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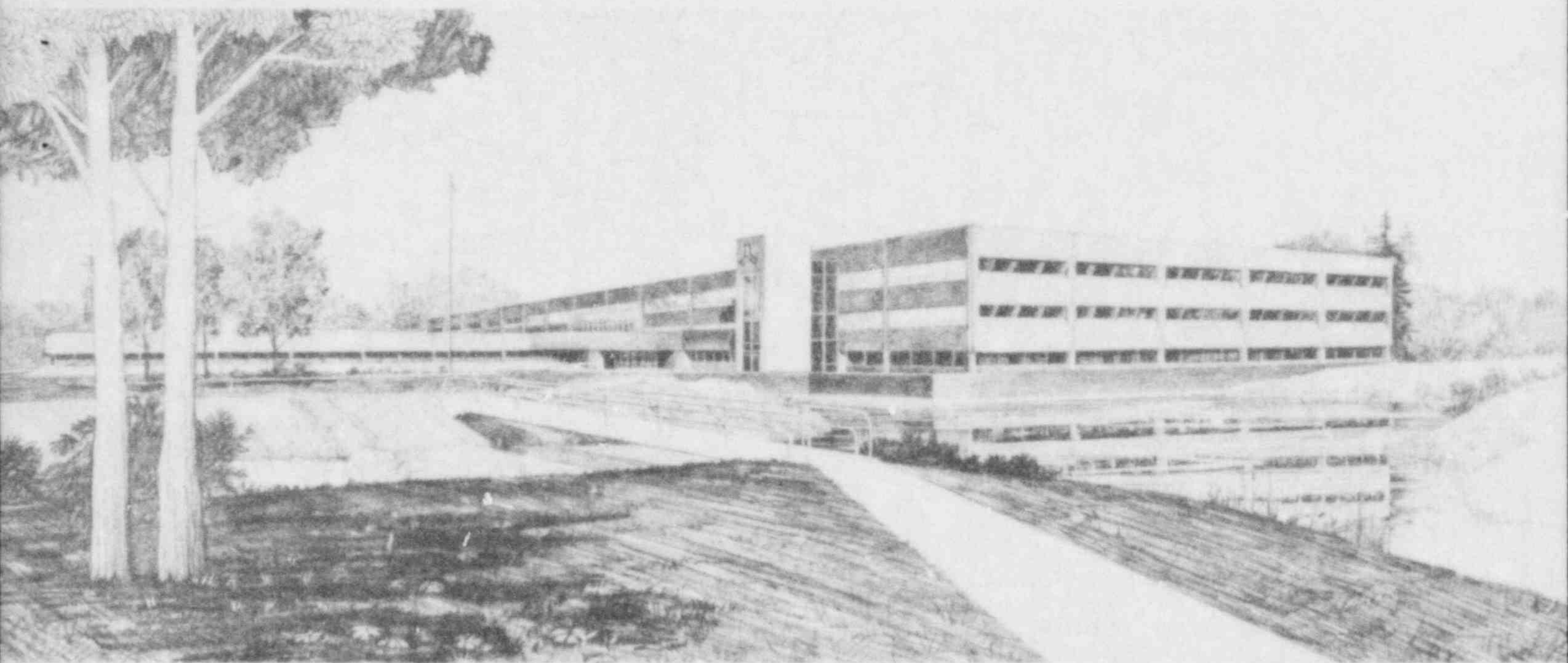

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Severe Accident Sequence Analysis (SASA) Program Sequence Event Tree: Boiling Water Reactor Anticipated Transient Without Scram

Sharon Z. Bruske
Ronald E. Wright

April 1984

Prepared for the
U.S. Nuclear Regulatory Commission
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SEVERE ACCIDENT SEQUENCE ANALYSIS (SASA) PROGRAM SEQUENCE EVENT TREE: BOILING WATER REACTOR ANTICIPATED TRANSIENT WITHOUT SCRAM

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ABSTRACT

The United States Nuclear Regulatory Commission is sponsoring an on-going safety research program to assess dominant risk events in boiling water reactors. As part of this program, a sequence event tree for a boiling water reactor anticipated transient without scram (ATWS) has been developed and quantified. The goal of the sequence event tree is to provide a logical representation of the systems that must respond to an ATWS, the required operator response to the event, operator actions that could be performed in response to multiple failures, and the phenomenological concerns. The purpose of the sequence event tree is to provide a basis upon which to perform additional deterministic thermal-hydraulic and core damage analyses in the most cost effective manner based on the most likely sequence of events that will lead to containment/core damage. The ATWS sequence event tree is based on the General Electric Owners Group emergency procedure guidelines and on preliminary deterministic thermal-hydraulic analyses performed by EG&G Idaho, Inc. personnel at the Idaho National Engineering Laboratory under direction of the Severe Accident Sequence Analysis Program.

The ATWS sequence event tree is based on main steam isolation valve (MSIV) closure as the initiating event. The ATWS sequence event tree logic is based on three means to achieve low power or subcriticality: (a) early boron injection, (b) level control to top of active fuel and pressure control without boron injection, or (c) late boron injection that has been preceded with either level or pressure control.

Out of ~200 potential containment/core damage sequences, additional deterministic thermal-hydraulic analyses can now be concentrated on the most likely containment/core damage sequences allowing results to be obtained quicker and in a cost effective manner.

FIN No. A6354—Severe Accident Sequence Analysis.

SUMMARY

Under the Accident Sequence Evaluation Program (ASEP), sponsored by the United States Nuclear Regulatory Commission (USNRC), the anticipated transient without scram (ATWS) at a boiling water reactor (BWR) has been identified as a dominant risk event. Areas of significant uncertainty associated with an ATWS were also identified. For example, in an existing probabilistic risk assessment, it was conservatively assumed that if the power conversion system is not available and if the reactor is not made subcritical, then the high pressure makeup systems will not be able to keep the core covered and core damage will occur. To address the ATWS and resolve as many of the uncertainties as necessary, deterministic thermal-hydraulic and core damage analyses are being performed under the Severe Accident Sequence Analysis (SASA) Program.

Preliminary deterministic thermal-hydraulic analyses have been performed for a loss of the power conversion system with a failure to scram. From these analyses it became apparent that there are many sequences of events that can lead to a stable plant condition or to containment/core damage following an ATWS. Without a structured approach, further deterministic thermal-hydraulic analyses of the numerous combinations of events leading to containment/core damage would be an expensive and time consuming process. To provide a structured approach, a BWR ATWS sequence event tree was developed. This sequence event tree logically represents the systems that could respond to an ATWS, the required operator response to this event, operator actions that could be performed in response to multiple failures, and the phenomenological concerns. By quantifying the sequence of events that result in containment/core damage, the most likely sequences are identified, and additional thermal-hydraulic analyses can be concentrated on these most likely sequences allowing for results to be obtained quicker and in a cost effective manner. A review of the most likely sequences also establishes the analytical models necessary to analyze important phenomena and provides direction for expanding calculational capability of the existing thermal-hydraulic analysis codes. The BWR ATWS sequence event tree is based on findings from preliminary thermal-hydraulic analyses and does not reflect any additional findings from thermal-hydraulic analyses that were performed after the sequence event tree was developed.

The BWR ATWS sequence event tree is based on main steam isolation valve (MSIV) closure as the initiating event. The BWR ATWS sequence event tree logic is based on three means to achieve low power or subcriticality that will be within the heat removal capability of the residual heat removal (RHR) system heat exchangers. These three means of achieving low power or subcriticality are not only based on the emergency procedure guidelines but are also based on insights gained from preliminary deterministic thermal-hydraulic analyses. The three means of achieving low power or subcriticality are: (a) early boron injection, (b) level control to top of active fuel and pressure reduction in a controlled manner without boron injection, or (c) late boron injection that has been preceded with either level or pressure control. The purpose of the thermal-hydraulic analyses is to examine the effects of failure to scram; therefore, manual rod insertion is not considered.

The two most likely sequences that result in containment/core damage as shown on the BWR ATWS sequence event tree (the quantified BWR ATWS sequence event tree is shown in Section 4) represent (a) a totally unmitigated transient (no early or late boron injection and no level control or pressure reduction in a controlled manner), and (b) a high-pressure boiloff transient.

