2 INITIATING EVENTS

Contents

List of Tables

2 INITIATING EVENTS

2.1 INTRODUCTION

Event trees are used in a PRA to model discrete accident sequences. An accident sequence is comprised of an initiating event followed by challenges to system successes and failures that lead to either safe, stable plant conditions or to a plant condition that is considered unacceptable. An initiating event is any occurrence that disrupts normal plant operations sufficiently to require a reactor trip by either automatic or manual action. An initiating event may occur due to a random component failure or a human action, thus requiring plant systems to respond to maintain the unit in a safe, stable condition. The initiating event marks the starting point of the accident sequence analysis.

One of the first and basic steps in a PRA is the identification and quantification of the initiating events to be used in the sequence analysis. Initiating events have historically been broadly classified as either "internal" or "external" events. The initiating events discussed in this section are limited to at-power internal initiating events, that is, those initiating events occurring during power operation either as a direct result of equipment failure, or as the result of errors while performing maintenance, testing, or any other operator action. External initiating events (e.g., seismic events, internal floods) and initiating events during shutdown are discussed in other sections.

Initiating event analysis involves the following two major steps:

(1) Identification and grouping, and

(2) Frequency quantification.

Each of these key steps is discussed below.

2.2 INITIATING EVENT IDENTIFICATION AND CATEGORIZATION

In order to identify the list of initiating events to be considered in the analysis, the following steps are taken:

(1) Identification of the set of initiating events applicable to the ESBWR plant by reviewing NUREG/CR-5750 (Ref. 2.2-1), and

(2) Analysis of the failure of individual ESBWR systems that could result in additional initiating events.

Human error induced initiating events are also considered for these analyses.

Individual initiating events that require similar response from front line and auxiliary systems and operators are combined into initiating event categories. Combining initiating events into categories reduces the number of event trees that need to be developed. The general categories of internal initiating events are:

(1) Transients,

(2) Loss of coolant accidents (LOCA),

(3) Interfacing-System LOCA (ISLOCA), and

(4) Special initiators.

Note that fire initiators are addressed in Chapter 12 and internal flooding initiators are addressed in Chapter 13. The categorization of initiating events within each of these major groups is discussed below.

The postulated initiating events to be addressed in the at-power PRA are derived from a review of nuclear power plant operating experience, as summarized in the NUREG/CR-5750. NUREG/CR-5750 builds upon previous industry studies with similar objectives, such as EPRI Report NP-2230 (Ref. 2.2-2) published in 1982 and NUREG/CR-3862 (Ref. 2.2-3), published in 1985. The NUREG/CR-5750 categories are applicable, in general, to all BWR and PWR plants that are currently in operation. The NUREG/CR-5750 initiating event categories are listed in Table 2.2-1 and are judged to represent the appropriate set of categories for the ESBWR.

Some systems in the ESBWR design differ from those for the operating BWR plants. In addition, the ESBWR design contains several innovative systems; as such, certain NUREG/CR-5750 categories are not directly applicable to the ESBWR. These are indicated in Table 2.2-1.

2.2.1 Transients

The transient initiating event categories are developed based on the following considerations:

- Effect of the initiator on feedwater and power conversion system (PCS) availability.
- Effect of the initiator on systems required to prevent core damage or containment failure.
- Successful configuration (e.g., mode of operation, number of pumps in operation) of mitigating systems.
- Effect of the initiator on operator response.

The resulting transient initiating event categories for the ESBWR PRA are as follows:

- General Transient,
- Transient with Power Conversion System (PCS) Unavailable,
- Loss of Feedwater,
- Loss of Preferred Power (LOPP), and
- Inadvertent Opening of SRV (IORV).

The results of this analysis and the NUREG/CR-5750 initiating events grouped into each of the ESBWR PRA initiating event categories are summarized in Table 2.2-2.

2.2.2 LOCAs

Loss of Coolant Accident (LOCA) initiator categories are defined based on approaches in previous BWR PRAs, plant specific configuration, and success criteria.

LOCA initiators are classified according to the following main factors:

(1) Level position: Steam breaks (for pipes above Level 3) or liquid breaks (for pipes below Level 3.) The exceptions are the RWCU/SDC pipe at elevation 17215 mm and the feedwater pipe which is at elevation 18915 mm which are below Level 3 at 19780 mm.

Thermal hydraulic calculation concluded that the RWCU/SDC and feedwater pipe break performed like a steam break.

(2) Break size: Large, medium, or small classifications are used.

(3) Pipe function: Emergency Core Cooling System (ECCS) line break or non-ECCS line break (the former affects ECCS operability).

(4) Location: Inside/outside containment breaks (breaks outside containment can potentially be isolated).

(5) Pipe class: LOCAs can be postulated for situations in which valves between the reactor coolant pressure boundary (RCPB) and non-RCPB segments fail, propagating high pressure to low pressure design piping.

(6) Coolability: Coolable LOCAs (all the above categories) or non-coolable LOCAs.

2.2.2.1 LOCAs Inside Containment

These initiators are failures of the RCPB that occur inside the containment during power operation. LOCAs inside containment are categorized into the following LOCA sizes:

- Large: RPV depressurizes sufficiently (without the need for additional depressurization methods) to allow GDCS injection prior to uncovering the top of active fuel (TAF).
- Medium (Steam): RPV depressurizes sufficiently (without the need for additional depressurization methods) to allow active low pressure injection prior to uncovering TAF.
- Medium (Liquid): CRD insufficient as a high pressure makeup source because the break flow exceeds CRD injection flow. RPV depressurization is required for low pressure injection.
- Small (Steam or Liquid): CRD is sufficient as a makeup source. RPV depressurization is required for low pressure injection.

The lower limit of LOCAs is defined as a RCPB leak small enough to be managed by normal operation water makeup and containment cooling systems. The Turbine Trip initiating event category considers this level of RCPB leakage.

An analysis is performed to establish the appropriate LOCA classification for each line connected to the Reactor Coolant Pressure Boundary (RCPB). Table 2.2-3 summarizes the number, the diameter, and the nozzle elevation of lines connected to the RCPB. For each line, Table 2.2-3 classifies the breaks into Large, Medium, and Small LOCA sizes according to the pipe cross-section area, and whether it is a liquid or steam break. Criteria for the classifications are:

- Pipes connected to the RPV at elevations above Level 3 are classified as steam breaks (even though a liquid phase is initially discharged through the break).
- Pipes connected to the RPV at elevations below Level 3 are classified as liquid breaks (RWCU/SDC pipe at elevation 17215 mm and feedwater pipe at elevation 18915 mm are considered as steam break).

The generic BWR emergency procedure guidelines direct the operators to maintain reactor vessel level between Level 3 and Level 8 after reactor trip. This action is considered applicable for ESBWR, so any LOCA below Level 3 results in the continual release of water through the break, provided that an injection system into the vessel is available.

As such, the LOCA categories in terms of location and diameter are as follows:

- Large (steam): >305 mm (12") dia. piping above Level 3,
- Large (liquid): >305 mm (12") dia. Piping below Level 3,
- Medium (steam): 264mm (10.4") – 305 mm (12") dia. piping above Level 3,

- Medium (liquid): 25 mm (1") – 305 mm (12") dia. piping below Level 3,
- Small (steam): <264 mm (10.4") dia. piping above Level 3, and
- Small (liquid): <25 mm (1") dia. piping below Level 3.

The diameter of an orifice or a nozzle in a line, when it exists, is considered in the break size classification.

The initiating event categorization for LOCAs inside containment is summarized below and in Table 2.3-3.

Large Break LOCA

The large break LOCA is sufficiently large to depressurize the reactor to permit low pressure system injection, without the need for additional depressurization methods. The ESBWR design does not contain any piping with diameters greater than 12" below Level 3, thus, all breaks analyzed are steam break. Inadvertent opening of all DPV/SRVs is also included in this group.

This initiating event category corresponds to the NUREG/CR-5750 category (G7) Large Pipe Break LOCA.

Medium Break LOCA

- Steam break case - The medium break steam LOCA is large enough to depressurize the reactor sufficiently to permit low pressure system injection (except GDCS) without the need for additional RPV depressurization methods. Inadvertent opening of a DPV is included in this group. No additional ESBWR lines fall into this category.
- Liquid break case - The flow rate at reactor pressure for a medium break liquid LOCA is greater than the CRD makeup capacity. Depressurization is needed for GDCS injection to be effective. A number of ESBWR lines fall into this category.

This initiating event category corresponds to the NUREG/CR-5750 category (G6) Medium Pipe Break LOCA.

