Initiating Event Data Sheets

2015 Update

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Update Notes

This file represents the second update to the original set of initiating event data sheets, which was completed in February 2007. The original set of initiating event data sheets were extracted from NUREG/CR-6928 [Reference 4] and generally contained data from the date range of 1988 to 2002. This edition generally represents results using a date range of 1988 to 2015.

This update is different from the original in the following respects:

1. The hierarchy of the report has been changed to facilitate finding sections
2. Several new initiating events have been added to support more detailed SPAR models.
    a. All of the high-energy line break events
    b. Two or more stuck open relief valves
    c. Calculated loss of multiple AC or DC busses
    d. Interfacing system Loss of Coolant Accident (LOCA)
    e. Reactor Coolant Pump Seal LOCA (RCPLOCA)
    f. LOOP in power operations and in shutdown.

The original NUREG/CR-6928 used some statistical adjustments to data that have been modified to be less arbitrary:

1. The use of the SCNID distribution (a simplified version of the CNID) has been discontinued. The Jefferies update replaces that distribution. The SCNID had the property of producing a result with a highly uncertain distribution, which was supposed to enhance the use of the reliability results as the prior to a plant-specific update. The primary use of these results is to support SPAR modeling, and the use of highly uncertain distributions leads to more uncertainty in the final CDF.
2. There was a decision made when the empirical Bayes (EB) analysis produced a result that had a low (<0.3) α parameter to the beta or gamma distribution, that the α parameter was reset to 0.3 and β and the mean were recalculated. This action was motivated since the EB could produce extremely wide distributions that nobody believed were valid. This update revises the decision-making and the alternative method of obtaining a reasonable distribution. The decision point is now whether the difference between the 5th percentile and the mean is greater than 4 orders of magnitude (this happens to approximate the decision point of $\alpha < 0.3$). When the decision point is reached, instead of creating an arbitrary distribution, the Jeffries distribution is used, which is the same decision that is made when the EB does not return a result.

1 Primary/Secondary Inventory Control

This category includes line breaks from both the primary and secondary systems.

1.1 High Energy Line Breaks

This category includes breaks of steam and feedwater lines greater than one inch in diameter. It does not have to be a complete break. Included are actuations or failure of rupture disks, splits, cracks, and failed welds.

1.1.1 Feedwater Line Break (BWR)

1.1.1.1 Initiating Event Description

From Reference 3, the Feedwater Line Break at Pressurized Water Reactors (FWLB (BWR)) initiating event is a break of a one-inch equivalent diameter or more in a feedwater or condensate line that contains main turbine working fluid at or above atmospheric saturation conditions. Examples include: breeches of a pipe caused by a split, crack, weld failure, or circumferential break.

1.1.1.2 Data Collection and Review

Data for the FWLB (PWR) baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for FWLB (BWR) is 1988–2015. The RADS database was used to collect the FWLB (BWR) data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. Table 1-1 summarizes the data obtained from RADS and used in the FWLB (BWR) analysis.

Table 1-1. FWLB (BWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
0	834	1988--2015	37	0.0%

1.1.1.3 Industry-Average Baselines

Table 1-2 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 1-2. Selected industry distribution of λ for FWLB (BWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	2.36E-06	6.00E-04	2.30E-03	Gamma	0.500	8.340E+02

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.1.2 Feedwater Line Break (PWR)

1.1.2.1 Initiating Event Description

From Reference 3, the Feedwater Line Break at Pressurized Water Reactors (FWLB (PWR)) initiating event is a break of a one-inch equivalent diameter or more in a feedwater or condensate line that contains main turbine working fluid at or above atmospheric saturation conditions. Examples include: breeches of a pipe caused by a split, crack, weld failure, or circumferential break.

1.1.2.2 Data Collection and Review

Data for the FWLB (PWR) baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for FWLB (PWR) is 1988–2015. Figure 1-1 shows the trend of the full FWLB (PWR) data set and the baseline period used in this analysis. The RADS database was used to collect the FWLB (PWR) data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. Table 1-3 summarizes the data obtained from RADS and used in the FWLB (PWR) analysis.

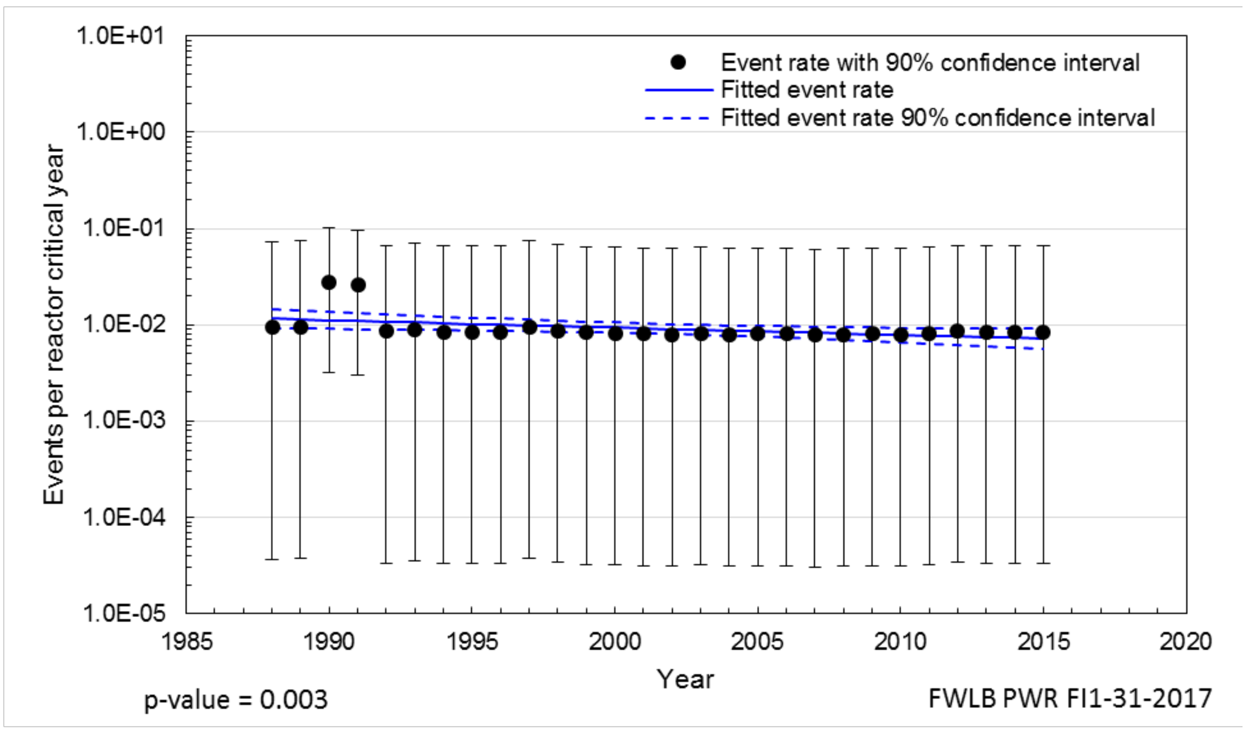

Figure 1-1. FWLB (PWR) trend plot.

Table 1-3. FWLB (PWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
2	1663	1988-2015	77	2.6%

1.1.2.3 Industry-Average Baselines

Table 1-4 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 1-4. Selected industry distribution of λ for FWLB (PWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	3.45E-04	1.50E-03	3.33E-03	Gamma	2.500	1.660E+03

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.1.3 Steamline Break inside Containment

1.1.3.1 Initiating Event Description

From Reference 3, the Steam Line Break inside Containment (PWR) (SLBIC (PWR)) initiating event is a break of one-inch equivalent diameter or more in a steam line located inside the primary containment that contains main turbine working fluid at or above atmospheric saturation conditions.

This category applies to PWRs only. Examples include: breeches of a pipe caused by a split, crack, weld failure, or circumferential break.

1.1.3.2 Data Collection and Review

Data for the SLBIC (PWR) baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for SLBIC (PWR) is 1988–2015. The RADS database was used to collect the SLBIC (PWR) data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. Table 1-5 summarizes the data obtained from RADS and used in the SLBIC (PWR) analysis.

Table 1-5. SLBIC (PWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
0	1663	1988-2015	77	0.0%

1.1.3.3 Industry-Average Baselines

Table 1-6 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 1-6. Selected industry distribution of λ for SLBIC (PWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	1.18E-06	3.01E-04	1.16E-03	Gamma	0.500	1.660E+03

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.1.4 Steamline Break outside Containment (BWR)

1.1.4.1 Initiating Event Description

From Reference 3, the Steam Line Break outside Containment at Boiling Water Reactors (SLBOC (BWR)) initiating event is a break of one-inch equivalent diameter or more in a steam line located outside the primary containment that contains main turbine working fluid at or above atmospheric saturation conditions.

Examples include: operation of rupture disks; and breeches of a pipe caused by a split, crack, weld failure, or circumferential break.

1.1.4.2 Data Collection and Review

Data for the SLBOC (BWR) baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for SLBOC (BWR) is 1988–2015. Figure 1-2 shows the trend of the full SLBOC (BWR) data set and the baseline period used in this analysis. The RADS database was used to collect the SLBOC (BWR) data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. Table 1-7 summarizes the data obtained from RADS and used in the SLBOC (BWR) analysis.