The action in the emergency procedure guidelines dealing with suppression pool heat capacity temperature limits was strictly interpreted in the preliminary thermal-hydraulic analyses as a pressure reduction that follows the suppression pool heat capacity temperature curve instead of the initiation of a rapid depressurization. In this report, pressure control (controlled pressure reduction) and depressurization (e.g., automatic depressurization) are two separate actions. Since strictly following the suppression pool heat capacity temperature curve is not specifically part of the emergency procedure guidelines, it was judged that operator training would not normally reflect this particular aspect of this action. Upon the basis of this and the fact that an MSIV closure with a failure to scram represents a significant challenge to systems and operator actions due to the relatively short available reaction time, some of the actions could be difficult to perform, the simulators are limited in realistic responses for extreme accident conditions, and based on the human error probabilities from

WASH-1400, a best estimate failure probability of 0.9 was assigned to most of the operator errors shown on the sequence event tree. A sensitivity analysis was also performed to determine if there would be any additional most likely containment/core damage sequences if the operator failure probabilities were varied. First a failure probability of 0.5 was assigned to failure of each operator action and the containment/core damage sequence probabilities were calculated. Then, a failure probability of 0.1 was assigned to failure of each operator action, and again, the containment/core damage sequence probabilities were calculated. In this way, operator failure probabilities span approximately one order of magnitude. The results of the sensitivity analysis show that if the operator failure probability is changed to 0.5, there are two additional most likely containment/core damage sequences. These two additional most likely sequences represent (a) a transient in which the only major successful action is level control, and (b) a transient in which the only major successful action is pressure control. The sensitivity analysis also shows that if the operator failure probability is changed to 0.1, there are no additional containment/core damage sequences that can be considered as most likely.

Out of ~200 potential containment/core damage sequences, additional thermal-hydraulic analyses can now be concentrated on the four most likely containment/core damage sequences allowing results to be obtained quicker and for a much more reasonable cost. Also, analysis of the four most likely risk sequences indicates the importance of level control and pressure control by the operator on accident mitigation.

The sequence event tree also provides additional insights. It helps one visualize what critical plant information an operator requires in order to make a decision under many plant conditions. Since the sequence event tree identifies required operator responses to an accident initiator and the operator actions that could be performed in response to multiple failures, it can be used as a tool to evaluate the adequacy of emergency procedures. The sequence event tree may also be used as a basis for developing control room staffing criteria based on the timing and the type of required operator actions that may be necessary.

ACKNOWLEDGMENTS

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CONTENTS

NOMENCLATURE

ADS	Automatic depressurization system
ASEP	Accident Sequence Evaluation Program
ATWS	Anticipated transient without scram
BWR	Boiling water reactor
CRD	Control rod drive
HPCI	High pressure coolant injection
LOCA	Loss-of-coolant accident
LPCI	Low pressure coolant injection
MSIV	Main steam isolation valve
NREP	National Reliability Evaluation Program
PRA	Probabilistic risk assessment
RCIC	Reactor core isolation cooling
RCS	Reactor coolant system
RHR	Residual heat removal
RPT	Recirculation pump trip
RPV	Reactor pressure vessel
SASA	Severe Accident Sequence Analysis Program
SLCS	Standby liquid control system
SRV	Safety relief valve
TAF	Top of active fuel
TVA	Tennessee Valley Authority
WASH-1400	Reactor Safety Study

SEVERE ACCIDENT SEQUENCE ANALYSIS (SASA) PROGRAM SEQUENCE EVENT TREE: BOILING WATER REACTOR ANTICIPATED TRANSIENT WITHOUT SCRAM

1. INTRODUCTION

Under the Accident Sequence Evaluation Program (ASEP), existing probabilistic risk assessment (PRA) studies of several boiling water reactors (BWRs) have been reviewed. These reviews identified the anticipated transient without scram (ATWS) as an important risk contributor. In addition, a number of areas were identified where high uncertainty exists as to what happens following an ATWS. For example, in an existing PRA, it was conservatively assumed that if the power conversion system were not available and if the reactor were not made subcritical, then the high pressure makeup systems would not be able to keep the core covered and core damage would occur. Under the Severe Accident Sequence Analysis (SASA) Program, deterministic thermal-hydraulic and core damage analyses are being performed at the Idaho National Engineering Laboratory to describe the physical phenomena associated with the ATWS for the purpose of resolving as many of the uncertainties as possible.

Preliminary deterministic thermal-hydraulic analyses have been performed for a loss of the power conversion system with a failure to scram.[1] From these analyses it became apparent that there are many ways or sequences of events that can lead to a stable plant condition or to containment/core damage following an ATWS. Since deterministic analyses of the many combinations of events leading to containment/core damage would be expensive, the need arose to develop a structured approach to the problem. What was necessary was to logically model the sequences of events that might occur following an ATWS, and by identifying the most likely sequences of events that result in containment/core damage, provide a basis for performing additional deterministic analyses in a cost effective manner. To meet this need, an ATWS sequence event tree was developed to logically represent the systems that could respond to an ATWS, the required operator response to the accident initiator, the operator actions that could be performed in response to multiple failures, and the phenomenological concerns. By quantifying the sequence of events that result in containment/core damage, the most likely risk sequences were identified providing a basis for limiting the number of additional thermal-hydraulic and core damage analyses required and thus, minimizing analytical costs. The BWR ATWS sequence event tree is based on findings from preliminary thermal-hydraulic analyses and does not reflect any additional findings from thermal-hydraulic analyses that were performed after the sequence event tree was developed.

The following sections discuss the sequence event tree methodology, the development and quantification of the ATWS sequence event tree, and the results and interpretation of the ATWS sequence event tree.

2. SEQUENCE EVENT TREE METHODOLOGY

The starting point in the sequence event tree construction process is the development of a functional event tree that depicts the complete set of critical safety functions that must be performed in response to a selected initiating event. The functional event tree logically defines the potential success and failure paths that can evolve from the accident initiator. An example of such a functional event tree is shown in Figure 1.

The next step is to develop the sequence event tree by depicting the systems, operator actions, and phenomenological concerns required to meet the needs of the critical safety functions identified on the functional event tree. Both emergency procedures[2] and findings from preliminary thermal-hydraulic analyses[1] are used to identify the required systems and the required operator response to the accident initiator, the operator actions that could

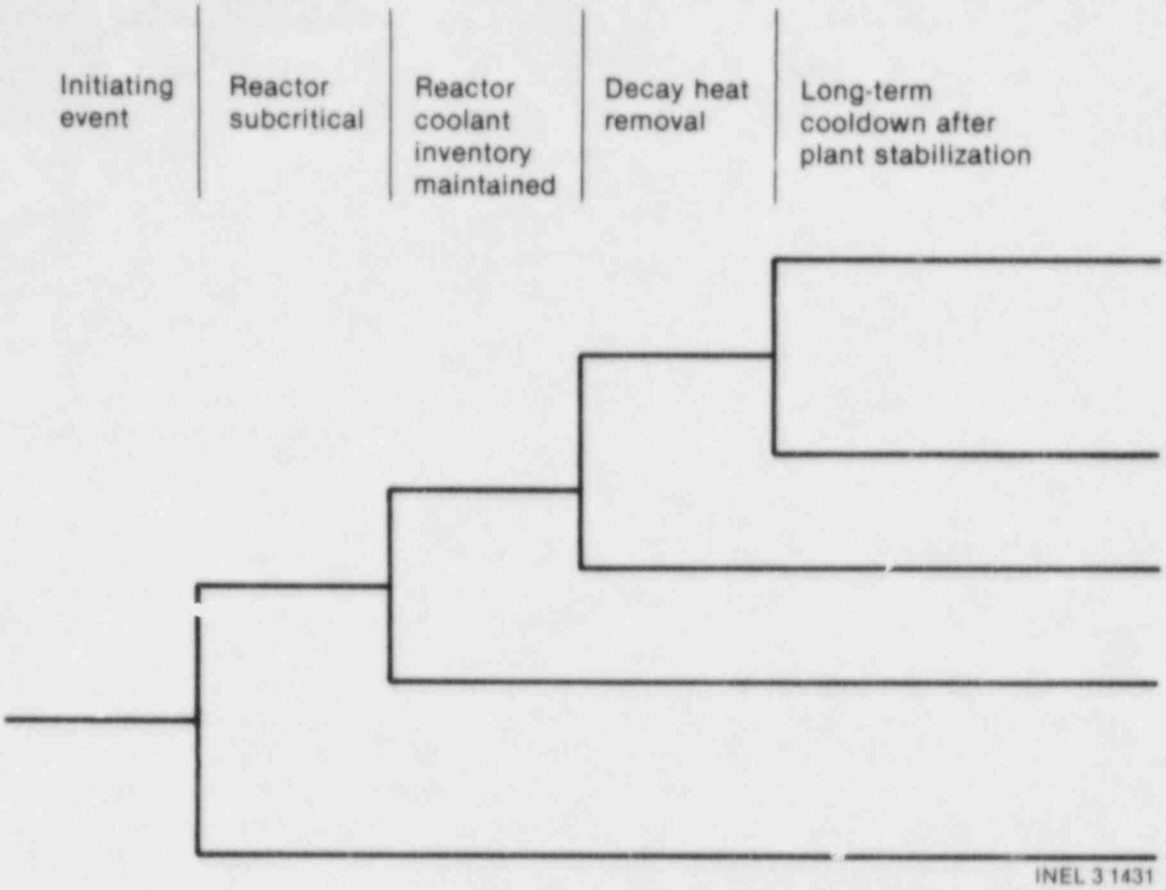

Figure 1. Example of a functional event tree.

be performed in response to multiple failure events, and the phenomenological concerns such as primary containment integrity. A system event tree identifies the specific systems that can operate to satisfy the critical safety functions shown in the functional event tree. System event trees (an example is shown in Figure 2) from existing PRAs can also be used since these already identify the systems that can respond to the accident initiator; however, these system event trees must be expanded to include the various operator tasks and the areas of phenomenological interest. The following section discusses the ATWS sequence event tree.