Small Break LOCA

- Steam break case - The small break steam LOCA is sufficiently small such that RPV depressurization is needed for effective low pressure injection (including LPCI and FPS). ESBWR instrument lines above Level 3 fall into this category.

  The consequences of an inadvertent opening of one SRV (IORV) are similar to the consequences of a small steam LOCA; however, IORV is analyzed separately as a transient because of the steam discharge to the suppression pool versus the drywell.
- Liquid break case - CRD is a viable high pressure injection source, as the flow rate for a small break liquid LOCA is less than the CRD makeup capacity. RPV depressurization is needed for effective low pressure system injection. ESBWR instrument lines below Level 3 fall into this category.

This initiating event category corresponds to the NUREG/CR-5750 category (G3) Small Pipe Break LOCA.

Excessive LOCA

An "Excessive LOCA" is defined as a LOCA that cannot be mitigated by any combination of engineered systems. LOCAs, such as vessel rupture have been considered excessive LOCAs in the past; however, the ESBWR design is such that, depending on location, the vessel rupture can be recovered. If the LOCA location is above the core, the LOCA is recoverable and transfers to the Reactor Vessel Rupture (RVR) Event tree. If the excessive LOCA location is below the core, no recovery is credited.

2.2.2.2 LOCAs Outside Containment

LOCAs outside containment are defined as breaks in high pressure systems outside the containment. The break can be isolated by closure of the isolation valves, and this is addressed in the accident sequence analysis.

This initiating event category corresponds to NUREG/CR-5750 categories (K1) Steam Line Break Outside Containment and (K2) Feedwater Line Break.

2.2.2.3 Interfacing System LOCAs

An interfacing system LOCA occurs at a breach in a system that interfaces with the RCS, where isolation between the breached system and the RCS fails. An ISLOCA is characterized by the over pressurization of a low pressure system when subjected to RCS pressure and can result in a containment bypass. This initiating event category corresponds to NUREG/CR-5750 category (N1) Interfacing System LOCA.

2.2.3 Special Initiators

Special initiating events refer to plant-specific malfunctions that could lead to a plant trip. A new initiating category is identified when the event is not associated with any of the categories defined before; otherwise, it is grouped into one of the defined categories and the frequency of the existing category is modified to account for the contribution of the systemic malfunction.

In general, systems that can influence scram or isolation signals (either directly or as a consequence) are analyzed. The list of systems analyzed is as follows:

Front line systems

- Control Rod Drive System
- Feedwater / Condensate
- Isolation Condenser System
- Depressurization System
- Gravity Driven Cooling System

- Fuel and Auxiliary Pool Cooling System
- Reactor Water Cleanup / Shutdown Cooling System
- Standby Liquid Control System

Support Systems

- Reactor Component Cooling Water System
- Turbine Component Cooling Water System
- Plant Service Water System
- Air Systems (High Pressure Nitrogen Supply, Service Air, Instrument Air)
- 13.8 kVAC and 6.9 kVAC bus system
- 250VDC bus system
- Reactor Water Level Instrumentation (RWLI) System
- Drywell Cooling System

For each of these systems, the possibility for the generation of inadvertent scram or isolation signals is investigated. The effects of spurious equipment actuation, human errors, hardware failures and pipe breaks in these systems are also considered.

A qualitative screening analysis is performed to eliminate errors or failures which are not credible or which do not contribute to a scram initiation or isolation function. This analysis is described in the subsections below.

2.2.3.1 Control Rod Drive System

The CRD system has one pump in operation during normal plant operation. The other pump is in standby and starts automatically upon a reactor water Level 2 signal.

In normal operation the system performs the following functions:

- Charging of hydraulic control units, and
- Supply of purging water to the control rod drives (CRD), RWCU/SDC pumps and reactor water level reference leg instrument lines.

Refer to Section 4.3 for a description and simplified diagram of the system.

The following malfunctions are considered:

(1) Trip of the operating pump: No major effects are expected because at the time of the pump trip the hydraulic control units are charged and the check valve on the charging line prevents their discharge. Also, loss of purge does not impair the Fine Motion Control Rod Drive (FMCRD) function in the short term. If the tripped pump or the standby pump cannot be restarted, a controlled reactor shutdown is pursued.

(2) Spurious actuation of the standby pump and opening of the injection valve: This malfunction results in additional inventory of cold water entering the reactor pressure

vessel (RPV) from the feedwater line. Because the feedwater system is able to regulate the injection flow (by level control) and the temperature reduction of the total flow entering the vessel is negligible when feedwater is operating, no consequences are expected from this event.

(3) Injection line breaks downstream of the check valves that isolate CRD from the RWCU line): This line is common to the injection of CRD and RWCU train A; because it belongs to RWCU, the consequences of this event are analyzed within RWCU below.

2.2.3.2 Feedwater / Condensate Systems

The Feedwater and Condensate systems are in operation at all times during normal plant operation. Refer to Section 4.9 for a description and simplified diagrams of these systems.

The following malfunctions are considered:

(1) FW controller malfunction (reactor water level increase): The FW controller is single failure proof. Therefore, the FW controller failure is much less likely to occur in the ESBWR than in current generation reactors.

In addition, there is the added protection of a FW runback, closure signals to the high pressure makeup (CRD) flow control valves on reactor water Level 8, and trip of the FW pumps on reactor water Level 9. A scram occurs on reactor water Level 8; this event is included in the General Transient initiating event category.

In case a high reactor water level trip signal does not operate and the operator does not perform backup action, the following Main Steam Line (MSL) flooding scenarios can be postulated:

- MSL break: this event is similar to a steam LOCA event.
- MSL does not break and ICS does not operate: in this case, the SRV opens and discharges water. A break of the SRV discharge might occur and cause a LOCA inside the drywell. The consequence of this event is bounded by the Small Steam LOCA event.
- No pipe break occurs, the ICS is not operating, and an SRV remains stuck open: The consequence of this event is bounded by the IORV event.

The frequency of all the above failures is estimated to be below any LOCA frequency.

(2) FW controller malfunctions (reactor water level decrease): A controller malfunction that results in reduced feedwater flow is included in the General Transient initiating event category. A controller malfunction that results in loss of all feedwater flow is included in the Loss of Feedwater initiating event category.

(3) Partial loss of feedwater: The design of the ESBWR is such that loss of a single train of feedwater or condensate will not cause a reactor trip. Loss of two or three trains is not considered credible because there are four independent power trains supplying the two trains of four condensate and feedwater pumps and it would take a common cause failure of two out of four pumps or three out of four pumps to lead to a partial loss of feedwater for two or three condensate or feedwater pumps. Therefore, malfunctions involving partial loss of the condensate or feedwater systems are not modeled.

(4) Complete loss of feedwater: Malfunctions that cause complete loss of condensate or complete loss of feedwater are included in the Loss of Feedwater initiating event category.

2.2.3.3 Isolation Condensers Systems

This system is in standby during normal plant operation. It is automatically actuated in case of reactor isolation or high reactor pressure. Refer to Section 4.2 for a description and simplified diagram of this system.

The following malfunctions are considered:

a. Spurious actuation of isolation condensers: The effect on the plant for spurious actuation of from one to four isolation condensers is an insertion of positive reactivity. The likely consequence of the event is a reactor trip. The water level changes are corrected by FW flow control. This event is included as part of the General Transient initiating event category.

b. Spurious opening of one to twelve of the twelve vent lines venting the IC to the suppression pool: Each isolation condenser is equipped with a top header ¾ inch vent line and a bottom header ¾ inch vent line. The bottom header ¾ inch vent line splits into two parallel vent lines. Each vent line has two, automatic, solenoid-operated valves in series making a total of six solenoid operated valves. All valves fail closed.

   The bottom header vent valves are signaled to open by high RPV pressure (above IC actuation value). The spurious opening of just one vent line produces a small LOCA with the steam flowing directly to the suppression pool and eventually leading to a high suppression pool temperature scram. This event is bounded by the IORV.

c. System pipe breaks upstream of the isolation valves: These breaks are not isolable and they produce LOCA events. This event is considered in the frequency evaluation of LOCAs inside containment (refer to Section 2.3).

d. System pipe breaks downstream from the isolation valves:

- Significant break inside the containment: Automatic isolation valves are provided on the suction lines; if a break is sensed (by measuring the flow in the line) these valves close rapidly, so that only a limited loss of RPV water occurs, and no transient initiation is possible. Pipe breaks, combined with failure to isolate the line, have a very low frequency. The consequences are similar to a LOCA inside the containment, but the frequency of the event is much lower. Such events are subsumed by the small steam LOCA initiators.
- IC tube or pipe break outside containment: Automatic or manual break isolation is possible through the multiple leak detection systems, depending on the break size. The IC pipe break outside containment is analyzed as a separate initiating event category.