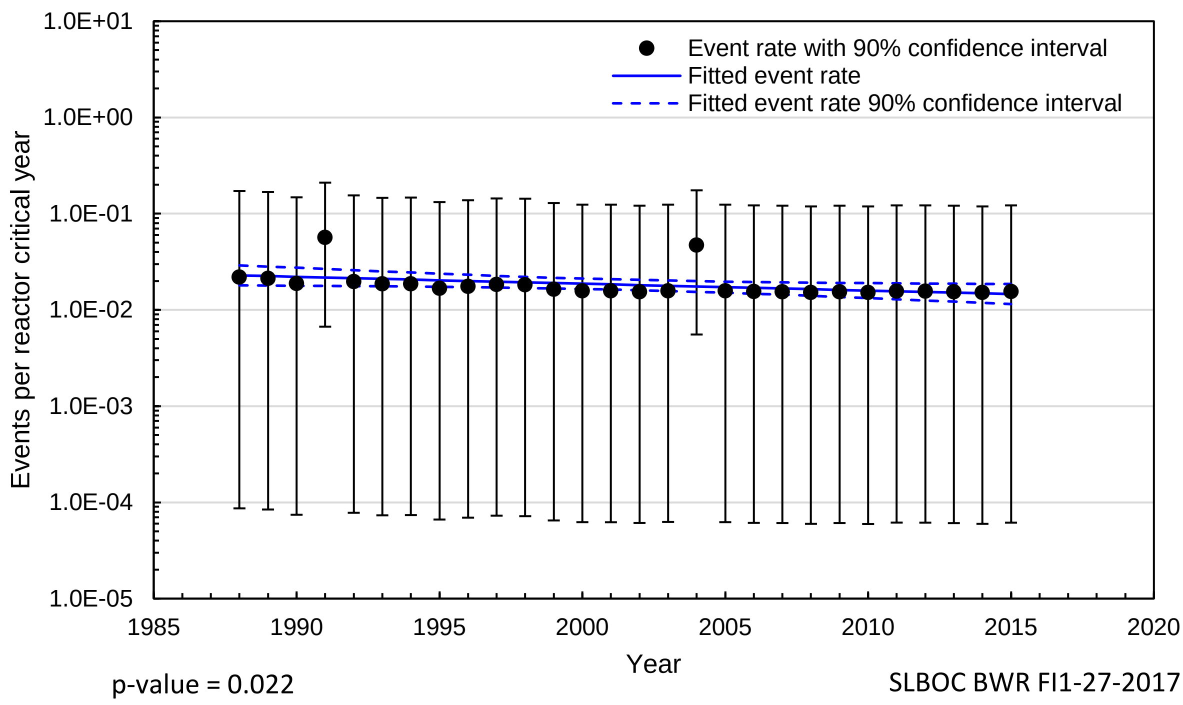

Figure 1-2. SLBOC (BWR) trend plot.

Table 1-7. SLBOC (BWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
2	834	1988-2015	37	5.4%

1.1.4.3 Industry-Average Baselines

Table 1-8 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 1-8. Selected industry distribution of λ for SLBOC (BWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	6.87E-04	3.00E-03	6.64E-03	Gamma	2.500	8.340E+02

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.1.5 Steamline Break outside Containment (PWR)

1.1.5.1 Initiating Event Description

From Reference 3, the Steam Line Break outside Containment at Pressurized Water Reactors (SLBOC (PWR)) initiating event is a break of one-inch equivalent diameter or more in a steam line located outside the primary containment that contains main turbine working fluid at or above atmospheric saturation conditions.

Examples include: operation of rupture disks; and breeches of a pipe caused by a split, crack, weld failure, or circumferential break.

1.1.5.2 Data Collection and Review

Data for the SLBOC (PWR) baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for SLBOC (PWR) is 1988–2015. Figure 1-3 shows the trend of the full SLBOC (PWR) data set and the baseline period used in this analysis. The RADS database was used to collect the SLBOC (PWR) data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. Table 1-9 summarizes the data obtained from RADS and used in the SLBOC (PWR) analysis.

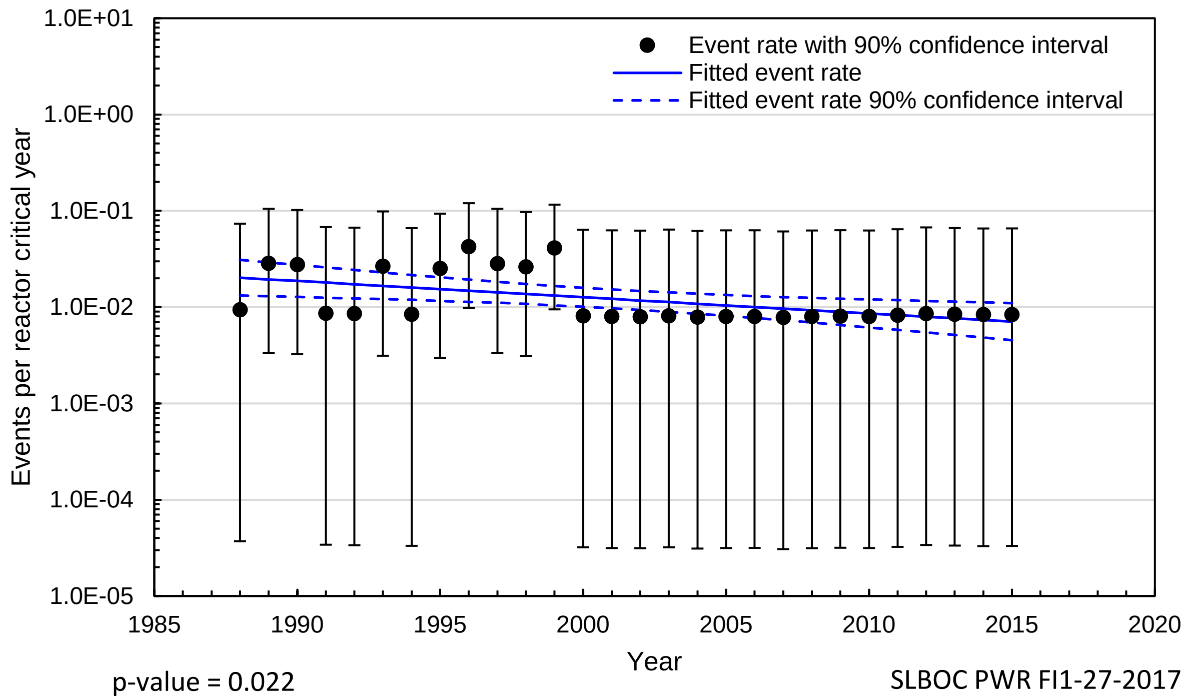

Figure 1-3. SLBOC (PWR) trend plot.

Table 1-9. SLBOC (PWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
10	1663	1988-2015	77	13.0%

1.1.5.3 Industry-Average Baselines

Table 1-10 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 1-10. Selected industry distribution of λ for SLBOC (PWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	3.49E-03	6.32E-03	9.84E-03	Gamma	10.500	1.660E+03

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.2 Steam Generator Tube Rupture (SGTR)

1.2.1 Initiating Event Description

From Reference 3, the Steam Generator Tube Rupture (STGR) initiating event is a rupture of one or more steam generator tubes that results in a loss of primary coolant to the secondary side of the steam generator at a rate greater than or equal to 100 gallons per minute (gpm). A SGTR can occur as the initial plant fault, such as a tube rupture caused by high cycle fatigue or loose parts, or as a consequence of another initiating event. The latter case would be classified as a functional impact. This category applies to pressurized water reactors (PWRs) only. This category includes excessive leakage caused by the failure of a previous SGTR repair (i.e., leakage past a plug).

1.2.2 Data Collection and Review

Two methodologies are summarized in this section. For one approach, information for the SGTR baseline was obtained from *Estimating Loss-of-Coolant Accident (LOCA) Frequencies through the Elicitation Process* (Ref. 5). In that document, the SGTR frequency was estimated based on an expert elicitation process "…to consolidate service history data and PFM [probabilistic fracture mechanics] studies with knowledge of plant design, operation, and material performance." Reference 5 is a draft document. Results obtained from that document could change when the final report is issued.

From Table 7.3 in Reference 5, the mean frequency for SGTR ((> 100 gpm) is 3.4E-3/reactor calendar year (rcy). To convert this to reactor critical years (rcry's), it was assumed that reactors are critical 90% of each year. Converting to rcry's, the result is

(3.40E-4/rcy)(1 rcy/0.9 rcry) = 3.78E-3/rcry.

The associated error factor (95th percentile divided by median) associated with the SGTR category from Reference 5 is

(8.2E-3/rcy)/(2.6E-3/rcy) = 3.2,

which converts to an α of 1.6.

For the other approach, data for the SGTR baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for SGTR is 1991–2015.

Figure 1-4 shows the trend of the full SGTR data set and the baseline period used in this analysis. The RADS database was used to collect the SGTR data for that period. Results include total number of events and total rcry's for the U.S. commercial nuclear power plant industry. Table 1-11 summarizes the data obtained from RADS and used in the SGTR analysis.

Table 1-11. STGR frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
2	1503	1991-2015	76	2.6%

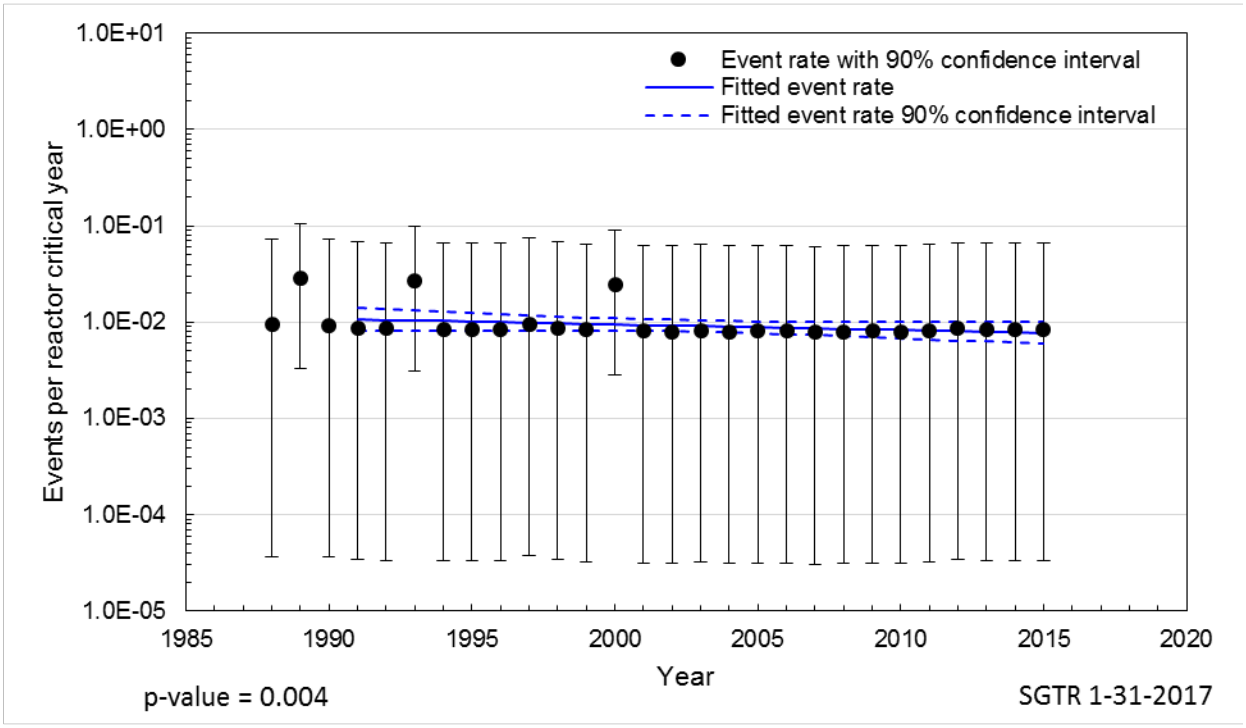

Figure 1-4. SGTR trend plot.