3. ATWS SEQUENCE EVENT TREE DEVELOPMENT

The goal of the ATWS sequence event tree is to produce a clear and logical representation of the systems that must respond to an ATWS event, the required operator response to the accident initiator, operator actions that could be performed in response to multiple failure events, and the phenomenological concerns. The purpose of the ATWS sequence event tree is to provide a basis to perform additional deterministic thermal-hydraulic and core damage analyses in the most cost effective manner based on the most likely sequences of events that will lead to containment/core damage.

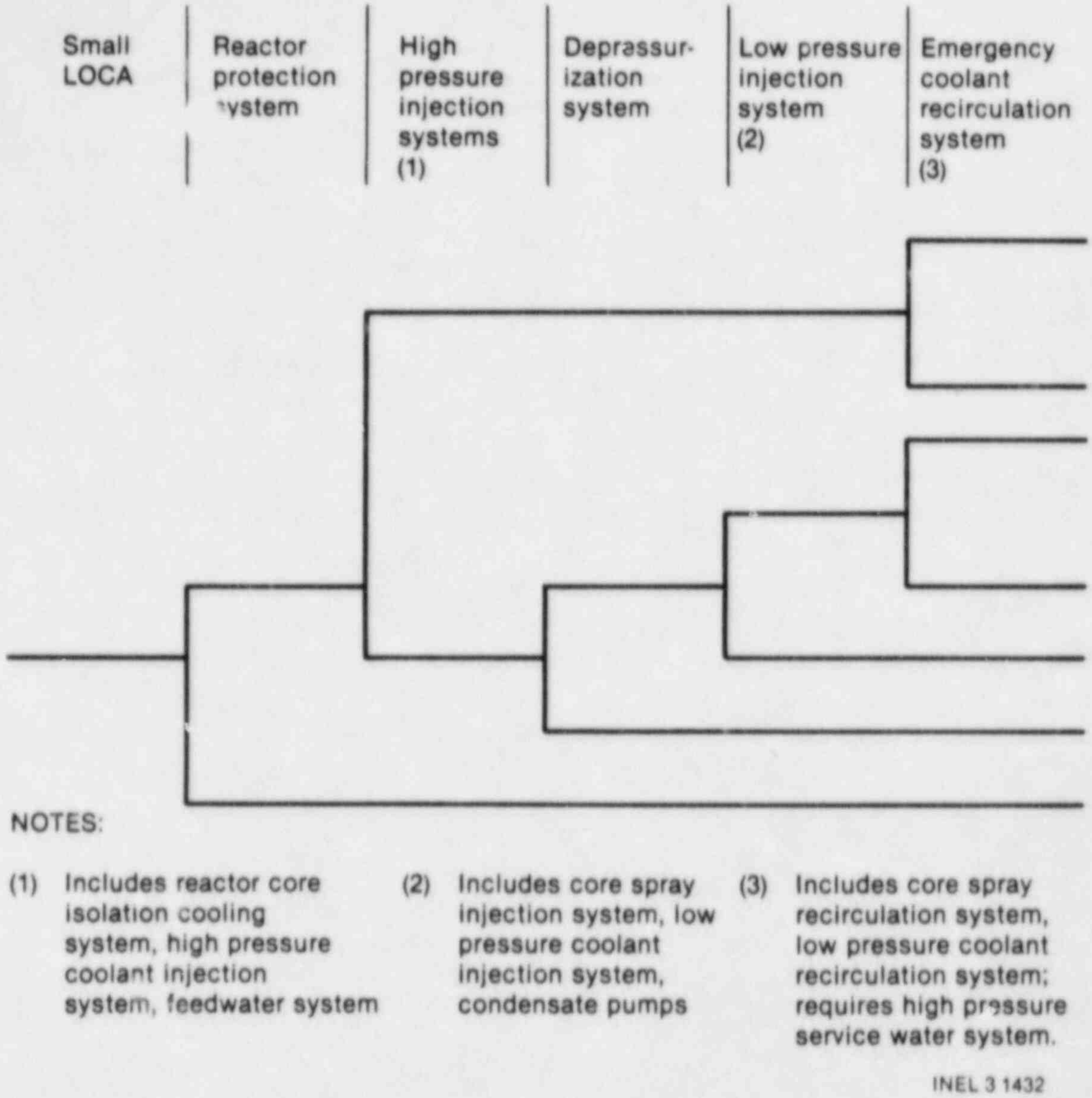

Figure 2. Example of a system event tree (for a small LOCA in a BWR).

The ATWS sequence event tree is shown in Figure 3. The sequence event tree is based on main steam isolation valve (MSIV) closure as the initiating event at the Browns Ferry Nuclear Plant. The major portion of the sequence event tree logic is based on three means to achieve low power or subcriticality that will be within the heat removal capability of the residual heat removal (RHR) system heat exchangers during an ATWS. These means are: (a) early boron injection, (b) level control to the top of the active fuel and pressure reduction in a controlled manner with no boron injection, or (c) late boron injection that has been preceded with either level or pressure control. These means of achieving low power or subcriticality during an ATWS are based on emergency procedures[2] and on insights gained from preliminary thermal-hydraulic analyses.[1] Since the purpose of the thermal-hydraulic analyses is to examine a failure to scram, manual rod insertion is not considered.

The action in the emergency procedure guidelines dealing with suppression pool heat capacity temperature limits was strictly interpreted in the preliminary thermal-hydraulic analysis as a pressure reduction that follows the suppression pool heat capacity temperature curve instead of the initiation of a rapid depressurization. In this report, pressure control (controlled pressure reduction) and depressurization (e.g., automatic depressurization)

