19.3 Internal Event Analysis

19.3.1 Frequency of Core Damage

This subsection describes the approach taken to assess accident event sequences and determine core damage frequency. Human and equipment reliability models and system descriptions provided the bases for constructing system fault trees. Results of these trees and applicable system success criteria were used to construct and evaluate accident event trees to determine the outcomes of accident sequence initiating events. The frequency of core damage was provided directly for Class I and III events by outcomes of the accident event trees. Determination of what fractions of Class II and IV events led to core damage required additional processing through containment event trees, discussed in Subsection 19.3.2, which were used to determine the final outcomes of those sequences involving loss of heat removal or ATWS, Class II and IV events. This approach to assessing core damage frequency and fission product releases is schematically illustrated in Figure 19.3-1.

In addition, a revised evaluation of common cause failures of various mechanical systems (evaluated as part of a PRA sensitivity study) was performed to address an issue identified by the NRC regarding the original evaluation.

19.3.1.1 Accident Initiators

This subsection describes the accident sequence initiating events documented in Subsection 19D.3. These initiators are separated into two general groups, transients and loss of coolant accidents (LOCAs). Table 19.3-1 provides a summary of these initiators and their expected frequency of occurrence.

The total frequency of transient initiators used in these evaluations is based upon a 2007 analysis of operating plant data (Reference 19.3-8). The frequency of transients is taken from this study and used directly as historical industry performance and is conservative. Apportioning of the expected transient frequency by initiating event was done on the basis of historical electrical grid and BWR performance data as described in Subsection 19D.3.

LOCA initiation frequencies are based upon NUREG/CR-6928 (Reference 19.3-8). After reviewing these values and their bases, their use in the ABWR PRA was judged appropriate.

19.3.1.2 Equipment Reliability and Availability

Not part of DCD (Refer to Section 19D.3)

19.3.1.3 Accident Sequence Analysis

19.3.1.3.1 Success Criteria

This subsection provides a discussion of the ABWR success criteria employed in this analysis. These criteria govern the construction of accident event trees which are used to model all accident sequences. The criteria are defined for both non–ATWS events and ATWS events.

(1) Success Criteria for Non–ATWS Events

The success criteria in this subsection are based on best–estimate predictions using approved computer models. Several BWR generic studies have determined that one motor–driven ECCS pump has sufficient reflooding flow to provide adequate core cooling.

(a) Core Cooling

A peak cladding temperature (PCT) of 1478 K (2200°F) was chosen as the criteria in determining the success of a coolant injection system. The resultant ABWR core cooling success criteria to prevent initial core damage for transient and Loss Of Coolant Accident (LOCA) events with scram initiated from the Reactor Protection System (RPS) are given in Table 19.3-2.

The high pressure core flooder (HPCF) pumps, which are the Emergency Core Cooling System (ECCS) high pressure pumps, have a large capacity for making up lost inventory. Following any LOCA or transient event, either one of the HPCF pumps can reestablish the water level and maintain the PCT below 1478 K (2200°F).

The residual heat removal (RHR) pumps, which can be used in the ECCS low pressure core flooding mode, are also large capacity pumps. Following a large LOCA, the Reactor Pressure Vessel (RPV) depressurizes sufficiently and any one of the three RHR pumps can reestablish the water level and maintain the PCT below 1478 K (2200°F). For small or medium LOCA, or transient events, RPV depressurization using at least three (3) depressurization valves is needed to permit timely use of an RHR pump.

The reactor core isolation cooling (RCIC) pump is also part of the ECCS. It is turbine driven and also provides high pressure water makeup to the RPV as long as steam is available at pressures greater than 0.345 MPa differential for the RCIC turbine. The RCIC requires high pressure steam from the vessel to drive the turbine; therefore, its ability to maintain adequate core cooling by itself is limited to the small liquid LOCA or transient (excluding IORV) events.

The capacity of non-safety-related systems, such as the feedwater, condensate booster, and condensate pumps, has been estimated based on the ECCS performance analyses. Non-safety-related systems which contribute to a successful conclusion of the event have been included in the success criteria. The Control Rod Drive (CRD) pumps which have limited capacity have not been included in the success criteria.

The condensate and condensate booster pumps are motor-driven pumps and their use depends on the RPV pressure and the availability of makeup water and electrical power. These pumps have higher shut–off heads than the RHR pumps, but still require depressurization before they can be used for core cooling. The source of makeup water for these pumps are the main condenser hotwell and the condensate storage tank. Sufficient makeup water is available to enable these pumps to maintain adequate core cooling for all events except large or medium liquid LOCAs.

A motor driven feedwater pump is combined in series with a condensate pump and condensate booster pump in order to provide a higher pressure system. Therefore, this option also depends on the availability of makeup water and electrical power. Sufficient makeup water is available to enable this series of pumps to maintain adequate core cooling for the small steam LOCA and transient events.

The fire protection system has two pumps which take suction from the firewater tanks and inject into the RPV through an RHR line. One pump is driven by an electric motor which requires AC power. The other is driven directly by a diesel engine. Once the reactor system has been depressurized, either pump can provide enough makeup water to restore and maintain the RPV water level following any transient (including IORV) event. The analysis to support this conclusion assumes a full ADS blowdown begins within 15 minutes after the vessel water level has reached the level 1 setpoint. The subsequent reactor system depressurization allows injection from the fire protection system about 7 minutes after the start of the blowdown. The ability of the fire protection system to mitigate the consequences of LOCA events is conservatively ignored. For more information about the fire protection system refer to Subsection 5.4.7.

It is conservative to use the 1204°C (2200 °F) PCT licensing limit as an acceptance criteria for the success criteria since tests have been performed which show that the core will remain in a coolable geometry with temperatures as high as 1482°C (2700°F).

A review of Table 19.3-2 shows that, for success, the inventory threatening events require the flow equivalent of only 1 RHR/LPFL or 1 HPCF pump available for large break cases and only 1 HPCF or 1 RHR/LPFL + 3 ADS available for small break cases. The resulting PCTs for the large break cases and transients were between 482°C (900°F) and 593°C (1100°F). For the small break cases with the flow equivalent of only 1 HPCF available the resulting PCTs were less than 538°C and with 1 RHR/LPFL + 3 ADS available the maximum PCT was 982°C (1800°F).

(b) Containment Heat Removal

Following the success of the core cooling function, heat must be removed from the containment. Containment heat removal is considered a success if the containment pressure is kept below the pressure at which loss of containment integrity is estimated to occur (Appendix 19F). Successful containment heat removal can be achieved by using the RHR System or, depending on the circumstances as defined in Table 19.3-2, the normal heat removal path or the CUW System. The resultant ABWR longterm heat removal success criteria to prevent initial core damage for transient and Loss of Coolant Accident (LOCA) events with RPS scram are given in Table 19.3-2.