1.2.3 Industry-Average Baselines

Table 1-12 lists the industry-average frequency distribution. Two different approaches to estimating the frequency for SGTR were discussed – the expert elicitation approach from Reference 5, and the data analysis using the IEDB. Because the expert elicitation process outlined in Reference 5 resulted in a mean frequency for SGTR (3.78E-3/rcry) which is higher than that obtained from optimizing the SGTR data from the IEDB (2.07E-03/rcry), the IEDB results were used. This industry-average frequency does not account for any recovery.

Table 1-12. Selected industry distribution of λ for SGTR.

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	3.82E-04	1.66E-03	3.69E-03	Gamma	2.500	1.500E+03

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.3 Loss of Coolant Accidents

Although there have not been any actual small LOCA or larger events recorded in the US operating experience data collected through 2015, there have been numerous instances of reactor coolant leakage events (e.g., break flow within the capacity of normal makeup systems). There have also been failures of smaller pressure boundary pipes (i.e., less than 2") that have not exceed the capacity of normal makeup systems. In general, most aging management and inspection programs focus on medium and large diameter piping (i.e., > 4" diameter). Such programs are more effective for larger diameter piping systems because these pipes are most likely to experiences leaks that can be detected and mitigated before component failure occurs. These factors lead to uncertainty in the small break LOCA frequency estimates, which are principally related to failure of smaller diameter piping (i.e., 2" to 4" diameter). It is therefore important that plant operators are cognizant of the reduced failure margins associated with small diameter piping and that they have aging management programs – including attributes related to inspection, monitoring, and mitigation – specifically targeted to provide reasonable assurance that failure will not occur in these systems.

1.3.1 Large Loss-of-Coolant Accident at Boiling Water Reactors (LLOCA (BWR))

1.3.1.1 Initiating Event Description

The Large Loss-of-Coolant Accident for Boiling Water Reactors (LLOCA (BWR)) is a break size greater than 6-inch inside diameter pipe equivalent for liquid and steam in the reactor coolant system pressure boundary.

1.3.1.2 Data Collection and Review

Information for the LLOCA (BWR) baseline was obtained from *Estimating Loss-of-Coolant Accident (LOCA) Frequencies through the Elicitation Process* (Ref. 5). The LLOCA frequency was estimated based on an expert elicitation process "…to consolidate service history data and PFM [probabilistic fracture mechanics] studies with knowledge of plant design, operation, and material performance."

Table 7.17 in Reference 5 presents frequencies for LOCAs exceeding various sizes by gallon per minute (gpm) break flow and effective pipe size break. Six different sizes are listed, ranging from 0.5-inch diameter (> 100 gpm) to 41-inch diameter (> 500,000 gpm). The frequencies presented for each size indicate the frequency of LOCAs of that size or greater occurring. In addition, frequencies for each size are presented for 25 years of fleet operation, and for end-of-life conditions (40 years of operation). Since much of the reactor fleet now has over 35 years of operation, and will be over 40 years by the next expected update, 40-year average fleet conditions were used.

Reference 8 provides details for determining the break sizes for use in the SPAR models and for obtaining the related frequency information from Reference 5. The LLOCA break threshold for the SPAR models is 6 inches which requires interpolation between rows in Table 7.17. The LLOCA frequency is provided in reactor calendar years (rcy's). To convert this to reactor critical years (rcry's), it was assumed that reactors are critical 90% of each year. Converting to rcry's and rounding using the NUREG/CR-6928 round off scheme results provided in Table 1-13.

Table 7.17 includes excessive LOCA data (>41.0 inch break diameter) which should be removed from the LLOCA result, but the frequency is so small as to be negligible and the interpolated result was used without removing the contribution from excessive LOCA.

Reference 5 provided an evaluation of industry conditions up to 2002. Additional operating experience has been recorded since then, and the NUREG-1829 result has been updated with 0 recorded events and 418 rcry of fleet operation for the date range 2003 to 2015. The updated frequency is provided in the second row of Table 1-13. The Bayes Update row is the recommended value for the SPAR models.

1.3.1.3 Industry-Average Baselines

Table 1-13 lists the industry-average frequency distribution.

Table 1-13. Selected industry distribution of λ for LLOCA (BWR).

Source	5%	Mean	95%	Distribution		
				Type	α	B
Ref. 5	1.28E-09	1.20E-05	5.49E-05	Gamma	0.300	2.500E+04
Bayes Update	1.26E-09	1.18E-05	5.40E-05	Gamma	0.300	2.542E+04

Note – The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.3.2 Large Loss-of-Coolant Accident at Pressurized Water Reactors (LLOCA (PWR))

1.3.2.1 Initiating Event Description

The Large Loss-of-Coolant Accident at Pressurized Water Reactors (LLOCA (PWR)) is a break in the primary system boundary with an equivalent inside pipe diameter greater than 6 inches.

1.3.2.2 Data Collection and Review

Information for the LLOCA (PWR) baseline was obtained from *Estimating Loss-of-Coolant Accident (LOCA) Frequencies through the Elicitation Process* (Ref. 5). The LLOCA frequency was estimated based on an expert elicitation process "…to consolidate service history data and PFM [probabilistic fracture mechanics] studies with knowledge of plant design, operation, and material performance."

Table 7.19 of Reference 5 presents frequencies for LOCAs exceeding various sizes by gallon per minute (gpm) break flow and effective pipe size break. Six different sizes are listed, ranging from 0.5-inch diameter (> 100 gpm) to 31-inch (> 500,000 gpm). The frequencies presented for each size indicate the frequency of LOCAs of that size or greater occurring. In addition, frequencies for each size are presented for average of 25 years of operation, and for end-of-life conditions (40 years of operation). Since much of the reactor fleet now has over 35 years of operation, and will be over 40 years by the next expected update, 40-year average fleet conditions were used.

Reference 8 provides details for determining the break sizes for use in the SPAR models and for obtaining the related frequency information from Reference 5. The LLOCA break threshold for the SPAR models is 6 inches which requires interpolation between rows in Table 7.19. The LLOCA frequency is provided in reactor calendar years (rcy's). To convert this to reactor critical years (rcry's), it was assumed that reactors are critical 90% of each year. Converting to rcry's and rounding using the NUREG/CR-6928 round off scheme results provided in Table 1-14.

Table 7.19 includes excessive LOCA data (>31.0 inch equivalent break diameter) which should be removed from the LLOCA result, but the frequency is so small as to be negligible and the interpolated result was used without removing the contribution from excessive LOCA.

Reference 5 was an evaluation of industry conditions up to 2002. Additional operating experience has been recorded since then, and the NUREG-1829 result has been updated with 0 recorded events and 797 rcry of fleet operation for the date range 2003 to 2015. The updated frequency is provided in the second row of Table 1-14. The Bayes Update row is the recommended value for the SPAR models.

1.3.2.3 Industry-Average Baselines

Table 1-14 lists the industry-average frequency distribution.

Table 1-14. Selected industry distribution of λ for LLOCA (PWR).

Source	5%	Mean	95%	Distribution		
				Type	α	B
Ref. 5	6.42E-10	6.00E-06	2.74E-05	Gamma	0.300	5.000E+04
Bayes Update	6.32E-10	5.91E-06	2.70E-05	Gamma	0.300	5.080E+04

Note – The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

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1.3.3 Medium Loss-of-Coolant Accident at Boiling Water Reactors (MLOCA (BWR))

1.3.3.1 Initiating Event Description

The Medium Loss-of-Coolant Accident for Boiling Water Reactors (MLOCA (BWR)) initiating event is defined as a break in the reactor coolant system boundary with size between 2- and 6-inch inside diameter pipe equivalent.

1.3.3.2 Data Collection and Review

Information for the MLOCA (BWR) baseline was obtained from *Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process* (Ref. 5). The MLOCA frequency was estimated based on an expert elicitation process "…to consolidate service history data and PFM [probabilistic fracture mechanics] studies with knowledge of plant design, operation, and material performance."

Table 7.17 in Reference 5 presents frequencies for LOCAs exceeding various sizes indicated by gallon per minute (gpm) break flow and effective pipe size break. Six different sizes are listed, ranging from 0.5-inch diameter (> 100 gpm) to 41-inch diameter (> 500,000 gpm). The frequencies presented for each size indicate the frequency of LOCAs of that size or greater occurring. In addition, frequencies for each size are presented for current day conditions (assuming an average of 25 years of fleet operation) and for end-of-life conditions (40 years of operation). For this estimate, frequencies appropriate for 40 years of fleet operation were used.

Reference 8 provides details for determining the break sizes for use in the SPAR models and for obtaining the related frequency information from Reference 5. The SPAR model break range is not provided in Table 7.17 of Reference 5 and must worked out by interpolation between the provided rows. Subtraction of the means from the interpolated results for 2-inch and 6-inch breaks gives the mean MLOCA frequency. The uncertainty distribution parameters are obtained from the difference in variances assuming lognormally-distributed difference in the means. A lognormal distribution with the resulting mean and variance is converted to an equivalent gamma distribution by setting means and error factors equal. Finally, the result is converted to reactor critical years (rcry's) assuming that reactors are critical 90% of each year, and rounded using the round off scheme provided in NUREG/CR-6928. The resulting MLOCA frequency is provided in Table 1-15.

Reference 5 was an evaluation of industry conditions up to 2002. Additional operating experience has been recorded since then, and the NUREG-1829 result has been updated with 0 recorded events and 418 rcry of fleet operation for the date range 2003 to 2015. The updated frequency is provided in the second row of Table 1-15. The Bayes Update row is the recommended value for the SPAR models.

1.3.3.3 Industry-Average Baselines

Table 1-15 lists the industry-average frequency distribution.