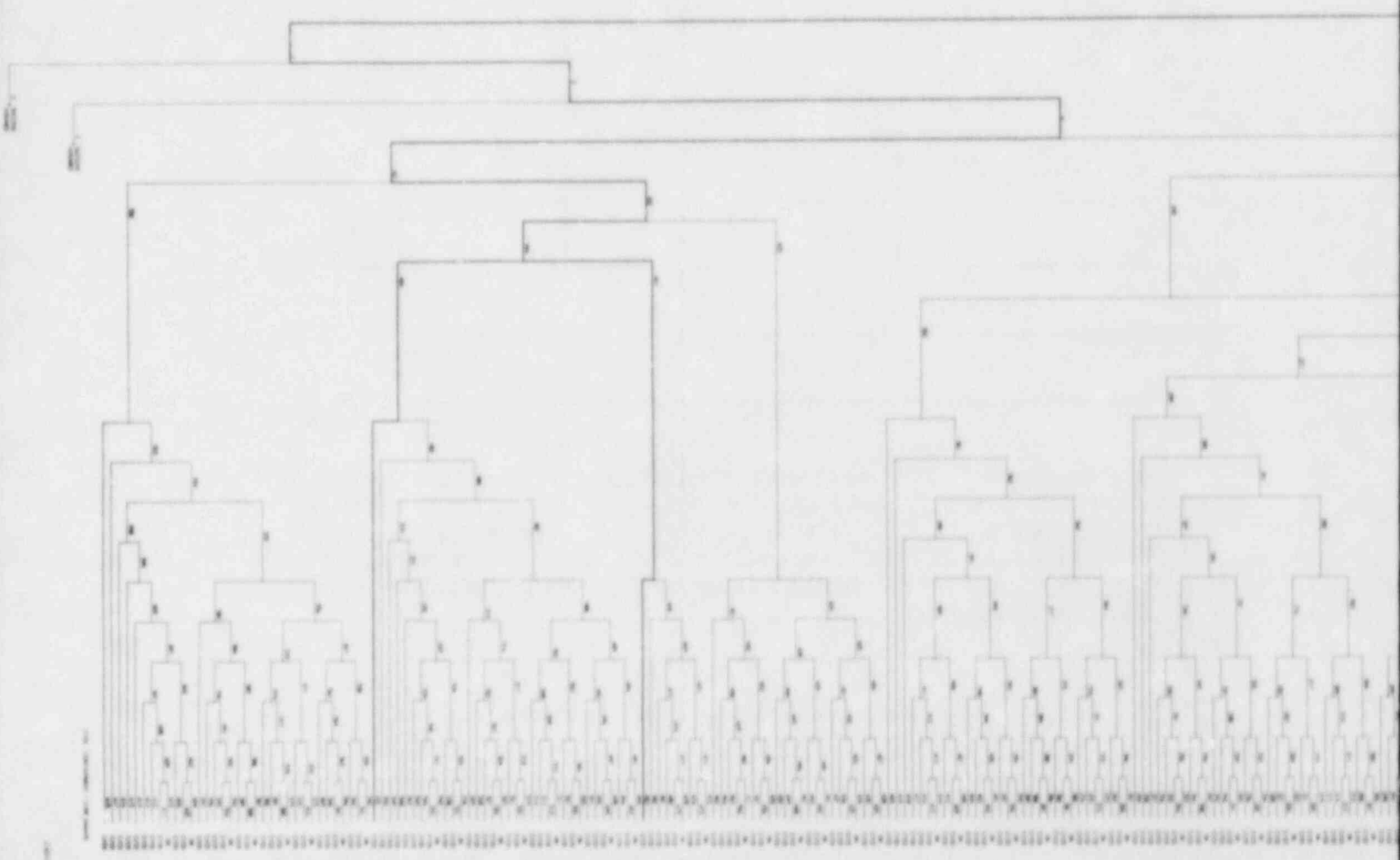

Figure 3. ATWS sequence event tree.

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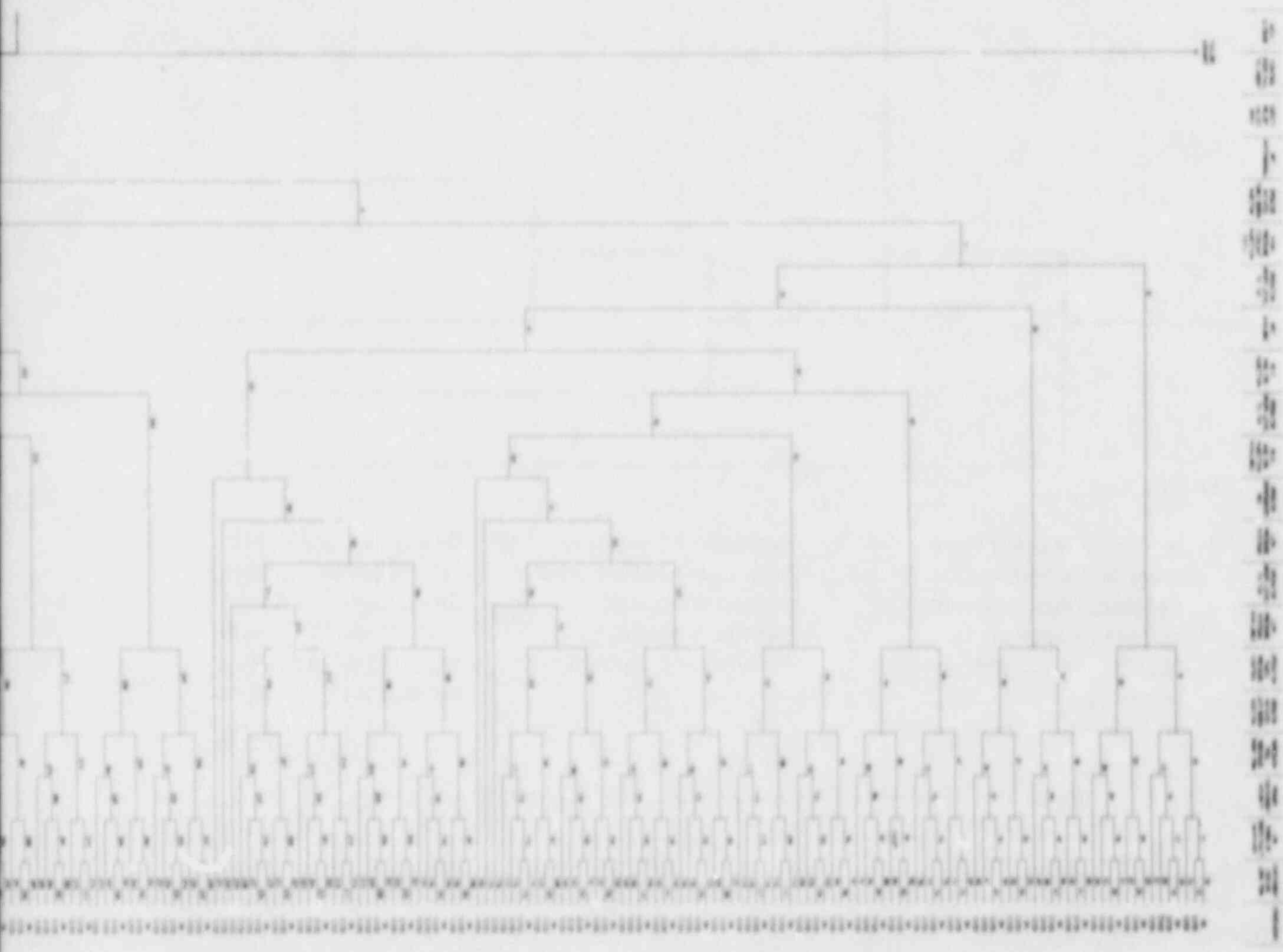

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are two separate actions. A rapid depressurization is only considered necessary when the high pressure injection systems fail to operate.

Sequence Event Tree Heading Success Criteria

Each heading on the ATWS sequence event tree identifies the systems and operator actions required to mitigate an ATWS and the areas of phenomenological concern. The following describes the initiating event and the success criteria for each of the sequence event tree headings.

MSIV Closure. The MSIV closure initiating event was chosen primarily because (a) the frequency is relatively high,[3] (b) the MSIV closure transient results in higher vessel pressures, more severe fuel duty, and a larger amount of steam discharged to the suppression pool than any other moderately frequent transient, (c) most of the operator actions to mitigate the accident are similar to other transient events, and (d) it represents a significant challenge to systems and operator response because of the relatively short available reaction time.

The MSIV closure initiated transient coupled with a failure to insert control rods is a severe challenge to the BWR systems since all the steam produced in the reactor is directed to the suppression pool and may result in sufficient increase in suppression pool temperature and pressure beyond its design limits to challenge containment integrity. In addition, some sequences may lead to core damage, not necessarily in conjunction with containment failure.

Control Rods Fail to Insert. In this study the scram and backup scram are defined to have failed leaving all control rods fully withdrawn. Backup scram consists of two solenoid valves that actuate to vent the air supply to all of the scram valves. Although several scram signals should have been initiated, it is assumed either its electronic circuitry or mechanical systems have malfunctioned and prevented a scram. It is also defined that recovery actions causing control rod insertion are either not initiated or are ineffective until after plant stabilization is achieved.

Safety Relief Valves (SRVs) Open. It has been assumed that adequate overpressure protection is available during an MSIV closure from 100% power with failure to scram given successful recirculation pump trip.

Failure of the SRVs to close, given successful opening, is also not considered in this study. If one or two SRVs stick open, the existing transient would not be significantly altered. The systems necessary to mitigate one or two SRVs sticking open are the same systems being used to mitigate the ATWS.

Recirculation Pump Trip (RPT) Auto/Manual. Success for this event is both recirculation pumps tripping. The signal to initiate the ATWS RPT is high reactor pressure vessel (RPV) pressure. The tripping of the pumps is assumed to immediately reduce reactor power by $\sim$50 to 70%. Failure to trip the recirculation pumps will result in a rapid pressure increase that has been conservatively assumed to exceed the capacity of the SRVs, and it is possible that a rupture of the primary system will occur causing a loss-of-coolant accident (LOCA). The sequence of events and consequences of failure of RPT are not considered in this analysis.

