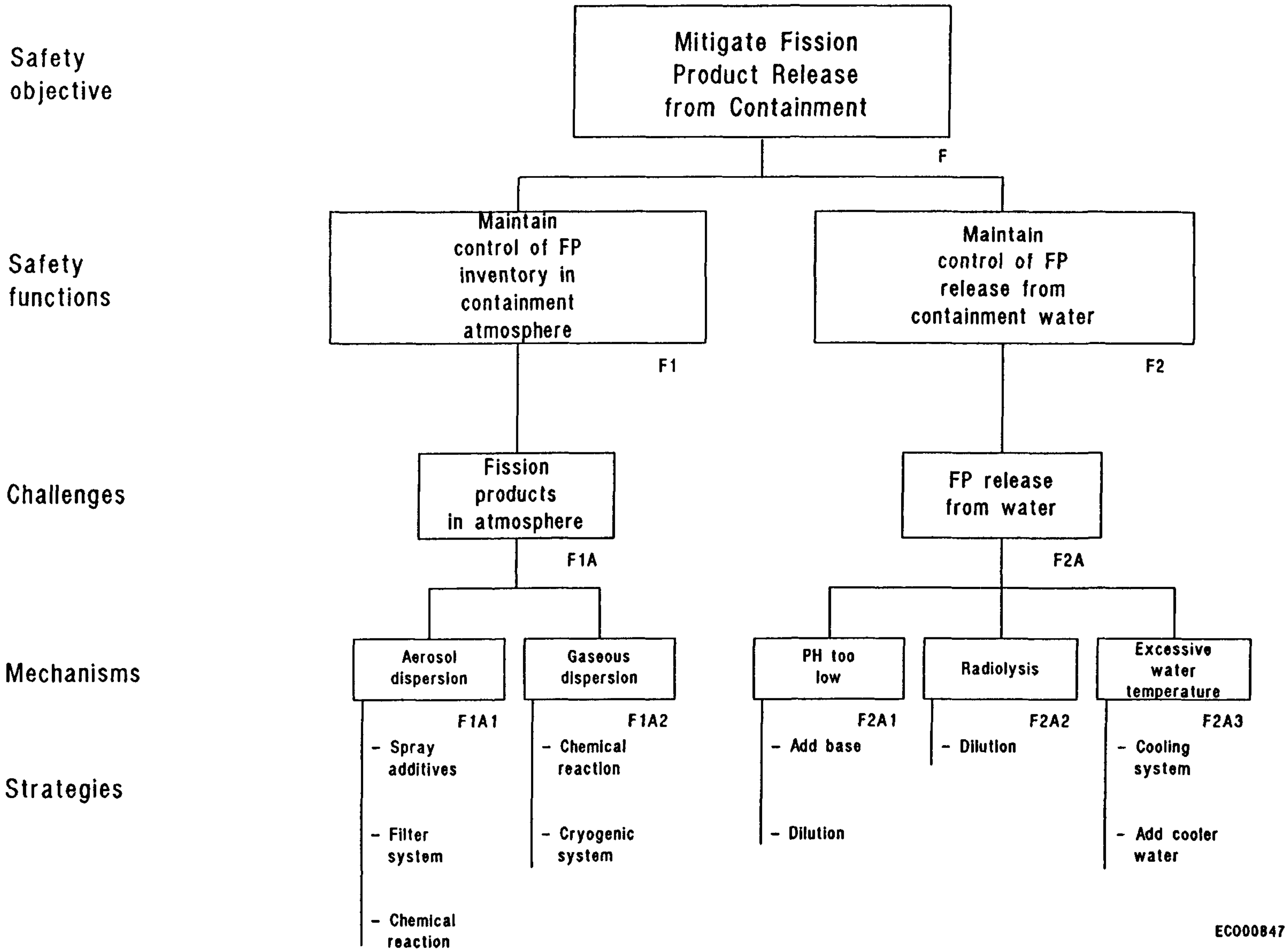

Figure 6. Safety objective tree: Mitigate Fission Product Release from Containment.

diverted to a location outside of the containment, e.g. the Auxiliary Building. The challenge to this safety function is the release of fission products from the water.

There are three mechanisms identified that can cause the release of fission products from the containment water: (1) if the pH of the water is too low, the capability to retain fission products is reduced, (2) radiolysis can cause the release of fission products from water, and (3) excessive water temperature will reduce the retention capability of the water. These mechanisms are shown in the Mitigate Fission Product Release from Containment safety objective tree respectively as: pH Too Low (F2A1), Radiolysis (F2A2), and Excessive Water Temperature (F2A3).

The strategies that can be used to address the first mechanism, a low pH in the containment water, are Add Base and Dilution.

The second mechanism that can result in the release of fission products from the containment water is radiolysis of the water in a high radiation field. The strategy identified as being capable of inhibiting radiolysis of the containment water is Dilution.

The third mechanism that results in the release of fission products from the containment water is excessive water temperature. Excessive water temperature can result in the vaporization of fission products. For example, excessive water temperature could have a large influence on the effect of the containment spray systems if the containment atmosphere or structures were at a sufficiently high temperature to cause some or all of the spray droplets to evaporate. Strategies that could be used to reduce the effects of excessive water temperature on this mechanism are Cooling System and Add Cooler Water.

3.2 Information Needs For a PWR With a Large, Dry Containment

The methodology for identifying information needs, described in Section 2.1.2, was applied to a PWR with a large, dry containment. The safety objective trees described in the previous section were used as the basis for development of the table–based format. This development was accomplished by personnel with both severe accident and operations experience, while the information was reviewed by personnel that administer operator examinations for PWRs with large, dry containments. Information needs on plant hardware status were generally not listed since it is recognized that such needs as switch positions, valve alignments, etc. would be required prior to the use of plant systems.

The information–needs tables that were developed for the PWR are extensive and, consequently, are presented in Appendix A. The first two pages from these tables were obtained from the Prevent Core Dispersal from Vessel (V) safety objective tree and are displayed for discussion purposes in Table 2. Included are the information needs for the Maintain RCS Heat Removal (V1) safety function, the Inadequate Secondary Inventory (V1A1) mechanism, and the Secondary Injection Methods strategy. The format of the table enables the reader to quickly scan the columns to determine the information needs, identify the sources of information, and ascertain whether existing instruments are available. The information need for the Maintain RCS Heat Removal safety function is the energy removal rate from the RCS. Since there are many ways that energy can be removed from the RCS and many of these are not measured with sufficient accuracy to derive an energy removal rate, it was concluded that a direct information source for energy removal rate does not exist. There are, however, numerous indirect information sources. Some of these sources, such as steam flow rate, could provide a reasonable measurement of energy removal rate. Others, such as PORV or atmospheric dump valve flow, are not measured but are indicated by such devices as acoustic monitors or temperature measurements downstream from the valves.

An indicator supplying the information need for the Inadequate Secondary Mechanism (V1A1) would be the liquid inventory in the secondary sides of the steam generators. This is considered to be an indicator for the information need because there is an information source that would identify that there is inadequate inventory. There is a direct information source for this information need, secondary side liquid level, which would indicate the inventory, and there is an instrument available to provide this information. An example of a precursor information need would be feedwater flow status. A sharp reduction in feedwater flow, under certain plant conditions, would provide early information that would alert the operator to the potential for an inadequate inventory situation to occur in the future.

The second page of Table 2 provides the information needs for the Secondary Injection Methods strategy. These information needs are relatively straight forward because all have direct information sources and available instruments.

Table 2. Example information needs table
[prevent core dispersal from vessel (v)—inadequate secondary inventory mechanism (v1a1)]

	Information Needs	Direct Information Source	Indirect Information Source	Available Instruments	Potential Instruments
Maintain RCS Heat Removal Safety Function (V1)	Energy removal rate	None		None	
			RCS fluid temperature	Hot or cold leg RTD	
			RCS pressure	Pressurizer pressure	
			Steam generator steam flow	Steam flow indicators	
			PORV discharge pipe noise	Acoustic monitor	
			RHR heat removal	RHR flows, temperature	
	Indicator				
Inadequate Secondary Inventory Mechanism (V1A1)	Secondary fluid inventory	Secondary liquid level		Secondary liquid level	
	Precursor				
	Feedwater flow status	Feedwater flow rate		Feedwater flow rate	MFW flow AFW flow

Table 2. (continued)

	Information Needs	Direct Information Source	Indirect Information Source	Available Instruments	Potential Instruments
	Selection Criteria				
Injection Methods Strategy	Inventory availability	Tank inventories		Tank levels	
	Pumping capability	Electrical power availability			
		Steam availability			
	Alignment capability	Valve alignments		Valve position indicators	
	Strategy Initiation				
	Feedwater flow status	Feedwater flow rate		Feedwater flow rate	MFW flow AFW flow
	Injection water inventory (decreasing)	Tank inventory		Tank level SI Tanks RWST	
	Strategy Effectiveness				
	RCS fluid temperature	RCS fluid temperature		Hot or cold leg RTDs	
	Secondary fluid inventory	Secondary liquid level		Secondary liquid level	

The results presented in Table 2 and in the extensive information–needs tables contained in Appendix A provide an indication of the redundancy and diversity of the plant instruments in supplying the information needs. An indication of the redundancy can be determined by evaluating the number of direct and indirect information sources that are available for each information need. An indication of diversity can be obtained by comparing the number of different types of information sources. These comparisons would not account for such considerations as common cause failures which could reduce the redundancy, or the ability of some diverse instruments to supply the needed information.

Since the information needs tables are lengthy and contain large quantities of data on information needs and available instruments, several methods of extracting and summarizing the important findings were considered. There were two major types of findings that were considered to be important:

1. Information needs for which neither direct nor indirect information sources exist

2. Information needs with only indirect sources of information.

These two types of findings are discussed in Sections 3.3 and 3.4, respectively.

3.3 Capability of Existing Instrumentation

An evaluation was made to determine whether the existing instrumentation has the capability to supply all of the identified information needs. This evaluation was accomplished using the results from the information needs tables presented in Appendix A by searching for information needs that do not have instrumentation identified to supply either the direct or the indirect sources of information. The information needs identified during this evaluation are listed in Table 3 and represent those information needs that cannot currently be satisfied by existing measurements in a PWR with a large, dry containment.

The strategies were generally not considered in this evaluation since they are intended as an example and may not represent a complete listing of information needs. One exception consisted of the strategy of adding water to the vessel cavity, since this strategy has been discussed extensively for reducing the effects of the molten core material on the containment basemat. Each of the seven information needs listed in Table 3 is discussed below.

Table 3. Information needs with no direct or indirect measurements

1.	Core relocation status
2.	Lower plenum coolant inventory
3.	Lower head temperature and integrity
4.	Containment penetration integrity
5.	Basemat integrity (amount of concrete ablated)
6.	Cavity level
7.	Presence of missile in containment

1. Core Relocation Status – There are no instruments currently installed that provide reliable information on the location of the fuel rod cladding, fuel, control rod material, or supporting structure once relocation begins. Information on the location of this material would provide personnel involved in accident management with the capability to concentrate efforts on trying to maintain the core within the vessel or initiate those efforts necessary to maintain the integrity of the containment. This information is considered to be very important. Some data on the response of the source range detectors and the self–powered neutron detectors during the Three Mile Island incident and from tests at the Loss–of–Fluid Test Facility indicate that, with additional evaluations, these instruments could be adapted to provide this information.

2. Lower Plenum Coolant Inventory – Information on the amount of water in the lower plenum during a severe accident would aid personnel involved in accident management by projecting the possible failure time of the vessel, which would aid in determining what repair and restoration strategies could be effective and what containment strategies could be appropriate. This information would be most useful when combined with a knowledge of the core relocation status. It is not considered to be as important as the core relocation status.

3. Lower Head Temperature and Integrity – A knowledge of the temperature of the lower head could provide important information on whether substantial core debris has relocated into the lower head and could aid the operator in determining the likelihood of failure of the

lower head. This information is considered to be very important because it could provide a good indication of the timing of possible vessel failure. This timing information could be used in decisions concerning repair and restoration of equipment and the prioritizing of resources that may be required in both the RCS and the containment.

4. Containment Penetration Integrity – Although the integrity of the containment penetrations is important for preventing release of fission products, there is little that can be done if possible failure was detected during a severe accident. Flooding at the failure location is a possible strategy if the breach is located.

5. Basemat Integrity (Amount of Concrete Ablated) – The escape of fission products into the soil and groundwater beneath the basemat could have serious long-term consequences. However, there are no apparent strategies that would utilize this information to aid in mitigating core-concrete interactions. The information would, therefore, be useful only in the emergency response process, and the information is not considered to be important to accident management.

6. Cavity Level – In some PWR containments, the sump level is not a reliable indicator of the reactor vessel cavity water level. This lack of information would be detrimental for those plants that would select strategies that involve intentionally filling the cavity. An example strategy that would involve water in the cavity consists of mitigating the effects of the core-concrete interactions.

7. Presence of Missile in Containment – Since there is nothing that can be done if missiles are detected, supplying this information need is not considered to be important.

For those information needs that are considered to be important, means of obtaining the information should be considered. These means could take several forms, such as adding protection from severe environmental conditions for the transducer, cabling, or electronics; using computational aids; or adding instrumentation. Determination of the optimum means for obtaining this information was beyond the scope of this project.

3.4 Potential to Mislead Accident Management Personnel

The potential for information needs with only indirect sources of information to mislead accident management personnel were discussed in Section 2.1.4. To determine the extent of this potential for a PWR with a large, dry containment, the tables in Appendix A were searched to identify those information needs in which direct sources of information are not measured and, therefore, can be only inferred from measurements of indirect information sources. Table 4 summarizes the findings of this search. For the purposes of discussion, the information needs have been categorized into those that could be important in misleading personnel, identified as Category 1, and those that would be much less important, or Category 2. The categorization process

Table 4. Information needs with only indirect information sources

<u>Category 1 – More Important</u>

1. Core damage status
2. RCS inventory
3. Fuel rod cladding temperature
4. Containment leak location
5. Location of ISLOCA Containment Bypass

<u>Category 2 – Less Important</u>

1. Energy removal rate from RCS
2. Steam generator atmospheric dump flow rate
3. RCS energy transport
4. Pressurizer (RCS) PORV flow rate
5. Core heat removal
6. Insufficient energy removal from containment
7. Steam explosion mechanism in containment
8. Containment leak rate
9. Presence of ISLOCA containment bypass
10. Containment shell temperature
11. Presence of aerosols in containment atmosphere
12. Presence of fission product gasses in containment
13. Amount of contaminated water
14. Presence of radiolytic products
15. Inadequate energy removal from sump water

considered the type and diversity of indirect information available and the degree of ambiguity that would be expected from these sources. In addition, consideration was given to the importance of the information in making decisions on the progress of an accident or on the selection or monitoring of corrective strategies. For example, information on the fuel rod cladding temperature could be very important in making decisions concerning the time available to repair equipment. The following discussion presents a brief description of the Category 1 information needs and the rational for why Category 2 information needs were less important.