Table 1-15. Selected industry distribution of λ for MLOCA (BWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
Ref. 5	1.04E-07	1.00E-04	4.15E-04	Gamma	0.400	4.000E+03
Bayes Update	9.39E-08	9.05E-05	3.76E-04	Gamma	0.400	4.418E+03

Note – The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.3.4 Medium Loss-of-Coolant Accident at Pressurized Water Reactors (MLOCA (PWR))

1.3.4.1 Initiating Event Description

The Medium Loss-of-Coolant Accident at Pressurized Water Reactors (MLOCA (PWR)) initiating event is defined for PWRs, as a pipe break in the primary system boundary with an inside diameter between 2 and 6 inches.

1.3.4.2 Data Collection and Review

Information for the MLOCA (BWR) baseline was obtained from *Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process* (Ref. 5). The MLOCA frequency was estimated based on an expert elicitation process "…to consolidate service history data and PFM [probabilistic fracture mechanics] studies with knowledge of plant design, operation, and material performance."

Table 7.19 in Reference 5 presents frequencies for LOCAs exceeding various sizes indicated by gallon per minute (gpm) break flow and effective pipe size break. Six different sizes are listed, ranging from 0.5-inch diameter (> 100 gpm) to 31-inch diameter (> 500,000 gpm). The frequencies presented for each size indicate the frequency of LOCAs of that size or greater occurring. In addition, frequencies for each size are presented for current day conditions (assuming an average of 25 years of operation) and for end-of-life conditions (40 years of operation). For this estimate, frequencies for 40 years of operation were used.

Reference 8 provides details for determining the break sizes for use in the SPAR models and for obtaining the related frequency information from Reference 5. The SPAR model break range is not provided in Table 7.19 and must worked out by interpolation between the provided rows. Subtraction of the means from the interpolated results for 2-inch and 6-inch breaks gives the mean MLOCA frequency. The uncertainty distribution parameters are obtained from the difference in variances assuming lognormally-distributed difference in the means. The resulting lognormal distribution is converted to an equivalent gamma distribution by setting means and error factors equal. Finally, the result is converted to reactor critical years (rcry's) assuming that reactors are critical 90% of each year, and rounded using the round off scheme provided in NUREG/CR-6928. The resulting MLOCA frequency is provided in Table 1-16.

Reference 5 was an evaluation of industry conditions up to 2002. Additional operating experience has been recorded since then, and the NUREG-1829 result has been updated with 0 recorded events and 797 rcry of fleet operation for the date range 2003 to 2015. The updated frequency is provided in the second row of Table 1-16. The Bayes Update row is the recommended value for the SPAR models.

1.3.4.3 Industry-Average Baselines

Table 1-16 lists the industry-average frequency distribution.

Table 1-16. Selected industry distribution of λ for MLOCA (PWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
Ref. 5	2.68E-08	2.50E-04	1.14E-03	Gamma	0.300	1.200E+03
Bayes Update	1.61E-08	1.50E-04	6.87E-04	Gamma	0.300	1.997E+03

Note – The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.3.5 Small Loss-of-Coolant Accident at Boiling Water Reactors (SLOCA (BWR))

1.3.5.1 Initiating Event Description

The Small Loss-of-Coolant Accident (SLOCA) initiating event is defined for a boiling water reactor (BWR) as a break size between 0.5-inch inside diameter pipe equivalent and 2-inch inside diameter pipe equivalent in the reactor coolant system pressure boundary.

1.3.5.2 Data Collection and Review

Information for the SLOCA (BWR) baseline was obtained from *Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process* (Ref. 5). The LOCA frequency was estimated based on an expert elicitation process "…to consolidate service history data and PFM [probabilistic fracture mechanics] studies with knowledge of plant design, operation, and material performance."

Table 7.17 of Reference 5 presents frequencies for LOCAs exceeding various sizes indicated by gallon per minute (gpm) break flow and effective pipe size break. Six different sizes are listed, ranging from 0.5-inch diameter (> 100 gpm) to 41-inch diameter (> 500,000 gpm). The frequencies presented for each size indicate the frequency of LOCAs of that size or greater occurring. In addition, frequencies for each size are presented for current day conditions (assuming an average of 25 years of fleet operation) and for end-of-life conditions (40 years of operation). For this estimate frequencies for 40 years of fleet operation were used.

Reference 8 provides details for determining the break sizes for use in the SPAR models and for obtaining the related frequency information from Reference 5. The SPAR model break range is not provided in Table 7.17 of Reference 5 and must worked out by interpolation between the provided rows. Subtraction of the means from 0.5-inch break and the interpolated 2-inch break gives the mean SLOCA frequency. The uncertainty distribution parameters are obtained from the difference in variances assuming lognormally-distributed difference in the means. A lognormal distribution with the resulting mean and variance is converted to an equivalent gamma distribution by setting means and error factors equal. Finally, the result is converted to reactor critical years (rcry's) assuming that reactors are critical 90% of each year, and rounded using the round off scheme provided in NUREG/CR-6928. The resulting SLOCA frequency is provided in Table 1-18.

Reference 5 was an evaluation of industry conditions up to 2002. Additional operating experience has been recorded since then, and the NUREG-1829 result has been updated with 1 recorded event and 418 rcry of fleet operation for the date range 2003 to 2015. The updated frequency is provided in the second row of Table 1-18. The Bayes Update row is the recommended value for the SPAR models.

Table 1-17. SLOCA (BWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
1	418	2003-2015	37	2.7%

1.3.5.3 Industry-Average Baselines

Table 1-18 lists the industry-average frequency distribution.

Table 1-18. Selected industry distribution of λ for SLOCA (BWR).

Source	5%	Mean	95%	Distribution		
				Type	α	B
Ref. 5	6.22E-07	6.00E-04	2.49E-03	Gamma	0.400	6.667E+02
Bayes Update	3.82E-07	3.69E-04	1.53E-03	Gamma	0.400	1.085E+03

Note –The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.3.6 Small Loss-of-Coolant Accident at Pressurized Water Reactors (SLOCA (PWR))

1.3.6.1 Initiating Event Description

The Small Loss-of-Coolant Accident (SLOCA) initiating event is defined for a pressurized water reactor (PWR) as a break in the primary system pressure boundary with an equivalent inside pipe diameter between 0.5 and 2 inches.

1.3.6.2 Data Collection and Review

Information for the SLOCA (PWR) baseline was obtained from *Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process* (Ref. 5). The LOCA frequency was estimated based on an expert elicitation process "…to consolidate service history data and PFM [probabilistic fracture mechanics] studies with knowledge of plant design, operation, and material performance."

Table 7.19 of Reference 5 presents frequencies for LOCAs exceeding various sizes indicated by gallon per minute (gpm) break flow and effective pipe size break. Six different sizes are listed, ranging from 0.5-inch diameter (> 100 gpm) to 31-inch diameter (> 500,000 gpm). The frequencies presented for each size indicate the frequency of LOCAs of that size or greater occurring. In addition, frequencies for each size are presented for current day conditions (assuming an average of 25 years of fleet operation) and for end-of-life conditions (40 years of operation). For this estimate frequencies for 40 years of fleet operation were used.

Reference 8 provides details for determining the break sizes for use in the SPAR models and for obtaining the related frequency information from Reference 5. The SPAR model break range is not provided in Table 7.19 and must worked out by interpolation between the provided rows. Subtraction of the means from 0.5-inch break and the interpolated 2-inch break gives the mean SLOCA frequency. The uncertainty distribution parameters are obtained from the difference in variances assuming lognormally-distributed difference in the means. A lognormal distribution with the resulting mean and variance is converted to an equivalent gamma distribution by setting means and error factors equal. Finally, the result is converted to reactor critical years (rcry's) assuming that reactors are critical 90% of each year, and rounded using the round off scheme provided in NUREG/CR-6928. The resulting SLOCA frequency is provided in Table 1-20.

Reference 5 was an evaluation of industry conditions up to 2002. Additional operating experience has been recorded since then, and the NUREG-1829 result has been updated with 0 recorded events and 797 rcry of fleet operation for the date range 2003 to 2015. The updated frequency is provided in the second row of Table 1-20. The Bayes Update row is the recommended value for the SPAR models.

Table 1-19. SLOCA (PWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
0	797	2003-2015	77	0.0%

1.3.6.3 Industry-Average Baselines

Table 1-20 lists the industry-average frequency distribution.

Table 1-20. Selected industry distribution of λ for SLOCA (PWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
Ref. 5	2.07E-06	2.00E-03	8.16E-03	Gamma	0.400	2.000E+02
Bayes Update	4.16E-07	4.01E-04	1.67E-03	Gamma	0.400	9.968E+02

Note –The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.3.7 Very Small Loss-of-Coolant Accident at Boiling Water Reactors (VSLOCA (BWR))

1.3.7.1 Initiating Event Description

From Reference 3, the Very Small Loss of Coolant Accident (VSLOCA) initiating event is a pipe break or component failure that results in a loss of primary coolant between 10 to 100 gallons per minute (gpm), but does not require the automatic or manual actuation of high-pressure injection systems. Examples include reactor coolant pump (for pressurized water reactors) or recirculating pump (for boiling water reactors) seal failures, valve packing failures, steam generator tube leaks, and instrument line fitting failures.

1.3.7.2 Data Collection and Review

Data for the VSLOCA (BWR) baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for VSLOCA (BWR) is 1992–2015. Figure 1-5 shows the trend of the full VSLOCA (BWR) data set and the baseline period used in this analysis. The RADS database was used to collect the VSLOCA (BWR) data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. Table 1-21 summarizes the data obtained from RADS and used in the VSLOCA (BWR) analysis.