Operator Initiates Early Boron Injection. Success is defined as early operator recognition that an ATWS has occurred and that the operator takes immediate action to activate the standby liquid control system (SLCS). The SLCS is not automatically actuated at the Browns Ferry Nuclear Plant. Successful SLCS operation requires that a sufficient quantity of boron is injected into the core to effect a sustained subcriticality. It is estimated that reactor shutdown by early boron injection can occur in as little as 30 min.[4] Therefore other mitigating systems must function to ensure that the critical functions, RPV water inventory and core and suppression pool cooling, are maintained.

High Pressure Coolant Injection. The high pressure coolant injection (HPCI) system is successful if it maintains adequate core makeup to provide sufficient core cooling. HPCI operates from $\sim$1120 to 150 psig.

Operator Drops Level to Top of Active Fuel (TAF). This heading is represented on the sequence event tree immediately following each coolant injection method [HPCI, reactor core isolation cooling (RCIC) and control rod drive (CRD) injection, or low pressure systems]. Success is defined as the operator maintaining the RPV water level at the TAF. Even though the emergency procedures require the operator to inject boron and lower RPV water level to TAF, preliminary thermal-hydraulic calculations using RELAP5 indicate that if the operator drops RPV water level to the TAF given RPT but no boron injection, then reactor power will be reduced to ~9% at 1050 psi.[1] As previously discussed, one of the means identified to achieve low power that will be within the heat removal capability of the RHR heat exchangers is for the operator to reduce RPV water level to the TAF and take pressure control. The effect of pressure control on reactor power given RPV level at TAF, is discussed later in this section.

HPCI Available. Similar headings (RCIC and CRD Injection Available and Low Pressure Systems Available) are represented on the sequence event tree following each coolant injection method. The purpose of these headings is to distinguish if coolant injection is still available or if the operator has terminated flow by improperly taking level control. For example, it is necessary to know if HPCI injection is available and if not, a decision can be made at the RCIC and CRD injection branch, if required. The headings HPCI Available, RCIC and CRD Injection Available, and Low Pressure Systems Available are dummy events for the purpose of removing logic loops in the event tree.

RCIC and CRD Injection. RCIC and CRD injection are successful if they are combined with SLCS to maintain adequate core makeup to provide sufficient core cooling. If HPCI fails, RCIC, CRD and SLCS injection are assumed to be sufficient to maintain core coverage.[1] SLCS is discussed under the Early Boron Injection heading.

Operator Depressurizes the RCS. Depressurization is successful if it reduces RCS pressure to below the point of the low pressure injection systems shutoff head (~295 psig for the low pressure coolant injection system and ~289 psig for the core spray system). Automatic depressurization can be actuated by the opening of 6 of the 13 SRVs, or depressurization can be manually actuated. Depressurization is required if the high pressure injection systems fail to operate.

Low Pressure Systems. The low pressure systems are successful if they maintain adequate core makeup to provide sufficient core cooling. The systems available for low pressure injection are the low pressure coolant injection (LPCI) system and the core spray system. If both high and low pressure injection systems fail, then core damage will occur.

Operator Takes Pressure Control. Success is defined as the operator using the SRVs to bring about a controlled pressure reduction to obtain a RPV pressure of ~250 psi. Preliminary thermal-hydraulic calculations using RELAP5 indicate that by dropping level to TAF and reducing pressure in a controlled manner to ~250 psi, the reactor power level will be reduced to ~3%[1] whether boron injection is initiated or not. This is within the heat removal capability of the RHR heat exchangers which are capable of removing 1.5 to 3.5% power in the RHR mode.

Operator Initiates Late Boron Injection. Success is defined as the operator recognizing that previous actions to achieve and maintain low power have failed and that now SLCS operation must be initiated. SLCS operation requires that a sufficient quantity of boron is injected, into the core to effect a sustained subcriticality. This operator action is applicable if the operator waived early boron injection, attempted reactor power reduction with level and pressure control, but was unsuccessful in adequately reducing the power level. Due to the time it may take for the boron injection to result in a reactor shutdown, it has been assumed that late boron injection will not be effective soon enough to prevent containment/core damage if the operator has failed to take both level and pressure control. Late boron injection will be effective if the operator has successfully controlled either RPV level or pressure.

Suppression Pool Subcooled Short Term. Success under this heading occurs if the pool remains at least 45°F subcooled during the large energy absorption in the early minutes of the accident. Above 170°F, a potential challenge results from steam breakthrough in the pool due to large localized energy absorption. Steam breakthrough eventually threatens the containment due to overpressurization and leads to increased torus structural loading.[5]

Primary Containment Integrity. If the pressure rise due to loss of suppression pool subcooling is insufficient to cause containment damage, primary containment integrity is achieved. If the suppression pool remains subcooled the heading Primary Containment Integrity is passed through. If the pool has not remained subcooled then the question is asked whether the primary containment integrity is violated or not. If the containment has not failed then the critical functions, RPV water inventory and long term heat removal need to be maintained. If containment is breached, then core damage is expected to occur due to loss of cooling water injection which results from either failure of the injection systems to operate under the environmental conditions that they are exposed to as a result of containment failure, or from loss of injection paths as a result of containment failure.

Low Power or Subcriticality Maintained. Subcriticality is maintained if insufficient positive reactivity is inserted during injection of cold water by either the high or low pressure injection systems, or from the possible boron concentration reduction by carryover to the suppression pool or plating out on the RPV internals. For the BWR ATWS sequence event tree, it has been defined that if criticality is achieved, reactor power will rise to a level that exceeds the heat removal capacity of the RHR system, which in turn results in containment/core damage.

Long Term Containment Cooling. Long-term containment cooling is successful if the RHR system removes the heat being generated by the reactor core. Given that low power or subcriticality is maintained, it is assumed that the heat generated is within the heat removal capability of the RHR system. If the RHR system fails to operate, this will eventually result in containment/core damage.

Consequences. The consequences are divided into two categories: core okay and containment/core damage. It should be understood that some of the sequences labeled containment/core damage may not actually result in damage but in partial failure potentially leaving an okay core condition. However, this event tree modeled binary decision points (success and failure, and therefore, partial success is not considered in the consequence column). The scope of this study does not include the determination of offsite releases or doses. The scope of this study is to identify the most likely containment/core damage sequences for additional examination using deterministic thermal-hydraulic techniques.

4. ATWS SEQUENCE EVENT TREE QUANTIFICATION

The quantified sequence event tree is presented in Figure 4. Typically, sequence quantification is the process of combining the initiator frequency with mitigating system and operator action failure probabilities as prescribed by the sequence event tree logic in order to determine the frequency of any containment/core damage sequence. For this sequence event tree, the sequence probabilities have been reduced (the MSIV closure frequency and the probability of the control rods failing to insert have not been included in the sequence quantification). In other words, the sequence probabilities are relative values, not absolute values. Unavailability values or failure rate data for each mitigating system or operator action depicted on the sequence event tree are based on experience data or similar supporting documentation, engineering judgment, or a combination of these. From this evaluation, the most likely sequences leading to containment/core damage are identified, providing the basis for additional thermal and core damage analyses.

Sequence Event Tree Heading Failure Probabilities

The following discusses the failure probabilities assigned to each sequence event tree heading.

MSIV Closure. The initiator frequency for MSIV closure (5×10^{-1}/year) is from the proposed data base for the National Reliability Evaluation Program (NREP).[3]

Control Rods Fail to Insert. The failure probability for this system (3×10^{-5}) is from NUREG-0460.[6]

Safety Relief Valves Open. Browns Ferry has 13 SRVs. It is considered unlikely that inadequate overpressure protection will exist during an MSIV closure from 100% power with failure to scram given RPT. Several events have been reported on failure of SRVs to open at set pressure; however, none of these events have resulted in reactor coolant pressure boundary overpressurization.[7] Preliminary thermal-hydraulic calculations assumed that all 13 SRVs operate.

Failure of the SRVs to close, given successful opening, is not considered in this study. If one or two SRVs stick open, the existing transient would not be significantly altered. The systems necessary to mitigate one or two SRVs sticking open are the same systems being used to mitigate the ATWS.

