INITIATING EVENT EXECUTIVE SUMMARY

This report presents an analysis of initiating event frequencies at United States (U.S.) nuclear power plants. The evaluation is based primarily on the operating experience from 1987 through 1995 as reported in Licensee Event Reports (LERs). The objectives of the study are: (1) provide revised, historical frequencies for the occurrence of initiating events in U.S. nuclear power plants, (2) compare these estimates based on operating experience to estimates used in probabilistic risk assessments (PRAs), individual plant examinations (IPEs), and other regulatory issues; and (3) review the operating data from an engineering perspective to determine trends and patterns of plant performance on a plant-type [i.e., pressurized water reactor (PWR) or boiling water reactor (BWR)], plant-specific, and industry-wide basis.

This study used as one of its sources of data the operating experience from 1987 through 1995 as reported in LERs. The Sequence Coding and Search System (SCSS) database was used to identify LERs for review and classification for this study. Each LER was reviewed from a risk and reliability perspective by an engineer with nuclear power plant experience. Based on the LER review, approximately 2,000 reactor trip events were analyzed with regard to their effect on plant performance.

For some initiators whose frequency is low enough that no events would be expected in the 1987–1995 period, additional operating experience and information from other sources were used to estimate their frequencies. These included operating experience from U.S. and foreign reactors, as well as evaluation of engineering aspects of certain rare events, such as loss-of-coolant accidents (LOCAs).

Major Findings

This report provides information on frequencies, trends, and between-plant variation for initiating events. An evaluation of the results indicates that:

– Combined initiating event frequencies for all initiators calculated from the 1987–1995 experience are lower than the frequencies used in NUREG-1150, *Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants*, and IPEs by a factor of five and four, respectively.

– General transients constitute 77% of all initiating events. Events that pose a more severe challenge to the plant's mitigation systems (non-general transients) constitute the remaining 23%.

– Over the nine-year span considered by this report, either a decreasing or constant time trend was observed for all categories of events. A decreasing trend was identified in approximately two-thirds of the more risk-significant categories that had sufficient data for trending analysis. The overall initiating event frequency decreased by a factor of two to three during the nine-year span. Most risk-significant initiator frequencies (such as total loss of feedwater flow, loss of instrument or control air, inadvertent closure of all main steam isolation valves (MSIVs), and total loss of condenser heat sink for BWRs) decreased at a faster rate than the overall initiating event frequency.

– Loss-of-coolant accident frequencies are lower than those used in NUREG-1150 and industry-wide IPEs.

– The frequencies (per critical year) estimated from the 1987–1995 experience for the risk-significant categories and general transients are the following. All but the first show a decreasing trend, and the values presented here apply to 1995.

Loss of Offsite Power (PWR and BWR)	4.6E-2
Total Loss of Condenser Heat Sink: PWR	1.2E-1
Total Loss of Condenser Heat Sink: BWR	2.9E-1
Total Loss of Feedwater Flow (PWR and BWR)	8.5E-2
General transients: PWR	1.2
General transients: BWR	1.5

For LOCA categories, the frequencies were evaluated using data and information prior to 1987 due to their relatively low frequency and the corresponding sparseness of data. No pipe break LOCA events were found in the U.S. operating experience. For the small pipe break LOCA frequency, the estimate from WASH-1400, *Reactor Safety Study*, was updated using U.S. reactor experience. For medium and large pipe break LOCAs, frequency estimates were calculated by using the frequency of leaks or through-wall cracks that have occurred which challenge the piping integrity. Further, conservative estimates were used for the probability of break given a leak (based on a technical review of information on fracture mechanics, data on high energy pipe failures and cracks, and assessment of pipe break frequencies estimated by others since WASH-1400). The pipe-break LOCA frequencies (per critical year) estimated from the experience are:

	Small LOCA	Medium LOCA	Large LOCA
PWR:	5E-4	4E-5	5E-6
BWR:	5E-4	4E-5	3E-5

No interfacing system loss-of-coolant accident (ISLOCA) events were identified in the U.S. operating experience.

Between-plant variation in initiating event frequencies was identified in the following categories: Total Loss of Condenser Heat Sink for PWRs, Loss of Condenser Vacuum for PWRs, Inadvertent Closure of All MSIVs for BWRs, Total Loss of Feedwater Flow, and General Transients for BWRs and for PWRs. Several plants whose uncertainty interval of the mean is statistically significantly higher than the industry average (i.e., the uncertainty interval is located completely to the right of the industry average mean) for several risk-significant categories have been identified. A listing of these plants is provided in Table 4-4 in the main report.

A comparison was made between initiating event frequencies based on the 1987–1995 operating experience for non-LOCA categories and the corresponding values from PRA/IPEs. Based on the cumulative mean frequency of the initiating events, the IPE-wide frequency is higher (approximately a factor of four) than the frequency estimated from operating experience. Table ES-1 provides a comparison of the operating experience to the average of the IPE population.

The mean frequencies calculated from the 1987–1995 operating experience for non-LOCA events have generally decreased by a factor of two as compared with the mean frequencies from NUREG/CR-3862, *Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments*, and NUREG-1150, which were based on experience at the time of the studies.

Table ES-1. Initiating event frequencies (per critical year) based on operating experience compared to the average of the IPE population.