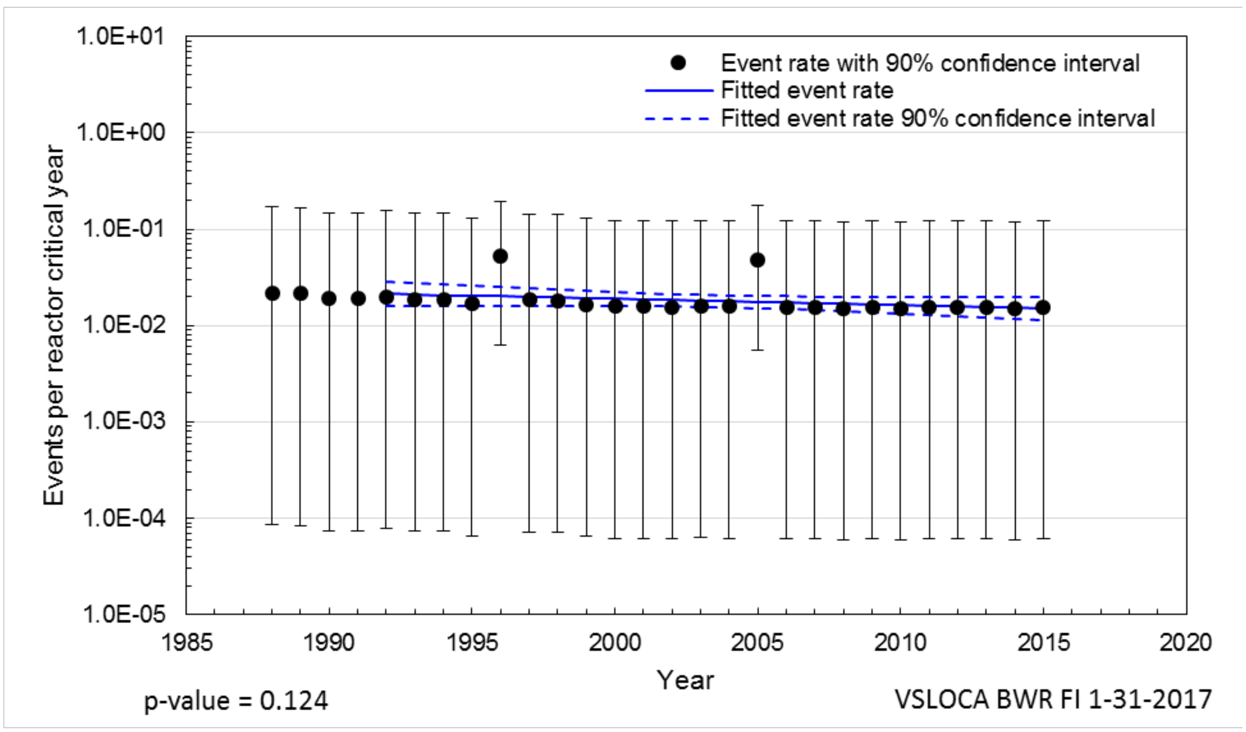

Figure 1-5. VSLOCA (BWR) trend plot.

Table 1-21. VSLOCA (BWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
2	735	1992--2015	37	5.4%

1.3.7.3 Industry-Average Baselines

Table 1-22 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 1-22. Selected industry distribution of λ for VSLOCA (BWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	7.79E-04	3.40E-03	7.53E-03	Gamma	2.500	7.350E+02

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.3.8 Very Small Loss-of-Coolant Accident at Pressurized Water Reactors (VSLOCA (PWR))

1.3.8.1 Initiating Event Description

From Reference 3, the Very Small Loss of Coolant Accident (VSLOCA) initiating event is a pipe break or component failure that results in a loss of primary coolant between 10 to 100 gallons per minute (gpm), but does not require the automatic or manual actuation of high-pressure injection systems. Examples include reactor coolant pump (for pressurized water reactors) or recirculating pump (for boiling water reactors) seal failures, valve packing failures, steam generator tube leaks, and instrument line fitting failures.

1.3.8.2 Data Collection and Review

Data for the VSLOCA baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for VSLOCA (PWR) is 1992–2015. The RADS database was used to collect the VSLOCA (PWR) data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. Table 1-23 summarizes the data obtained from RADS and used in the VSLOCA (PWR) analysis.

Table 1-23. VSLOCA (PWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
0	1445	1992-2015	75	0.0%

1.3.8.3 Industry-Average Baselines

Table 1-24 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 1-24. Selected industry distribution of λ for VSLOCA (PWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	1.36E-06	3.46E-04	1.32E-03	Gamma	0.500	1.450E+03

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.3.9 Stuck Open Relief Valve at Boiling Water Reactors (SORV (BWR))

1.3.9.1 Initiating Event Description

From Reference 3, the Stuck Open Relief Valve at Boiling Water Reactors (SORV (BWR)) initiating event is a failure of one primary system safety and/or relief valve (SRV) to fully close, resulting in the loss of primary coolant. The valves included in this category are main steam line safety valves (BWR) and automatic depressurization system relief valves (BWR). The stuck open SRV may or may not cause the automatic or manual actuation of high-pressure injection systems.

This category includes a stuck open valve that cannot be subsequently closed upon manual demand or does not subsequently close on its own immediately after the reactor trip. The mechanism that opens the valve is not a defining factor. The different mechanisms than can open an SRV are transient-induced opening, manual opening during valve testing, and spurious opening.

1.3.9.2 Data Collection and Review

Data for the SORV (BWR) baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for SORV (BWR) is 1993–2015. Figure 1-6 shows the trend of a single SORV (BWR) data set and the baseline period used in this analysis. There were no events for 2 or more SORV (BWR) failures.

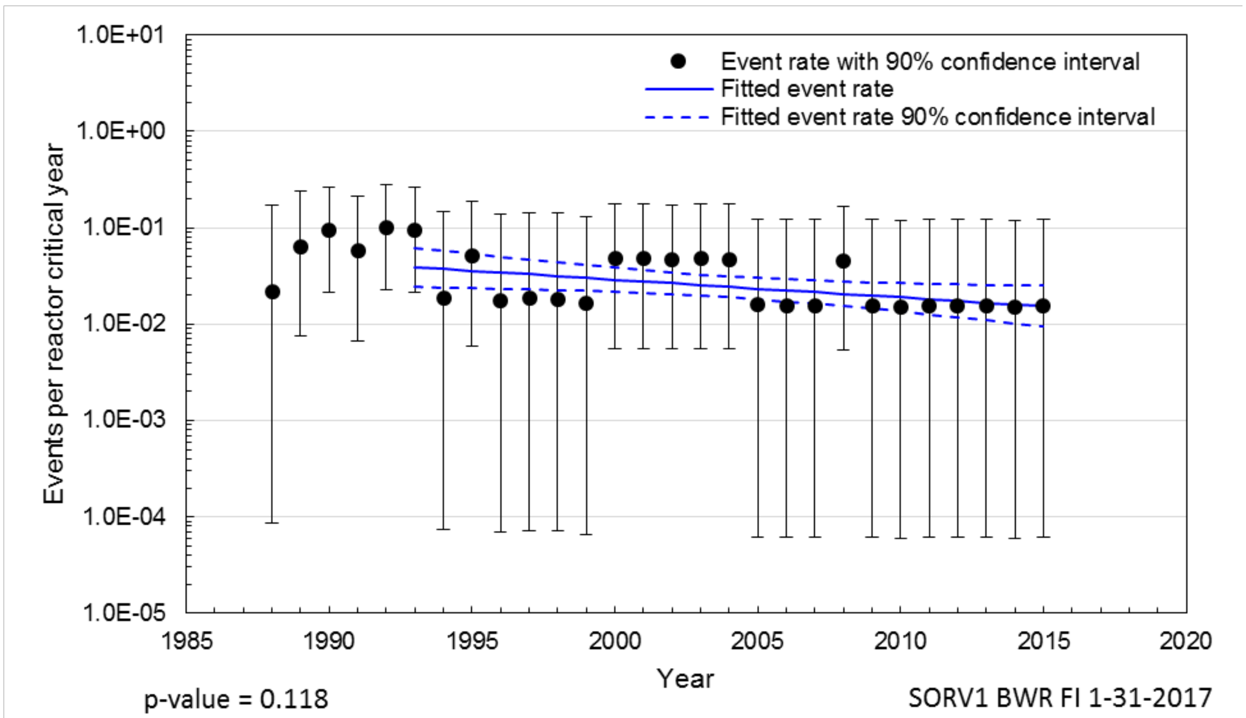

Figure 1-6. SORV (BWR) trend plot.

The RADS database was used to collect the SORV (BWR) data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. The SPAR models use two SORV initiating events in the models; a single SORV (SORV1) and two or more SORVs (SORV2). The single SORV has empirical Bayes results at the plant level. Table 1-25 summarizes the data obtained from RADS and used in the SORV (BWR) analysis.

Table 1-25. SORV (BWR) frequency data for baseline period.

Event Type	Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
	Events	Reactor Critical Years (rcry)			
SORV1	9	710	1993-2015	37	18.9%
SORV2	0	710	1993-2015	37	0.0%

1.3.9.3 Industry-Average Baselines

Table 1-26 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 1-26. Selected industry distribution of λ for SORV (BWR).

Event Type	Source	5%	Mean	95%	Distribution		
					Type	α	β
SORV1	EB/PL/KS	5.31E-04	1.26E-02	3.88E-02	Gamma	0.927	7.350E+01
SORV2	JNID/IL	2.77E-06	7.05E-04	2.71E-03	Gamma	0.500	7.100E+02

Note – EB/PL/KS is an empirical Bayes analysis at the plant level with the Kass-Steffey adjustment. JNID/IL is a Jeffrey’s noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.3.10 Stuck Open Relief Valve at Pressurized Water Reactors (SORV (PWR))

1.3.10.1 Initiating Event Description

From Reference 3, the Stuck Open Relief Valve at Pressurized Water Reactors (SORV (PWR)) initiating event is a failure of one primary system safety and/or relief valve (SRV) to fully close, resulting in the loss of primary coolant. The valves included in this category are pressurizer code safety valves (PWR). The stuck open SRV may or may not cause the automatic or manual actuation of high-pressure injection systems.

1.3.10.2 Data Collection and Review

Data for the SORV (PWR) baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for SORV (PWR) is 1988–2015. Figure 1-7 shows the trend for a single SORV (PWR) data set and the baseline period used in this analysis. There were no events of 2 or more SORV (PWR) failures.