1. Core Damage Status – There are no direct measurements that would provide an unambiguous indication of the status of the core during the period when core damage is occurring. Without this information, personnel involved in accident management could have difficulties making proper decisions on the use of personnel and resources to accomplish effective accident management strategies. For example, there may not be sufficient information to decide if repair of failed equipment can be completed in sufficient time to be effective or whether to continue to attempt to add water to the RCS rather than to concentrate efforts on maintaining the integrity of the containment. For these reasons, the lack of more direct measurements of core damage status could result in personnel being mislead.

2. RCS Inventory – This information need could be considered as supporting the core damage status information need discussed above. There are several instruments installed in a PWR that provide an indication of the RCS inventory, such as the pressurizer level and reactor vessel level monitoring systems. Together, these two measurements would adequately indicate the RCS inventory to the time of core uncovery. Although there are situations in which these measurements would provide ambiguous indications, these situations are now understood. Arguments have been made that a knowledge of the inventory of the core is not needed since the core exit thermocouples provide an indirect measurement of core inventory. There are situations in which these arguments may not be correct and include, for example, conditions when the reactor coolant pump is operating and during periods of injection to the vessel from the hot legs. For these conditions, there is the potential for the operator to be mislead since the core exit thermocouples could be misinterpreted to indicate a cooled core during these situations. A knowledge of the lower plenum inventory could also be important in making decisions on whether to continue efforts on adding water to the RCS or to concentrate efforts on preparing the containment to receive molten core material.

3. Fuel Rod Cladding Temperature – This information need could also be considered as supporting the core damage status information need, and similar reasons for its importance could be given. Although it is doubtful that a direct measurement of fuel rod cladding temperature would be developed and installed in a PWR, the absence of this information could make decisions on the management of accidents less certain.

4. Containment Leak Location – Without direct information on the location of containment leakage, the personnel involved in accident management are not likely to be able to quickly effect isolation.

5. Location of ISLOCA Containment Bypass – The capability to terminate an ISLOCA by closing the containment isolation valves or to mitigate the consequences through such strategies as flooding the break location depends on the capability to determine where the break has occurred. Although there may not be a reasonable instrument to provide a good, direct measurement, the indirect measurement capabilities could be improved to allow accident management personnel to identify and diagnose the break location quickly. Additional use and display of the temperatures and pressures in lines that could be over–pressurized would lessen the likelihood of misleading personnel.

The information needs in Category 2 from Table 4 were considered to be less important for several reasons. First, some of the indirect measurements provide a good indication of the information that is needed. For example, although the energy removal rate from the RCS is not measured directly (Item 1 in Category 2, Table 4), the RCS fluid temperature and pressure measurements provide a good indication that there is insufficient energy removal. Second, in some cases, the lack of direct information would not alter the approach to accident management and would, therefore, not mislead personnel involved in accident management. An example would be the presence of either aerosols or fission product gases in the containment atmosphere (Items 11 and 12 in Category 2, Table 4). Current strategies for mitigating the effects of fission products in the containment are not sufficiently sophisticated to motivate the need for different strategies for aerosols and gases. In these cases, a direct measurement would provide better information than an indirect measurement, but it would not make accident management more effective.

The results of this evaluation of the potential to mislead personnel involved in accident management has identified five information needs for which direct or improved indirect measurements would be beneficial. These information needs should be evaluated further to determine the acceptable means of providing the needed information or to provide a clearer understanding of the limitations on accident management. More detailed evaluations will take place as part of the assessment of accident management strategies being conducted by the NRC.

Other conditions which could mislead the operators includes instrument failures as a result of severe environmental conditions and instrument range limitations. These additional instrument failure modes are important but are pertinent to a specific plant and will be discussed in the following section, in which information–needs methodology is evaluated for a specific plant and severe accident sequence.

4. METHODOLOGY EVALUATION FOR A SPECIFIC SEVERE ACCIDENT SEQUENCE

This section presents an application of the safety objective models to a particular severe accident sequence to assess the information needs, the available instruments to supply these needs, and the potential strategies used to effectively manage the course of a specific severe accident sequence. The accident sequence is discussed first, followed by a description of the safety objective models, information needs, and instrument availability during the selected severe accident sequence.

4.1 Severe Accident Sequence

The sequence consists of a small break loss–of–coolant accident (LOCA) induced by a rupture of the reactor coolant pump (RCP) seals and which is also accompanied by failure of the emergency core cooling (ECC) system high and low pressure injection pumps. The containment sprays fail due to an assumed failure to switch to the sump during recirculation. The containment fan coolers are initially operable but also fail at the time of failure of the lower head of the reactor vessel. A slow, long–term pressurization of the containment ensues through the later portion of the event as a result of the slow boil–off of the water which entered the reactor vessel cavity during the initial injection phase of the event.

The sequence parameters were selected from an analysis of the Zion Nuclear Power Plant which is a Westinghouse PWR with a large, dry containment.[3] The sequence was designated as S_2DC_r with the key events summarized in Table 5.

The event begins with the RCP seal failure. With no ECC injection pumps available, the primary system inventory decreases, producing uncovery of the core at about 65 minutes into the event. The continued lack of injection produces a core melt at 108 minutes, followed by a core slump and collapse. With the core relocated in the lower plenum of the reactor vessel and further absence of ECC injection, the lower head fails at 133 minutes. The discharge of the molten core into the cavity then initiates attack of the concrete structures. The containment fan coolers are assumed to fail upon failure of the lower head. The containment sprays continue to operate and quickly reduce the pressure increase following the hydrogen burn at 162 minutes. The failure of containment sprays, upon the switch to the recirculation mode, results in the long–term pressurization of the containment, which eventually fails at 1444 minutes into the event. Also note that because the Reference 2 evaluation of this sequence did not include the effects of direct containment heating, it is not included herein although lower head failure occurs at a high RCS pressure.

Table 5. Timing of key events

Zion S_2DC_r	
Event	Time in Minutes
Containment cooler "on"	0.0
Core uncovery	64.9
Start melt	94.2
Core slump	108.4
Core collapse	108.9
Bottom head dryout	125.3
Bottom head failure	133.0
Containment cooler "off"	133.0
Accumulators empty	133.0
Start concrete attack	133.1
Containment spray injection "on"	133.4
Corium layers invert	161.1
Hydrogen burn	162.1
Containment spray injection "off"/recirculation failure	184.1
Containment failure	1444.0
End of calculation	1633.1

This sequence is divided into four phases to conveniently separate the major phenomenological events. This division allows for an ideal separation of the information needs, which differ throughout the event as the various challenges to the major safety functions develop. The specific phases include the initial blowdown and loss of RCS inventory including the initial core uncovery, the complete uncovery of the core and core melt, the core relocation and lower head failure, and finally, the slow pressurization and heatup of the containment. The RCS and containment parameters pertinent to this accident sequence are presented in Figure 7. The phases are also indicated in Figure 7 and are summarized here:

Phase 1 This phase represents the initial 80 minutes of the transient. This portion of the transient includes the depletion of mass in the RCS and the initial uncovery of the core. This portion of the event is covered by the existing plant Emergency Operating Procedures (EOPs).

Phase 2 This phase includes the continuation of core uncovery and the subsequent fuel clad heatup. During this phase, the fuel heatup results in significant clad oxidation, clad distortion, and pellet/clad melting with relocation of the molten core materials into the lower plenum. This phase represents the portion of the severe accident (i.e., 80 to 110 minutes) in which the existing plant EOPs end.

Phase 3 Phase 3 begins with the relocation of the core materials into the lower plenum and ends with the failure of the lower head. This phase includes the time frame from 110 minutes to 133 minutes.

Phase 4 This phase represents the long-term heatup and pressurization of the containment (133 to 1444 minutes). A late containment failure occurs due to the excessive pressurization at the end of the sequence caused by failure of the containment fan cooler and spray systems.

With the key parameters for the sequence defined in Figure 7, the pressure and temperature ranges describing the environmental conditions in the RCS and containment are identified and presented in Table 6. This table will be used to identify the failure conditions and instruments available during each phase of the event, which is discussed in more detail in Section 4.4 below entitled "Instrument Failure Conditions."

Before the instrument failure conditions are defined, the remaining tabular information specific to a PWR with a large, dry containment are developed. The analysis of information needs for accident management contained in this section is, therefore, designed to identify existing measurements that could be used to provide the previously identified information needs, to identify the depth and diversity of existing sources of information, and to identify potential additional measurements that could be used to satisfy these information needs.

Qualification requirements were also determined for the existing measurements, based on Regulatory Guide 1.97 (see Reference 2). This section, therefore, also builds upon the results of the steps presented in Section 2.1. In review, Step 1 included the development of safety objective trees to identify the safety functions that are required to support the safety objectives, to show the challenges to these safety functions that may be encountered, and to present strategies that could potentially be employed to prevent or mitigate the safety function challenges. The safety objective trees are shown in Figures 4, 5, and 6. Step 2 then identified, for each component of the safety objective trees, the information needs for accident management. The information needs are those that are necessary to monitor the status of the safety functions, detect the mechanisms of the safety function challenges, select a strategy for addressing the challenges, and monitor the implementation and effectiveness of the strategy. The results of this analysis are presented in Tables A1, A2, and A3 of Appendix A. The tables include categories for description of the information needs and potential direct and indirect sources of the information. Columns are also included in the table for noting available measurements that could be employed to provide each information source and for noting potential measurements that could provide information sources that are not currently instrumented or measured in a PWR with a large, dry containment. For Step 2, a preliminary list of the available measurements was also included.

The identification of applicable measurements for providing accident management information needs was accomplished by building upon the results of the previous steps. As such, new tables were developed to summarize the availability and qualification of measurements for accident management. Figure 8 shows the work flow and identifies the new tables that have been developed to summarize information needs and measurements for accident management. First,

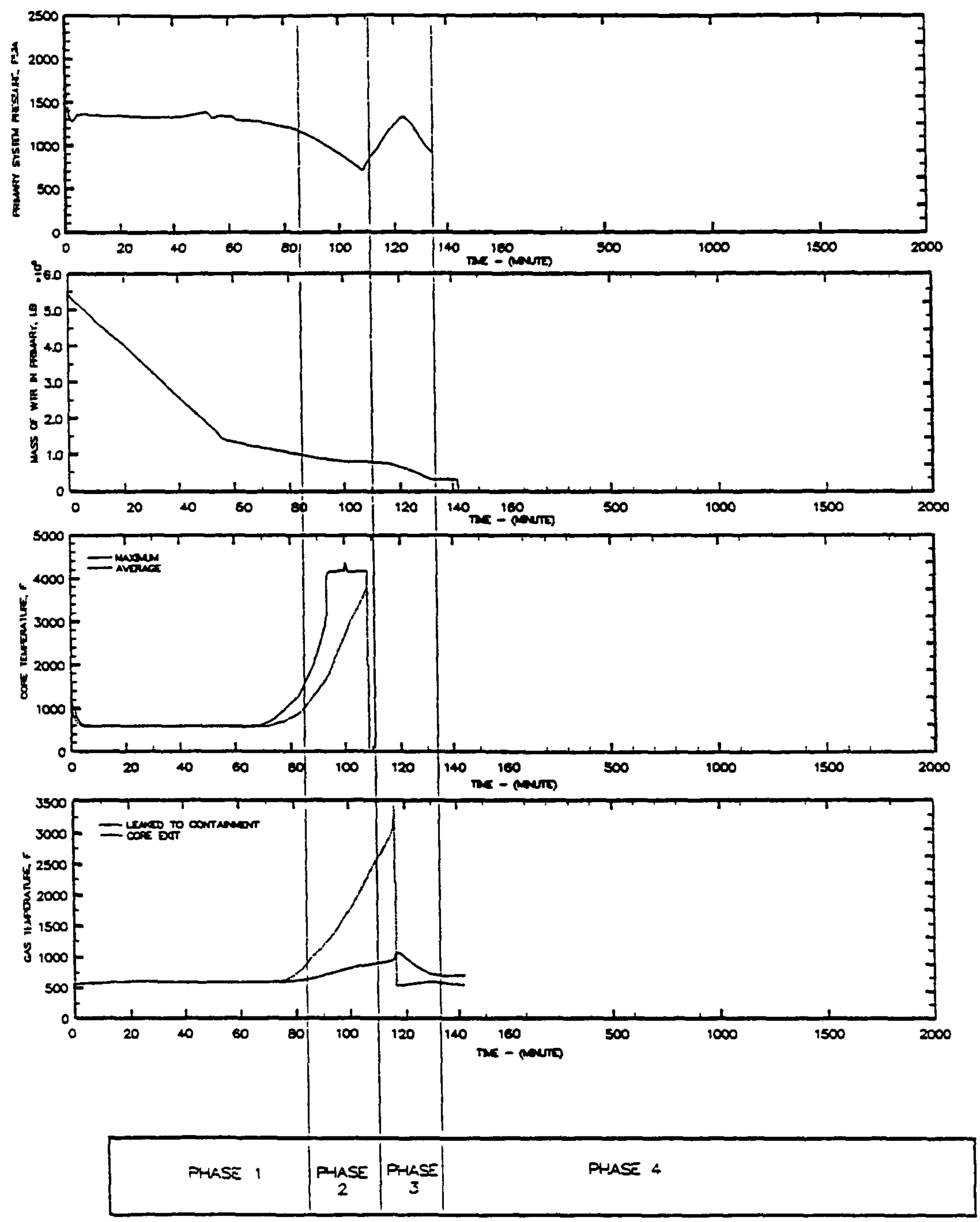

Figure 7. Key RCS and containment response parameters for S_2DC_r.