The RHR has four major modes of operation and heat is removed from the containment in each of these modes. During the core cooling mode which is initiated automatically, the RHR heat exchanger is in the loop and the heat removal process is established. If core cooling is accomplished without the use of an RHR System, and the suppression pool begins overheating, the suppression pool cooling mode of the RHR will be automatically or manually initiated by the operator. Once initiated, an RHR System will begin removing heat from the containment and eventually terminate the pool heatup.

The normal heat removal path is through the main condenser. This path can be used under transient and accident conditions when the MSIVs or the main steam drainlines are open (or re-opened if closed earlier during the event) and the condensate can be removed. If the RPV is depressurized, the main steam drainline option is not viable since it will not pass enough steam to remove the decay heat energy.

The Reactor Water Cleanup (CUW) System is capable of removing the energy due to decay heat (at greater then 4 hours after scram) at high RPV pressures if the return water bypasses the regenerative heat exchanger. Therefore, its ability to maintain adequate longterm cooling is also limited to the small liquid LOCA or transient events where the reactor system can be maintained at high pressure and temperature producing a large temperature differential across the CUW non–regenerative heat exchanger.

(c) RPV Pressure Relief

A pressure of 150% of the reactor-coolant pressure-boundary design pressure [8.719 MPa], the faulted limit, was chosen as the criterion in determining the success of the system to prevent overpressure failure of the reactor primary system during moderately frequent events. The turbine bypass system and safety/relief valves represent the two success paths in the ABWR overpressure protection success criteria for the events given in Table 19.3-2.

For events resulting in isolation of the primary system, only the safety/relief valves are available to minimize the reactor system pressure rise. The most severe ABWR transient with scram for these events is a closure of all MSIVs. For this case, six of the eighteen safety relief valves are required to open to limit the peak primary system pressure to below 13.029 MPa.

For events which do not result in isolation of the primary system, both the turbine bypass valves and the safety/relief valves are available to minimize the reactor system pressure rise. The most severe ABWR transient with scram for these events is a turbine trip at full power with at least some turbine bypass capability available. For this bounding case, the acceptable combinations of turbine bypass and relief valves available are given in Table 19.3-2.

(2) ATWS Success Criteria

For anticipated transient without scram (ATWS) events, the success criteria are defined in terms of the system mitigation capability to meet the following criteria:

(a) The reactor pressure vessel must be protected from overpressurization.

(b) A coolable core geometry must be maintained.

(c) Long term heat removal must be adequate to preserve containment integrity.

For a postulated system failure, an ATWS event is considered successful (or acceptable) if the resulting ATWS consequences meet these criteria.

Three classes of ATWS may be considered. The first class when Reactor Protection System (RPS) scram does not operate but the rods are inserted by alternate means. Electric rod run in is an automatic function which is initiated by the RPS. However, because the rods are inserted more slowly than with scram, the successful operation of this method of reactivity insertion is considered an ATWS. The rods may also be inserted with alternate rod insertion (ARI). High vessel pressure or low water level initiate ARI. Additionally, the operator may manually insert the rods.

The second class of ATWS sequences are those for which there is no rod insertion, but the reactor is brought to a subcritical condition by standby liquid control (SLC) injection. The requirements for this system are discussed below.

The third class of ATWS events has neither rod insertion nor SLC injection. Work performed by General Electric and Idaho National Engineering Laboratory (Reference 19.3-7) has shown that a single high pressure system can maintain adequate core cooling. Maintaining the containment pressure below service level C and the containment overpressure protection (COPS) setpoint is the appropriate success criteria for the containment heat removal system in this highly degraded scenario. If three heat exchangers are available in this case, the containment pressure can be maintained below these levels. In addition, the pressure can be maintained below service level C if the COPS actuates, thus COPS is also an adequate method of containment cooling. However, because the probability of ATWS is very low, no credit is taken for mitigation of this class in the internal events analysis.

The following discussion determines the system and operator requirements necessary to ensure adequate RPV pressure relief, core cooling, and containment cooling during ATWS.

(a) RPV Pressure Relief

As for the non–ATWS transients, a pressure of 150% of the reactor–coolant pressure–boundary design pressure, the faulted limit, was chosen as the criteria in determining the success of the system to prevent over pressure failure of the reactor primary system during moderately frequent events. The turbine bypass system and safety/relief valves represent the two success paths for the ABWR. Overpressure protection success criteria for ATWS events are included in Table 19.3-3.

For events resulting in isolation of the primary system, only the safety/relief valves are available to minimize the reactor system pressure rise. The most severe ABWR ATWS event for overpressure protection is a closure of all main steam isolation valves (MSIVs). For this case fifteen of the eighteen safety relief valves are required to open to limit the peak primary system pressure to below 13.029 MPa.

For events which do not result in isolation of the primary system both the turbine bypass valves and the safety/relief valves are available to minimize the reactor system pressure rise. The most severe ABWR transient for these events is a turbine trip at full power, but in this case some turbine bypass capability is available. For this bounding case, the acceptable combinations of turbine bypass and relief valves are given in Table 19.3-3.

(b) Core Cooling

Adequate core cooling is necessary to prevent fuel failure. Assuming some form of reactivity control is operated, an ATWS event does not substantially differ from its associated transient. Any one high pressure system is adequate to provide core cooling for all initiators except inadvertent opening of a relief valve (IORV). For the case of IORV, the vessel will continually depressurize through the stuck open valve, therefore, at least one High Pressure Core Flooder (HPCF) System must inject to the vessel. These requirements are summarized in Table 19.3-3.

Additionally, for ATWS events where the rods are eventually inserted, the low pressure systems may be used for core cooling. For these events, the core cooling success criteria are identical to those given in Table 19.3-2.

(c) Containment Integrity

Assuming the success of the core cooling function, heat must be removed from the containment. Containment heat removal is considered a success if the containment pressure is kept below the pressure at which loss of the containment integrity is estimated to occur. Successful containment heat removal can be achieved by using the Residual Heat Removal (RHR) System, or, depending on the circumstances, the normal heat removal path.

For ATWS events, the ability to maintain containment integrity is dependent not only on the availability of the RHR System, but also on the rate at which the reactor is brought to shutdown conditions. Therefore, it is necessary to consider each type of ATWS initiating transient separately in order to determine the energy being passed to the pool as a function of time.

The speed at which poison is injected determines the energy which is passed to the suppression pool. Analysis shows that once the reactor has been shut down one RHR loop is capable of maintaining containment integrity.