Recirculation Pump Trip (RPT). The RPT ATWS trip for each pump depends on the operation of

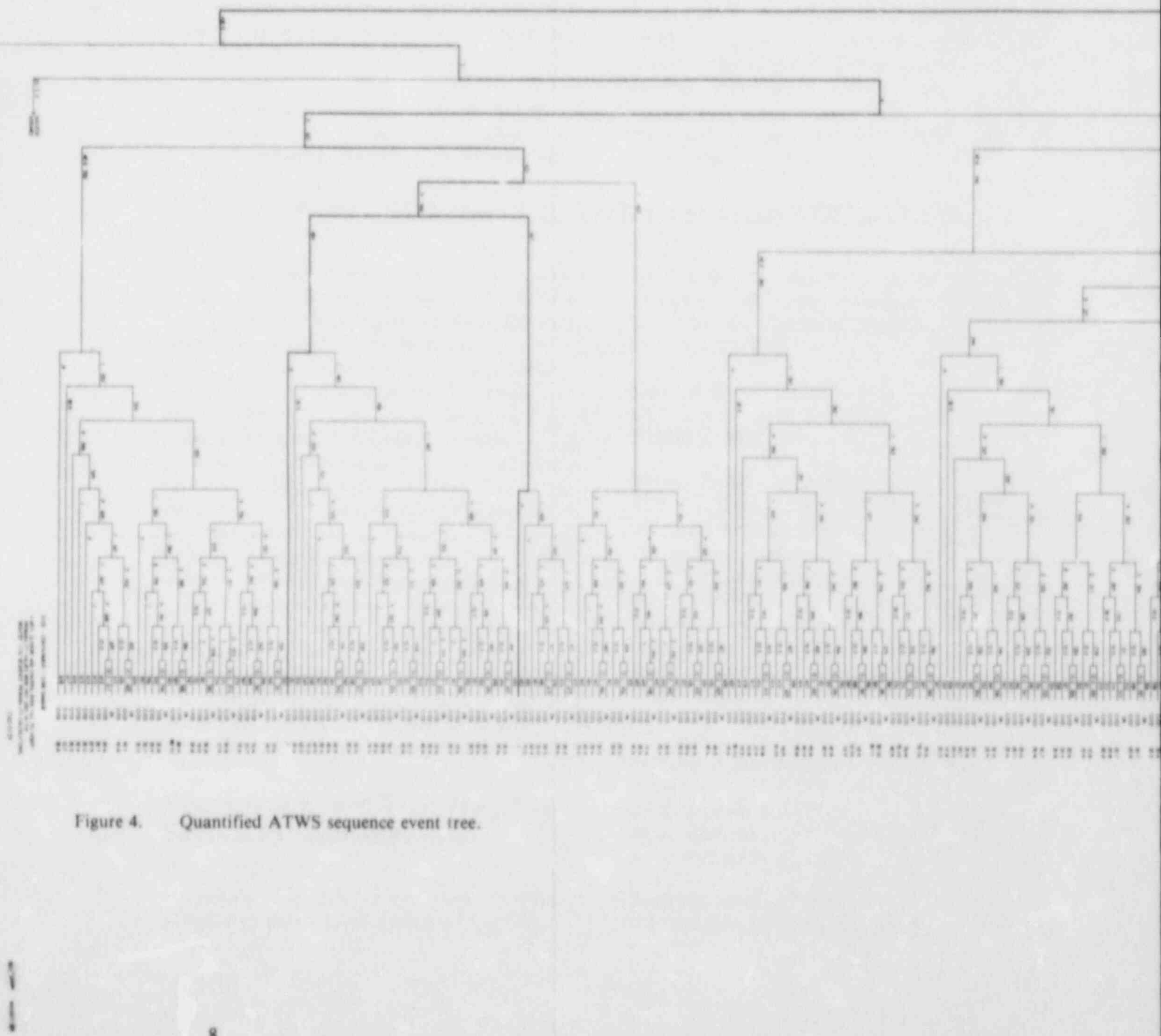

Figure 4. Quantified ATWS sequence event tree.

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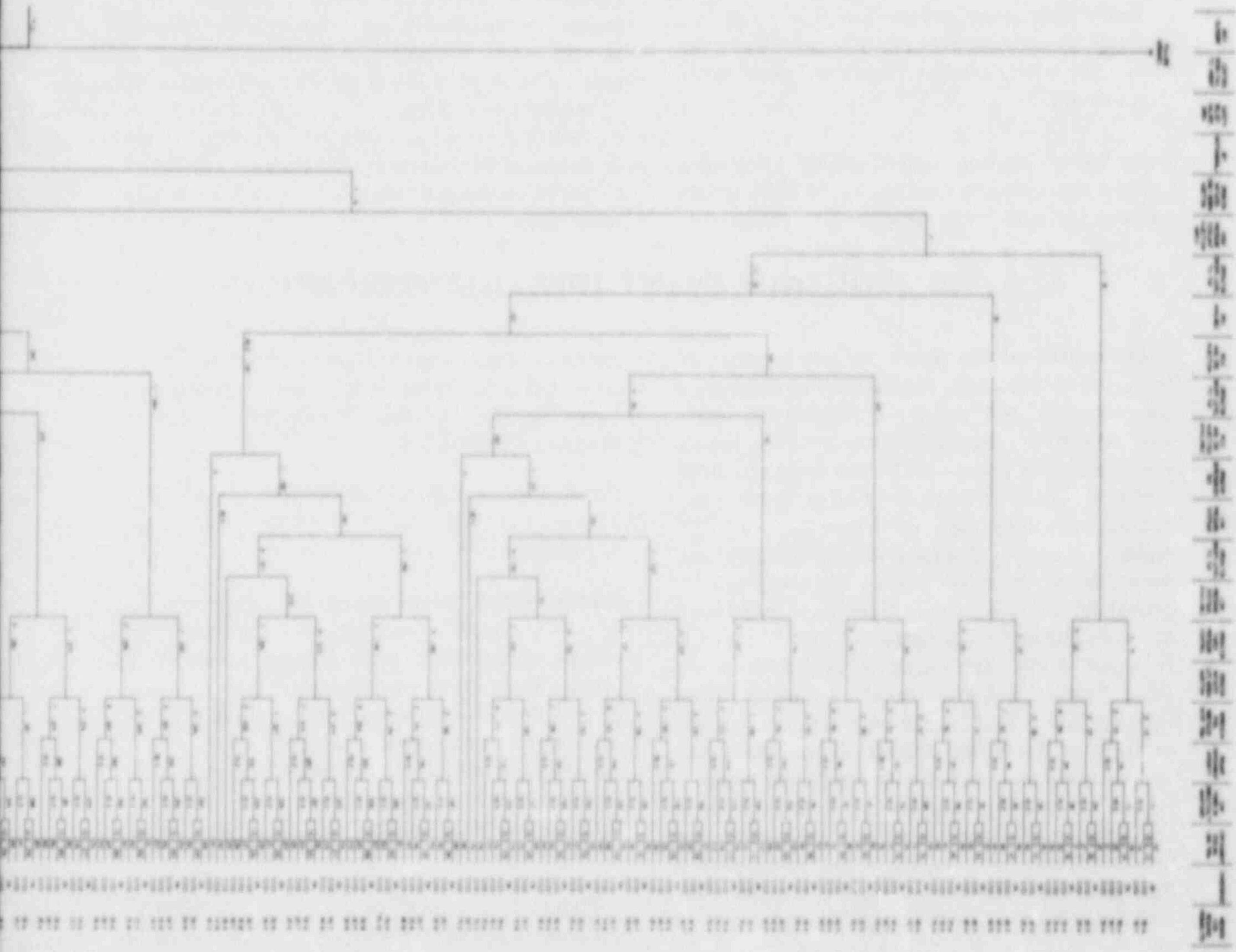

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pressure sensors to close a contact that energizes a coil that opens the circuit breaker to the motor-generator set. For one pump, failure of the contact to close, the coil to energize, and the breaker to open is $\sim 1 \times 10^{-3}$ based on failure rates from the proposed data base for the NREP Guide.[3] Failure of both recirculation pumps to trip is therefore $\sim 2 \times 10^{-3}$.

Operator Initiates Early Boron Injection. Reported data on human behavior indicates that the error rate for a task has a relationship to the stress level perceived by the operator. Following a major accident, human error will be high due to a probable incredulity response. Since the probability of a major accident is small, for some moments after the onset of the accident, a potential operator response would be to disbelieve panel indicators. Under such conditions no action might be taken at all for at least one min and if any action were taken it would likely be inappropriate. WASH-1400 assessed the human error rate to be 1.0 for the first min, 0.9 five min after an accident, 0.1 after thirty min, and 0.01 after several hours.[8] Based on this information, the probability of the operator failing to initiate early boron injection was chosen as 0.9.