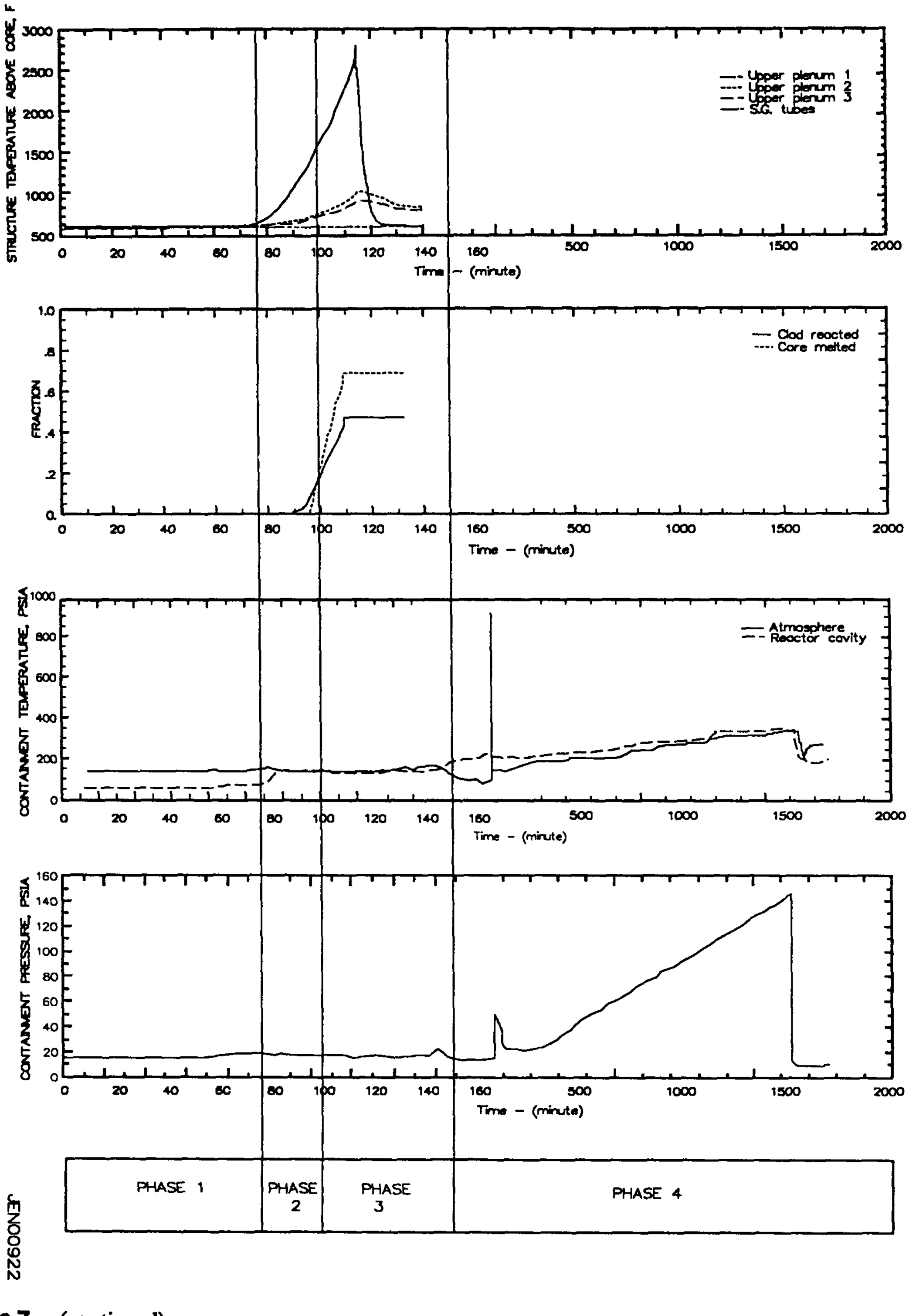

Figure 7. (continued).

Table 6. Key parameter ranges

Key RCS/ Containment Parameters (RCS)	Phase 1: Blowdown/Boiloff/ Start of Core Uncovery (0–80 min)	Phase 2: Continued Core Heatup/ Damage/Slump (80–110 min)	Phase 3: Core Relocation/ Vessel Attack Head Failure (110–133 min)	Phase 4: Head Failure/Slow Containment Overpressurization (133–1444 min)
Pressure	2250–1300 psia	1300–700 psia	700–1250 psia	1250–900 psia
Core exit temperature	600–1000°F	1000–3500°F	3500°F – failure	
Vessel level	100% – Partial uncovery	Partial uncovery to total uncovery	Water in lower plenum to complete voiding	None
Containment				
Pressure	14.7–20 psia	20–25 psia	20–30 psia	20–150 psia Spike to 55 psia at 162 min
Temperature	110–175°F	160–170°F	170–195°F	195–344°F Spike to 861°F at 162 min

Tables A1, A2, and A3 from Appendix A were completed by filling in the columns for available and potential measurements that could be used to provide information needs. Available measurements are those that are typically available in the current generation of PWRs with large, dry containments. Potential measurements are those that could possibly be installed to provide sources of information but which are not currently available.

Following the completion of Tables A1, A2, and A3, additional tables were developed to summarize the availability and qualification requirements for measurements needed for accident management. These additional tables were developed to also highlight potential concerns regarding the diversity and depth of information sources for the accident management information needs.

In order to illustrate these potential concerns with the depth and diversity of accident management information sources, the remainder of this section discusses the additional tables that were developed. As discussed in Section 3.3, Table 3 listed those information needs for which no direct or indirect measurements are currently available. The items in Table 3 were extracted from Tables A1, A2, and A3 by searching for information needs that do not have direct or indirect sources of information that are currently measured. Table 3, therefore, represents those information needs that cannot currently be satisfied by existing measurements in a PWR with a large, dry containment. Unless it can be demonstrated that these information needs are not important for accident management, it may be necessary to develop measurements so that the operating staff can access this information during a severe accident.

Table 4, from Section 3.4, listed the information needs which are currently supplied by indirect information sources only. The items in Table 4 were extracted from Tables A1, A2, and A3 by searching for those information needs that currently have no measured direct sources of information. These information needs can only be inferred by monitoring measurements of indirect information sources. There may be a substantial potential for operator error when attempting to infer information from only indirect information sources. It may be necessary to develop additional measurements to provide direct indication of these conditions. Additional investigations will be required to assess the relative importance of these information needs and to assess the degree of ambiguity that results when using the indirect information sources to infer an information need. Table 4 contains information needs

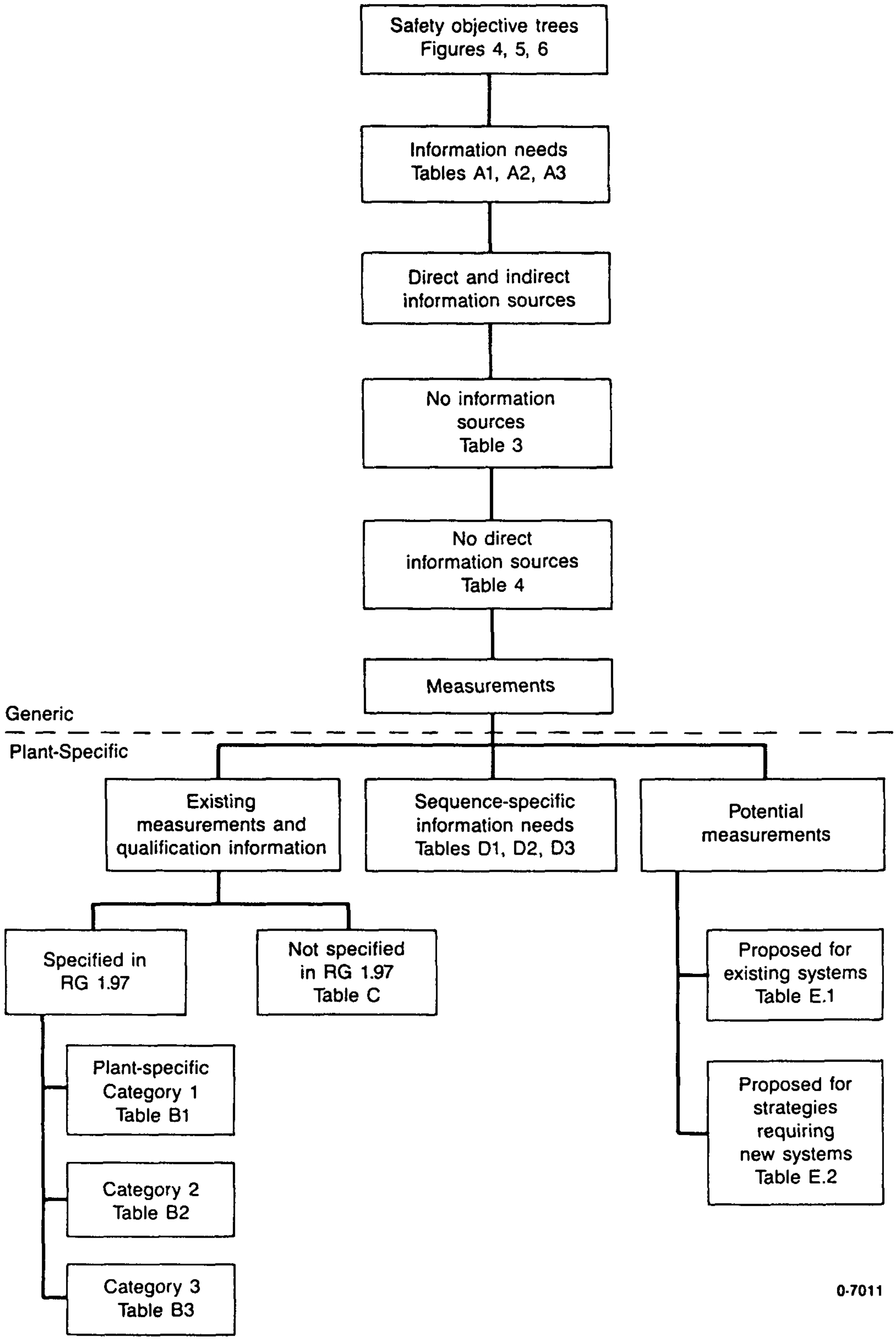

Figure 8. Table development for identification of information needs, potential measurements, and qualification information.

regarding major groups of systems including fuel, vessel, reactor coolant system, containment water and radiation monitoring, and containment boundary.

Tables B1, B2, and B3 in Appendix B were developed to summarize the availability of plant-specific measurements for accident management and the requirements for their qualification. Tables B1, B2, and B3 list the existing measurements according to those contained in Regulatory Guide 1.97 (see Reference 2). Regulatory Guide 1.97 specifies a method acceptable to the NRC staff for providing instrumentation to monitor plant variables during and following an accident in a light-water-cooled nuclear power plant. Regulatory Guide 1.97 specifies three different categories of qualification:

> Category 1 provides for full qualification, redundancy, and continuous real-time display and requires onsite (standby) power. Category 2 provides for qualification but is less stringent in that it does not (of itself) include seismic qualification, redundancy, or continuous display and requires only a high reliability power source (not necessarily standby power). Category 3 is the least stringent. It provides for high quality commercial-grade equipment that requires only offsite power.

Table B1 in Appendix B lists those existing measurements from Category 1 as specified in Regulatory Guide 1.97. Similarly, Table B2 lists existing measurements from Category 2 and Table B3 lists the existing measurements from Category 3. These tables represent a Combustion Engineering 2700 MW(t) plant such as Calvert Cliffs Units 1 and 2 or Millstone Unit 2. Columns are provided in these tables to list information regarding current requirements for environmental qualification for the specific measurement and the location of the sensor (reactor, auxiliary building, containment, turbine building, etc.). It is important to note that representative containment harsh environments include a maximum pressure of 60 psia and a maximum temperature of 300°F and are generally associated with the design basis LOCA. Some instruments are required for other design basis accidents (DBAs) which do not include harsh environments. To identify the two potential environments, these requirements are denoted as DBA Harsh and DBA Non-Harsh in the table. This information will be used to determine the survivability of the instruments in the environment that can be expected during a severe accident sequence. This analysis will be performed as part of the effort discussed later in this section.

Table C in Appendix C lists those existing measurements that could be used for accident management that are not specified in Regulatory Guide 1.97. This table includes columns to list the current measurement range of the instrument, the severe accident measurement range, and the location of the sensor.

Tables D1, D2, and D3 in Appendix D present sequence-specific information needs for a PWR with a large, dry containment, which will be discussed below in Section 4.5.

Table E1 in Appendix E lists proposed or potential measurements that could be used to supply information needs in cases where measurements are not currently available. Table E2 lists measurements that would be required for existing systems that could be used as part of a strategy for accident management. The columns that are contained in this table are as follows:

> *Potential Measurements Identified:* the measurements identified from Tables A1 through A3 that may be required for existing systems to be used for particular accident management strategies
>
> *Information Need:* the information needs that would be provided by the potential measurement
>
> *Potential Severe Accident Measurement Range:* the range of conditions the instrument might experience during a severe accident
>
> *Location of the Instrument:* the general location of the instrument sensor.

Table E2 lists those measurements that could be used to monitor proposed systems that would be needed for new strategies for accident management. This table has columns to list the potential measurement, the information needs it would provide, the measurement range it would need to cover, and the location of the sensor. The last two columns will be completed during the analysis of a specific severe accident presented later in this section.

4.2 Use of Information-Need Tabular Information

The purpose of this section is to determine the ability of existing measurements to satisfy the information needs for a PWR with a large, dry containment. This determination can be made by evaluating the results contained in the Tables in Appendices B and E. Comparison of Tables B1, B2, and B3 with Tables E1 and E2 shows the relative contribution of existing measurements to supply accident management information needs with regard to additional measurements that may be required. This evaluation determines the number of information needs that are currently measured. In order to fully assess the ability of the existing measurements to satisfy the information needs, it is necessary to evaluate whether individual instruments are expected to function during a severe accident. This can be accomplished by comparing the qualification and range information contained in Appendix B with the conditions expected for a severe accident. This will be performed for a specific accident sequence as part of the application of the method to a specific severe accident sequence presented below.