In order to determine the maximum time available for poison injection, four events are considered below. The required timing for the events shown below was determined assuming that containment cooling is not initiated until the reactor is shut down. Therefore, only one RHR loop must operate. However, for those ATWS initiators for which isolation is not an immediate consequence of the initiator, the main condenser was assumed to be available. Also assumed in the analysis below is that both HPCF Systems and the Reactor Core Isolation Cooling (RCIC) System will automatically initiate if their initiation conditions are reached. This has the effect of predicting the maximum power generation, and therefore, leads to shorter operator action times than a case in which core cooling injection is limited. The results of these considerations is summarized in Table 19.3-3.

(i) MSIV Closure

The sequence of events and anticipated operator actions for an MSIV Closure with failure of automatic rod insertion is as follows. Upon closure of the MSIVs, the reactor pressure will increase sharply and the safety relief valves (SRVs) will open. This will cause 4 of the reactor internal pumps (RIPs) to trip. The water level will rise to level 8, causing a feedwater trip. The operator will observe the failure to insert rods based on the rod position switches, and attempt to insert the rods manually approximately 30 seconds into the transient. If the rods are inserted, then the sequence is successfully terminated.

If the rods are not inserted, then the water level will begin to fall. When the water level reaches level 2, in about 30 seconds, the RCIC will initiate and the remaining 6 RIPs will trip. The water level will continue to decrease, initiating the two HPCF Systems about one minute into the transient. ADS initiation is automatically inhibited due to the ATWS event. Assuming all three water injection systems come on, the power level will drop to approximately 20% of rated. The operator is then instructed to allow the water level to drop by terminating and preventing all injection into the RPV except from the CRD until at least one of the following conditions exist:

- The reactor power drops below the APRM downscale trip.
- The water level in the vessel reaches the top of active fuel.
- All the SRVs are closed and the drywell pressure remains below 0.014 MPaG.

The operator will initiate the SLC System based upon a suppression pool temperature versus power criterion. Assuming that one SLC will operate, approximately 10 minutes are allowed for him to complete this procedure. In order to bring the transient to a complete stop, the RHR System must be initiated in suppression pool cooling or drywell spray mode. A minimum of 30 minutes is allowed for this action.

(ii) Turbine Trip

The sequence of events for the turbine trip is quite different from that for an MSIV closure. When the turbine trips, the turbine bypass valves will open, permitting approximately one third of rated steam flow to pass through them. The remainder of the steam generated will pass through the SRVs into the suppression pool. The initial pressure rise will cause 4 of the RIPs to trip, and the core flow will drop to about 80% of rated. The feedwater pump will attempt to maintain normal water level. The power will equalize at approximately 80% of the rated value. Therefore, about 50% of rated power will be directed to the suppression pool. The operator will be aware of ATWS conditions within 30 seconds. The operator is instructed to attempt to take actions to shut down the reactor. These actions include:

- Placing the mode switch in Shutdown.

- Tripping all RIPs.
- Inserting rods by various means.

If these actions fail to shut down the reactor, the feedwater will run back and prevent RPV injection in preparation for SLC injection. The water level will fall to level 2, and the remaining RIPs will trip if not previously tripped manually. This will cause the power level to drop to about 20%. Both feedwater runback and SLC initiation are assumed to occur within 10 minutes of the initiation of the event. Finally, in order to maintain the pool temperature, the RHR System must be initiated within 30 minutes.

(iii) IORV

This sequence is quite different from the remainder of the sequences considered here. The failure of rod insertion for an IORV event is based on a manual operation taken when the suppression pool temperature limit is reached. It is assumed that one SRV is stuck open. This allows 4.5% of the power to flow to the pool. The turbine and the main condenser are available to remove the remaining energy from the plant. For this sequence, the operator has the capability of shutting down the plant in an orderly fashion.

The RHR will be initiated to attempt to keep the suppression pool cool. The operator will attempt to insert the rods. If that fails, the RIPs will trip, the feedwater will be run back and SLC will be initiated. Even if RHR is not operating, the power being directed to the pool is very low since the bypass valves can be used to divert steam flow to the main condenser, and there will be ample time for SLC initiation.

(iv) Loss of Offsite Power

The loss of offsite power (LOOP) initiated ATWS is similar to the MSIV closure transient. The water level will fall, generating a scram signal. if the water level drops below level 1.5, the MSIVs will close. The feedwater and RIPs will trip. Thus, the power level in the reactor will drop even more quickly than will the MSIV closure case. As for the case of an MSIV closure, the operator will first attempt to insert the rods manually. If this fails, the ADS actuation must be inhibited and the SLC injection must begin within about 10 minutes, and RHR must be started in 30 minutes.

19.3.1.3.2 Accident Sequence Event Trees

This subsection describes construction of event trees used in the analysis to determine accident sequence frequencies. These sequences lead to core damage, safe reactor shutdown, or to intermediate states which require additional treatment in the containment event trees of Subsection 19D.5 to establish final core states. Separate trees have been developed, as shown in Figures 19D.4–1 through 19D.4–15, for each of the initiating events considered. All accident event tree sequences other than those leading to safe reactor shutdown are further treated in the containment event trees of Subsection 19D.5 to determine frequencies of radiation release to the environment.

For purposes of illustration, consider Figure 19.3-2, the event tree for the reactor shutdown initiating event. The initiating event frequency is given as the first branch of the far left column of the tree. The initiating event name and symbol are provided at the top of the column. The tree is developed by identifying the system functions required, in the approximate chronological order of occurrence, for successful reactor shutdown. Success and failure states of each system function are represented by branches in the tree, where the upper branch represents success and the lower branch failure. If a prior system function leads directly to success or failure in the accident sequence, analysis of the remaining system functions is unnecessary.

Information given at the top of the column for each system function consists of an abbreviated definition of success and the symbol for conditional failure probability. The value for each failure probability is shown on the lower branch. Each accident sequence terminates in the column labeled “FREQ” which contains the frequency of occurrence of that sequence. The final column contains the classification of each sequence; either successful termination (OK), core damage, or a sequence which is developed further in another accident tree or transferred to the appropriate containment event tree.

Accident event trees developed in this analysis contain branches which address the primary safety functions of reactivity control, reactor pressure control, core cooling, and containment heat removal. These four functions are considered in all event trees except the reactor shutdown event in which reactivity control is, by definition, provided by event initiation. Success criteria provide the bases for defining minimum combinations of those functions required to bring the plant to a safe stable shutdown condition. Success criteria are presented and discussed in Subsection 19. 3.1.3.1.

19.3.1.3.3 Classification of Accident Classes

Accident sequences identified and evaluated in the event trees were examined and classified on the basis of similarity of timing, potential for fission product release, and containment response. Accident sequence classes used in the analysis are described in Subsection 19D.5.