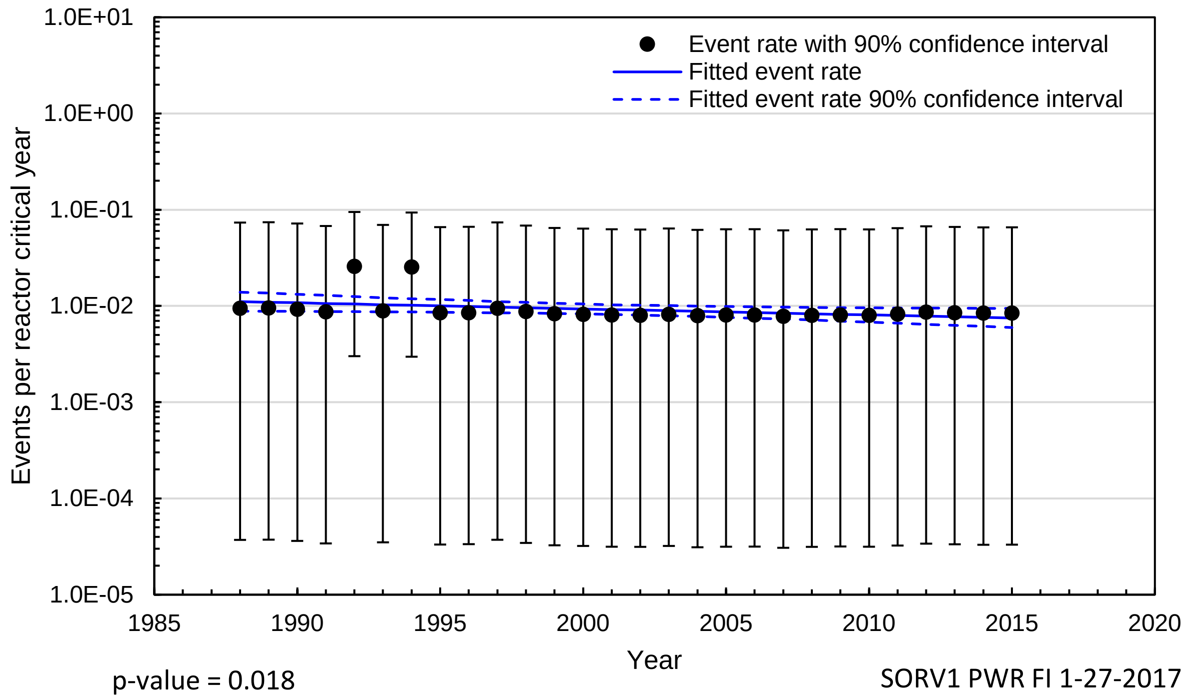

Figure 1-7. SORV (PWR) trend plot.

The RADS database was used to collect the SORV (PWR) data for that period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. Results are shown for two SORV initiating events; a single SORV (SORV1) and two or more SORVs (SORV2). Table 1-27 summarizes the data obtained from RADS and used in the SORV (PWR) analysis.

Table 1-27. SORV (PWR) frequency data for baseline period.

Event Type	Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
	Events	Reactor Critical Years (rcry)			
SORV1	2	1663	1988-2015	77	2.6%
SORV2	0	1101	1998-2015	69	0.0%

1.3.10.3 Industry-Average Baselines

Table 1-28 lists the industry-average frequency distribution. With only two events, an empirical Bayes analysis could not be performed. Therefore, the SCNID analysis results were used. This industry-average frequency does not account for any recovery.

Table 1-28. Selected industry distribution of λ for SORV (PWR).

Event Type	Source	5%	Mean	95%	Distribution		
					Type	α	β
SORV1	JNID/IL	3.45E-04	1.50E-03	3.33E-03	Gamma	2.500	1.660E+03
SORV2	JNID/IL	1.79E-06	4.54E-04	1.75E-03	Gamma	0.500	1.100E+03

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.3.11 Interfacing System Loss-of-Coolant Accident at Boiling Water Reactors

1.3.11.1 Initiating Event Description

From Reference 3, the Interfacing System LOCA (ISLOCA) initiating event is a backflow of high-pressure coolant from the primary system through low-pressure system piping which results in the breach of the pipe or component.

1.3.11.2 Data Collection and Review

Data for the ISLOCA baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for ISLOCA (BWR) is 1988–2015. The RADS database was used to collect the ISLOCA (BWR) data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. Table 1-29 summarizes the data obtained from RADS and used in the ISLOCA (BWR) analysis.

Table 1-29. ISLOCA (BWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
0	834	1988-2015	35	0.0%

1.3.11.3 Industry-Average Baselines

Table 1-30 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 1-30. Selected industry distribution of λ for ISLOCA (BWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	2.36E-06	6.00E-04	2.3E-03	Gamma	0.500	8.34E+02

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.3.12 Interfacing System Loss-of-Coolant Accident at Pressurized Water Reactors

1.3.12.1 Initiating Event Description

From Reference 3, the Interfacing System LOCA (ISLOCA) initiating event is a backflow of high-pressure coolant from the primary system through low-pressure system piping which results in the breach of the pipe or component.

1.3.12.2 Data Collection and Review

Data for the ISLOCA baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for ISLOCA (PWR) is 1988–2015. The RADS database was used to collect the ISLOCA (PWR) data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. Table 1-31 summarizes the data obtained from RADS and used in the ISLOCA (PWR) analysis.

Table 1-31. ISLOCA (PWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
0	610	1988-2015	69	0.0%

1.3.12.3 Industry-Average Baselines

Table 1-32 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 1-32. Selected industry distribution of λ for ISLOCA (PWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	1.18E-06	3.01E-04	1.16E-03	Gamma	0.500	1.66E+03

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.3.13 Reactor Coolant Pump Seal LOCA (RCPLOCA)

1.3.13.1 Initiating Event Description

From Reference 3, the Reactor Coolant Pump Seal LOCA (RCPLOCA) initiating event is a catastrophic failure the reactor coolant pump seal assembly that results in a primary coolant leak into the primary containment at a rate greater than 100 gpm. This category applies to PWRs only.

1.3.13.2 Data Collection and Review

Data for the RCPLOCA baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for RCPLOCA is 1988–2015. The RADS database was used to collect the RCPLOCA data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. Table 1-31 summarizes the data obtained from RADS and used in the RCPLOCA analysis.

Table 1-33. RCPLOCA frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
0	610	1988-2015	69	0.0%

1.3.13.3 Industry-Average Baselines

Table 1-32 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 1-34. Selected industry distribution of λ for RCPLOCA.

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	1.18E-06	3.01E-04	1.16E-03	Gamma	0.500	1.66E+03

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

1.3.14 Excessive Loss of Coolant Event (Vessel Rupture) (XLOCA)

1.3.14.1 Initiating Event Description

Excessive Loss of Coolant Event (Vessel Rupture) (XLOCA). This event represents a LOCA of such size as to be beyond the capacity of safety systems to protect the reactor core. This is considered to be a break of equivalent pipe diameter of greater than 41 inches for BWRs and 31 inches for PWRs.

1.3.14.2 Data Collection and Review

Reference 7 provided the 1.0E-7 per rcry estimate currently used in the SPAR models. A more current estimate is provided by *Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process* (Ref. 5). The LOCA frequency was estimated based on an expert elicitation process "…to consolidate service history data and PFM [probabilistic fracture mechanics] studies with knowledge of plant design, operation, and material performance."

Tables 7.17 and 7.19 of Reference 5 present frequencies for LOCAs exceeding various sizes indicated by gallon per minute (gpm) break flow and effective pipe size break. XLOCA is represented by the last entry in the tables, 41-inch breaks for BWRs and 31-inch diameter for PWRs. The frequencies are presented both for current day conditions (assuming an average of 25 years of fleet operation) and for end-of-life conditions (40 years of operation). For this estimate, frequencies for 40 years of fleet operation were used. The frequencies are provided in reactor calendar years (rcy) and are converted to reactor critical years (rcry) assuming that reactors are critical 90% of each year, and rounded using the round off scheme provided in NUREG/CR-6928. The resulting XLOCA frequencies are provided in Table 1-35.

The ACRS result is still the recommended value. The other values are provided for reference.

1.3.14.3 Industry-Average Baselines

Table 1-35 lists the industry-average frequency distribution.

Table 1-35. Selected industry distribution of λ for XLOCA.

Source	Plant Type	5%	Mean	95%	Distribution Type	α	β
Ref. 5	BWR	1.02E-14	1.00E-08	5.15E-08	Gamma	0.200	2.000E+07
Ref. 5	PWR	8.16E-14	8.00E-08	4.12E-07	Gamma	0.200	2.500E+06
Ref. 7	ALL		1.00E-07				

Note –The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

2 Transients

The general transient categories result in automatic or manual reactor trips but do not degrade safety system response.

2.1 General Transient

2.1.1 General Transient at Boiling Water Reactors (TRANS (BWR))

2.1.1.1 Initiating Event Description

From Reference 3, the General Transient at Boiling Water Reactors (TRANS (BWR)) initiating event is a general transient that results in automatic or manual reactor trips but does not degrade safety system response.

2.1.1.2 Data Collection and Review

Data for the TRAN (BWR) baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for TRANS (BWR) is 1997–2015. Figure 2-1 shows the trend of the full TRANS (BWR) data set and the baseline period used in this analysis. The RADS database was used to collect the TRANS (BWR) data for the baseline period. Only initial plant fault events as defined in Reference 3 were used. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. These results also include the individual plant results for the same period. Table 2-1 summarizes the data obtained from RADS and used in the TRANS (BWR) analysis.