Another goal of this effort was to identify alternate means of supplying the information needs. Alternate means for identifying the information needs can be identified by evaluating the results in Tables A1, A2, and A3 of Appendix A to see where additional direct and indirect information sources can be used to supply the same information need. Also, Appendix E lists those measurements that could be implemented as additional means for satisfying those information needs that currently have no available sources of information.

The final application of the results of this portion of the effort was to indicate the diversity and depth of measurements for each identified information need. This can be accomplished by reviewing Tables 3 and 4 in Section 3. These tables highlight those information needs in which insufficient diversity and depth of measurements may be a concern. Table 3 identifies information needs for which there are no existing sources of information. Table 4 lists the information needs that can be inferred only from indirect sources of information. There is always a substantial opportunity for error when an information need can be inferred only from an indirect information source rather than from a direct information source.

4.3 Status of Instrument Qualification Information

Development of qualification information for the measurements discussed in the previous sections was determined to be very complex for two reasons. First, current qualification requirements are generally based on individual plant conditions for the design basis accident (DBA). The qualification conditions, therefore, vary somewhat from plant to plant depending on the calculated DBA conditions and the location of the instrumentation. Second, the envelope of severe accident plant conditions for PWRs with large, dry containments was not found to be readily available. Developing this envelope was determined to be a large undertaking since the performance of a uniform set of new calculational results would be both expensive and time consuming, and because the screening of existing severe accident calculations would require review of a large number of documents by a select group of people with an awareness of the adequacy of past and present analyses. For example, the Industry Degraded Core Rulemaking Program (IDCOR) examined a limited number of pieces of equipment from four reference plants to determine survivability when exposed to severe accident conditions. While the conclusion of the IDCOR study (Technical Report 17, "Equipment Survivability in Degraded Core Environment") stated that the subject equipment could withstand the effects of most degraded core accident environments, the study investigated a limited number of severe accident sequences and did not address complete instrument systems including transducer, cabling, terminal blocks, etc. Furthermore, the IDCOR report concluded that use of a bounding envelope of environmental conditions in the containment was useful only for general scoping and sensitivity studies but not for assessment of the survivability of specific equipment. In light of these circumstances, development of an envelope is considered to be well beyond the scope of the current information needs program.

4.4 Instrument Failure Conditions

This section presents the application to a severe accident sequence. In order to identify the instruments that may fail during each phase of the sequence, plant–specific instrument qualification information was used. It should be mentioned that plant–specific instrumentation from the Calvert Cliffs Plants were used. Although the severe accident sequence selected for this application is not directly applicable to the Calvert Cliffs instrumentation qualification needs, use of the Zion accident sequence is considered appropriate since the purpose of this task is to demonstrate use of the information needs methodology and not to present an evaluation of a specific plant.

Since many instruments available in the plant are required during the course of DBAs, the plant instruments can be subdivided into those which are qualified to DBA environmental conditions and those which are not qualified, since some instruments are not required for accident conditions. Since both types of instruments can experience harsh or non–harsh environments, four instrument qualification categories can be identified. These categories include instruments qualified for:

1. DBAs with harsh environments
2. DBAs with non–harsh environments
3. Non–DBAs with harsh environments
4. Non–DBAs with non–harsh environments.

The qualification of the plant instrumentation corresponding to these categories can be readily obtained from documents presenting a review of the plant instruments listed in Regulatory Guide 1.97. This information provides the measurement ranges and the qualification level of each instrument required for DBA events. This information can be used to establish the failure conditions for the various sensing and monitoring instrumentation in the reactor coolant system and containment building locations. The instruments to be used for the example sequence described herein are qualified to the following DBA harsh environmental conditions:

1. Instruments within the RCS
   - Maximum temperature = 2300°F
   - Maximum pressure = 2500 psia
2. Instruments within the containment building
   - Maximum temperature = 300°F
   - Maximum pressure = 60 psia

No other limitations are placed on the instrumentation for the purposes of this evaluation.

From the RCS and containment pressure and temperature profiles given in Table 6 and the instrument environmental qualification information discussed above, the timing of the instrument failures due to the severe accident conditions are summarized below:

1. All unqualified instruments fail in Phase 1
2. All qualified RCS instruments fail in Phase 2
3. All qualified containment instruments fail in Phase 4
4. All instruments located in the auxiliary and turbine building are available for all phases of the event since these locations do not experience harsh environmental conditions.

The list below summarizes which systems fail due to specific, assumed support system failures:

1. All Regulatory Guide 1.97, Category 3 instruments fail upon loss of offsite power at the initiation of the event
2. The high and low pressure ECC injection pumps are assumed to fail due to a support system failure
3. The fan coolers fail due to failure of the reactor vessel lower head
4. The containment spray pumps fail due to a failure to switch to the recirculation mode.

It is important to note that a detailed analysis of the survivability of the instrumentation was not performed to address equipment performance beyond the DBA qualification limits. If such an analysis was performed, it may indicate that some instruments would survive well beyond the time when the RCS or containment conditions exceed the DBA qualification conditions. As a consequence, the information available to the operating staff which results from this task can be considered to represent a conservative lower limit.

4.5 Identification of Capability of Available Instruments for a Specific Plant Type

This section presents the process by which the information needs, from the three safety objective trees, can be compared to the information available in each phase of the event so that the operating staff can effectively manage the course of a severe accident. This process was previously discussed in Section 2.2 and is now demonstrated through application to the severe accident sequence described earlier.

Given the accident sequence and the instrument failure conditions described above, the following steps from Section 2.2 are used to determine the information needs and available instruments.

Step 1 Identify instrument availability during each phase of the accident using Table 6, given the instrument failure conditions of the previous section. The instrument availability is then indicated in Tables B1, B2, and B3 of Appendix B, where the status of each instrument for each phase of the sequence is denoted by the following acronyms:

- FSS – failed due to support system failure
- FEC – failure due to environmental conditions
- OOR – out of range
- AVI – available
- NI – not installed.

Step 2 Transfer the instrument availability determined above in Step 1 to Tables D1, D2, and D3 of Appendix D to identify the status of the information needs for each of the safety objective trees. Information need status is indicated by the following acronyms:

- ADI – available directly or indirectly
- ADO – available directly only
- AIO – available indirectly only
- NAV – not available.

The status of the information needs is indicated in brackets in the information needs column in Tables D1, D2, and D3 of Appendix D.

Step 3 Identify information needs by using the safety objective trees (see Figures 9 through 16) and finding the safety function status, the existence of mechanisms for various challenges, and the relevant strategies for each phase of the sequence. The status of the safety objective tree levels is indicated in brackets. The acronyms for the status of the safety functions, state of mechanisms causing the challenges, and status of the strategies are given below:

Status of Safety Functions

- OKA – status is okay
- CAA – challenged
- LOS – lost
- IOF – information deficiencies exist

State of Mechanisms

- PRE – present
- POT – potential
- IDM – information deficiencies
- NPE – not present or not expected

Status of Strategies

- UFE – unavailable failed equipment
- IDS – information deficiencies
- AVS – available
- ACT – active
- NI – not installed.

Step 4 The final step includes developing Tables F1 through F8 in Appendix F to compare the information needs to the information available in each phase. This table is referred to as the "Accident Management Information Assessment" table and is discussed in detail in the following section.

4.6 Accident Management Information Assessment

The purpose of the Accident Management Information Assessment table (Table F of Appendix F) is to present the final product, which is an assessment of the availability of plant instrumentation for accident management during each phase of the specific accident sequence. The assessment was made by applying the plant conditions for each phase of the sequence to the Appendix D tables that were developed as part of the methodology to identify the information needs for management of an accident, and which was summarized through the four-step process given in Section 4.5 above.

The accident management information assessment discussed below was performed in the context of the

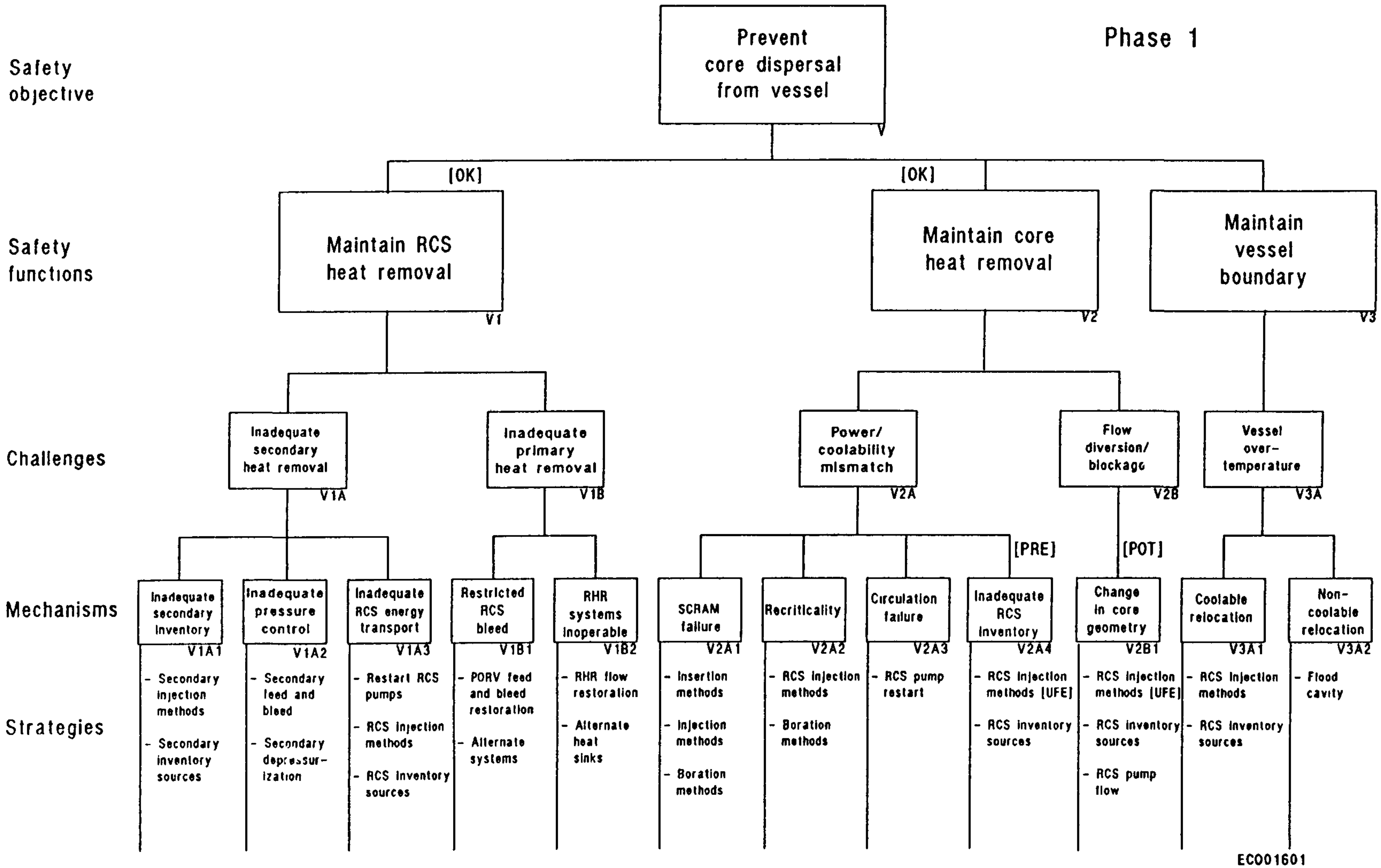

Figure 9. Safety objective tree: Phase 1 of Prevent Core Dispersal from Vessel.

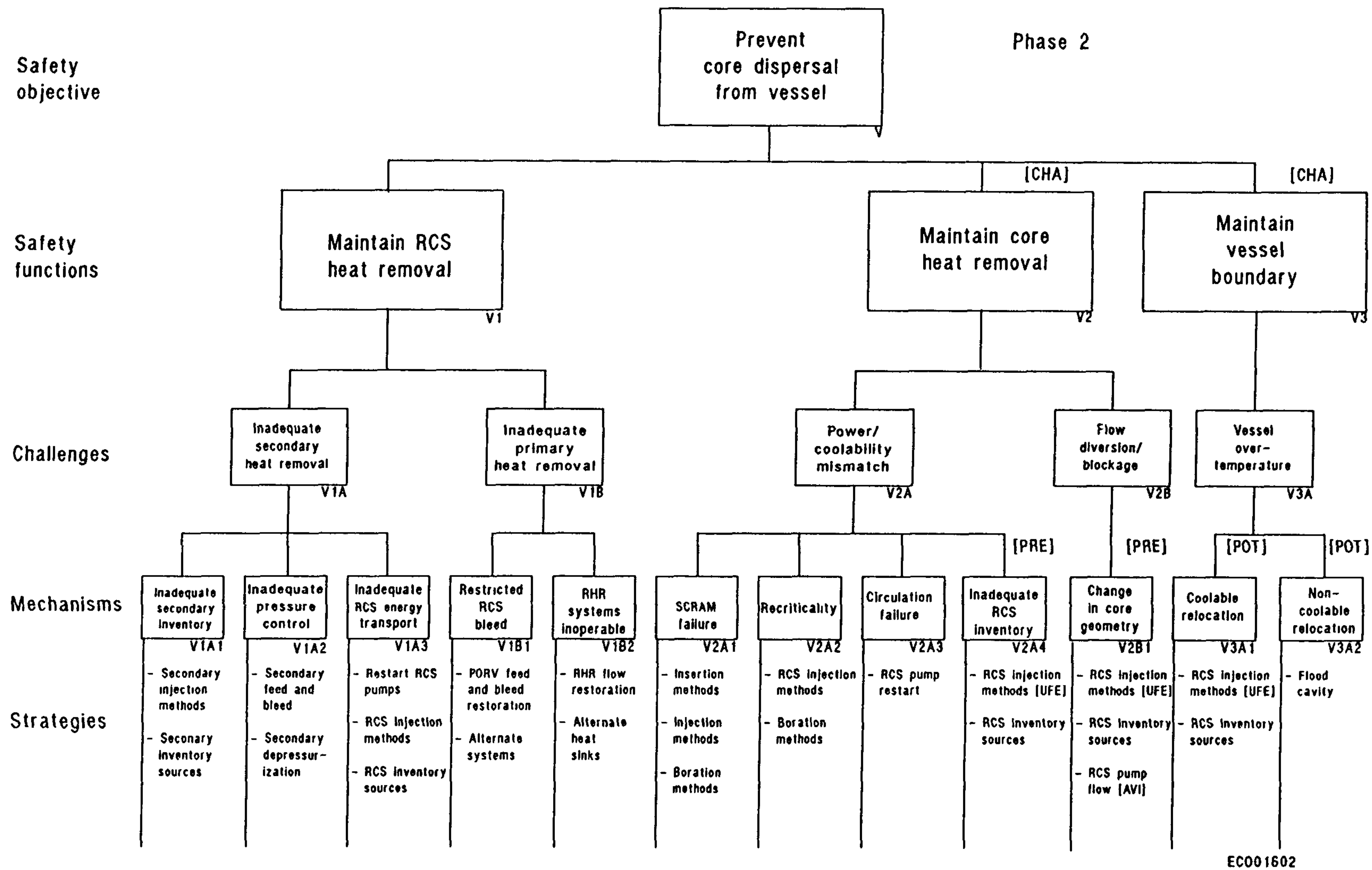

Figure 10. Safety objective tree: Phase 2 of Prevent Core Dispersal from Vessel.