2.2.3.4 Depressurization and Safety/Relief Valves

Refer to Section 4.1 for a description and simplified diagram of this system.

For this system, only spurious actuation resulting either from operator error or from hardware failure is considered. Breaks of piping between valves and RPV nozzles are taken into account in the LOCA frequency evaluation (see Table 2.2-3).

The spurious actuation of an SRV constitutes an IORV event. An IORV is similar to a small break LOCA, but a specific event tree is used. Its frequency is evaluated in Section 2.3.

The spurious actuation of only one DPV is included in the medium steam LOCA initiating event category. Only the valve mechanical failure contributes to this event because the logic failure causes more than one valve to open, and thus, a different event.

The spurious actuation of two or more SRVs and DPVs is accounted for within the large LOCA category because of the large total flow area through the valves. The main causes for this event are discussed and quantified in Section 2.3.

2.2.3.5 Gravity Driven Cooling System

The GDCS is in standby and is automatically started by low reactor water level or by drywell high temperature. Refer to Section 4.6 for a description and simplified diagram of this system.

The following malfunctions are considered:

(1) Spurious actuation of a squib valve due to Instrumentation and Control failures: The spurious actuation of a squib valve due to Instrumentation and Control failures is equivalent to the total frequency of inadvertent opening of one or more DPVs due to Instrumentation and Control failures which was calculated in DCD 15A.3.9.3 to be 5.75E-04/year and the failure of the in-line check valve to close (1.0E-03/demand) produce an event similar to a GDCS line break. The probability of occurrence is then 5.75E-04 * 1.0E-03 = 5.75E-07/year. This is added to the medium liquid LOCA initiating event frequency.

(2) Spurious actuation of all squib valves: This event can occur for the same reasons as the spurious opening of all DPVs; however, the frequency of this event is lower because, in this case, the check valves on the GDCS lines would need to fail to close. This event is not explicitly considered in the analysis.

(3) Injection line break between the nozzle and the check valve: This event is a LOCA and its frequency is evaluated in Section 2.3.

2.2.3.6 Fuel and Auxiliary Pools Cooling System

This system is normally in operation with one pump providing spent fuel pool cooling during plant operation. Its reconfiguration into the suppression pool cooling mode is automatic upon a high Suppression Pool temperature signal. The operator initiates the LPCI mode manually. Refer to Section 4.7 for a description and simplified diagram of this system.

The following malfunctions are considered:

(1) Trip of the operating pump: Tripping of the operating pump results in an increase of the spent fuel pool water temperature over the long term; however, the standby pump can be aligned. This will not cause a reactor trip. No effect on plant safety is expected.

(2) Spurious startup of the standby pump: The startup of the standby pump has no effect on plant safety. If the related suction valves are not open, then the pump could be damaged due to overtemperature. A reactor trip will not be generated. No effect on plant safety is expected.

(3) Injection line breaks downstream of the check valves (isolating FAPCS from RWCU line): This line is common to the injection of FAPCS and RWCU train B; because it belongs to RWCU, the consequences of this event are analyzed within RWCU below.

2.2.3.7 Reactor Water Clean-Up System

This system is in operation during normal plant operation with one pump working at a percentage of the rated speed to carry out the reactor water clean-up function. No automatic signals are provided for standby pump actuation or for increase of the operating pump speed. Refer to Section 4.8 for a description and simplified diagram of this system.

The following malfunctions are considered:

(1) Trip of the operating pump: Tripping of the operating pump does not produce an immediate effect on plant safety because only the RWCU function is impaired. The loss of the RWCU system for a long time might result in high water conductivity and require a controlled plant shutdown. However, there is sufficient time available to correct high water conductivity using the standby pump or other functions, and this malfunction is not included as an initiator.

(2) Spurious startup of the standby pump: No effect on plant safety is expected. No reactor trip will be generated if there is a spurious startup of the standby pump, not included as an initiator.

(3) Injection line break downstream from the check valves (isolating RWCU from FW line): This event causes loss of FW because of the diverted flow, unavailability of the RWCU trains due to high temperature in the Main Steam Tunnel, loss of the FAPCS in half of the cases (FW line A break), and loss of the CRD injection in half of the cases (FW line B break). The event behaves like a loss of FW with the unavailability of some mitigating systems.

(4) RWCU line break outside of containment within Reactor Building (other than 3 above): This event causes loss of RWCU/SDC because of the LD&IS isolating signals, RPV Level 2 containment IS isolating signal, and unavailability of both RWCU/SDC trains.

Breaks in the RWCU return line from the RPV lead to LOCA events. The frequency of such an event is evaluated in Section 2.3.

2.2.3.8 Standby Liquid Control System

The system is in standby during normal plant operation and is automatically actuated in response to ATWS signals. Refer to Section 4.4 for a description and simplified diagram of this system.

The following malfunctions are considered:

(1) Spurious system actuation: If one of the redundant squib valves spuriously actuates, boron injection into the RPV occurs. Plant availability is affected, but not plant safety. This event is considered as part of the General Transient initiating event category.

(2) Injection line break: This event causes a LOCA. Its contribution to LOCA frequency is evaluated in Section 2.3.

2.2.3.9 RCCW System

The Reactor Component Cooling Water System supplies cooling water to the following systems:

- CRD
- FAPCS
- RWCU/SDC
- Chilled Water System, and
- Instrument Air System.

Refer to Section 4.10 for a description and simplified diagram of this system.

The consequences of total failure of RCCW are less severe than total failure of PSWS because BOP systems cooled by TCCW/PSWS remain available for the loss of RCCW event. However, because the consequences of this event are bounded by the Complete Loss of PSWS initiating event category, loss of RCCW is conservatively grouped together with the Complete Loss of PSWS initiator.

2.2.3.10 TCCW System

Refer to Section 4.9 for a description and simplified diagram of this system.

The loss of this system results in loss of balance-of-plant (BOP) systems (i.e., a turbine trip with feedwater and condensate system failure). Given the functional consequences of this event, loss of TCCW is grouped together with the Loss of Feedwater initiating event category.

2.2.3.11 Plant Service Water System

Refer to Section 4.11 for a description and simplified diagram of this system.

The loss of this system produces a loss of RCCW and TCCW. The event results in the impairment of several mitigating systems (all non-safety systems, including complete loss of AC power to battery chargers). The consequences are similar to a station blackout event. This event is modeled with a separate initiating event category.

For partial loss of Service Water system, see Note 3 to Table 2.2-1.

2.2.3.12 Air Systems

This event consists of the simultaneous loss of HPNSS and both the instrument and service air systems. Refer to Sections 4.12 and 4.13 for descriptions and simplified diagrams of these systems. As a result of complete loss of air, the MSIVs will eventually close because of the discharge of the accumulators due to normal leakage. After the accumulators discharge, the event proceeds like an MSIV closure event. This event is modeled with a separate initiating event category.

2.2.3.13 Loss of a Single 13.8 kVAC or 6.9 kVAC Safeguards Bus

Refer to Section 4.14 for a description and simplified diagram of the AC system.

In accordance with DCD chapter 8.1.5.2.2.1, there is no direct Class 1E AC power source required for safety related loads. Because there are no 13.8 kVAC or 6.9 kVAC vital buses they are not included as an initiator at ESBWR. Nonsafety-related Bus loss (QC5) is discussed in general transients and as discussed in Subsection 2.3.1.1does not cause a reactor scram.

2.2.3.14 Loss of a Single 250 VDC Bus

Refer to Section 4.17 for a description and simplified diagram of the 250 DC system.

The Class 1E 250 VDC power distribution system provides four independent and redundant on-site sources of power for operation of DC loads. Because of the N - 2 design of the ESBWR, a single 250 VDC Bus can be taken out of service without shutting down the plant. Therefore, loss of a single 250 VDC Bus will not cause a plant shutdown and is not considered as a credible initiating event for the ESBWR.

2.2.3.15 Reactor Water Level Instrumentation Failures

Failures of the RWLI system can initiate a transient event and at the same time jeopardize the operation of mitigating systems. The failures considered are those involving an erroneous high level signal.