19.3.1.4 Frequency of Core Damage

Of the ten accident classes defined in Table 19.3-4, eight lead directly to core damage. The remaining two classes can lead initially to loss of the containment heat removal function and subsequently, possible core damage. For these latter sequences, outcomes in event trees documented in Subsection 19D.4 do not necessarily lead to core damage. Detailed analyses of the frequency of core damage following loss of containment integrity is presented in

Subsection 19D.5 where it is addressed relative to containment release paths. That analysis is based on the accident event tree outcomes, containment overpressure capability discussed in Subsection 19.3.2, and on the containment event trees of Subsection 19D.5.

Table 19.3-4 summarizes the frequencies of core damage as a function of accident class. As explained above, eight of the ten frequencies (all Class I and III events) are obtained directly from the outcomes of the accident sequence event trees of Subsection 19D.4. Frequencies of the Class II and IV events, where containment heat removal is lost, were determined by processing the loss of heat removal outcomes of the accident event trees through the containment event trees to determine the probability of failure to prevent core damage for these events. The bypass study in Subsection 19E.2.3.3 concluded that the core damage frequency and risk associated with Class V events is negligible. Therefore, the Class V frequencies are not given in the table.

Table 19.3-5 provides a different perspective by showing the breakdown of core damage frequency by initiating event. Expected frequencies are given both in terms of events per year and percent of total. Loss of offsite power and station blackout are the dominant contributors to expected core damage. In the loss of offsite power sequence, dependence on diesel generators alone for electric power results in lower availability and reliability of the emergency core cooling and heat removal systems. This situation is aggravated in the station blackout sequences since all diesel generators are also lost, and adequate core cooling is totally dependent on successful performance of the RCIC System.

19.3.1.5 Results in Perspective

The estimated core damage frequencies are extremely low. It is impossible to calculate such low numbers with a high degree of confidence using the PRA models developed here. For example, a number of potential common cause failures of components such as similar valves have not been included in the fault tree models, on the expectation that such failures are negligible contributors to overall core damage frequency.

In addition, although the ABWR PRA has addressed those initiating events and event sequences identified as potentially significant contributors to core damage risk, it is impossible to be certain that all initiators and event sequences leading to core damage at such low levels of expected core damage frequency have been identified.

19.3.1.6 Positions and Assumptions Implicit in the Analysis

A number of positions were taken and assumptions made at the outset of the internal events analysis which affect the results obtained and conclusions drawn. Included among these was the decision to apply GESSAR II information to the extent possible to the ABWR PRA. As a result, ABWR ECCS test and surveillance intervals were assumed the same as GESSAR II. In addition, estimated unavailabilities of systems not modeled by fault trees, a number of component failure rates, and certain human error probabilities were taken from GESSAR II and used in the ABWR PRA when judged applicable.

No credit was taken for the Firewater Addition System in the level one analysis for several reasons. First, the core damage frequency is very low as discussed in Subsection 19.3.1.5. Second, the containment event tree analysis is significantly simplified if no credit is taken for the firewater system in the accident event tree. And finally, a relatively short period of time is available for the operator to take the necessary actions.

19.3.2 Frequency of Radioactive Release

19.3.2.1 Overview

Accident event trees developed for each of the accident initiators are described as part of the core damage frequency evaluation. These trees model the event progression for the various accident initiators, and provide the classification and frequency of accident sequences. In these event trees, the sequences which are terminated safely without core damage are designated as “OK”. The event sequences which are not successfully terminated could either directly lead to core damage or in some cases could lead to containment structural failure which in turn could lead to core damage. These event sequences are “binned” into various accident classes depending upon the expected event progression, timing and mode of containment failure and the amount of fission product release to the environment.

There are five basic classes (I through V), and a total of ten classes including subclasses such as IA, IB, IC, etc. A Class IA event, for example, is a transient event with loss of high pressure water makeup systems followed by a failure to depressurize the reactor.

The accident event progression for each of the accident classes was analyzed using the MAAP code (Modular Accident Analysis Program). A detailed description of the analysis is given in Subsection 19E.2. For each accident class, these analyses provide the time of RPV failure, containment pressure and time history, and the time at which radioactivity is released to the environment. Also evaluated are the amount of fission products released to the environment.

The event progressions for each of the ten subclasses of events are modeled in the containment event trees (CETs). The CETs model recovery actions which could prevent core damage or arrest core damage if already initiated. Where recovery actions are unsuccessful, the CETs model core melt leading to reactor vessel rupture, containment structural failure and fission product release to the environment. The CET models are based on core–melt progression analysis discussed in Subsection 19E.2. The mode and location of containment failure is modeled based on a study of the containment capability discussed in Appendix 19F.

There is one CET for each of the ten accident classes. The end states of CETs are either states with insignificant or no release (i.e., core damage prevented or core melt arrested), or states with a release path to the environment resulting from the failure of the containment. Associated with each release path in each of the containment event trees, is a frequency of occurrence and a magnitude of fission product release. The frequencies are calculated by the CETs, and the fission product releases are evaluated using the fission product transport analysis discussed in Subsection 19E.2. The numerous release paths can be consolidated or “binned” into release categories by grouping them based on the expected amount of fission product release to the environment.

The consolidated release categories and the associated frequencies are used as input to the consequence analysis discussed in Subsection 19E.3.

19.3.2.2 Accident Classes

Accident event trees developed for each of the accident initiators are described as part of the core damage frequency evaluation. The end states of these accident event trees are "binned" (grouped) into five basic accident classes based on similarities in the subsequent core melt event progression and the containment response. The key factors that influence the definition of the accident classes are as follows:

- Type of initiating event (transient, LOCA, etc.).
- Relative times of core melt and containment failure.
- Whether suppression pool is bypassed.

The type of initiating event is significant because it determines the speed of the event progression. For instance, when no core cooling is available, core melt occurs faster for the LOCA event than for the transient event because of the faster depletion of the coolant inventory.

The relative times of core melt and containment structural failure are important because if core melt occurs first, the time between core melt and containment structural failure is available for decay and removal of radioactive material released in the accident. This time is also available for enabling the operator to recover failed water makeup systems in order to get water on top of the molten core or to regain suppression pool cooling if it had been lost.

The significance of the suppression pool bypass event is that, following core melt, the fission products are released to the environment without the beneficial effects of passing through the suppression pool.

Five basic accident classes, I through V, have been identified. A brief summary of these five classes is provided below:

Class I:	Transient followed by loss of core cooling.
Class II:	Transient with successful core cooling followed by loss of containment heat removal systems.
Class III:	Loss of coolant accident followed by loss of core cooling.
Class IV:	Anticipated transients without scram (ATWS) events with no mitigation.
Class V:	Events in which suppression pool is bypassed (e.g., LOCA outside the containment).

Some of the accident classes are further divided into subclasses in order to facilitate more accurate modeling of the event progression in the containment event trees (CETs). A brief summary of the Class I and III subclasses is provided below. The remaining classes were not subdivided.