High Pressure Coolant Injection (HPCI). A failure probability of 4×10^{-2} for this system was obtained from the Interim Reliability Evaluation Program on the Browns Ferry Unit 1.[9] Even though this system is automatically actuated, the operator must manually control HPCI flow later in the sequence of events in order to drop the reactor vessel water level to TAF. Thus, the probability of 0.9 for failure of the operator to properly drop level to the TAF was also assigned to failure of this system as a result of this operator interface with HPCI operation.

Operator Drops Level to Top of Active Fuel (TAF). A value of 0.9 was chosen for failure of the operator to drop the reactor water level to TAF. This value is from WASH-1400 as discussed above.[8]

As previously discussed, preliminary thermal-hydraulic calculations indicate that if the operator drops RPV water level to the TAF given RPT but no boron injection, then reactor power will be reduced to $\sim 9\%$ at 1050 psi. One of the means to achieve low power that will be within the heat removal capability of the RHR heat exchangers in the RHR cooling mode is for the operator to reduce RPV water level to the TAF and take pressure control. The effect of pressure control on reactor power, given RPV water level at TAF and no boron injection, will be to lower reactor power to within the heat removal capability of the RHR heat exchangers.

HPCI Available. This heading and the headings RCIC and CRD Injection Available and Low Pressure Systems Available all serve the same purpose: to distinguish if coolant injection is still available or if the operator has terminated flow by improperly taking level control. For example, it is necessary to know if HPCI injection is still available or not so a decision can be made at RCIC and CRD injection, if required. In other words, the headings HPCI Available, RCIC and CRD Injection Available, and Low Pressure Systems Available are dummy events; thus, a probability value was not assigned to these headings.

RCIC and CRD Injection. It has been assumed that if HPCI fails, then RCIC and CRD injection combined with SLCS are required to provide core cooling given that early boron injection has occurred. An unavailability for CRD injection was not available from the Interim Reliability Evaluation Program on the Browns Ferry Nuclear Plant; however, since CRD injection is normally operating the probability of CRD injection failing was considered to be small when compared to the probability of the RCIC failing to operate. Therefore, only the RCIC unavailability was used in the sequence quantification. The RCIC unavailability, 2×10^{-4}, is from the Interim Reliability Evaluation Program on the Browns Ferry Nuclear Plant PRA.[9] As discussed under HPCI operation, the probability of 0.9 for failure of the operator to properly drop level to the TAF was also assigned to failure of RCIC and CRD injection as a result of this operator interface with system operation.

Operator Depressurizes the RCS. A value of 0.9 was chosen for failure of the operator to depressurize the RCS. This value is from WASH-1400 as discussed above.[8] Depressurization is required if the high pressure injection systems fail to operate.

Low Pressure Systems. Low pressure systems refer to the LPCI system and the core spray system. The unavailability value of 3×10^{-5} is from the Interim Reliability Evaluation Program on the Browns Ferry Nuclear Plant and is for failure of both LPCI and the core spray system.[9] As discussed under HPCI operation, the probability of 0.9 for failure of the operator to properly drop level to the TAF was also assigned to failure of low pressure systems as a result of this operator interface with system operation.

The low pressure systems injection valves automatically open on a combination of low reactor vessel pressure and high dry well pressure, but there is not automatic closing logic. Check valves between the RPV and the low pressure systems provide the last level of protection against overpressurization of the low pressure systems once the RPV has been depressurized. However, at low pressures, the RPV is susceptible to sudden power spikes and repressurization which may result in increasing the probability of a LOCA outside containment.

Operator Takes Pressure Control. A value of 0.9 was chosen for failure of the operator to take pressure control provided that the operator has not had to depressurize the RCS to allow for low pressure injection. This value is from WASH-1400 as discussed above.[8] For the case where the operator has depressurized the RCS for low pressure injection, it was judged that failure of the operator to take pressure control would be less likely than for the case where RCS depressurization is not required; therefore, a value of 0.1 was assigned.

As previously discussed, preliminary thermal-hydraulic calculations indicate that by lowering RPV water level to the TAF and reducing pressure in a controlled manner to ~250 psi, the reactor power level will be reduced to ~3% whether boron injection is initiated or not. This power level is within the heat removal capability of the RHR heat exchangers.

Operator Initiates Late Boron Injection. This action has been assigned two values based on what actions the operator has already taken. If the operator has taken level and pressure control, then the operator has correctly recognized the accident and has control of the plant. Also if the operator has taken level and pressure control of the plant, the operator can safely take more time to decide what would be the next best action. Thus, for this case a value of 0.1 was used for failure of the operator to initiate late boron injection. This value is from WASH-1400.[8]

For the case where the operator has failed to take either level or pressure control, it is more likely that the operator does not fully recognize the accident that has occurred and that the operator does not have control of the plant. If the operator has not taken level or pressure control there is also less time available for the operator to make the decision to initiate boron injection. For this case a value of 0.9 was used for failure of the operator to initiate late boron injection. This value is also from WASH-1400.[8]

Suppression Pool Subcooled Short Term. Suppression pool subcooled short term is a phenomenological concern. Various values for failure of the suppression pool to remain subcooled have been assigned based on a perception of the plant conditions due to the success or failure of preceding mitigating actions.

For the sequences where early boron injection has been initiated, then if the operator has taken level and pressure control, a value of 0.5 was judged to be appropriate for failure of the suppression pool to remain subcooled. If the operator has only taken level control and not pressure control, the power level will remain high (~10% or greater) making it more likely that the suppression pool will not remain subcooled; thus, a value of 0.8 was assigned. If the operator has only taken pressure control and not level control, a value of 0.7 was judged to be appropriate for failure of the suppression pool to remain subcooled. If the operator has not taken level and pressure control, power will remain high until the boron becomes effective; thus, a value of 0.9 was assigned.

For the sequences where early boron injection has not been initiated, then if the operator has taken both level and pressure control, a value of 0.5 was assigned whether the operator initiates late boron injection or not. If the operator has taken level control and initiated late boron injection, a value of 0.8 was assigned since the power will remain high (~10% or greater) until late boron becomes effective. If the operator has only taken pressure control and initiated late boron injection, then it was judged only slightly less likely that the suppression pool would not remain subcooled: 0.7.

Primary Containment Integrity. Primary containment integrity is also a phenomenological concern and is coupled to the suppression pool remaining subcooled as well as with other previous actions.

For the sequences where the operator has initiated early boron injection and has taken both level and pressure control, a value of 5×10^{-3} was judged to be appropriate for failure of primary containment integrity. If the operator has only initiated early boron injection or early boron injection with level or pressure control, then a value of 1×10^{-2} was assigned.

For the sequences where early boron injection has not been initiated, then if the operator has taken both level and pressure control, a value of 1×10^{-2} was assigned whether the operator initiates late boron injection or not. If the operator has taken either level or pressure control and initiated late boron injection, a value of 1×10^{-1} was assigned.

Low Power or Subcriticality Maintained. It was judged that if boron injection has occurred, then failure to remain subcritical would be 1×10^{-3}. If the operator is maintaining low power with level and pressure control only, then it was judged to be much more likely that low power could not be maintained; thus, a value of 0.7 was assigned.

Containment Cooling Long Term. Whenever low-low-low reactor vessel water level is indicated, the RHR system automatically aligns to vessel injection. Successful RHR torus cooling is highly dependent on the operator switching the RHR system back to the torus cooling mode after each time low-low-low reactor vessel water level is obtained. Therefore, the probability of containment long term cooling failing was judged to be 1×10^{-2}.

Sensitivity Analysis

For most of the operator actions required following an ATWS, a failure probability of 0.9 was assigned as previously discussed. A sensitivity analysis was performed to determine if there would be any additional most likely containment/core damage sequences if the operator failure probabilities were varied. First a failure probability of 0.5 was assigned to failure of each required operator action and the containment/core damage sequence probabilities were calculated. Then a failure probability of 0.1 was assigned to failure of each operator action, and again, the containment/core damage sequence probabilities were calculated. In this way, operator failure probabilities span approximately one order of magnitude. Table 1 shows the State number, the reduced sequence probability values from Figure 4, the reduced sequence probability value if each operator failure probability is changed to 0.5, and the reduced sequence probability value if each operator failure probability is changed to 0.1. For example, the reduced sequence probability for Sequence 483 from Figure 4 is 7×10^{-1}. If each operator failure within Sequence 483 is assigned a value of 0.5, the reduced sequence probability is 1×10^{-1}. If each operator failure within Sequence 483 is assigned a value of 0.1, the reduced sequence probability is 1×10^{-3}. Table 1 shows only those sequences that have reduced sequence probability values of 10^{-2} or 10^{-1} from either Figure 4 or after the operator failure probability has been changed to 0.5 or 0.1. The results of the sensitivity analysis show that if the operator failure probability is changed to 0.5, there are two additional most likely containment/core damage sequences. These sequences are represented by States 465 and 482 on the BWR ATWS sequence event tree. The sensitivity analysis also shows that if the operator failure probability is changed to 0.1, there are no additional containment/core damage sequences that can be considered as most likely.