The experience of operating plants shows that an RWLI failure could be caused by leaks in the instrumentation leg or by extreme environmental conditions, such as a high drywell temperature. The following discussion examines the effect of such phenomena, assuming the occurrence of a single leg failure.

(1) Leaks: Leaks occurring in the instrumentation legs could alter the level, inducing an error in all of the connecting instruments. The ESBWR plant design incorporates one separate leg for each level instrumentation division; this means that if only one leg fails, the RWLI system can still assure correct performance (i.e., one scram signal is not generated and mitigating systems are not prevented from starting).

(2) Extreme environmental conditions: A high DW temperature could be caused by accidents such as 1) LOCAs, 2) inadvertent opening of one DPV, or 3) loss of the drywell cooling system. However, the instrumentation is assumed to be designed for the maximum temperature attainable in the DW, so no instrumentation effects should occur. A possible local effect could be generated by a LOCA if the break is close to the instrumentation piping. For this case, only one division is expected to be affected, so no significant contribution to core damage frequency is expected. Various level sensors actuate the safety systems (IC, GDCS, SRV).

Given the above, RWLI failures are not included as an initiator for accident sequence quantification.

2.2.3.16 Drywell Cooling System

A scram is initiated when the drywell temperature increases above the predetermined limit. No mitigating systems are impaired. The event is similar to a spurious trip and is included in the General Transient initiating event category.

2.2.3.17 Summary of Special Initiators

The following are the special initiators that require no further consideration for initiating events:

- Control Rod Drive System
- Feedwater/Condensate
- Isolation Condensers
- Depressurization and Safety/Relief Valves
- Gravity Driven Cooling system
- Fuel and Auxiliary Pools Cooling System
- Reactor Water Clean-up System

- Standby Liquid Control System
- Complete Loss of RCCW
- Complete Loss of TCCW System
- Loss of a Single 13.8 kVAC or 6.9 kVAC Safeguards Bus
- Loss of a Single 250 VDC Bus
- Reactor Water Level Instrumentation Failures
- Loss of Drywell Cooling System.

The following are special initiators that are treated as separate initiating event categories:

- Complete Loss of Plant Service Water System
- Complete Loss of Air Systems.

2.2.4 References

2.2-1 Idaho National Engineering and Environmental Laboratory, "Rates of Initiating Events at U. S. Nuclear Power Plants: 1987-1995", NUREG/CR-5750, February 1999.

2.2-2 Electric Power Research Institute, "EPRI-ALWR Utility Requirements Document", EPRI ALWR URD, Revision 4, April 1992.

2.2-3 Idaho National Engineering Laboratory, "Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments", NUREG/CR-3862, May 1985.

Table 2.2-1

NUREG/CR-5750 Initiating Event Categories

A	(Reserved)[8]	
B	Loss of Offsite Power	
	(B1)	Loss of Offsite Power (LOSP)
C	Loss of Safety-Related Bus	
	(C1)	Loss of Vital Medium Voltage AC Bus[2]
	(C2)	Loss of Vital Low Voltage AC Bus[2]
	(C3)	Loss of Vital DC Bus
D	Loss of Instrument or Control Air	
	(D1)	Loss of Instrument or Control Air
E	Loss of Safety-Related Cooling Water	
	(E1)	Total Loss of Service Water
	(E2)	Partial Loss of Service Water[3]
F	Steam Generator Tube Rupture	
	(F1)	Steam Generator Tube Rupture[1]
G	Loss of Coolant Accident (LOCA)/Leak	
	(G1)	Very Small LOCA/Leak
	(G2)	Stuck Open: 1 Safety/Relief Valve
	(G3)	Small Pipe Break LOCA
	(G4)	Stuck Open: Pressurizer PORV[1]
	(G5)	Stuck Open: 2 or More Safety/Relief Valves
	(G6)	Medium Pipe Break LOCA
	(G7)	Large Pipe Break LOCA
	(G8)	Reactor Coolant Pump Seal LOCA: PWR[1]
H	Fire	
	(H1)	Fire
J	Flood	
	(J1)	Flood
K	High Energy Line Break	

Table 2.2-1

NUREG/CR-5750 Initiating Event Categories

	(K1)	Steam Line Break Outside Containment
	(K2)	Feedwater Line Break
	(K3)	Steam Line Break Inside Containment: PWR[1]
L	Total Loss of Condenser Heat Sink	
	(L1)	Inadvertent Closure of All MSIVs
	(L2)	Loss of Condenser Vacuum
	(L3)	Turbine Bypass Unavailable
M	(Reserved)[8]	
N	Interfacing System LOCA	
	(N1)	Interfacing System LOCA
P	Total Loss of Feedwater Flow	
	(P1)	Total Loss of Feedwater Flow
Q	General Transients	
	(QC4)	Loss of AC Instrumentation and Control Bus
	(QC5)	Loss of Non-safety-related Bus
	(QG9)	Primary System Leak
	(QG10)	Inadvertent Open/Close: 1 Safety/Relief Valve
	(QK4)	Steam or Feed Leakage
	(QL4)	Loss of Nonsafety-Related Cooling Water
	(QL5)	Partial Closure of MSIVs[4]
	(QL6)	Condenser Leakage
	(QP2)	Partial Loss of Feedwater Flow[5]
	(QP3)	Total Loss of Condensate Flow
	(QP4)	Partial Loss of Condensate Flow[6]
	(QP5)	Excessive Feedwater Flow
	(QR0)	RCS High Pressure (RPS Trip)
	(QR1)	RCS Low Pressure (RPS Trip): PWR[1]
	(QR2)	Loss of Primary Flow (RPS Trip): PWR[1]
	(QR3)	Reactivity Control Imbalance
	(QR4)	Core Power Excursion (RPS Trip)

Table 2.2-1

NUREG/CR-5750 Initiating Event Categories

(QR5) Turbine Trip (QR6) Manual Reactor Trip (QR7) Other Reactor Trip (Valid RPS Trip) (QR8) Spurious Reactor Trip (QR9) Spurious Engineered Safety Feature Actuation[7]

Notes to Table 2.2-1:

1 PWR event. Not applicable to ESBWR.
2 These event categories are not applicable to the ESBWR, as the Low and Medium Voltage AC Buses in the ESBWR design are nonsafety-related (with the exception of a portion of the uninterruptible power supply system that is backed up by batteries). Low and Medium voltage AC initiators for the ESBWR are addressed by the NUREG/CR-5750 category QC5 (Loss of Non-safety-Related bus).
3 In accordance with DCD, 9.2.1.2 and 9.2.1.5, loss of a single PSWS train does not cause a reactor trip for the ESBWR design.
4 In accordance with DCD, 9A.6.4.5 and 15.2.2.6.2, closure of one MSIV does not cause a reactor trip for the ESBWR design.
5 In accordance with DCD, 10.4.7.1.2, 10.4.7.2.1, 14.2.8.2.24, Table 6.3-1 and 7.7.5.2.3, loss of a single feedwater pump train does not cause a reactor trip for the ESBWR design.
6 In accordance with DCD, 10.4.7.1.2 and 10.4.7.2.1, loss of a single condensate pump train does not cause a reactor trip for the ESBWR design.
7 The ESBWR design does not have high pressure standby safety injection systems, and the ESBWR low pressure safety systems cannot inject into the RPV when it is at power. Spurious IC actuation is considered under Special Initiators, and spurious ADS are considered in the LOCA initiator categories.
8 “Reserved” is used for this category in NUREG/CR-5750 and does not mean it has been reserved for ESBWR.