Class IA:	Transient, followed by failure of high pressure core cooling system coupled with failure to depressurize the reactor.
Class IB:	Events are broken into three categories:
Class IB-1:	Station Blackout (SBO) event with RCIC failure, onsite power is recovered in eight hours.
Class IB-2:	SBO event, RCIC is available for operation and keeps the core cooled for eight hours at the end of which RCIC is assumed to fail. The suppression pool continues to heat up during RCIC operation.
Class IB-3:	SBO event similar to Class IB-1 but the onsite power is not recovered in eight hours.
Class IC:	ATWS, followed by failure of boron injection and core cooling.
Class ID:	Transients followed by loss of both high and low pressure core cooling systems, reactor at low pressure.
Class IIIA:	Similar to Class IA but for a LOCA initiator.
Class IIID:	Similar to Class ID but for a LOCA initiator.

19.3.2.3 Accident Event Progression

The accident event progression for the above accident classes were analyzed using the MAAP code. A detailed description of the analyses is given in Subsection 19E.2.

A typical core melt sequence may include the following 8 steps:

(1) Core melt in the RPV.

(2) RPV failure.

(3) Discharge of corium (i.e., a mixture of molten metal and core material) and lower plenum water in the lower drywell (LDW) area.

(4) Evaporation of water in LDW producing steam.

(5) Core-concrete interaction producing non-condensible gases.

(6) Drywell heat up causing actuation of a passive flooder system causing suppression pool water to flow to the lower drywell, quenching the corium and terminating the interaction with concrete.

(7) Containment leakage (high containment temperature and pressure discussed in Subsection 19.3.2.4).

(8) Containment overpressure leading to fission product release (see Subsection 19.3.2.4 for limiting pressure).

Each accident sequence is unique with respect to timing of the above events, rate of containment pressurization and pressure rise and the order in which they occur.

The “Passive Mitigation” discussed above in (6) is a unique feature of ABWR containment configuration which allows for core melt arrest without the use of active components.

For purposes of illustration, the timing of a typical Class ID sequence is given below:

RPV failure:	1.8 hour
Water in LDW boils off:	2.7 hours
Passive Mitigation	5.4 hours
Rupture disk opens:	20.2 hours

Class II sequences which involve successful core cooling but no containment heat removal are significantly different. Rupture disk opening occurs in about 20 hours following which core cooling continues for a long time (> 100 hours) before it is necessary to replenish suppression pool water inventory

Associated with each accident sequence is the amount of fission products released to the environment. This depends upon factors such as the amount of release through the suppression pool prior to RPV failure, timing and location of containment structural failure, core decay heat at the time of accident. The fission products released are documented in Subsection 19E.2.

19.3.2.4 Containment Structural Capability

The ABWR containment design pressure is 0.412 MPa. Past stress analyses preformed for other PRAs have shown that the containments are capable of withstanding much higher pressure (typically 2 to 3 times the design pressure). A discussion of the ABWR containment capability is provided in Appendix 19F. The containment structural capability is limited by that of the drywell head. The drywell pressure capability depends upon the containment temperature. At 533 K (500°F), (which is a typical temperature for most accident sequences), the drywell median ultimate strength is evaluated to be 1.025 MPa.

19.3.2.5 Containment Structural Failure Modes and Location

In Appendix 19F it is concluded that when the containment is pressurized, the most likely mode of failure is the plastic yield of the drywell torispherical dome. Containment rupture which impairs the ability of the containment to provide structural support is not judged to be a credible mode of failure. Containment leakage at pressures below the failure pressure is judged to be not significant (i.e., not sufficient to depressurize the containment). However, at high temperatures [i.e., >533 K (500°F)] there is a potential for degradation of seals in the large operable penetrations such as the equipment hatch and personnel air locks. A conservative evaluation shows that leakage is expected to occur only when the containment pressure exceeds 0.460 MPa.

The following failure modes are explicitly modeled in the CETs.

(1) Containment leakage occurs when the temperature exceeds 533 K (500°F) and pressure exceeds 0.460 MPa.

(2) Drywell head failure occurs when the containment pressure exceeds 1.025 MPa and temperature is below 533 K (500°F).

(3) Containment high temperature failure occurs when the containment experiences a very high temperature [>811 K (1000°F)].

There are a number of containment structural failure modes which have been shown to be negligible contributors to plant risk in past PRAs and, therefore, are not included in the ABWR CETs. Examples of such failure modes are steam explosion, basemat penetration, pressure vessel rupture leading to containment failure, etc. Hydrogen detonation is not modeled because the ABWR containment is inerted and hydrogen detonations are, therefore, judged to be negligible contributors to ABWR risk.

19.3.2.6 Suppression Pool Bypass Events

The ABWR suppression pool plays a key role in reducing the fission products released to the environment following a severe accident. Fission products released through the suppression pool benefit from the “scrubbing” action which traps most of the fission products such as cesium iodide. However, if the accident sequence results in bypass of the suppression pool, the magnitude of the associated release could be a factor of 100-10000 more than that for a sequence which discharges through the suppression pool.

It is, therefore, important to study the suppression pool bypass paths and evaluate its impact on the PRA results. There are a number of ways the ABWR suppression pool can be bypassed. Most of them involve some combination of pipe and valve failures, or leakage through closed isolation valves. Examples of suppression pool bypass paths are as follows:

(1) Failure of MSIVs and turbine bypass valves

(2) Failure of MSIVs and main steamline break outside containment

(3) Wetwell-drywell vacuum breaker failure.

A separate study of these suppression pool bypass paths was conducted and it was concluded that the contribution of these paths to ABWR risk was small. With the exception of the wetwell-drywell vacuum breakers not including these paths in the CET, models explicitly will affect the risk results by a small percentage of the total risk. Only the vacuum breakers were modeled in the CETs. However, CETs also model the suppression pool bypass paths resulting from the structural failure of the containment. For instance, the three containment failure modes discussed in Subsection 19.3.2.5 and modeled in the CETs (leakage, overpressure and high temperature failure) all lead to suppression pool bypass.

The suppression pool bypass study is documented in Subsection 19E.2.3.3.

19.3.2.7 Recovery of Failed Systems

Recovery of failed systems, onsite and offsite power has been modeled in the CETs.

System recovery probabilities are generally calculated using the exponential recovery formula:

$$P_f = \text{Exponential}(-T/MTTR)$$

where:

P_f	=	Probability of failure to recover
T	=	Available repair time
MTTR	=	Mean time to repair

For accident sequences in which core melt had proceeded to the point of RPV failure, it was judged that high radiation might make it difficult to carry out some repair activities. For events involving station blackout, the recovery data was based on historical data.

The time available for repairing or recovering each system was determined by the time within which the system had to be operating to prevent the occurrence of failure (core recovery, containment overpressure, etc.). The available repair times were obtained based on the core melt progression analysis discussed in Subsection 19E.2.