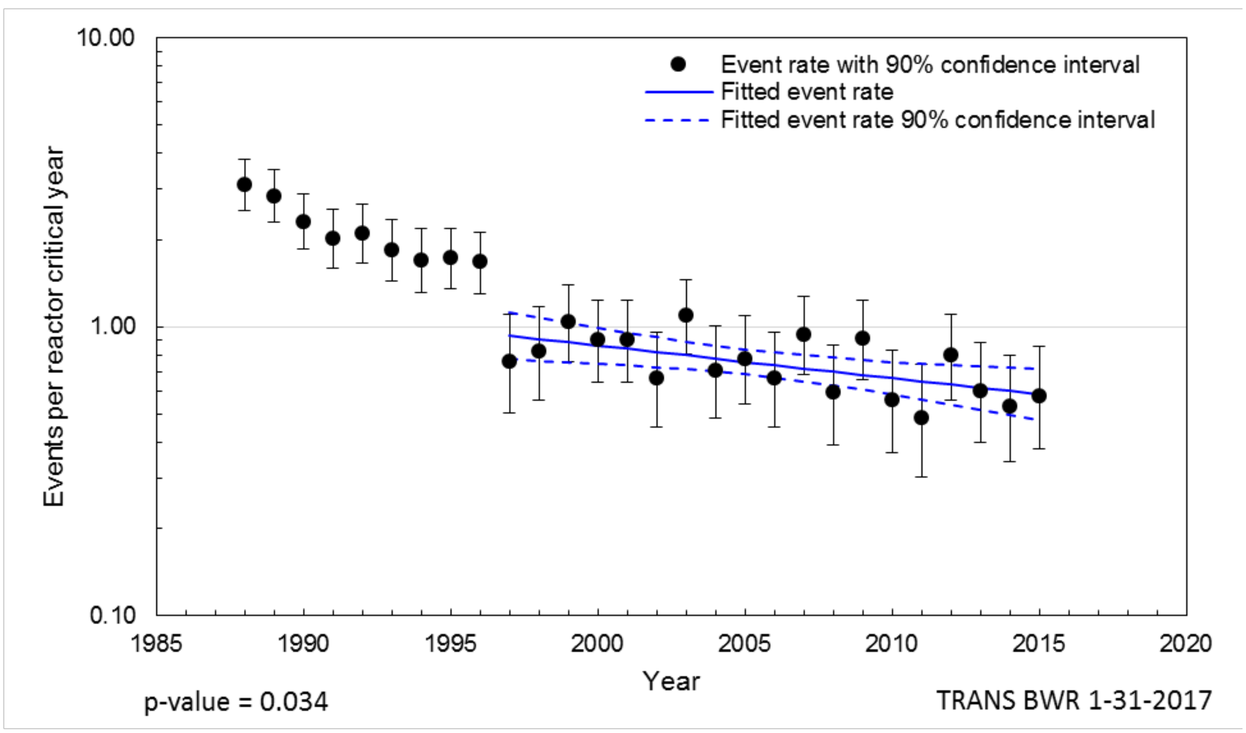

Figure 2-1. TRANS (BWR) trend plot.

Table 2-1. TRANS (BWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
441	598	1997-2015	36	97.2%

2.1.1.3 Industry-Average Baselines

Table 2-2 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 2-2. Selected industry distribution of λ for TRANS (BWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
EB/PL/KS	3.35E-01	6.76E-01	1.12E+00	Gamma	7.860	1.160E+01

Note – EB/PL/KS is an empirical Bayes analysis at the plant level with the Kass-Steffey adjustment. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

2.1.2 General Transient at Pressurized Water Reactors (TRANS (PWR))

2.1.2.1 Initiating Event Description

From Reference 3, the General Transient at Pressurized Water Reactors (TRANS (PWR)) initiating event is a general transient that results in automatic or manual reactor trips but does not degrade safety system response.

2.1.2.2 Data Collection and Review

Data for the TRANS (PWR) baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for TRANS (PWR) is 1998–2015. Figure 2-2 shows the trend of the full TRANS (PWR) data set and the baseline period used in this analysis. The RADS database was used to collect the TRANS (PWR) data for the baseline period. Only initial plant fault events as defined in Reference 3 were used. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. These results also include the individual plant results for the same period. Table 2-3 summarizes the data obtained from RADS and used in the TRANS (PWR) analysis.

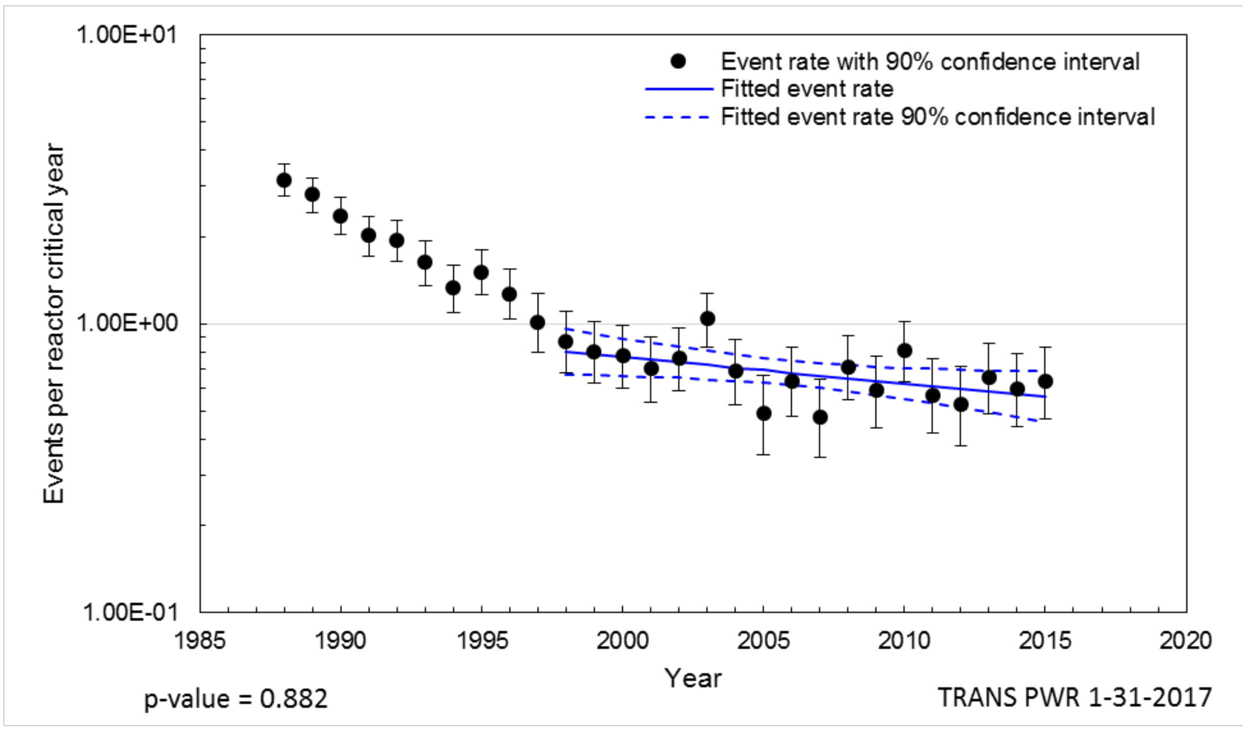

Figure 2-2. TRANS (PWR) trend plot.

Table 2-3. TRANS (PWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
743	1101	1998-2015	69	100.0%

2.1.2.3 Industry-Average Baselines

Table 2-4 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 2-4. Selected industry distribution of λ for TRANS (PWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
EB/PL/KS	3.35E-01	6.76E-01	1.12E+00	Gamma	7.860	1.160E+01

Note – EB/PL/KS is an empirical Bayes analysis at the plant level with the Kass-Steffey adjustment. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

2.2 Loss of Condenser Heat Sink

2.2.1 Loss of Condenser Heat Sink at Boiling Water Reactors (LOCHS (BWR))

2.2.1.1 Initiating Event Description

From Reference 3, the Loss of Condenser Heat Sink at Boiling Water Reactors (LOCHS (BWR)) initiating event is defined as at least one of the following:

1. A complete closure of at least one main steam isolation valve in each main steam line.
2. A decrease in condenser vacuum that leads to an automatic or manual reactor trip, or manual turbine trip; or a complete loss of condenser vacuum that prevents the condenser from removing decay heat after a reactor trip. In addition, reactor trips that are the indirect result of a low condenser vacuum, such as a loss of feedwater caused by condensate pumps tripping on high condensate temperature because of loss of vacuum, are counted.
3. The failure of one or more turbine bypass valves to maintain the reactor pressure and temperature at the desired operating condition.

2.2.1.2 Data Collection and Review

Data for the LOCHS (BWR) baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for LOCHS (BWR) is 1996–2015. Figure 2-3 shows the trend of the full LOCHS (BWR) data set and the baseline period used in this analysis. The RADS database was used to collect the LOCHS (BWR) data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. Table 2-5 summarizes the data obtained from RADS and used in the LOCHS (BWR) analysis.

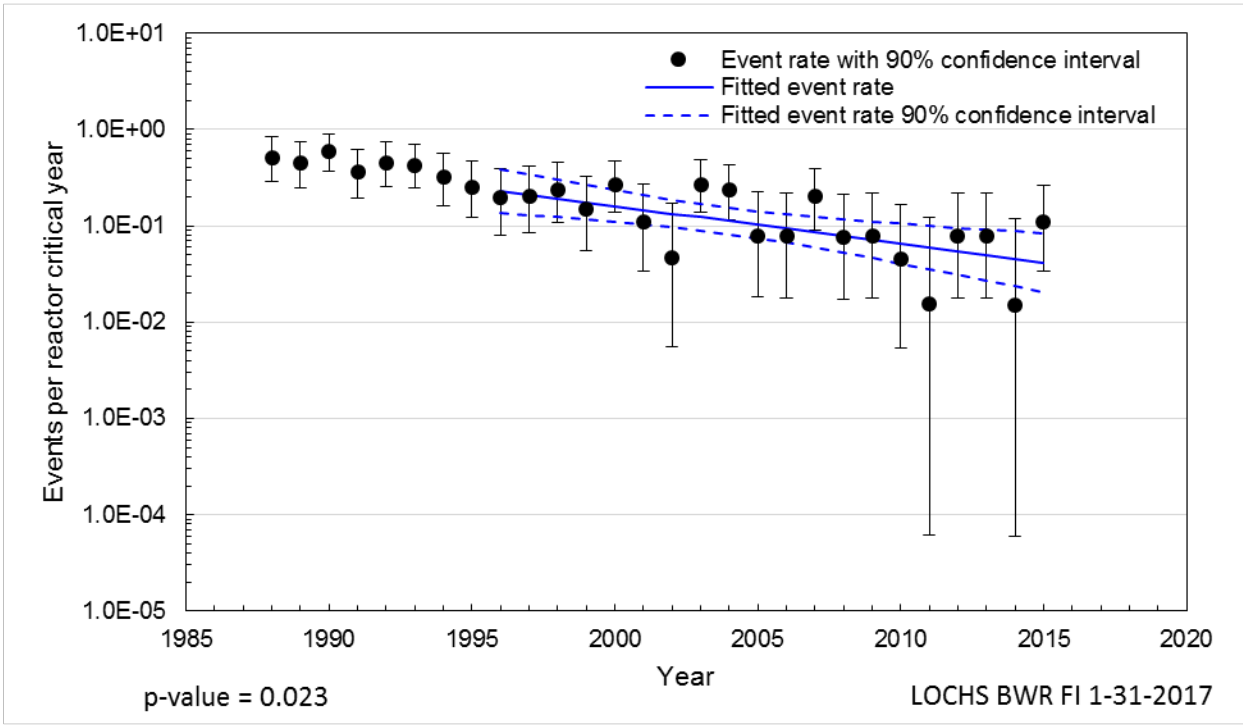

Figure 2-3. LOCHS (BWR) trend plot.