Table 2.2-2

ESBWR Transient Initiating Event Categorization

ESBWR Transient Initiating Category	**Description**	**NUREG/CR-5750 Categories**
General Transient	Scram occurs. FW remains available or is promptly recoverable. Main condenser is available, if the mode switch is positioned in shutdown.	a. (QC4) Loss of ac Instrumentation and Control bus b. (QC5) Loss of Nonsafety-related bus c. (QG9) Primary System Leak d. (QK4) Steam or Feed Leakage e. (QL6) Condenser Leakage f. (QP5) Excessive Feedwater Flow g. (QR0) RCS High Pressure (RPS Trip) h. (QR3) Reactivity Control Imbalance i. (QR4) Core Power Excursion (RPS Trip) j. (QR5) Turbine Trip k. (QR6) Manual Reactor Trip l. (QR7) Other Reactor Trip (Valid RPS Trip) m. (QR8) Spurious Reactor Trip
Transient with PCS Unavailable	Scram occurs. FW remains available for RPV inventory makeup. Main condenser is unavailable.	a. (L1) Inadvertent Closure of All MSIVs b. (L2) Loss of Condenser Vacuum c. (L3) Turbine Bypass Unavailable
Loss of Feedwater	Scram occurs. Feedwater is not available. Main condenser is unavailable due to MSIVs closure on Level 2.	a. (P1) Total Loss of Feedwater Flow b. (QL4) Total Loss of TCCW (subset of QL4) c. (QP3) Total Loss of Condensate Flow
IORV	Scram occurs. FW remains available. One or more safety/relief valves initially stuck open	a. (G2) Stuck Open: 1 Safety/Relief Valve b. (QG10) Inadvertent Open/Close: 1 Safety/ Relief Valve

Table 2.2-3

Lines Connected to Reactor Coolant Pressure Boundary (RCPB)

Line	Number of lines	mm (inches) diameter	Nozzle Elevation (mm)	Break[7,] Type	Notes
Main Steam (MSL) (N3)	4	700 (28)[1]	22840	L	Steam break
DPV/IC (N5)	4	450 (18), 350 (14)[2]	21910	L	Steam break
DPV/MSL [9]	4	N/A[3]	22840	M or L[6]	Steam break
SRV/MSL[9]	18	N/A	22840	M or L[6]	Steam break
FW-12" (N4)	6	300 (12)	18915	L	Steam break
22"	2	550 (22)	__	L	Steam break
RWCU/SDC (N8)	2	300 (12)	17215	L[8]	Steam break
IC return lines	4	200 (8)	13025	M	Liquid break
GDCS (N6)	8	150 (6) [4]	10453	M	Liquid break
Equalizing lines	4	150 (6) [5]	8453	M	Liquid break
RWCU/RPV Drain Lines (N15)	4	50 (2)	0	M	Liquid break
SLCS (N16)	2	50 (2)		M	Liquid break
Instrument lines above L3	4	50 (2)	>L3	S	Steam break
Instrument lines below TAF	4+2	25 (1)	<TAF	S	Liquid break
Instrument lines above TAF and below L3	4	25 (1)	TAF-L3	S	Liquid break

Notes:
1 9.75 10-2 m2 (1.05 sq. ft) throat diameter
2 450 mm common pipe, 350 mm IC branch pipe
3 DPV directly mounted on MSL pipe
4 75 mm (3 inches) throat diameter
5 50 mm (2 inches) throat diameter
6 Break size depends upon whether single or multiple valves spuriously open (single DPV or SRV is Medium; multiple DPVs or SRVs is Large)
7 L = Large; M = Medium; S = Small
8 In accordance with thermal hydraulic analysis the RWCU/SDC responds like a large steam break
9 Use same elevation on main steam lines because penetration is branch off main steam line.

2.3 INITIATING EVENTS FREQUENCY QUANTIFICATION

The initiating events are identified and grouped into initiating event categories as discussed in Section 2.2.

This section documents the frequencies calculated for each of the initiating event categories. Loss of coolant accidents (LOCAs) are divided into further subcategories for more effective event tree modeling and quantification.

2.3.1 Transients

All transient frequencies are determined based on NUREG/CR-5750 (Ref. 2.3-1) results, which are based on U.S. nuclear power plant operational experience. If the 2005 data in NUREG/CR-5750 is available it is used instead of the Ref. 2.3-1 data. Where NUREG/CR-5750 does not provide information on a particular initiator, the source of the frequency for the initiator is provided in the appropriate section. The resulting frequencies are summarized in Table 2.3-4 and the details of the calculations are presented below.

2.3.1.1 General Transient

The nominal frequency for BWR General Transient is 1.5 /yr. However, several adjustments are required to this number for the ESBWR. In accordance with Table 2.2-1, closure of a single MSIV does not cause a reactor trip for the ESBWR. Loss of two or three MSIVs out of eight in which the failed closed MSIVs are not on the same line is not considered credible since there were 11 events for partial closure of MSIVs, QL5, for BWRs included in the Q events for BWRs. The common cause failure for two or three of four MSIVs would not be credible given there were only 11 partial closures of MSIVs in the database. In accordance with Table 2.2-1, loss of a single feedwater train does not cause a reactor trip for the ESBWR design. There were 45 events for partial loss of feedwater, QP2, for BWRs included in the Q events for BWRs. In accordance with Table 2.2-1, loss of a single train of condensate does not cause a reactor trip for the ESBWR design. There were 13 events for partial loss of condensate, QP4, for BWRs included in the Q events for BWRs. The loss of two or three condensate trains is not considered credible since there are four independent power trains supplying the four condensate pumps and common cause would not be a significant contributor given there were only 13 partial losses of condensate. In accordance with Table 2.2-1, note 7, spurious Engineered Safety Feature Actuation either does not cause a reactor trip or has been included under other initiators and therefore does not get counted as a contributor to Q for the ESBWR PRA. There were 14 events for spurious Engineered Safety Feature Actuation, QR9, for BWRs included in the Q events for BWRs. The partial loss of TCCW would not cause a reactor trip in the ESBWR design since loss of one feedwater train or loss of one condensate train will not cause a reactor trip in the ESBWR design. Loss of two or three trains out of four trains is not considered credible since all feedwater pumps have independent power trains and common cause of two or three out of four given there was only 45 events.

There were 16 events for partial and total loss of non-safety related cooling, QL4. Finally, loss of a nonsafety-related bus, QC5, will not cause a reactor trip in the ESBWR design because loss of one feedwater train, loss of one condensate train, or partial loss of PSW will not trip the unit. To summarize, the following categories will be removed from the Q events:

- Partial closure of MSIVs (QL5) – 11 events,
- Partial loss of feedwater (QP2) – 45 events,
- Partial loss of condensate (QP4) – 13 events,
- Spurious Engineered Safety Feature Actuation (QR9) – 14 events,
- Loss of TCCW (QL4) – 16 events (see Loss of Feedwater also), and
- Loss of Nonsafety-related Bus (QC5) – 5 events.

Additional adjustments required to Q for the ESBWR are removal of the Q contributors that are included in other ESBWR PRA transient initiating event categories. As can be seen from Subsections 2.3.1.3 and 2.3.1.4, these are:

- Total Loss of Condensate Flow (QP3) – 5 events (Loss of Feedwater),
- Total Loss of TCCW (subset of QL4) – 8 events (0.5 * 16 events) (Loss of Feedwater), and
- Inadvertent Open/Close: 1 Safety/Relief Valve (QG10) – no events (IORV).

Summing all of the Q events for BWRs from Table D-4, pages D-5 and D-6 of Ref. 2.3-1, the total number of Q events for BWRs was 541. The total events to be subtracted from this for the ESBWR design are 117 from above. The calculated value of Q to be used for the ESBWR PRA initiating event is then:

$$Q = (541-117)/(541) * 1.5/\text{yr}$$

$$Q = 1.18/\text{yr}$$

2.3.1.2 Transient With PCS Unavailable

The frequency for BWR loss of heat sink is 1.97E-1/yr. This is conservatively taken as the frequency of the ESBWR PRA Transient with PCS Unavailable initiating event category.

2.3.1.3 Loss of Feedwater

As shown in Table 2.2-2, the following NUREG/CR-5750 categories are grouped into the ESBWR PRA Loss of Feedwater initiating event category:

- Total Loss of Feedwater Flow (P1),
- Total Loss of Condensate Flow (QP3), and
- Total Loss of TCCW (subset of QL4).

The frequency of the “Total Loss of Feedwater Flow (P1)” initiating event category is 9.59E-2/yr. Appendix A of NUREG/CR-5750 notes that QP3 events are already included in the calculation of the “Total Loss of Feedwater Flow (P1)” frequency (as well as separately in the “General Transient (Q)” frequency Summing all of the Q events for BWRs from Table D-4, the total number of Q events for BWRs was 541. Of these, 5 were QP3 (Total Loss of Condensate Flow) events for BWRs . This means the percentage of QP3 events for BWRs, which contribute to Q events for BWRs, is about 1% (5/541). So, the QP3 event initiator for BWRs is Q (1.5/yr) * 0.01 = 1.5E-02/yr.