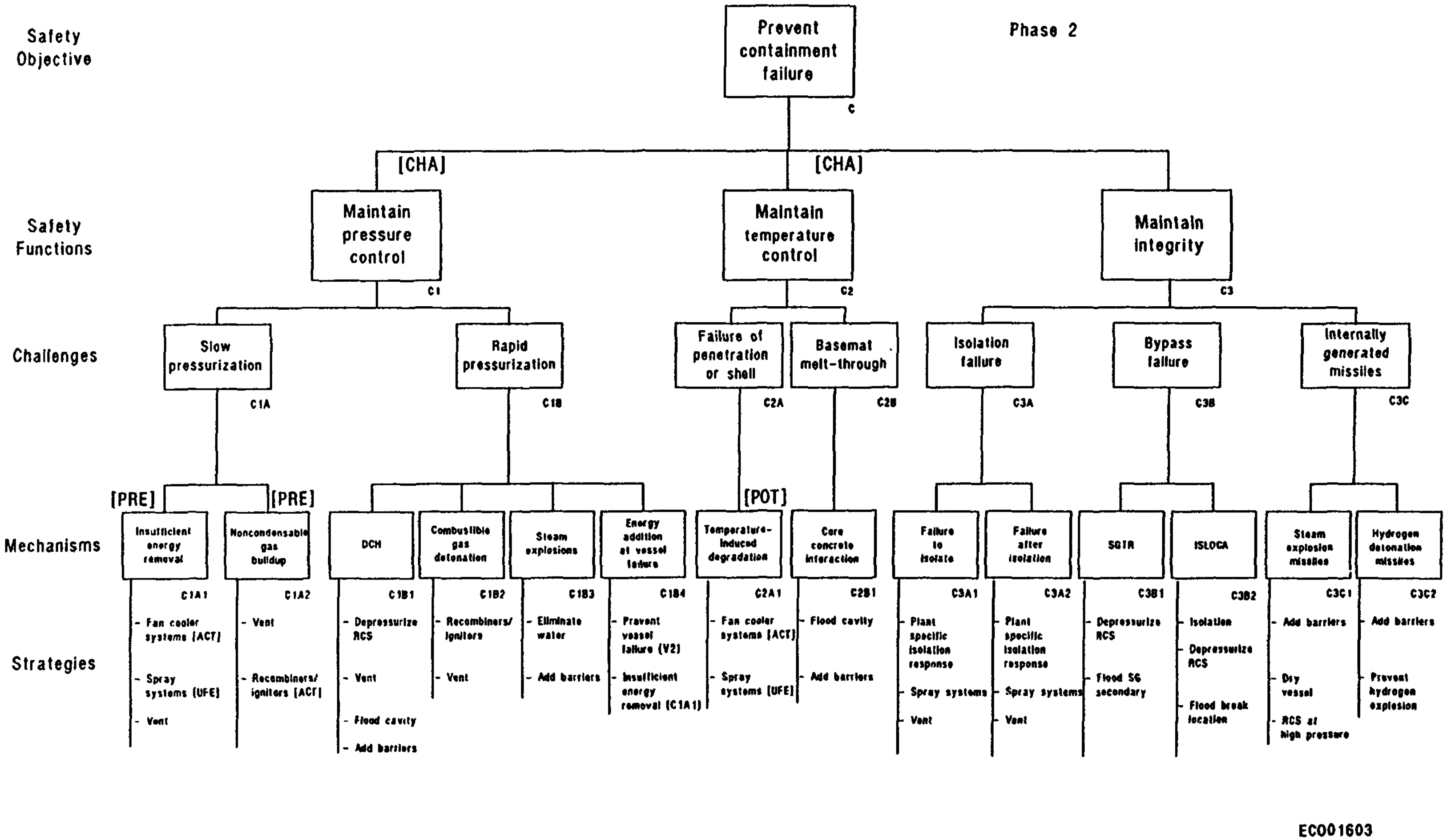

Figure 11. Safety objective tree: Phase 2 of Prevent Containment Failure.

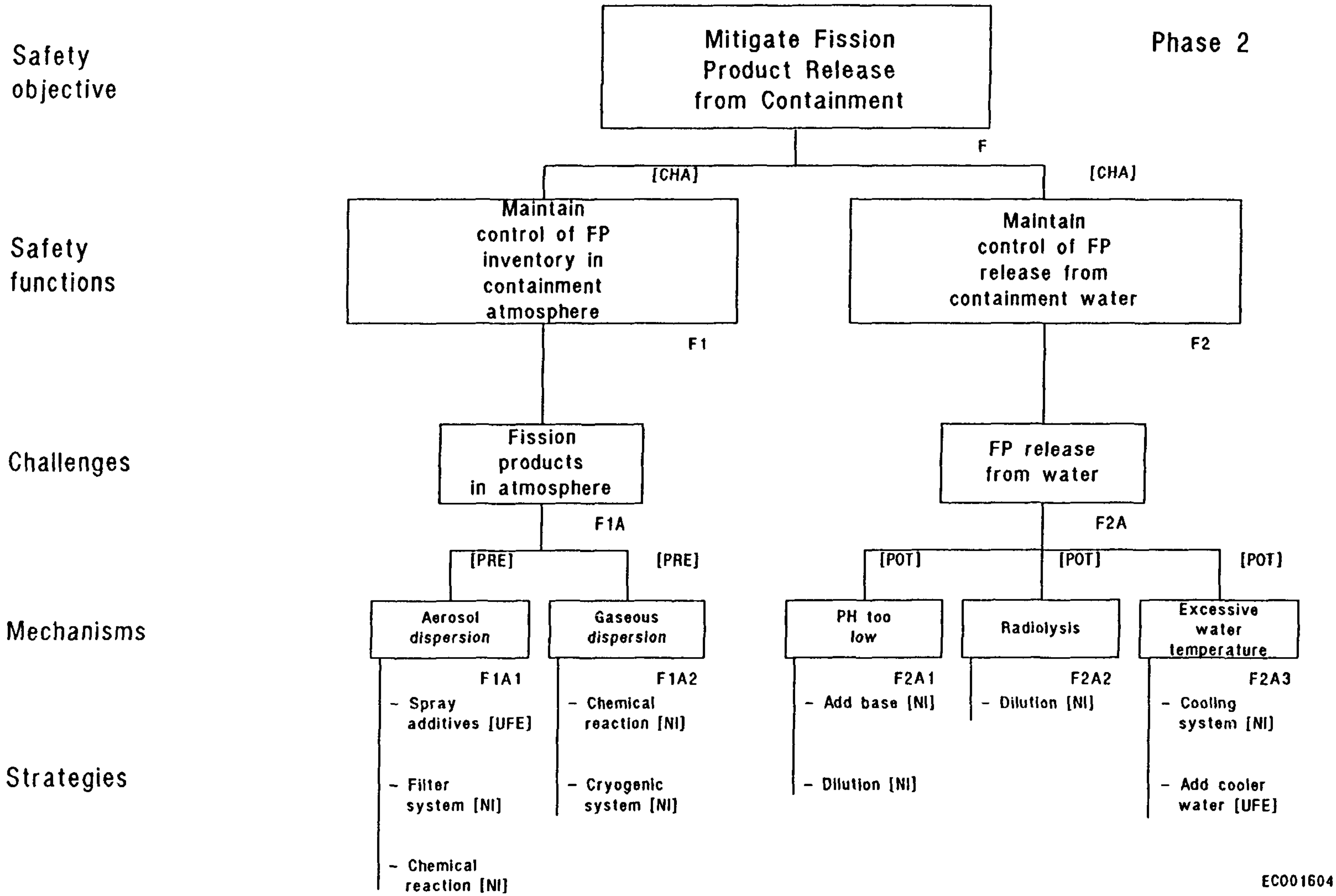

Figure 12. Safety objective tree: Phase 2 of Mitigate Fission Product Release from Containment.

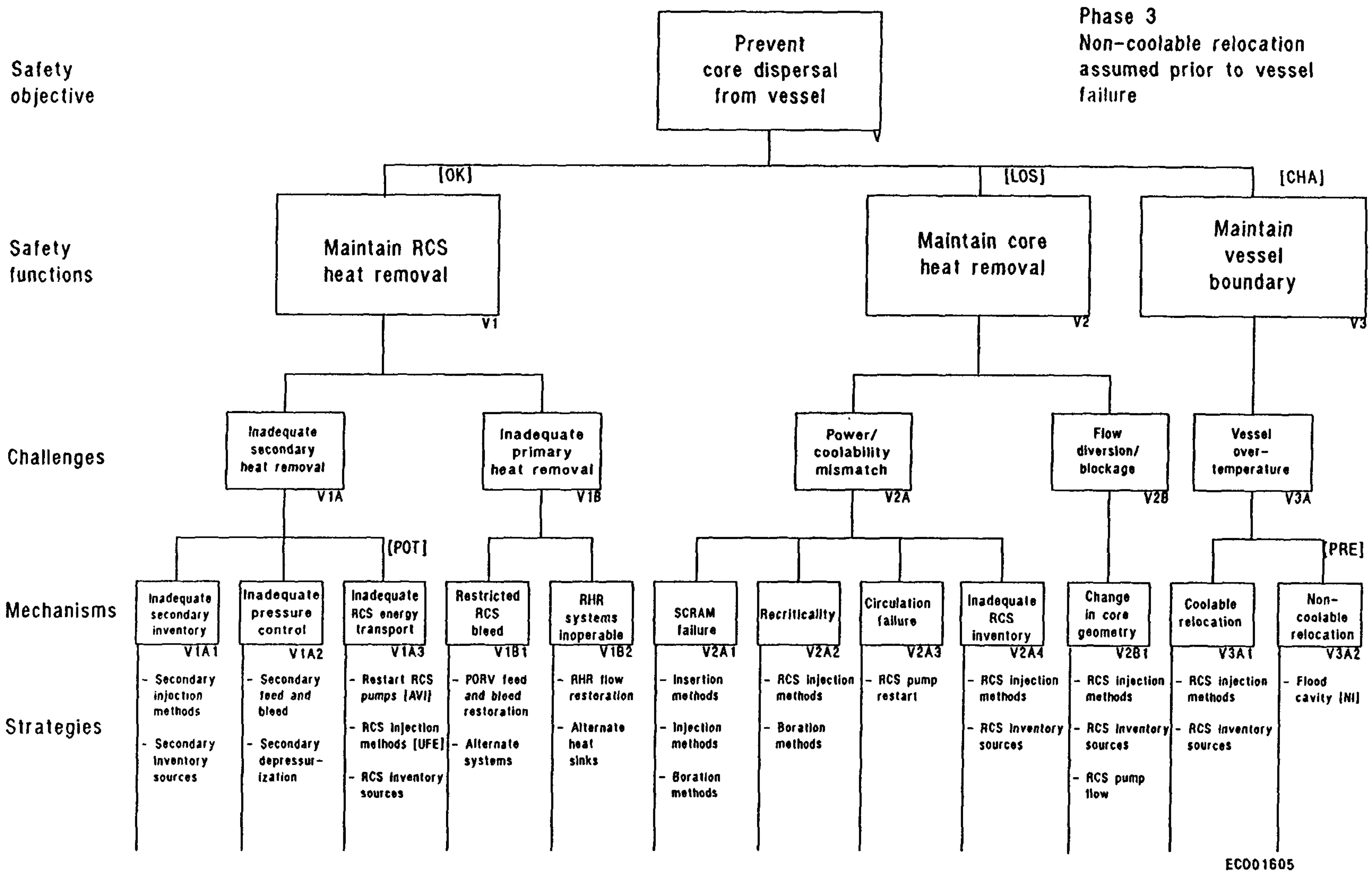

Figure 13. Safety objective tree: Phase 3 of Prevent Core Dispersal from Vessel.