19.3.2.8 Core Melt Arrest Success Criteria

The accident event progression analysis described in Subsection 19.3.2.3 shows that core melt can be arrested by quenching the molten corium. The core melt arrest can take place within the RPV in the early stages of the accident if core cooling can be recovered in time. If this does not occur, then the core melt proceeds to RPV failure and the molten corium is discharged into the LDW. The core melt can be arrested in the containment if the core cooling is recovered before the containment experiences structural failure due to overpressure, leakage or high temperature. In many sequences, the core melt is arrested by passive flooder system operation. In Class IA accident sequences, (i.e., loss of high pressure core cooling system coupled with failure to depressurize reactor), the RPV failure depressurizes the reactor making the low pressure core cooling systems available for arresting core melt in the containment.

In addition to core melt arrest, one containment heat removal system must also be re-established to prevent overpressurizing the containment.

The core melt arrest success criteria is discussed in detail in Subsection 19D.5.8.

19.3.2.9 Containment Overpressure Protection

Sensitivity studies in Subsection 19E.2.8.1.4 were conducted to determine the value of providing a containment overpressure relief feature. The results show a substantial reduction in offsite dose.

19.3.2.10 Containment Release Categories

The amount of radioactive release to the environment depends upon a number of factors such as the timing of containment failure and the location of containment failure. Ideally, there is a specific radioactive release associated with each outcome of the containment event trees. However, evaluating the source terms for each event tree output is very time consuming. Therefore, the releases with similar characteristics are grouped (“binned”) together to define release categories as discussed in Subsection 19E.2.2.

19.3.2.11 Containment Event Trees

The results of the accident event trees were grouped into ten accident classes. In general, one CET was developed for each of the accident classes. However, two of the accident classes IC and IV, had negligibly low occurrence frequencies and CETs were not developed for these accident classes. Class IC event frequencies were added to the class IA frequencies and the class IV frequencies were assumed to result directly in core damage and early containment failure.

The CETs model recovery actions and containment failure modes. The end states of CETs are either states with insignificant or no release (i.e., core damage prevented or core melt arrested), or states with a release path to the environment resulting from the failure of the containment. The end states are assigned a source term category grouping which depends on key containment performance criteria as shown in Figure 19D.5-3. These results are then binned into release categories as discussed in 19.2.3.10.

19.3.2.12 Results

The results (discussed in detail in Subsection 19D.5.12) indicate core damage frequency is extremely small for internal events. These results, together with the associated source terms, form the input for the consequence analysis in Subsection 19E.3.

19.3.3 Magnitude and Timing of Radioactive Release

The evaluation of the fission product release was performed as discussed in Subsection 19E.2. Representative accident sequences were chosen for study on the basis of the core damage and containment event trees. Each accident sequence was then evaluated for the timing and magnitude of release.

There are three important considerations for the timing of fission product release when considering the consequences of a potential severe accident.

(1) The time available for fission product decay affects the maximum source which could be released. In an extreme case, if all of the fission products were released after an infinite period of time, the offsite dose would be zero because all the fission products would have decayed to stable states. In the ABWR, the COPS ensures that the noble gasses are the only significant release from the containment for most sequences. The potential dose associated with the release of noble gasses drops to less than 10% of its initial value within 7 hours of shutdown. Twelve hours after shutdown, the potential dose has dropped to 5% of its initial value, and it decreases very slowly thereafter. For cases without COPS actuation, the potential dose can be dominated by iodine species. These species decay very slowly retaining two-thirds of their potential dose after 40 hours.

(2) The time between the release of fission products from the core and the time of release from containment (residence time) affects the removal in containment. For releases through the COPS, this term is not important since noble gasses are not retained, and the suppression pool effectively scrubs the remaining fission products as they pass through the pool. This time can be important for the few accidents which have drywell releases. However, for most sequences, a time delay of a few hours after

release from the fuel brings the airborne fission product concentration to its equilibrium value. This is primarily the result of the submergence of the debris with water from the Firewater Addition System or the passive flooder.

(3) The time available for offsite evacuation, should it be necessary, is also important. Discussions with several utilities indicate that evacuation of their Emergency Planning Zones (EPZ) can be completed in less than 8 hours, even in the worst weather conditions. Experience has also indicated that ad hoc planning can successfully evacuate a region in about 24 hours (Appendix 6J of Reference 19.3-4).

Based on the forgoing, four time frames were selected in determining the time of fission product release, either via the rupture disk or directly from the drywell. Table 19.3-6 summarizes the results which were obtained by using the probabilities summarized in Table 19.3-4 and assigning them to a time and mode of release based on the accident analysis contained in Subsection 19E.2.2.

19.3.4 Consequence of Radioactive Release

The evaluation for consequences of potential radioactive releases was performed using the CRAC-2 computer code as is detailed in Subsection 19E.3. Based upon the evaluation of plant performance, accident classes were defined in terms of their associated release characteristics and fission product releases. Each accident class was then evaluated by the CRAC-2 code at five sites, one representing each major geographical region of the United States. Each site was chosen as representative of its geographical region based upon meteorological calculations and was further defined as average in terms of population density for that geographical region. The results for the five sites were averaged and compared to three goals, two based upon the NRC safety goal policy of minimizing risk to an individual and the public near a plant, and the third based upon an industry goal of minimizing the dose close to the plant. The results of this study show that the ABWR Standard Plant satisfies these goals.

19.3.5 References

19.3-1 Not Used.

19.3-2 Not Used.

19.3-3 Not Used.

19.3-4 "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", WASH-1400, NUREG-75/014, United States Atomic Energy Commission, October 1975.

19.3-5 Not Used.

19.3-6 Not Used.

19.3-7 "Analysis of a High Pressure ATWS with Very Low Makeup Flow", DOE/ID-10211, Idaho National Engineering Laboratory, October 1988.

19.3-8 Eide, S.A., Wierman, T.E., Gentillon, C.D., et al., Industry Average Performance for Components and Initiating Events at U.S. Commerical Nuclear Powerplants. USNRC, NUREG/CR-6928, February 2007.

Table 19.3-1 Initiating Event Frequencies

Initiating Event	Frequency Per Reactor Year*
Manual Shutdown	
Isolation/Loss of Feedwater	
MSIV Closure	
Loss of Condenser Vacuum	
Press. Reg./Bypass Valves Closed	
Loss of Feedwater	
Non-Isolation Event (Trip with bypass)	
Inadvertent (Stuck) Open Relief Valve	
Loss of Offsite Power	
Less than 30 minutes	
30 Minutes to 2 Hours	
2 to 8 Hours	
Greater than 8 Hours	
Small LOCA	
[Liquid Break 5.063 cm^2 (0.00545 ft^2) or less] [Steam Break <278.7 cm^2 (<0.3 ft^2)]	
Medium LOCA	
[Liquid Break greater than 5.063 cm^2 (0.00545 ft^2) and less than 278.7 cm^2 (0.3 ft^2)]	
Large LOCA	
[Liquid Break 278.7 cm^2 (0.3 ft^2) or greater] [Steam Break 278.7 cm^2 (0.3 ft^2) or greater]	

* Not part of DCD (refer to Section 19D.3).