NUREG/CR-5750 does not provide a frequency estimate for the QL4 contributor (it is part of the larger NUREG/CR-5750 category “General Transient (Q)”); however, the frequency of QL4 can be estimated from the contribution breakdowns provided in NUREG/CR-5750. QL4 (Loss of nonsafety-related cooling water) events are included in the “General Transient (Q)” frequency. Summing all of the Q events for BWRs from Table D-4, the total number of Q events for BWRs was 541. Of these, 16 were QL4 events for BWRs . The percentage of QL4 events for BWRs that contribute to Q for BWRs is about 3% (16/541). So, the QL4 event initiator for BWRs is Q (1.5/yr) * 0.03 = 4.5E-02/yr. This is the total contribution of both partial and total losses of non-safety related cooling water. Since only the total loss of TCCW will lead to a scram, only the total loss need be considered for an initiating event. Assuming the ratio of total losses of nonsafety-related cooling water to partial and total losses of cooling water is the same as it is for safety related cooling water, in accordance with NUREG/CR-5750, there are 1 total loss of Service Water and 6 partial losses of service water. The split fraction of total losses to partial and total losses is 1/(6+1) = 0.14. Then multiplying the 0.14 times the total for QL4, which from above is 4.5E-2/yr, yields an initiating event frequency for QL4 total losses of 4.5E-02/yr * 0.14 = 6.3E-03/yr.

Therefore, the frequency of the ESBWR PRA Loss of Feedwater initiating event category is calculated as follows:

$$9.59\text{E-}2/\text{yr} + 1.5\text{E-}2/\text{yr} + 6.3\text{E-}3/\text{yr} = 1.17\text{E-}1/\text{yr}$$

2.3.1.4 IORV

As shown in Table 2.2-2, the ESBWR PRA IORV initiating event category is comprised of the following contributors:

- Stuck Open: 1 Safety/Relief Valve (G2),
- Inadvertent Open/Close: 1 Safety/Relief Valve (QG10), and
- Spurious opening of two vent valves in series from IC to suppression pool. (This item does not have a NUREG corresponding category, however, its frequency is discussed below and added to the total IORV frequency).

The frequency of "BWR Stuck Open: SRV" initiating event category is 2.23E-2/yr.

There were no occurrences of Inadvertent Open/Close: 1 Safety/Relief Valve (QG10) for BWRs. This would make the QG10 frequency contribution a non-significant contributor and will not be evaluated further.

There are three vent lines from each IC to the suppression pool. There are four IC condensers. Each vent line contains two normally closed vent valves in series. Both valves in each line would have to spuriously open to provide a path from the isolation condenser to the suppression pool. A typical SOV spurious operation rate is 5.7E-07/hr. Multiplying by the number of hours in a year, 8760, provides an annual spurious SOV rate of 5.0E-03/year. Independent spurious operation of two solenoid valves in series would be a negligible contributor. The common cause failure for two out of two solenoid valves failing would be the failure rate times beta where the beta factor is assumed to be 0.1. The annual common cause failure rate for two out of two of the solenoid vent valves would be 5.0E-03/year * 0.1 = 5.0E-04/year. There are a total of 12 lines resulting in an annual initiating event frequency for vent line open from IC to suppression pool of 5.0E-04/year * 12 = 6.0E-03/yr. This will be added to the Stuck Open 1 Safety/Relief Valve frequency of 2.23E-02/yr to provide the PRA IORV initiating event frequency of 2.83E-02/yr.

Therefore, the frequency of the ESBWR PRA IORV initiating event category is estimated at 2.83E-02/yr.

2.3.2 Loss of Preferred Power (LOPP)

The frequency of the Loss of Preferred Power (LOPP) initiating event category is based on the "Loss of Offsite Power (B1)" initiating event category. The frequency for Loss of Offsite Power (B1) is 3.59E-02/yr. The frequency for Loss of Offsite Power in NUREG/CR-5750 is based on the frequency reported in NUREG/CR-6890 (Ref. 2.3-2). This was further subdivided in NUREG/CR-6890 into the following LOOP categories in the critical operation mode:

- Plant Centered – 2.07E-03/rcry
- Switchyard Centered – 1.04E-02/rcry
- Grid Related – 1.86E-02/rcry
- Weather Related – 4.83E-03/rcry

Where the frequency units, /rcry, is per reactor critical years.

While NUREG/CR-6890 did address special topics such as seasonal effects on frequencies, consequential LOOPs, and modeling of sites with more than one plant, ESBWR uses the basic categories for Loss of Offsite Power. The following frequencies will be used for the Loss of Preferred Power (LOPP) initiation event subcategories:

- Plant Centered Loss of Preferred Power – 2.07E-03/rcry
- Switchyard Centered Loss of Preferred Power – 1.04E-02/rcry
- Grid Related Loss of Preferred Power – 1.86E-02/rcry
- Weather Related Loss of Preferred Power – 4.83E-03/rcry

2.3.3 LOCAs

The LOCA initiators considered in the ESBWR Probabilistic Risk Assessment are:

- Breaks of the reactor coolant pressure boundary (RCPB) during normal operation,

- Breaks of pipes with design pressure lower than the reactor pressure due to the malfunction of the interfacing valves (interfacing breaks), and
- Vessel rupture.

2.3.3.1 LOCAs Inside Containment

The frequencies of LOCAs inside containment are quantified using of the NUREG/CR-5750 mean frequencies (based on BWRs operational experience) for large, medium and small pipe breaks. A very small LOCA (G1) is defined as a pipe break or component failure that results in a loss of primary coolant between 10 to 100 gpm (6.3E-4 to 6.3E-3 $m^3 \cdot s^{-1}$). It is assumed that the condensate/feedwater makeup capacity is at least 100 gpm (6.3E-3 $m^3 \cdot s^{-1}$) . A reactor trip would not be required for a very small LOCA. For BWRs, a very small LOCA has traditionally been a recirculation pump seal leak. ESBWR does not have recirculation pumps, thus further reducing the probability of a very small LOCA. Therefore, a very small LOCA will not be considered an initiator for ESBWR. For each group of ESBWR lines, the associated NUREG/CR-5750 LOCA frequency is apportioned proportionally using three different methods:

- As a function of line length,
- As a function of the number of lines, and
- As a function of the number of line segments.

The results from the three different methods are averaged to determine the final frequency used in this analysis.

Table 2.3-1 summarizes the break frequency calculations for each group of lines. Table 2.3-1 summarizes for each pipe group: the number of lines, the number of sections (assessed on the basis of layout drawings), the frequency apportionments, and the final averaged frequencies. These data are binned into the LOCA initiator classes, as summarized in Table 2.3-2. Note that contributions from events other than pipe breaks are also included in some of the LOCA categories.

Additional LOCA Contributions

In addition to pipe breaks, the following additional events contribute to the LOCA frequencies:

	Event	Contribution To	Frequency
a.	Spurious actuation of one DPV	Medium Steam LOCA	6.19E-4
b.	Spurious actuation of two or more SRVs/DPVs	Large LOCA	3.2E-4
c.	Spurious actuation of one GDCS Squib Valve and failure of check valve	Medium Liquid LOCA	5.75E-7

Spurious actuation of a single SRV is addressed as a separate initiating event (IORV). Refer to Subsection 2.3.1.4.

For frequency of spurious actuation of one GDCS Squib valve and check valve see Section 2.2.3.5.

The first two additional LOCA contributors can result from the following causes:

- Valve(s) mechanical failure
- Operator error
- Spurious actuation

Chapter 15A.3.9.2.1.2 of the DCD examined the possibility of an operator error leading to the inadvertent opening of a DPV and concluded that operator errors were an insignificant contributor when compared to the probability of a spurious actuation signal. Operator errors are judged to be a non-significant contribution. The other causes contribute as follows.

(1) Opening of only one DPV:

There are two causes which contribute to this event:

- Spurious opening
- Shear disk rupture

The DPV spurious opening frequency was calculated in DCD Chapter 15A.3.9.3 as 5.75E-04/year. The check valve internal rupture (shear disk rupture) failure rate is assumed; considering a verification of the integrity of the nozzle is performed at every refueling, the annual failure rate is 4.4E-5, based on data in Appendix A of Chapter 1 of the EPRI ALWR URD (Ref. 2.3-3).

Summing these two values results in the frequency of 6.19E-4 events/year for only one of eight DPVs spuriously opening. This is an additional contributor to the medium steam break LOCA (no FW or GDCS line break) initiator category.

(2) Opening of two or more SRVs/DPVs:

Failure of the logic common to the other valves contributes to this event. Mechanical failure of all valves can be postulated, but is statistically a non-significant contributor.

The operational experience of BWRs indicates that no events of more than one SRV stuck open have been observed. NUREG/CR-5750 estimates frequency of 3.2E-4 events/year for this initiating event (NUREG/CR-5750 category G5). This is an additional contributor to the large steam break LOCA (no FW line break) initiator category.