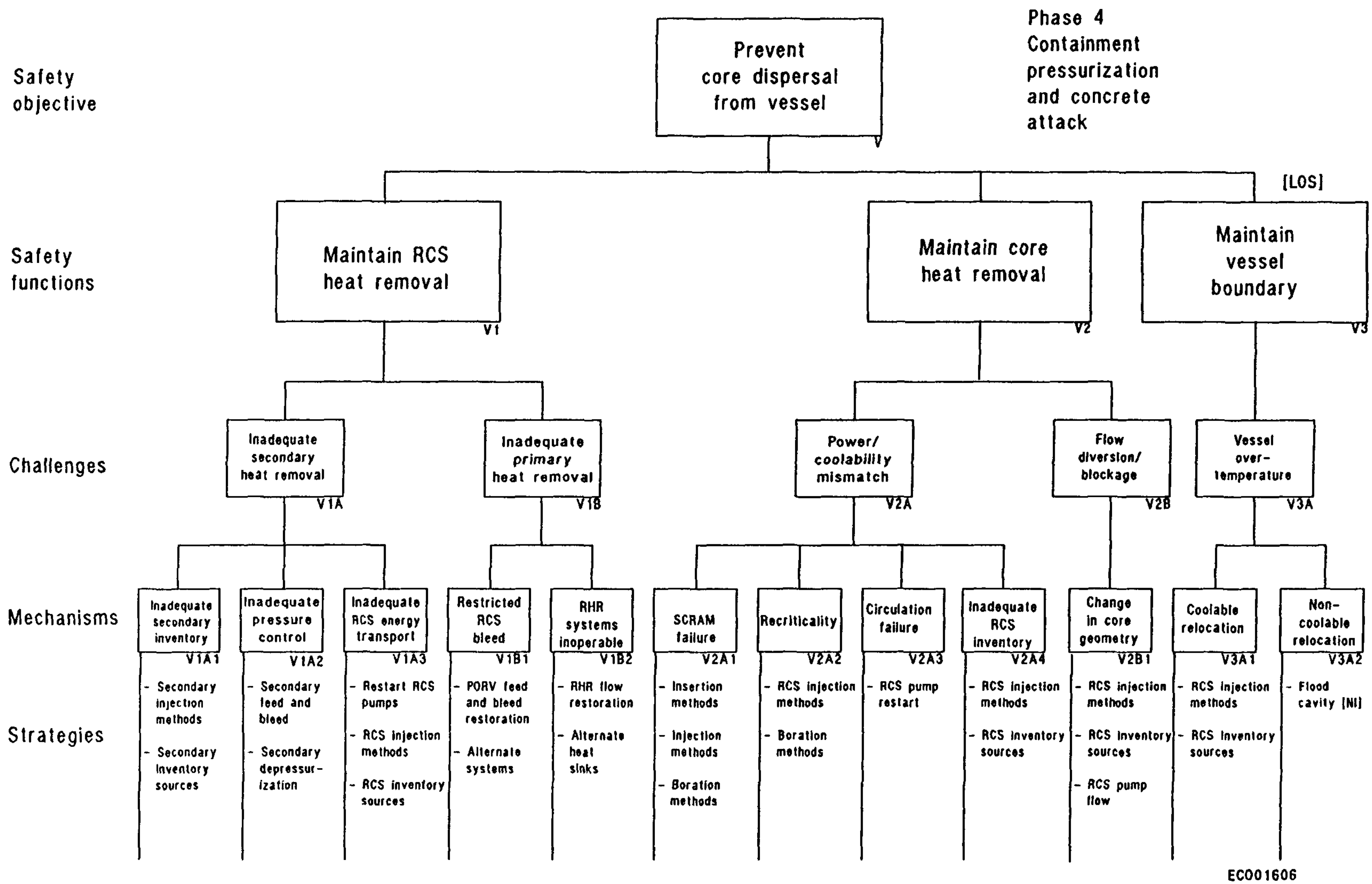

Figure 14. Safety objective tree: Phase 4 of Prevent Core Dispersal from Vessel.

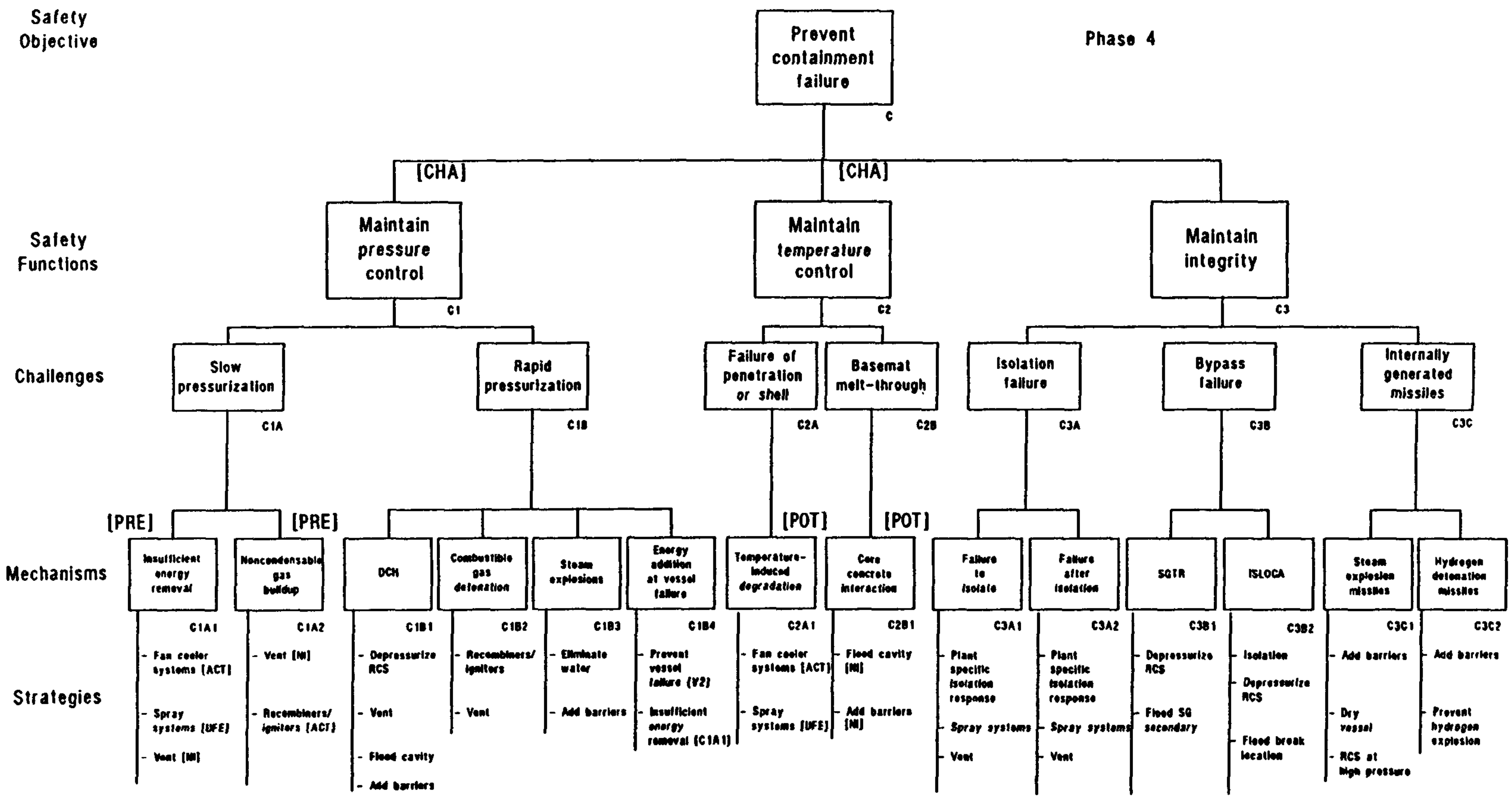

Figure 15. Safety objective tree: Phase 4 of Prevent Containment Failure.

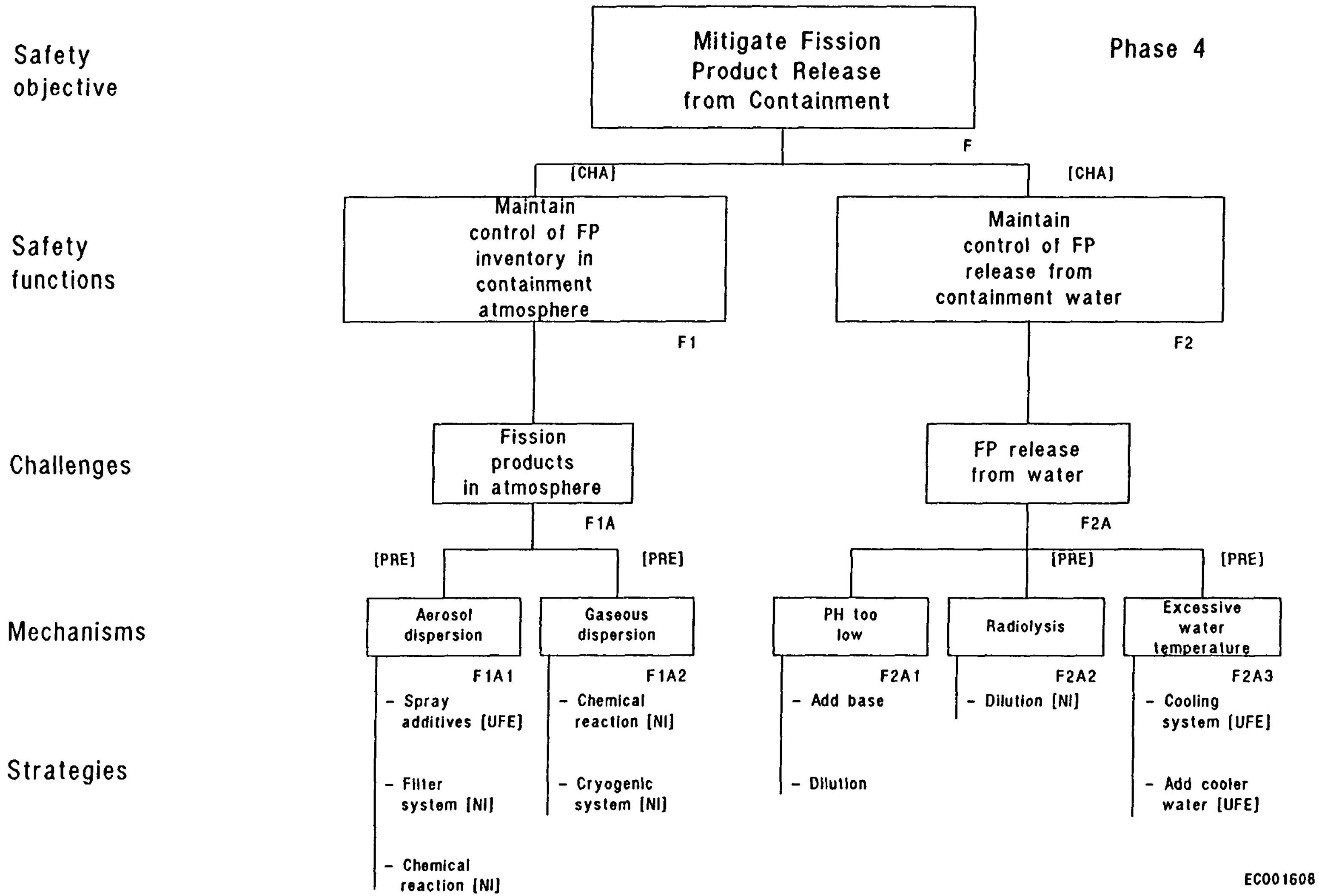

Figure 16. Safety objective tree: Phase 4 of Mitigate Fission Product Release from Containment.

three safety objective trees (i.e., Prevent Core Dispersal from Vessel, Prevent Containment Failure, and Mitigate Fission Product Release). The safety functions on each of the three trees were addressed in a left-to-right sequence and, therefore, the first assessment table addresses the instrument information needs necessary to control the Maintain RCS Heat Removal safety function, which supports the Prevent Core Dispersal from Vessel safety objective.

The information assessment table addresses the information needs and instruments available for each of the challenges and mechanisms under each safety function, starting with Inadequate Secondary Heat Removal for the Maintain RCS Heat Removal safety function. The strategies that may be available to combat a challenging mechanism are addressed immediately after addressing the individual mechanisms. The safety objective tree was reviewed from top to bottom and from left to right for this assessment of information needs, and the results are tabulated accordingly throughout Table F in Appendix F.

Tables F1 and F2, the information assessment tables for the Prevent Core Dispersal from Vessel safety objective, includes the information needed to determine the status of the Maintain RCS Heat Removal safety function. The information needed is that which would detect the energy removal rate from the RCS. There are no direct sources of this information; however, there are several indirect sources of information related to RCS heat removal and include RCS fluid temperatures and pressures, secondary steam flow, secondary safety valve discharge pipe temperature, and atmospheric dump valve position. If the plant is in a shutdown mode with RHR activated, information regarding RCS heat removal can be obtained from RHR flow and temperature indications.

For this severe accident sequence, the Maintain RCS Heat Removal safety function is not challenged by either inadequate secondary or inadequate primary heat removal during the initial stage of the event. The mechanisms causing the challenges to this safety function are not present in Phase 1; however, they may pose challenges later in the sequence in Phases 2 and 3 when noncondensable gases develop as a result of fuel-clad heatup and failure. However, since these challenges will not be present at the initial portion of this event, the challenges, associated mechanisms, and potential strategies will not be discussed here for this safety function, but will be addressed later in this section when Phase 3 is discussed. The status of the Maintain RCS Heat Removal safety function is important here only because the operator has sufficient information to confirm that this safety function is being maintained during the initial phase of the event.

Challenges to the Maintain Core Heat Removal safety function will develop during Phase 1 due to a power/coolability mismatch as a result of an inadequate RCS inventory. The operator will have sufficient information from the pressurizer level and, more importantly, from the reactor vessel level indications that the Inadequate RCS Inventory precursor and mechanism is present. The core exit thermocouples (CETs) and hot leg resistance temperature detectors (RTDs) will signal the initiation of core uncovery at the end of Phase 1. The operators will be attempting to restore ECC injection or utilize an alternate source of injection water if the refueling water storage tank cannot be used. Certainly, the operators will also be attempting to depressurize the RCS if insufficient ECC flow exists to possibly activate the safety injection tanks or other available systems which would actuate at low RCS pressures and inject coolant. During Phase 1, there is sufficient information available to the operators to identify challenges to the Maintain Core Heat Removal safety function and to attempt strategies to restore RCS inventory.

If the loss of RCS inventory continues, core uncovery will occur and the fuel will heat up, resulting in fuel rod clad deformation. If injection is not restored at this stage of the accident, major fuel rod damage will ensue with the top third of the core experiencing rod ballooning and rupture. A significant release of the fuel rod fission gas will also occur, and the generation of large amounts of hydrogen will begin. These "beyond" DBA conditions will mark the beginning of the next phase of the event, or Phase 2. The Change in Core Geometry mechanism at the beginning of Phase 2 will pose a Flow Blockage/Flow Diversion challenge to the Maintain Core Heat Removal safety function. With the continued absence of ECC injection, the clad will continue to heat up and eventually melt. As discussed earlier, there are no instruments for PWRs which could provide either direct information regarding changes in core geometry or information on the movement of fuel or cladding material within the core following melt. (For example, the Direct Instrument column contains the word "None.") There are, however, two instruments which provide indirect indication on the movement of fuel from the core to the lower plenum. The source range nuclear instruments (SRNIs) and self-powered neutron detectors (SPNDs) are listed under "Indirect Instruments" since movement of large amounts of fuel-bearing material across the core boundary can cause significant variation in the readings from the SRNIs and the SPNDs. However, changes in the water density within the reactor vessel can also cause changes in the signals from these instruments, as noted in the Phase 1 – Blow–down column. A plant–specific analysis would be required to determine the possibility of using these instruments to monitor this safety function procedurally to allow the operators to distinguish between the two different responses. These instruments and others with detectors and cabling in the containment are assumed to fail during Phase 2, due to the adverse containment conditions because these instruments are not qualified for harsh environments.