Table 19.3-2 Success Criteria to Prevent Initial Core Damage for Transient and LOCA Events With RPS Scram

Event	Success Criteria
CORE COOLING:	
Large Liquid LOCA [>=278.7 cm^2 (0.3 ft^2)]	HPCF-B or C or LPFL(1) – A or B or C
Large Steam LOCA [>=278.7 cm^2 (0.3 ft^2)]	HPCF-B or C or LPFL(1) – A or B or C or 1 Condensate Pump + 1 Condensate Transfer Pump(2)
Medium Liquid LOCA [>=278.7 cm^2 (0.3 ft^2) >5.063 cm^2 (0.00545 ft^2)]	HPCF-B or C or ADS3(3) + LPFL(1) – A or B or C
Small Liquid LOCA [>=5.063 cm^2 (0.00545 ft^2)]	RCIC(4) or HPCF-B or C or ADS3(3) + LPFL(1) – A or B or C or ADS3(3,5) + 1 Condensate Pump + 1 Condensate Booster Pump + 1 Condensate Transfer Pump(2)
All Transients (including IORV)	RCIC(4) or HPCF-B or C or 1 Feedwater Pump + 1 Condensate Pump + 1 Condensate Booster Pump + 1 Condensate Transfer Pump(2) or ADS3(3) + LPFL(1) – A or B or C or ADS3(3,5) + 1 Condensate Pump + 1 Condensate Booster Pump + 1 Condensate Transfer Pump(2) or ADS8(6) + 1 Firewater Addition System Pump

Table 19.3-2 Success Criteria to Prevent Initial Core Damage for Transient and LOCA Events With RPS Scram (Continued)

Event	Success Criteria
CORE COOLING (Cont.)	
Small Steam LOCA [<278.7 cm^2 (0.3 ft^2)]	HPCF-B or C or 1 Feedwater Pump + 1 Condensate Pump + 1 Condensate Booster Pump + 1 Condensate Transfer Pump(2) or ADS3(3) + LPFL(1) – A or B or C or ADS3(3,5) + 1 Condensate Pump + 1 Condensate Booster Pump + 1 Condensate Transfer Pump(2)
LONG-TERM HEAT REMOVAL:	
All Transients or Small Liquid LOCA	RHR-A or B or C(7) or Normal Heat Removal(8) or CUW(9)
All Steam LOCAs or IORV or Liquid LOCA (Large or Medium)	RHR-A or B or C(7) or Normal Heat Removal(8)

Table 19.3-2 Success Criteria to Prevent Initial Core Damage for Transient and LOCA Events With RPS Scram (Continued)

Event	Success Criteria
PRESSURE RELIEF:	
Isolation Events	6 Safety/Relief Valves
Non-Isolation Events	3 Turbine Bypass Valves or 2 Turbine Bypass Valves + 2 Safety/Relief Valve or 1 Turbine Bypass Valve + 4 Safety/Relief Valves or 6 Safety/Relief Valves

Notes:

(1) The term “LPFL” refers to the low pressure core flooding mode of the Residual Heat Removal (RHR) System.

(2) The condensate pumps take suction from the hotwell which is a limited water source. Therefore, if the MSIVs are not open, a condensate transfer pump is necessary to pump water from the condensate storage tank to the hotwell in order to replenish the water in the hotwell.

(3) The term “ADS3” implies that at least 3 automatic depressurization valves are automatically actuated on low level and high drywell pressure or the same number of SRVs are manually opened when the ADS would have actuated. For transients, the high drywell pressure signal is not present.

(4) The RCIC turbine needs sufficient steam generation at or above the required minimum pressure to drive the pump. For the IORV the RCIC will provide adequate cooling for at least 2 hours.

(5) If none of the motor driven ECCS pumps are running, the ADS will not automatically initiate.

(6) The term “ADS8” implies that the 8 automatic depressurization valves or the same number of SRVs are manually opened within one minute after the reactor vessel water level has decreased to the lower water level 1 setpoint.

(7) If the reactor system is at high pressure, the RHR System would be operated in the pool cooling mode. If the reactor system is at low pressure, the RHR System can be operated in either the pool cooling or the shutdown cooling mode.

(8) The MSIVs or the main steam drainlines must be manually opened, if previously closed, for this system to work. This requires the availability of a nitrogen gas supply. If the RPV is depressurized, the main steam drainline option is not viable since it will not pass enough steam to remove the decay heat energy. Furthermore, the circulating water pumps are required to cool the main condenser. Also, a condensate pump is required to transfer excess water from the hotwell to the suppression pool or the condensate storage tank.

(9) The Reactor Water Cleanup (CUW) System is capable of removing the energy due to decay heat (at greater than 4 hours after scram) at high RPV pressures if the return water bypasses the regenerative heat exchanger. Manual override of CUW isolation signals would be necessary if a level 3 isolation signal is present.

Table 19.3-3 Success Criteria and Required Operator Actions For ATWS Events

Initiator	Success Criteria	Time of Operator Action
PRESSURE RELIEF:		
Isolation Initiators	15 Safety/Relief Valves	
Non-Isolation Initiators	3 Turbine Bypass Valves + 9 Safety/Relief Valves	
	or	
	2 Turbine Bypass Valves + 11 Safety/Relief Valves	
	or	
	1 Turbine Bypass Valve + 13 Safety/Relief Valves	
	or	
	15 Safety/Relief Valves	
CORE COOLING:		
All events with rod insertion	Table 19.3-2	
IORV without rod insertion	1 Feedwater Pump	
	or 2 of RCIC or HPCF-B or HPCF-C	
All other events without rod insertion	1 Feedwater Pump	
	or	
	RCIC	
	or	
	HPCF-B or C	
POWER REDUCTION:		
With Rod Insertion:		
All events	Electric Rod Run In	Automatic
	or	
	ARI	Automatic
	or	
	Manual Insertion	10 minutes

Table 19.3-3 Success Criteria and Required Operator Actions For ATWS Events (Continued)

Initiator	Success Criteria	Time of Operator Action
With Boron Insertion		
Isolation Events(1)	RIP Trip	Automatic
	ADS Inhibit	5 minutes
	Feedwater Runback	10 minutes
	1 SLC	10 minutes
Non–Isolation Events(1)	ADS Inhibit	10 minutes
	Feedwater Runback	10 minutes
	Manual RIP Trip	10 minutes
	1 SLC	10 minutes
LONG-TERM HEAT REMOVAL		
Isolation Events	RHR–A or B or C	30 minutes
Non–Isolation Events	Normal Heat Removal(2) or RHR-A or B or C	30 minutes

Notes:

(1) MSIV Closure and LOSP initiators generate RIP Trip. IORV and Turbine Trip Events require Manual RIP Trip.