2.3.3.2 Breaks Outside Containment

In addition to loss of coolant accidents inside containment, lines outside containment that are exposed to the high pressure of the reactor coolant pressure boundary could lead to a LOCA if they experience a break. The high pressure lines where this event can occur are:

- Main steam
- Feedwater

- RWCU
- Isolation condenser (outside containment).

The largest pipes are the main steam lines. These lines isolate automatically if a break occurs downstream from the MSIVs. The frequency of a high energy line break (K) is 1.3E-02/yr. This is conservative since it includes pipe sizes for which the top event success criteria in the event tree are conservative. The frequency for a feedwater line break is 3.4E-03/yr. The frequency for a RWCU break outside containment is assumed to be the same as that for a feedwater line break and is 3.4E-03yr. The frequency of a main steam line break is the frequency of a high energy line break minus the frequencies of feedwater line breaks and RWCU line breaks. The frequency for a steam break/leak outside containment is 6.2E-03/yr. This frequency applies to experiencing the steam line break. Isolation of the break is addressed in the accident sequence analysis.

The feedwater pipes isolate automatically if a break occurs upstream from the containment isolation valves, but result in the unavailability of those systems/trains that inject through these lines. The frequency for this initiating event (K2) is 3.4E-03/yr. This frequency applies to experiencing the FW line break; isolation of the break is addressed in the accident sequence analysis. The initiating event frequency for a pipe break in the A feedwater train would be half of this or 1.70E-03/yr. Likewise, the initiating event frequency for a pipe break in the B feedwater train would be 1.70E-03/yr. The RWCU pipes isolate automatically if a break occurs downstream from the containment isolation valves, but result in the unavailability of the train affected by the break. The frequency for this initiating event is assumed to be similar to the frequency of the feedwater lines. The frequency for an RWCU line break outside containment is 3.40E-03/yr.

The IC pipes outside the containment isolate automatically if a break occurs downstream from the IC isolation valves, but result in the unavailability of the IC affected by the break. Because a majority of the ICS piping at RPV pressure is inside containment, the percentage of ICS piping outside containment is estimated to be 10%. An estimate of the frequency for this initiating event is 10% of the LOCA frequency apportioned for IC outside the containment in Table 2.3-1 and is 10% of the sum of DPV line/IC (7.59E-06/yr) and IC return lines (7.66E-06/yr) or 1.53E-06. This frequency applies to experiencing the line break; isolation of the break is addressed in the accident sequence analysis.

2.3.3.3 Vessel Rupture

Pressurized Thermal Shock (PTS) is one possible vessel failure mode of interest and for bounding the possibility of BWR vessel ruptures, PWRs will be used since the operating pressures in BWRs are far lower than in PWRs. The probability of PTS in a pressurized water reactor with a forged vessel with a 60 year life in accordance with NUREG-1806 (Ref. 2.3-3) is between 10^{-14} and 10^{-9} per reactor year.

Two additional observations are necessary. First, the potential of vessel rupture from PTS in a BWR is generally accepted as being substantially less than for a PWR. The fact that BWRs operate at a lower pressure reduces the hoop stress and the design of the vessel allows natural circulation, which reduces thermal stresses during overcooling transients. Second, improved materials, the absence of nozzles and welds at the core level, and the improved in-service inspection program all justify a lower value for the ESBWR.

Other failure modes might exist such as inadvertent overpressurization, weld failures, multiple control rod mechanism failures, among others. Plant design (e.g., relief valve capacity), periodic vessel inspections and primary system leakage monitoring help ensure that catastrophic failure of the vessel is an extremely unlikely event. Based on the above considerations, the frequency for a vessel rupture is 1.00E-10/year.

2.3.3.4 Interfacing Systems LOCA

No value is provided for ISLOCA in NUREG/CR-5750. Therefore, an ISLOCA evaluation has been performed which identifies two possible ISLOCA paths. The evaluation uses industry guidance from NUREG/CR-5744 (Ref. 2.3-4) and NUREG/CR-5102 (Ref. 2.3-5). One potential ISLOCA path is the main steam line drains from upstream of the MSIVs inside containment. The evaluation provides a probability of occurrence for this ISLOCA path of 1.75E-9/year. Because of the low probability of this event, it is not considered further. The other ISLOCA path that is evaluated is the FAPCS to RWCU/SDC piping. The evaluation provides a probability of occurrence for this event of 6.42E-8/year. The location of the FAPCS to RWCU/SDC piping interfacing system LOCA would generate a plant response similar to a Train A feedwater line break outside containment. Because the initiating event frequency for a Train A feedwater line break outside containment is significantly larger than a FAPCS to RWCU/SDC ISLOCA frequency, no adjustment to the Train A feedwater line break outside containment is required.

2.3.4 Special Initiators

As discussed in Section 2.2.3.17, the only systemic malfunctions maintained as separate special initiators are a Complete Loss of PSWS and Complete Loss of Air Systems. All other systemic malfunctions are either included in other existing initiator categories or do not cause a plant trip.

Complete Loss of PSWS

The ESBWR Plant Service Water Systems (PSWS) is designed with a high level of redundancy and diversity. Reactor Component Cooling (RCCW), which is grouped within the Complete Loss of PSWS initiator category (refer to Subsection 2.2.3.9), also has a design with a high level of redundancy considering the normal operational requirements (two out of six pumps and heat exchangers).

Given the design of these ESBWR systems, the frequency based on NUREG/CR-5750 is 9.7E-04/yr.

Complete Loss of Air System

The ESBWR Instrument Air System is designed with a high level of redundancy (two 100% capacity trains in parallel) and is supported by the Service Air System (SAS) compressors, which also consists of two identical trains in parallel. Air receivers and an air receiver/surge tank are provided to maintain air supply pressure if all of the IAS and SAS compressors fail. The High Pressure Nitrogen Supply System (HPNSS) provides distribution piping from the CIS to the nitrogen loads in containment including the inboard MSIVs. The HPNSS backup consists of two bottle-rack trains in parallel.

Given the design of these ESBWR systems, the frequency for the Complete Loss of Air Systems initiating event category for ESBWR will be 1.02E-02/yr (Ref. 2.3-6).

2.3.5 Summary of Initiating Event Frequencies

Table 2.3-3 summarizes the frequencies of the internal events initiators used in the event tree quantification.

The initiating event frequencies developed in this analysis are based on BWR historical experience. The total initiating event frequency estimated in this analysis is judged conservative because it is greater than the design goal of less than 1 event/year, to be demonstrated in a later stage for the ESBWR by the Reliability Availability Maintenance (RAM) program.

Table 2.3-4 provides a comparison of the ESBWR PRA internal events initiator frequencies to the following industry studies:

- EPRI ALWR Utility Requirements Document (Ref. 2.3-7),
- NUREG/CR-5750 (Ref. 2.3-1), and
- NUREG/CR-6928 (Ref. 2.3-6)

2.3.6 References

2.3-1 Idaho National Engineering and Environmental Laboratory, "Rates of Initiating Events at U. S. Nuclear Power Plants: 1987-1995", NUREG/CR-5750, February 1999.

2.3-2 Idaho National Laboratory, "Reevaluation of Station Blackout Risk at Nuclear Power Plants Analysis of Loss of Offsite Power Events: 1986-2004", NUREG/CR-6890, Vol. 1, December 2005.

2.3-3 U. S. Nuclear Regulatory Commission, "Technical Basis for Revision Of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)", NUREG/CR-1806, Manuscript Completed May 24, 2006.

2.3-4 Idaho National Engineering Laboratory, "Assessment of ISLOCA Risk-Methodology and Application to a Westinghouse Four-Loop Ice Condenser Plant", NUREG/CR-5744, April 1992.

2.3-5 Brookhaven National Laboratory, "Interfacing Systems LOCA: Pressurized Water Reactors", NUREG/CR-5102, February 1989.

2.3-6 Idaho National Laboratory, "Industry-Average Performance for Components and Initiating Events at Commercial Nuclear Power Plants, NUREG/CR-6928, February 2007.

2.3-7 Electric Power Research Institute, "EPRI-ALWR Utility Requirements Document", EPRI ALWR URD, Revision 4, April 1992.