5. RESULTS AND INTERPRETATION

Under the SASA Program, preliminary deterministic thermal-hydraulic analyses have been performed for a loss of the power conversion system with a failure to scram at a BWR. From these preliminary thermal-hydraulic analysis, three means of achieving low power or subcriticality that will be within the heat removal capability of the RHR heat exchangers were identified. These three means of achieving low power or subcriticality are not only based on the emergency procedure guidelines,[2] but are also based on insights gained from the preliminary thermal-hydraulic analyses.[1] The three means of achieving low power or subcriticality are: (a) early boron injection, (b) level control to TAF and pressure reduction in a controlled manner without boron injection, or (c) late boron injection that has been preceded with either level or pressure control.

From the preliminary thermal-hydraulic analyses, it became apparent that a structured approach was needed to logically develop the many sequences of events that lead to containment/core damage. The BWR ATWS sequence event tree (Figure 3) represents this structured approach. This sequence event tree logically depicts the systems that could

Table I. BWR ATWS sequence event tree sensitivity analysis

Sequence	Reduced Sequence Probability Value (from Figure 4)	Reduced Sequence Probability Value (if each operator failure probability is changed to 0.5)	Reduced Sequence Probability Value (if each operator failure probability is changed to 0.1)
178	7×10^{-3}	3×10^{-2}	8×10^{-4}
180	7×10^{-2}	6×10^{-2}	9×10^{-4}
241	3×10^{-3}	1×10^{-2}	4×10^{-4}
359	3×10^{-3}	1×10^{-2}	4×10^{-4}
465	7×10^{-2}	1×10^{-1}	1×10^{-2}
482	7×10^{-2}	1×10^{-1}	1×10^{-2}
483	7×10^{-1}	1×10^{-1}	1×10^{-3}
547	6×10^{-2}	3×10^{-2}	7×10^{-4}
548	7×10^{-3}	3×10^{-2}	9×10^{-5}
549	6×10^{-2}	6×10^{-2}	9×10^{-4}
551	7×10^{-1}	1×10^{-1}	1×10^{-3}
616	3×10^{-3}	1×10^{-2}	4×10^{-4}
618	3×10^{-2}	1×10^{-2}	4×10^{-4}

respond to an ATWS, the operator response to this event, the operator actions that could be performed in response to multiple failures, and the phenomenological concerns. The BWR ATWS sequence event tree logic is based on the three means identified to achieve low power or subcriticality when an ATWS has occurred. The purpose of the thermal-hydraulic analyses is to examine a failure to scram; therefore, manual rod insertion is not considered. The BWR ATWS sequence event tree is based on findings from preliminary thermal-hydraulic analyses[1] and does not reflect any additional findings from later thermal-hydraulic analyses.[10]

Figure 3 shows that there are many sequences that lead to containment/core damage, and thus, many deterministic thermal-hydraulic analyses could be performed at great expense. By quantifying the BWR ATWS sequence event tree, Figure 4, the most likely risk sequences are identified, and as a result, the thermal-hydraulic and core damage analyses can be concentrated on these most likely sequences. A review of the most likely sequences also establishes the analytical models necessary to analyze important phenomena and provides direction for expanding calculational capability of the existing thermal-hydraulic analysis codes.

The most likely sequences shown in Figure 4 are represented by States 483 and 551. The most likely sequences are described as follows:

State 483—MSIV closure has occurred, the control rods have failed to insert, but the safety relief valves are providing overpressure protection, and recirculation pump trip has reduced reactor power by 50 to 70%. Early boron injection has not been initiated. The operator has failed to take level control, but HPCI operation is successful. Since the reactor vessel water level has not been reduced to the TAF, reactor power is >9%. To achieve a stable plant condition, the operator must take pressure control and initiate late boron injection. In this sequence, the operator fails to take pressure control.

State 551—MSIV closure has occurred, the control rods have failed to insert, but the safety relief valves are providing overpressure protection, and recirculation pump trip has reduced reactor power by 50 to 70%. Early boron injection has not been initiated. HPCI operation is successful; however, the operator subsequently fails HPCI by improperly taking level control. A decision cannot be made at RCIC and CRD injection since early boron injection was not initiated. In this sequence, the operator fails to recognize the need to depressurize the RPV. This sequence represents a high pressure boiloff event without boron injection.

Since an MSIV closure with a failure to scram represents a significant challenge to both systems and operator responses due to the relatively short available reaction time, some of the actions could be difficult to perform, simulators are limited in realistic responses for extreme accident conditions, and based on human error probabilities from WASH-1400, a best estimate failure probability of 0.9 was assigned to most of the operator failures shown in Figure 4. A sensitivity analysis was performed to determine if there would be any additional most likely containment/core damage sequences if the operator failure probabilities were varied. First, a failure probability of 0.5 was assigned to failure of each operator action and the containment/core damage sequence probabilities were calculated. Then a failure probability of 0.1 was assigned to failure of each operator action and the containment/core damage sequence probabilities were calculated again. In this way, operator failure probabilities span approximately one order of magnitude. The results of the sensitivity analysis (see Section 4) show that if the operator failure probability is changed to 0.5, there are two additional most likely containment/core damage sequences. These sequences are represented by States 465 and 482 in Figure 4. The sensitivity analysis also shows that if the operator failure probability is changed to 0.1, there are no additional containment/core damage sequences that can be considered as most likely. States 465 and 482 are subsets of State 483 and are briefly described below.

State 465—MSIV closure has occurred, the control rods have failed to insert, but the safety relief valves are providing overpressure protection, and recirculation pump trip has reduced reactor power by 50 to 70%. The operator has not initiated early boron injection but has successfully dropped the reactor vessel water level to the TAF with use of the HPCI system. By dropping the reactor vessel water level to the TAF, the reactor power has been reduced to ~9% at 1050 psi. To reduce power to ~3% at 250 psi the operator must take pressure control, or to shutdown the reactor the operator must initiate late boron injection. In this sequence of events, the operator fails to take pressure control and fails to initiate late boron injection.

State 482—MSIV closure has occurred, the control rods have failed to insert, but the safety relief valves are providing overpressure protection, and recirculation pump trip has reduced reactor power by 50 to 70%. Early boron injection has not been initiated. The operator has failed to take level control but HPCI operation is successful. Since the reactor vessel water level has not been reduced to the TAF, reactor power is >9%. To achieve a stable plant condition, the operator must take pressure control and initiate late boron injection. In this sequence, the operator successfully takes pressure control but fails to initiate late boron injection.

Out of ~200 potential containment/core damage sequences, additional deterministic thermal-hydraulic analyses can now be concentrated on the four most likely containment/core damage sequences allowing results to be obtained quicker and for a much more reasonable cost. Also, analysis of these four most likely risk sequences indicates the importance of level control and pressure control by the operator on accident mitigation.

The sequence event tree can also provide additional insights. It can help one visualize what critical plant information an operator requires in order to make a decision under many plant conditions. Since the sequence event tree identifies required operator responses to an accident initiator and the operator actions that could be performed in response to multiple failures, it can be used as a tool to evaluate the adequacy of emergency procedures. The sequence event tree can also be used as a basis for developing control room staffing criteria based on the timing and the type of required operator actions and on the potential number of operator actions that may be necessary.

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The United States Nuclear Regulatory Commission is sponsoring an on-going safety research program to assess dominant risk events in boiling water reactors. As part of this program, a sequence event tree for a boiling water reactor anticipated transient without scram (ATWS) has been developed and quantified. The goal of the sequence event tree is to provide a logical representation of the systems that must respond to an ATWS, the required operator response to the event, operator actions that could be performed in response to multiple failures, and the phenomenological concerns. The purpose of the sequence event tree is to provide a basis upon which to perform additional deterministic thermal-hydraulic and core damage analyses in the most cost effective manner based on the most likely sequence of events that will lead to containment/core damage. The ATWS sequence event tree is based on the General Electric Owners Group emergency procedure guidelines and on preliminary deterministic thermal-hydraulic analyses performed by EG&G Idaho, Inc. personnel at the Idaho National Engineering Laboratory under direction of the Severe Accident Sequence Analysis Program.

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