(2) Adequate normal heat removal will be provided through the Turbine Bypass valves.

Table 19.3-4 Frequency of Core Damage by Accident Class

Accident Class	Description	Frequency (Events/year)*
IA	Transients followed by failure of high pressure core cooling and failure to depressurize the reactor.	
IB-1	Station blackout events (short term) with RCIC failure.	
IB-2	Station blackout events with RCIC available for core cooling for approximately eight hours.	
IB-3	Station blackout events (long term) with RCIC failure.	
IC	ATWS events without boron injection coupled with loss of core cooling.	
ID	Transients followed by loss of high pressure core cooling, successful depressurization, and loss of low pressure core cooling.	
II	Transient, LOCA, and ATWS (with boron injection) events, with successful core cooling but with possible failure of containment.	
IIIA	Small or medium LOCAs with failure of high pressure core cooling followed by failure to depressurize the reactor.	
IIID	LOCAs followed by loss of high pressure core cooling, successful depressurization, and loss of low pressure core cooling.	
IV	ATWS events without boron injection but with core cooling available.	
Total		

* Not part of DCD (Refer to Subsection 19D.5).

Table 19.3-5 Frequency of Core Damage by Initiating Event

Initiating Event	Description	Frequency* (Events/year)	% of Total
TM	Reactor Shutdown		
TT	Non-Isolation (Turbine Trip)		
TIS	Isolation/Loss of Feedwater		
TE2	Loss of Offsite Power for Less Than Two Hours		
TE8	Loss of Offsite Power for Two to Eight Hours		
TE0	Loss of Offsite Power for More Than Eight Hours		
BE2	Station Blackout for Less Than Two Hours		
BE8	Station Blackout for Two to Eight Hours		
BE0	Station Blackout for More Than Eight Hours		
TIO	Inadvertent Open Relief Valve		
S2	Small Break LOCA		
S1	Medium Break LOCA		
S0	Large Break LOCA		
ATWS	Anticipated Transient Without Scram		
Total			

* Not part of DCD (Refer to Subsection 19D.3).

Table 19.3-6 Frequency Of Fission Product Release

Time of Release	Release Frequency*	
No Release		
	Release via Rupture Disk*	**Release via Drywell***
> 24 hours		
16 to 24 hours		
8 to 16 hours		
< 8 hours		

* Not part of DCD (Refer to Subsection 19D.5).

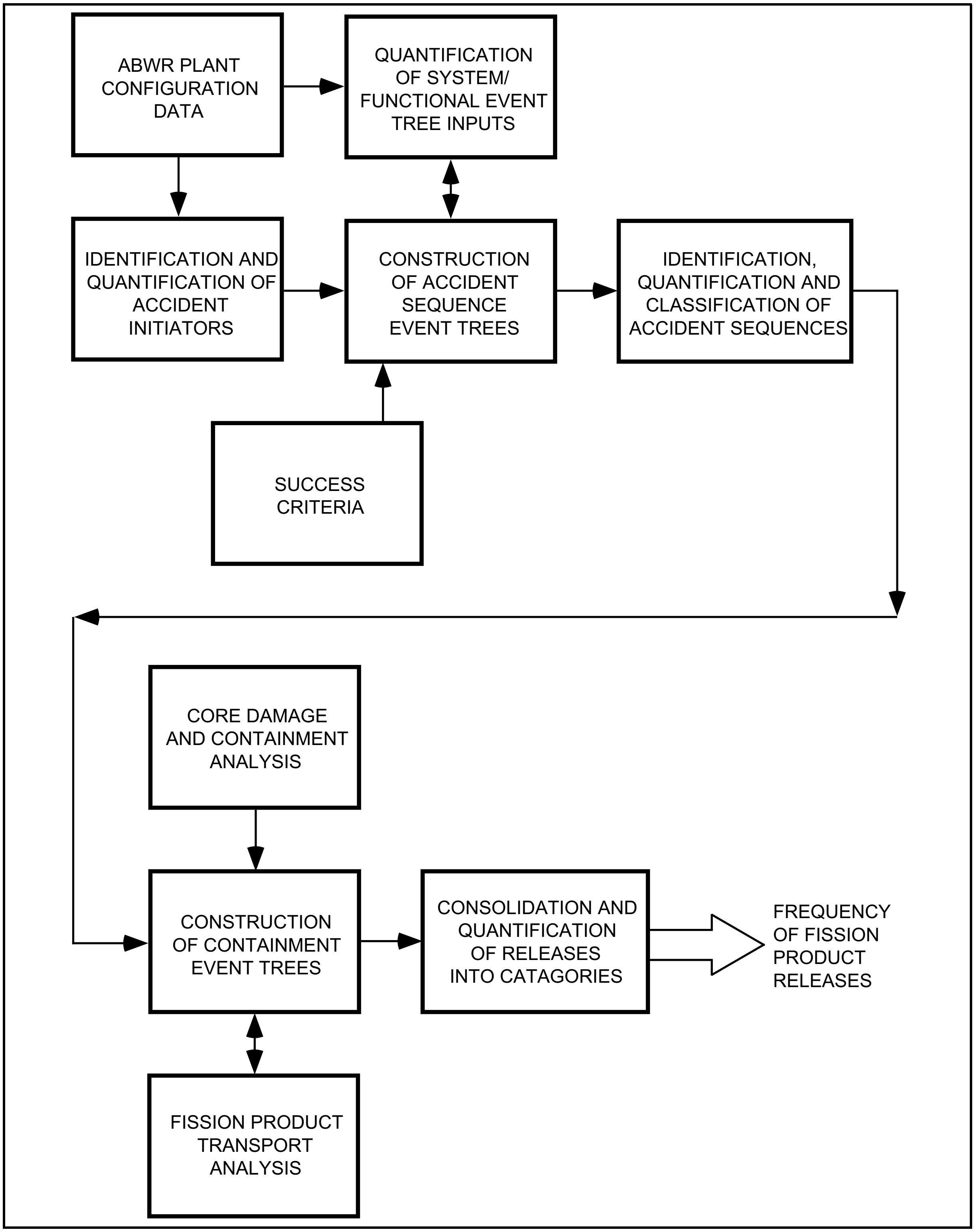

Figure 19.3-1 Overview of Methodology for Assessing Frequency of Core Damage and Fission Product Releases

Figure 19.3-2 Reactor Shutdown Event Tree

Not Part of DCD (Refer to Subsection 19D.4)