Table 2.3-1

Summary of Lines Connected to RPV And Quantification Of Frequency Apportionment

ID	Line	Length (m)	No. Lines	No. Sections	Apportionment Method: By Length[1]	By No. Lines[2]	By No. Sections[3]	Final Frequency[4] (events/yr)	Notes
a	Main Steam (MSL)	92	4	44	9.32E-6	6.67E-06	6.20E-6	7.40E-6	Large Steam break
b	DPV line/IC	70	4	64	7.09E-6	6.67E-6	9.01E-6	7.59E-6	Large Steam break
c	FW	95	8	74	9.63E-6	1.33E-5	1.04E-5	1.11E-5	Large Steam break
d	RWCU/SDC	39	2	31	3.95E-6	3.33E-6	4.37E-6	3.88E-6	Large Steam break
e	IC return lines	12	4	40	3.78E-6	7.27E06	1.19E-5	7.66E-6	Medium Liquid break
f	GDCS	44	8	32	1.39E-5	1.45E-5	9.55E-6	1.27E-5	Medium Liquid break
f1	Equalizing lines	11	4	12	3.46E-6	7.27E-6	3.58E-6	4.77E-6	Medium Liquid break
g	RWCU/RPV Drain Lines	20	4	20	6.30E-6	7.27E-6	5.97E-6	6.51E-6	Medium Liquid break
h	SLCS	40	2	30	1.26E-5	3.64E-6	8.96E-06	8.40E-6	Medium Liquid break
i	Instrument lines above L3	50	4	100	1.43E-4	1.43E-4	1.43E-4	1.43E-4	Small Steam break
i1	Instrument lines below TAF	75	6	150	2.14E-4	2.14E-4	2.14E-4	2.14E-4	Small Liquid break

Table 2.3-1

Summary of Lines Connected to RPV And Quantification Of Frequency Apportionment

i2	Instrument lines above TAF and under L3	50	4	100	1.43E-4	1.43E-4	1.43E-4	1.43E-4	Small Liquid break

Notes to Table 2.3-1:

1 Calculated as: [(length of line) / (total length of all lines corresponding to LOCA size category)] x 3E-5 per year Large Pipe Break LOCA size category frequency

2 Calculated as: [(number of individual lines) / (total number of lines corresponding to LOCA size category)] x 4E-5 per year Medium Pipe Break LOCA size category frequency

3 Calculated as: [(number of sections) / (total number of sections corresponding to LOCA size category)] x 5E-4 per year Small Pipe Break LOCA size category frequency

4 Final frequency is the average of the three apportionment methods.

Table 2.3-2

Attributions of Line Frequencies to LOCA Categories

LOCA Category	Contributors*	Frequency (events/yr)
Large steam LOCA (no FW line break)	a + b + d + (A)	3.39E-4
Large steam LOCA in FW line	c	1.11E-5
Medium liquid LOCA	e + f + f1 + h + (C) + g	4.06E-5
Medium/small steam LOCA	i + (B)	7.62E-4
Small liquid LOCA	il + i2	3.57E-4
Main Steam Line Break (Outside Containment)	--	6.200E-3
Feedwater Line Break (Outside Containment)	--	3.40E-3
RWCU Line Break (Outside Containment)	--	3.40E-3
IC Line Break (Outside Containment)	--	1.53E-6
ISLOCA	--	6.42E-8

*The additional LOCA contributors are as follows:

(A) Spurious actuation of two or more SRVs and DPVs (3.2E-4/yr)

(B) Spurious actuation of a single DPV (6.19E-4/yr)

(C) Spurious actuation of a single GDSC squib valve (5.75E-7/yr)

Table 2.3-3

Internal Events Initiator Frequencies

Initiating Event	PRA Basic Event ID	Frequency (events/year)
Transients		
General Transient	%T-GEN	1.18
Transient with PCS Unavailable	%T-PCS	1.97E-1
Loss of Feedwater	%T-FDW	1.17E-1
IORV	%T-IORV	2.83E-2
Loss of Preferred Power (LOPP)		
Plant Centered Loss of Preferred Power	%T-LOPP-PC	2.07E-3
Switchyard Centered Loss of Preferred Power	%T-LOPP-SC	1.04E-2
Grid Related Loss of Preferred Power	%T-LOPP-GR	1.86E-2
Weather Related Loss of Preferred Power	%T-LOPP-WR	4.83E-3
LOCAs Inside Containment		
Large Steam LOCA (no FW line break)	%LL-S	3.39E-4
Large Steam LOCA in FW line	%LL-S-FDWA	5.55E-6
	%LL-S-FDWB	5.55E-6
Medium Liquid LOCA	%ML-L	4.06E-5
Medium/Small Steam LOCA	%SL-S	7.62E-4
Small Liquid LOCA	%SL-L	3.57E-4
Vessel Rupture	%RVR	1.00E-10
LOCAs Outside Containment		
Main Steam Line Break (Outside Containment)	%BOC-MS	6.20E-3
Feedwater Line Break (Outside Containment)	%BOC-FDWA	1.70E-3
	%BOC-FDWB	1.70E-3
RWCU Line Break (Outside Containment)	%BOC-RWCU	3.40E-3
IC Line Break (Outside Containment)	%BOC-IC	1.53E-6
ISLOCA	%ISLOCA	6.42E-8
Special Initiators		
Complete Loss of PSWS	%T-SW	9.70E-4
Complete Loss of Air Systems	%T-IA	1.02E-2

Table 2.3-4

Comparison of ESBWR PRA Internal Events Initiating Event Frequencies to Other Studies

Initiating Event	Frequency (per year)			
	ESBWR PRA	EPRI ALWR URD	NUREG/CR-5750	NUREG/CR-6928
Year Published	n/a	1999	2006	2007
Transients				
General Transient	1.18	2.3	8.3E-1	8.3E-1
Transient with PCS Unavailable	1.97E-1	4.9E-1	1.97E-1	n/a
Loss of Feedwater	1.17E-1	3.7E-1	9.59E-2	9.59E-2
IORV	2.83E-2	n/a	n/a	2.23E-2
Loss of Preferred Power (LOPP)	n/a	3.5E-2	3.59E-2	3.59E-2
Plant Centered Loss of Preferred Power	2.07E-3	n/a	n/a	n/a
Switchyard Centered Loss of Preferred Power	1.04E-2	n/a	n/a	n/a
Grid Related Loss of Preferred Power	1.86E-2	n/a	n/a	n/a
Weather Related Loss of Preferred Power	4.83E-3	n/a	n/a	n/a
LOCAs Inside Containment				
Large Steam LOCA (no FW line break)	3.39E-4	5.8E-4	3.0E-5[1]	6.78E-6
Large Steam LOCA in FW line	1.11E-5			
Medium Liquid LOCA	4.06E-5	n/a	4.0E-5[1]	1.04E-4
Medium / Small Steam LOCA	7.62E-4	5.1E-3	5.0E-4[1]	5.0E-4
Small Liquid LOCA	3.57E-4			
Vessel Rupture	1.00E-10	n/a	n/a	n/a
LOCAs Outside Containment				
Main Steam Line Break (Outside Containment)	6.20E-3	n/a	n/a	n/a
Feedwater Line Break (Outside Containment)	3.40E-3			
RWCU Line Break (Outside Containment)	3.40E-3			
IC Line Break (Outside Containment)	1.53E-6			
ISLOCA	6.42E-8	n/a	n/a	n/a
Special Initiators				
Complete Loss of PSWS	9.70E-4	n/a	9.7E-4[1]	3.94E-4
Complete Loss of Air Systems	1.02E-2	n/a	1.02E-2	1.02E-2

Notes:

1. From NUREG/CR-5750, published in February, 1999.

2.4 ASSUMPTIONS

- Based on thermal hydraulic analysis, it was assumed that RWCU/SDC pipe break at elevation 17215 mm and feedwater pipe break at elevation 18915 mm behave like steam break, though generally, it was assumed that pipe breaks below level 3 are treated as liquid break.
- Evaluation of special initiators, for the most part, was based on qualitative assumptions for screening. For example, it was assumed that partial loss of feedwater, FAPCS or RWCU will not cause a reactor trip..
- Loss of two or three MSIVs out of eight in which the failed closed MSIVs are not on the same line is not considered as initiator based on industry events data.
- The common cause failure for two or three of four MSIVs is not considered as initiator based on industry events data.
- The loss of two or three condensate trains is not considered as credible initiating events.
- It is assumed that the condensate/feedwater makeup capacity is at least 100 gpm (6.3E-3 m3.s-1). A reactor trip would not be required for a very small LOCA. A very small LOCA is not considered an initiator for ESBWR.

2.5 INSIGHTS

- Current plant level initiator identification, classification and quantification provide applicable, reasonable and adequate insight to the early stage ESBWR design and decision-making. Section 7 and 11 present risk insights and contributions of initiators to core damage.