There is no direct indication available for detection of the Flow Diversion/Blockage challenge. Since flow blockage would occur initially as a result of fuel–clad ballooning and eventual rupture, manual sampling and analysis of RCS coolant activity should provide an indirect indication from which inferences of flow blockage and clad rupture can be made. The indications provided by this instrument should remain available through all phases of this accident sequence, although sampling of RCS coolant after a high level of activity will be difficult and hazardous.

The CETs are another example of indirect instruments, and their responses are available in the control room. A temperature reading corresponding to super–heated steam is a strong, early indication of core uncovery and incipient clad ballooning and blockage, while readings of the CETs in later stages of the accident may be more directly related to fuel clad and pellet melting than they would to flow blockage. When thermocouple temperatures exceed 2300°F during Phase 2, these instruments are assumed to fail.

The first strategy listed for forestalling the Inadequate RCS Inventory mechanism is injection into the RCS. The specific scenario for this study includes failures of all high and low pressure ECC pumps. However, the control room staff would be concerned with the obvious need for RCS injection, and the information needs and availability of instruments is relevant and readily available to confirm injection if the equipment failures are corrected.

As the accident progresses, Phase 3 is initiated as the core melt continues with a major relocation of core material into the lower plenum. As stated above, the SPNDs and SRNIs may provide an indirect indication of a core relocation, which will present a Vessel Over–Temperature challenge to the Maintain Vessel Boundary safety function. The mechanisms for the Vessel Over–Temperature challenge include a Coolable or Noncoolable relocation. There is no information to the operators to distinguish between a coolable versus a noncoolable relocation. The CETs provide an indirect indication to infer core damage status; however, this information will be lost due to the severe RCS environmental conditions that will develop during Phase 3. The operators will continue to attempt to add water and depressurize the RCS during this phase of the accident; however, with the presence of a significant amount of noncondensables in the RCS, the steam generators will become ineffective at this stage of the accident. In fact, during Phase 3, the ability of the steam generators to remove heat will degrade as the noncondensable gases accumulate in the RCS and enter the steam generator primaries. An Inadequate Primary Heat Removal challenge to the Maintain RCS Heat Removal safety function may develop with precursor indications late in Phase 2 and challenges to the Maintain RCS Heat Removal safety function in Phase 3. This challenge would be due to the Inadequate RCS Energy Transport mechanism caused by the noncondensable gases in the steam generator primaries. The degradation in condensation in the steam generator primaries will be indirectly indicated by an increase in RCS pressure and would be most dramatic for the smaller break sizes in which RCS heat removal is a prerequisite for ensuring core coolability. Thus, the Maintain RCS Heat Removal safety function may be challenged during the later portion of Phase 2 or during Phase 3. These developments may confuse the operators since adequate level, pressure, and secondary emergency feedwater flow is being maintained during this phase. In this situation, the operators may waste valuable time addressing the increasing RCS pressure by checking secondary emergency feedwater flow, inventory, and depressurization capability when the steam generators are in a state that will not enable RCS depressurization. It should be noted that there are no increases in RCS pressure late in Phase 2 nor in the first half of Phase 3 as shown in Figure 7. This potential situation is mentioned since smaller break sizes will show more sensitivity to steam generator heat removal than the small break illustrated in this sequence.

The eventual total core uncovery in Phase 3 will, however, result in faster RCS depressurization due to the large reduction in energy addition to the RCS from the presence of steam in the core. However, if injection is not restored during this phase, the core will relocate into the lower plenum with eventual lower head failure. Toward the end of Phase 3, the operators will have no direct information to determine if vessel integrity is challenged or if lower head failure will occur or when. If injection is restored at this stage, the operators will have no means of assessing whether the core can be cooled in the lower plenum, thus resulting in a possible waste of a limited supply of injection water. No information is available to assess whether injection will exacerbate the accident if it is restored at this time.

If the relocation has achieved a noncoolable state, the only available strategy is to flood the cavity in hopes of cooling the core in this region and thereby limit or prevent basemat attack. As indicated in Table F, there are no means of flooding the cavity nor are there means of inventory indication in this region. Phase 4 begins with lower head failure and, without the restoration of containment heat removal, the containment instrumentation will fail leaving the operators with neither direct nor indirect information, which could have alerted them to a challenge to containment integrity.

The remainder of Table F (Tables F3 through F8) presents the remaining information pertinent to both the Prevent Containment Failure and the mitigate Fission Product Release from containment safety objective trees for all phases of the event and will not be discussed further in this report.

While the tables in Appendix F present a summary of the assessment of the information needs and available instruments for the four phases of the event, it is instructional to also discuss some of the key information contained in the safety objective trees as the sequence progresses through each phase. This discussion provides some of the thought processes and potential concerns of an operating staff during a severe accident and is presented below for each phase using the three safety objective trees.

Phase 1

During Phase 1, the first 80 minutes of the event, opening of the break causes a depressurization and mass depletion of the RCS, resulting in uncovery of the core at about 65 minutes into the event. Referring to the Prevent Core Dispersal from Vessel safety objective tree in Figure 9, sufficient instrumentation is available to supply the information needs for management of the initial portion of the event. This is not surprising since the plant EOPs accommodate DBA sequences which represent Phase 1 of this accident. The needed instrumentation is qualified to the conditions of Phase 1.

Referring to Figure 9, the status of the Maintain Core Heat Removal safety function is being maintained and is labeled "OK," as is the Maintain RCS Heat Removal safety function. There is, however, a precursor present to the Power/Coolability Mismatch challenge since inadequate RCS inventory will develop with a subsequent core uncovery if ECC injection is not instituted. The information indicating the approach to core uncovery is available and includes:

Instrument	Trend
RCS subcooling	Decreasing or lost
Pressurizer level	Decreasing level
Reactor vessel level monitoring system (RVLMS)	Decreasing level
Sump Level	Increasing.

The precursor to the change in core geometry will also develop during this phase and will include an increase in steam super–heat due to the initial uncovery of the core. The instrumentation necessary to supply this precursor information is available and includes:

Instrument	Trend
Hot leg RTD	Increasing temperature (beyond T_{sat})
CETs	Increasing temperature (beyond T_{sat}).

The steam super–heat can be readily determined knowing the RCS saturation pressure obtained from the pressurizer pressure instrumentation.

One of the strategies to mitigate the potential challenge to the change in core geometry is restoration of injection into the RCS. Because the ECC high and low pressure injection pumps have failed, the operator's only recourse at this time is to depressurize the RCS using the steam generators atmospheric dump system or primary system PORVs. Since core uncovery will not occur for at least one hour into this event, the operators will have initiated a cooldown at probably no later than thirty minutes into the event. While the sequence does not include this operator action, it is mentioned here since use of the atmospheric dump valves or primary system PORVs will enable a cooldown to allow injection of the safety injection tanks, which would then actuate and provide core cooling (see Reference 3). The accumulators will act only to delay the potential core melt, but because the level and pressure instrumentation for the accumulator tanks are not qualified, the operator will have to rely on the CET response to determine if the accumulators discharge. Once the accumulators empty, core uncovery will occur again, indicated by an increasing CET temperature. With the increasing core exit temperatures, the next challenge on the tree, Flow Diversion/Blockage, will develop.

During Phase 1, neither the Prevent Containment Failure safety objective nor the Mitigate Fission Product Release From Containment safety objective is threatened by any challenges because containment sprays and heat removal fans are operating to control containment pressure and temperature.

Because there is at least one hour before core uncovery, sufficient time exists for the operator to attempt preventative actions to delay or mitigate uncovery of the core.

Phase 2

During Phase 2, the core experiences complete uncovery, with the fuel pellet and cladding temperatures reaching melting conditions (approximately 3500 to 4500°F). Referring to Figure 10, the Maintain Core Heat Removal safety function is challenged and is labeled "CHA." During this phase, challenges exist if the following potential mechanisms are present:

1. Clad rupture/relocation
2. Rubble bed formation
3. Inadequate core cooling
4. Non–coolable in–core geometry.

The information necessary for the operators to identify the above challenge mechanisms is available only indirectly through the response of the CETs, SRNIs, and SPNDs, which will fail in this phase because the qualification range will be exceeded. The hot leg RTDs are inadequate for these information needs since these instruments will be "pegged" at their maximum temperature range of 700°F. The temperature of super–heat must be determined from knowledge of the RCS pressure from the pressurizer pressure instrument and CET response, which is available only during the early portion of this phase.

At the end of Phase 2, the top portion of the core undergoes a gross melt with a slump and collapse occurring at about 108 minutes into the event. During this phase, which lasts from 80 to 110 minutes, the operators have about thirty minutes to institute ECC pumped injection to prevent a core relocation into the lower plenum for this particular sequence. At this stage of the event, the amount of available time to take action to depressurize the RCS, if not already attempted, would be too short in duration to prevent the partial core melt. While the Maintain Vessel Boundary safety function is potentially challenged (and is labeled "CHA" in Figure 10), the precursors also exist for a possible over–temperature failure of the lower head if the failed ECC pumps are not repaired quickly. The over–temperature challenge of the vessel lower head could result from a relocation mechanism which is non–coolable. The excessive CET temperatures and SPND alarms that exist during Phase 2 provide indirect information that a major core heatup and melt is in progress.

Although the containment spray pumps and fan cooling systems are operating to control pressure and temperature during Phase 2, the operators will have indication of potential challenges to the Maintain Pressure Control and Maintain Temperature Control safety functions, shown in the Prevent Containment Failure safety objective tree in Figure 11. The challenges will result from a buildup of non–condensable gases in the containment along with the increasing containment pressure and temperature due to the continued addition of RCS inventory through the break. The containment pressure and temperature instrument will provide the information to monitor over–pressure control. The post–accident sampling system will provide indication of the increasing hydrogen in the containment, and as required by current EOPs, the hydrogen recombiners will be placed in service at this time. Because of rapid core–wide oxidation occurring at this time, the buildup of hydrogen will continue during this phase, both in the RCS and the containment.

As shown in the Mitigate Fission Product Release from Containment safety objective tree in Figure 12, the Maintain Control of FP Inventory in Containment Atmosphere safety function and the Maintain Control of FP Release from Containment Water safety function are both challenged and are labeled "CHA." The presence of fission products in the atmosphere will be indicated by the containment radiation monitors; however, there will be no means of controlling these challenges other than the use of containment sprays currently in use to control containment pressure. At the end of Phase 2, the core begins relocation into the lower plenum.

Phase 3

At the beginning of Phase 3, which lasts from 110 to 133 minutes, the core begins relocating into the lower plenum. The Maintain Core Heat Removal safety function is lost and is labeled "LOS" in Figure 13. Meanwhile, the non–coolable relocation mechanism is present to create a Vessel Over–Temperature challenge to the vessel lower head. Also, the Maintain Vessel Boundary safety function is labeled "CHA" in Figure 13. Because the SRNIs located in the reactor cavity are not qualified for the DBA LOCA environment, these instruments are not available because they failed earlier in Phase 2. Because the entire core relocates into the vessel lower plenum, those remaining CETs and SPNDs that may have survived Phase 2 are also lost during this phase. As a consequence, the operators have little or no means of meeting their information needs to follow or determine the behavior of the core during Phase 3. The RCS pressurization (indicated by the pressurizer pressure instrument response) that occurs during Phase 3 as a result of the molten core contacting the water in the lower plenum is the only indirect means of potentially indicating that the core is relocating. The operator has no other information to determine whether the core has relocated into the lower plenum, whether it is coolable if injection is restored at this time, or whether the lower head is in danger of failing and within how much time.

At this time, the operator would still be attempting to reduce RCS pressure if he had not successfully reduced it during prior phases. Because of the noncondensable gases produced during Phase 2 and 3, it would probably be ineffective to use the steam generator secondary system to depressurize the RCS (using the atmospheric dump system if power is unavailable or the bypass system if power is available) since the ability to effectively condense steam on the primary system would be severely inhibited by the gases present. If the steam generators are to be used in this situation, it would be necessary to relieve the noncondensable gas from the system by using the hydrogen vents and by restarting the main coolant pumps to clear the gasses from the generators. Use of PORVs during Phase 3 may be a more effective way to depressurize; however, at this stage, if depressurization has not been initiated earlier, Phase 3 is probably too late for this strategy in this particular sequence. The acoustic flow monitoring system and PORV discharge pipe temperatures along with RCS pressure instrumentation may provide the verification that the PORVs were actuated. Furthermore, depressurization with PORVs could induce failure of the surgeline. While this failure would assist in depressurization, the pressurizer pressure instrument would no longer provide reliable information on RCS pressure. If RCS pressure indication is needed to place the plant in a long–term cooling mode later during the event, loss of pressure instrumentation could make recovery operations more difficult since RHR systems require pressure/temperature information for operation.

While successful depressurization of the RCS would actuate the accumulators during Phase 3, only a delay in the eventual failure of the lower head would be realized by this action. However, more time would be available which could be used to affect repair of the ECC pumps. If ECC pumped injection cannot be reinstated during this phase (if it is known to be beneficial at this time), failure of the lower head will result. The operator will have no instrumentation to determine that head failure is about to occur.

At this time in the event, flooding of the cavity may be a strategy to prepare for eventual head failure and ejection of the core materials into this region. However, there is no instrumentation nor means for the operators to verify this strategy nor is there special equipment to perform this action. If continued operation of the containment sprays permits partial filling of the cavity, then continued operation of the sprays to control containment pressure may also allow water to enter this region and permit cooling of core materials upon head failure.

The challenges for the Prevent Containment Failure and the Mitigate Fission Product Release From Containment safety objectives are the same as those discussed for Phase 2 above, except the conditions are increasing in severity.