Description	PWR Frequency—Mean (per critical year)[b]		BWR Frequency—Mean (per critical year)[b]	
	Operating Experience[a]	IPE[c]	Operating Experience[a]	IPE[c]
Small Pipe Break LOCA (G3)	5E-4[d]	9.2E-3	5E-4[d]	1.0E-2
Steam Generator Tube Rupture (F)	7.0E-3	2.0E-2	—	—
Loss of Offsite Power (B)	4.6E-2[d]	1.0E-1	4.6E-2[d]	1.3E-1
Total Loss of Condenser Heat Sink (L)	1.2E-1	3.0E-1	2.9E-1	4.3E-1
Total Loss of Feedwater Flow (P)	8.5E-2[d]	1.0E+0	8.5E-2[d]	5.7E-1
General Transients (Q)	1.2E+0	4.0E+0	1.5E+0	6.0E+0

a. 1987–1995 experience except for Small Pipe Break LOCA category which included total U.S. operating experience (1969–1997).

b. Units are in per critical year. One critical year equals 8,760 hours of reactor criticality.

c. The values are the mean of the IPE population for the plant type (PWR or BWR). The units stated in the IPE are per calendar year. For comparison purposes, the per calendar year was converted to critical year. One critical year equals one calendar year divided by the fraction of time the reactor was critical; 75% criticality factor was used based on the results of this study. Therefore the rate per critical year equals the rate per calendar year divided by 0.75.

d. The estimate did not differentiate with respect to plant type (i.e., PWR and BWR); therefore the value is same for either plant type.

The mean frequencies based on 1987–1995 experience are lower than the means from NUREG/CR-3862 and NUREG-1150 by a factor of four or more for the following categories: Loss of Offsite Power for BWRs and PWRs, and General Transients for BWRs and PWRs. (Note: NUREG-1150 used frequencies for non-LOCA categories from NUREG/CR-3862.)

The total initiating event frequency for BWRs and PWRs has decreased by about a factor of five and eight, respectively, since the NUREG/CR-3862 study was published in 1985.

Table ES-2 provides a comparison of the operating experience to the values reported in NUREG/CR-3862 and NUREG-1150.

A comparison was made with the frequencies used in the Anticipated Transients Without Scram (ATWS) Events Rulemaking (SECY-83-293). The frequency of ATWS transient initiators calculated from the 1987–1995 operating experience has decreased since the ATWS Rulemaking analysis was completed in 1983. This decrease indicates that the frequency of challenges that could result in a severe ATWS event has declined. The SECY-83-293 ATWS initiating frequencies would be reduced approximately by a factor of three for the PWR vendors while the BWR vendor is reduced by about a factor of four when updated with initiating event frequencies from this study. Assuming the average failure to scram probability used in SECY-83-293, the probability of ATWS per calendar year for PWRs and BWRs based on 1987–1995 experience and SECY-83-293 are as follows:

	PWR	BWR
1987–1995 experience	8.4E-6	3.3E-6
SECY-83-293	2.4E-5	1.2E-5

Table ES-2. Initiating event frequencies (per critical year) based on operating experience compared to NUREG/CR-3862 and NUREG-1150.

		Mean Frequency (per critical year)[b]	
Description	Operating Experience[a]	NUREG/CR-3862[c]	NUREG-1150[c]
Small Pipe Break LOCA (G3)	5E-4[d]	—	1.3E-3[d]
Steam Generator Tube Rupture (F)	7.0E-3	—	1.0E-2
Loss of Offsite Power (B)—PWR	4.6E-2[d]	1.9E-1	1.9E-1
Loss of Offsite Power (B)—BWR	4.6E-2[d]	1.1E-1	1.1E-1
Total Loss of Condenser Heat Sink (L)—PWR	1.2E-1	2.4E-1	2.4E-1
Total Loss of Condenser Heat Sink (L)—BWR	2.9E-1	9.1E-1	9.1E-1
Total Loss of Feedwater Flow (P) —PWR	8.5E-2[d]	2.2E-1	2.2E-1
Total Loss of Feedwater Flow (P) —BWR	8.5E-2[d]	9.3E-2	9.3E-2
General Transient—PWR (Q)	1.2E+0	1.0E+1	1.0E+1
General Transients—BWR (Q)	1.5E+0	8.6E+0	8.6E+0
Total of all events—PWR	1.4E+0	1.1E+1[e]	1.1E+1[e]
Total of all events—BWR	1.8E+0	9.7E+0[e]	9.9E+0[e]

a. 1987–1995 experience except for Small Pipe Break LOCA category which included total U.S. operating experience (1969–1997).

b. Units are in per critical year. One critical year equals 8,760 hours of reactor criticality.

c. The units stated in the report are per reactor year (i.e., numbers of years from start of commercial operation). For comparison purposes, the per reactor year was converted to critical year. One critical year equals one calendar year divided by the fraction of time the reactor was critical; 75% criticality factor was used based on the results of this study. Therefore the rate per critical year equals the rate per calendar year divided by 0.75.

d. The estimate did not differentiate with respect to plant type (i.e., PWR and BWR); therefore the value is same for either plant type.

e. This total represents the sum of all frequencies presented in the referenced report.