Without the restoration of pumped ECC injection during Phase 3 (assuming the relocated core could be cooled at this time), lower head failure will occur. The duration of this phase is about 23 minutes, so there is little or no time at this point in the event for the operators to take action to prevent head failure.

Phase 4

This phase begins with failure of the lower head and includes injection of the molten core into the containment. The Maintain Vessel Boundary safety function is lost and is labeled "LOS" in Figure 14. In addition, Figure 15 (the Prevent Containment Failure safety objective tree) shows that both the Maintain Pressure Control and the Maintain Temperature Control safety functions continue to be challenged in this phase. Because of the failure to switch to recirculation, containment sprays are lost during Phase 4. The hydrogen burn that occurs at 162 minutes violates the qualification conditions of the instruments in the containment; therefore, all of the containment instruments are assumed to fail during Phase 4. Because the containment pressure and temperature conditions exceed the qualification of the instruments without the hydrogen burn, these instruments would be lost early in Phase 4 anyway. As a consequence, if the operators were not successful in restoring the containment sprays early in Phase 4, eventual loss of all remaining instruments, in addition to failure of the containment boundary, would occur. Because there is a long period of time before containment failure occurs at 1444 minutes, there should be sufficient time to restore containment spray, or as a last option, vent the containment to prevent containment failure if equipment to vent the containment was installed. In any event, the loss of instrumentation for containment pressure and temperature would severely inhibit the operator's ability to determine whether any strategies are effective to control containment pressure and temperature during Phase 4. If some containment pressure instrument indications were limited in range to the design pressure, there would be no information available to the operators to indicate containment failure or to verify strategy effectiveness during this phase.

Figure 16 shows that both safety functions in the Mitigate Fission Product Release From Containment safety objective tree also continue to be challenged. With the exception of containment sprays, there is little or no equipment affording strategies to mitigate these challenges.

4.7 Evaluation of Instrument Shortcomings

This section discusses the instrumentation shortcomings identified in Tables 4 and 5 of Section 3, which also identified the information needs for which there are no associated available instruments and information which can only be supplied indirectly. It is important to note again that the instruments were assumed to fail once their qualification limits were exceeded. The instrument availability, therefore, represents a conservative lower limit. Table 7 shows, for each phase of the event, information needs and the instrumentation for the accident sequence in which shortcomings exist because the instrument does not exist or information is supplied only by indirect means and could mislead or confuse the operating staff.

As stated earlier in this section, there is sufficient information to effectively manage the accident through Phase 1 since the environmental conditions do not exceed the instrument qualification limits. This portion of the event does not pose any shortcomings and is included here for completeness. Should ECC injection be restored during this phase, the consequences of the accident will be terminated.

Phase 2 begins with continued core uncovery and heatup. Major fuel damage will begin with core melt also occurring during this phase. The only sources of information regarding core damage status are the CETs and SPNDs, which have not been proven to supply reliable information for this purpose. Since the SPNDs are not qualified for harsh environments, they are not expected to function for this portion of this accident. Failure of the SPNDs and CETs will accompany gross core melt, while the indication of the passage of core material into the lower plenum is provided by the SRNIs. However, with the harsh containment environment at this time, the SRNI indications may be erroneous with eventual failure occurring during this phase since these instruments are not qualified for harsh environmental conditions. This relocation may also be indicated by a step-like increase in the RCS pressure and hot/cold leg RTD responses, but only if the relocation has occurred with a refilled RCS. Otherwise, the hot/cold leg RTDs are of no use because their instrument range is limited to maximum temperatures of about 750°F. These instruments are expected to fail at the end of Phase 2 due to the severe environmental conditions in the RCS and reactor vessel cavity.

Of particular importance to severe accident management is the need to know if the core has relocated into the lower plenum and whether it poses a threat to vessel integrity. Such information is important even if the operating staff has refilled the RCS with ECC water because of the potential to establish an upper crust layer that will insulate molten material in the lower head from the RCS coolant. Because there are no indications of lower head temperature or the status of the core, whether in the core boundary or in the lower plenum, it may not be possible with present plant information and current support staff capabilities to effectively manage this type of severe accident behavior following a noncoolable relocation of the core into the lower plenum.

Some of the information needed to detect core relocation could be inferred from indirect information sources, but because some of these sources are not qualified to DBA environmental conditions, there may be accident conditions and instrument responses which could mislead the operators. For example, the passage of core material into the lower plenum can be inferred only from the SRNIs. The source range monitors are not qualified for DBA adverse environmental conditions and, therefore, could be subject to large errors and failure. In addition, because the passage of core material into the lower plenum could also result in an increase in energy addition to the coolant, an increase in RCS pressure and temperature may also occur. Furthermore, should the large amount of hydrogen present

Table 7. Instrument limitations for event phases

Phase	Event	Instrument	Limitation
1	Core uncovery, fuel rod heatup, fuel rod swell, and rupture	RCS subcooling, CETs, hot/cold RTDs, pressurizer Pzr, level, RVLMS	None, fuel clad swell and rupture/failure indirectly indicated by CETs and coolant activity. Approach to core uncovery indicated by RVLMS.
2	Core melt, core damage status, coolable/noncoolable core geometry	SRNIs, SPNDs, Pzr pressure, and hot/cold leg RTDs	SRNIs are not qualified for harsh environs. Failure or error will eliminate possible detection of relocation of core. No direct information available on coolability status.
3	Relocation of core materials to lower head, coolable/ noncoolable geometry, vessel integrity	SRNIs, Pzr pressure, and hot/cold leg RTDs	SRNIs will fail. No other information available RCS pressure, temperature (P,T) increase may be confused with noncondensable gases entering steam generator primaries. No information on approach to head failure.
4	Vessel head failure, containment integrity, hydrogen accumulation and detonation	Containment P,T instrumentation, radiation monitors	Containment P,T instrumentation may fail due to severe P,T environs. Information on H_2 may not be timely or possible from sampling system since range is limited to 10%. No information on basemat attack/failure.

in the RCS enter the steam generators during the sequence, the degradation in heat removal capability in the steam generators would also cause an increase in RCS pressure and temperature. If it took place over a relatively short period of time, this event could be confused with a relocation of core material into the lower plenum and mislead the operators, diverting their attention to actions associated with mitigating a core relocation when they should be addressing a loss of the heat capability to remove. If the source range monitors incurred large errors as a result of a harsh containment environment, the response of these instruments could confuse the operators. The potential for conflicting information is high for these two similar RCS responses.

If injection is restored before the lower head has failed and if single phase natural circulation has also been recovered, the ECC injection pumps will cause an increase in RCS pressure for the small breaks sizes indicative of this sequence. For these small break sizes, the RCS pressure will stabilize at a pressure where the break flow equals the injection flow. Because high–pressure injection pumps have high–shut–off heads, the RCS could be pressurized to pressures above 1500 psia. Since there is no information available to determine the approach to lower head failure, the operators could be mislead to believe that the core material has been recooled. With the RCS at elevated pressures, there is the potential that a challenge which could lead to direct containment heating is in progress with no information to identify these conditions.

Should vessel failure occur, no information will be available to the operators to monitor the attack of the molten material on the vessel cavity, including the basemat. Operation of the containment sprays are the only means of adding inventory to the cavity; however, present containment instrumentation generally does not provide operators with sufficient information to know if the cavity has been flooded, should this strategy prove to be effective. Furthermore, the use of permanent pool seals could further inhibit cavity flooding in some designs.

To address the mentioned instrument shortcomings, in lieu of adding additional instrumentation, plant organizations could consider development of computational aids, where appropriate, to fill the information needs which are not currently supplied by instrumentation. For example, the lack of information on lower head integrity could be supplied by a computational aid. Technical support teams could perform calculations initialized with information from the reactor vessel level indications and CETs. Information from these instruments could provide an input for the vessel failure computation based on the trend of these data prior to failure. Valuable information such as an estimate of the earliest time at which head failure could occur would be provided. If repair of equipment could be facilitated prior to lower head failure, an estimate of the head failure time could be used to prioritize which equipment, such as pumps and inventory supplies, should be repaired first. Such computational aids could provide a backup to the indirect information supplied to identify a core relocation and would currently provide the only source of information for head failure, since no instrumentation is now available to provide such information.

Because the containment environmental conditions can exceed the containment instrument qualification limits, analysis or computational aids could be used to back up the loss of vital instruments, such as containment pressure and hydrogen concentration. However, use of computational aids *should not* replace the need for assessing and understanding instrument behaviors during a severe accident. At a minimum, computational aids could supply key information which is not supplied directly by existing instrumentation, but which is essential to the operating staff for effective accident management.

5. CONCLUSIONS AND RECOMMENDATIONS

The importance of this evaluation is that it demonstrates the applicability of safety objective models for accident management, including information needs, instrument availability, and environmental qualification through application of the methodology to a severe accident sequence for a PWR with a large, dry containment. Functional models were exercised using data from an existing accident phenomenological calculation to further determine whether the required information will be available for effective accident management, while also accounting for the effects of environmental conditions on equipment and instrument availability.

In summary, this work demonstrates that:

1. The methods developed herein can be used to identify the information needed by plant personnel for management of severe accidents. The methods will also identify areas in which information needs will not be met and where difficulties may occur in understanding plant status, evaluating strategy selection, and determining strategy effectiveness for severe accident sequences.

2. The approach in this work would be useful in providing the means to incorporate new information, as it becomes available from the Severe Accident and Accident Management Research Programs, into utility accident management programs.

3. The safety objective trees display plant severe accident information in a manner that promotes understanding of plant information needs. Although the severe accident conditions presented on the trees are not new or unique, the structure of the trees allows easy visualization of what is important for a broad spectrum of severe accidents. This broad perspective can also be used to provide insights to the manner in which challenges to plant safety can be identified and what alternate means may be available to prevent or mitigate a severe accident.

4. The assessment of information needs for a PWR with a large, dry containment indicates that there is not sufficient instrumentation to understand the status of the core, such as temperatures and general location of core material, once core melting has begun. There is also insufficient information to determine whether the vessel lower head is being attacked and whether failure should be expected. The lack of information in these two areas increases the difficulty in making a decision of whether to use resources for strategies relating to preserving vessel integrity or to use strategies dealing with the preservation of containment integrity.

5. There are about twenty information needs in which personnel involved in accident management have the potential to be mislead because they must rely on interpretation of instruments that do not directly supply the needed information. Five of these were judged to be important and could be grouped into two types of information needs: (a) those related to the inability to understand core status following the initiation of melt, discussed above, and (b) those related to an inability to determine the location of a bypass to or leaks from the containment, such as the location of an interfacing system loss-of-coolant accident.

6. There may be instrumentation or other means that can be used to supply information that is not currently available. For example, source range instruments and self-powered neutron detectors may provide information on core status during relocation, but this information would require the capability for special interpretation which does not exist for this purpose. Computational aids that interpret or extend the use of instrumentation have the potential to provide the additional needed information.

From the application of methods to a specific sequence, some observations and recommendations can be summarized regarding management of a selected, severe accident in light of existing instruments in PWRs with a large, dry containment. It should be mentioned that instrument availability is considered to be a conservative lower limit since instruments were assumed to be unavailable during the sequence once its qualification limits were exceeded. The observations and recommendations include:

1. There is sufficient information to manage the selected, severe accident up to the time just prior to the onset of fuel/cladding melt. The measurement range of existing instruments

up to this time appear adequate. Following the onset of fuel/cladding melt, measurement ranges for some instruments would be exceeded, such as containment hydrogen concentration, CETs, and cold and hot leg RTDs. Should the CETs fail, the cold and hot leg RTDs could provide important information regarding core coolability if the upper ranges of these instruments were not limited from 600 to 750°F.

2. Assuming all instruments fail immediately after their qualification conditions are exceeded, there is no direct information available to the operators to follow the relocation of the core during the time period between the initiation of core melt and core relocation into the reactor vessel lower head. The source range nuclear instruments may provide indirect information, but are not qualified for harsh containment environments and can develop large errors which could mislead the operators prior to failure. There is also no information to detect the approach to lower head failure.

3. In the containment, little or no information would be available following lower head failure because all of the measurement qualification limits would be exceeded. Some instrumentation may survive; however, the effects of the harsh environmental conditions on instrument inaccuracy has not been evaluated. Should the containment pressure/temperature instruments fail, there are no backup sources of information.

4. A more detailed evaluation of measurement error and survivability for severe accidents is needed. This is not to suggest that a more extensive environmental qualification program is required; rather, it is important to understand how the instruments will behave so that the operating staff will know what information is reliable for use in understanding plant status and managing the accident effectively.

5. A more detailed evaluation of the capability of certain instruments to provide information on severe accident behaviors is needed. For example, use of source range nuclear instruments to indirectly indicate that a core relocation is occurring needs evaluation.

6. Computational aids could be developed to provide the needed information, which is unavailable during a severe accident sequence due to support system failures, instrument inaccuracy, or instrument unavailability. These analysis or computational aids could include: lower head failure, containment hydrogen buildup and combustion/detonation, basemat attack and failure, and containment failure. These sources of information should be made available to the operating staff since this information (a) is not provided by present plant instrumentation, (b) is supplied by unqualified instruments which are only indirect indicators, and (c) are qualified but are susceptible to failure due to the severe environmental conditions with no backup sources available.

7. The development of analysis and computational aids should not replace the need to evaluate and understand plant instrumentation response during severe accident conditions. Rather, they should be used in conjunction with a thorough understanding of plant instrument limitations and shortcomings to enable the operating staff to effectively manage severe accidents.

It is conservative to assume that all instruments will fail when their qualification conditions are exceeded. If a more realistic assessment of this situation is desired, it is recommended that a detailed evaluation of measurement survivability for severe accidents be performed to determine the conditions for which the instruments will function beyond their qualification limits. This evaluation would need to consider the entire instrument system that is exposed to the harsh conditions, including the transducer, cabling, electronics, etc. Also, based on the recommendations of this study, an environmental envelope encompassing risk–important severe accident conditions will be developed in FY–1990 for a PWR and BWR to provide insight for the conditions that would be needed for the qualification of instruments for severe accidents.

Each utility is now addressing the severe accident issue for its plants, which will include (a) completion of individual plant examinations (IPEs) and (b) eventual development and implementation of a severe accident management plan. Use of the methodology described and demonstrated in this document will provide tools to assist in the development, implementation, and evaluation of effective accident management programs.

6. REFERENCES

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