Table 2-5. LOCHS (BWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
69	627	1996-2015	36	77.8%

2.2.1.3 Industry-Average Baselines

Table 2-6 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 2-6. Selected industry distribution of λ for LOCHS (BWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
EB/PL/KS	3.51E-02	1.10E-01	2.18E-01	Gamma	3.650	3.320E+01

Note – EB/PL/KS is an empirical Bayes analysis at the plant level with the Kass-Steffey adjustment. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

2.2.2 Loss of Condenser Heat Sink at Pressurized Water Reactors (LOCHS (PWR))

2.2.2.1 Initiating Event Description

From Reference 3, the Loss of Condenser Heat Sink at Pressurized Water Reactors (LOCHS (PWR)) initiating event is defined as at least one of the following:

1. A complete closure of at least one main steam isolation valve in each main steam line.
2. A decrease in condenser vacuum that leads to an automatic or manual reactor trip, or manual turbine trip; or a complete loss of condenser vacuum that prevents the condenser from removing decay heat after a reactor trip. In addition, reactor trips that are the indirect result of a low condenser vacuum, such as a loss of feedwater caused by condensate pumps tripping on high condensate temperature because of loss of vacuum, are counted.
3. The failure of one or more turbine bypass valves to maintain the reactor pressure and temperature at the desired operating condition.

2.2.2.2 Data Collection and Review

Data for the LOCHS (PWR) baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for LOCHS (PWR) is 1995–2015. Figure 2-4 shows the trend of the full LOCHS (PWR) data set and the baseline period used in this analysis. The RADS database was used to collect the LOCHS (PWR) data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. Table 2-7 summarizes the data obtained from RADS and used in the LOCHS (PWR) analysis.

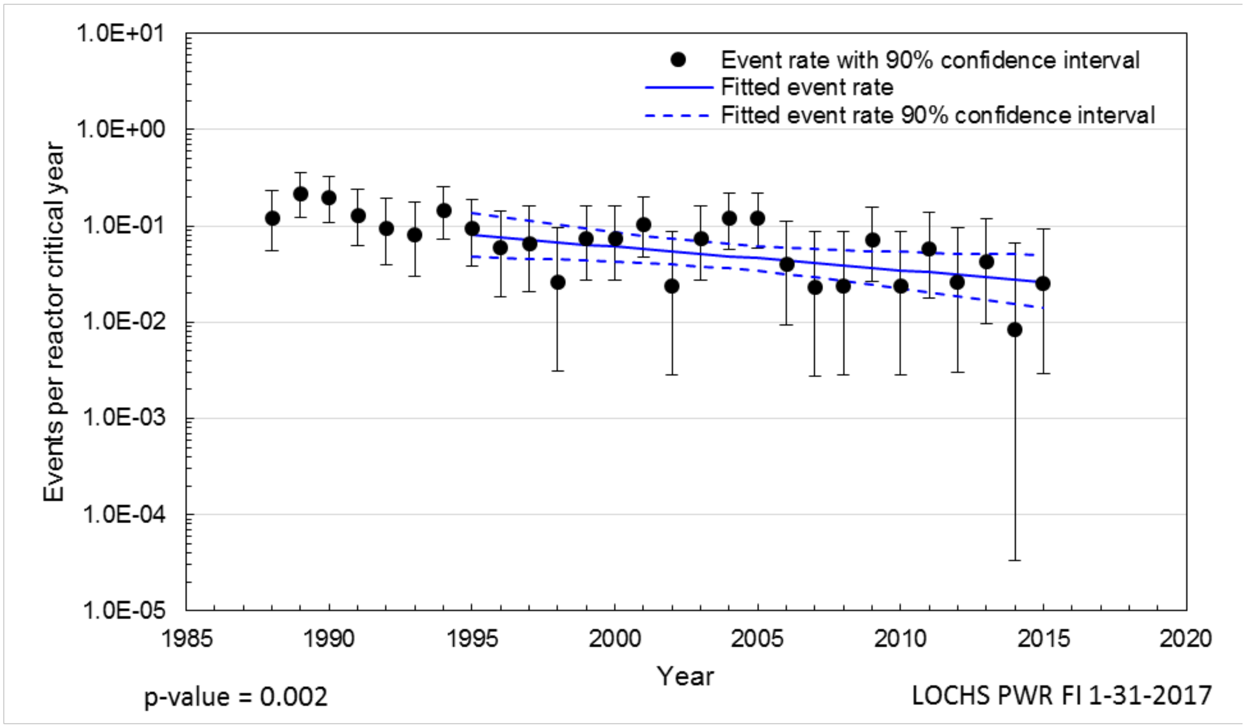

Figure 2-4. LOCHS (PWR) trend plot.

Table 2-7. LOCHS (PWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
61	1271	1995-2015	73	47.9%

2.2.2.3 Industry-Average Baselines

Table 2-8 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 2-8. Selected industry distribution of λ for LOCHS (PWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
EB/PL/KS	1.11E-02	4.82E-02	1.07E-01	Gamma	2.510	5.210E+01

Note – EB/PL/KS is an empirical Bayes analysis at the plant level with the Kass-Steffey adjustment. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

2.3 Loss of Main Feedwater (LOMFW)

2.3.1 Initiating Event Description

From Reference 3, the Loss of Main Feedwater (LOMFW) initiating event is a complete loss of all main feedwater flow. Examples include the following: trip of the only operating feedwater pump while operating at reduced power; the loss of a startup or an auxiliary feedwater pump normally used during plant startup; the loss of all operating feed pumps due to trips caused by low suction pressure, loss of seal water, or high water level (boiling water reactor vessel level or pressurized water reactor steam generator level); anticipatory reactor trip due to loss of all operating feed pumps; and manual reactor trip in response to feed problems characteristic of a total loss of feedwater flow, but prior to automatic reactor protection system signals. This category also includes the inadvertent isolation or closure of all feedwater control valves prior to the reactor trip; however, a main feedwater isolation caused by valid automatic system response after a reactor trip is not included. This category does not include the total loss of feedwater caused by the loss of offsite power.

2.3.2 Data Collection and Review

Data for the LOMFW baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for LOMFW is 1993–2015. Figure 2-5 shows the trend of the full LOMFW data set and the baseline period used in this analysis. The RADS database was used to collect the LOMFW data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. Table 2-9 summarizes the data obtained from RADS and used in the LOMFW analysis.

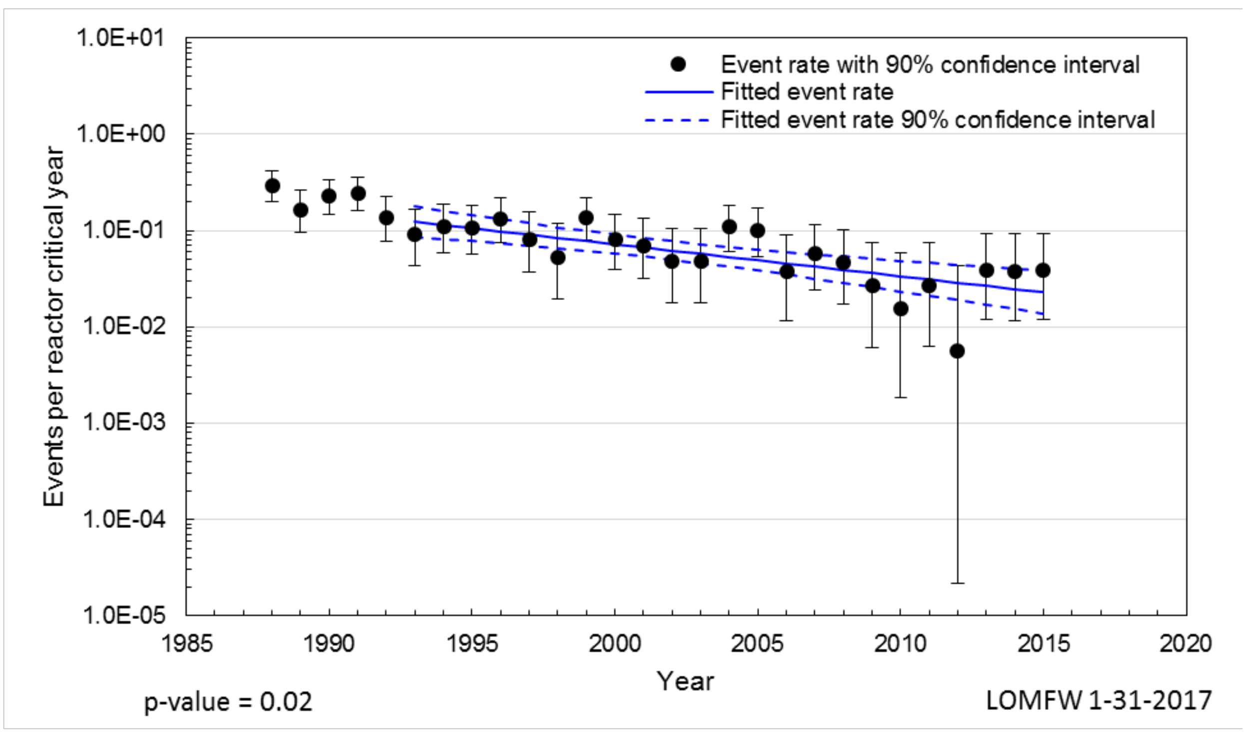

Figure 2-5. LOMFW trend plot.

Table 2-9. LOMFW frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
124	2096	1993-2015	110	55.5%

2.3.3 Industry-Average Baselines

Table 2-10 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 2-10. Selected industry distribution of λ for LOMFW.

Source	5%	Mean	95%	Distribution		
				Type	α	β
EB/PL/KS	1.11E-02	5.94E-02	1.39E-01	Gamma	2.080	3.500E+01

Note – EB/PL/KS is an empirical Bayes analysis at the plant level with the Kass-Steffey adjustment. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

3 Loss of Support Systems

3.1 Loss of Safety-Related Cooling Water

3.1.1 Loss of Standby (Emergency) Service Water (LOSWS)

3.1.1.1 Initiating Event Description

From Reference 3, the Loss of Service Water System (LOSWS) initiating event is a total loss of service water flow. The service water system (SWS) can be an open-cycle or a closed-cycle cooling water system. An open-cycle SWS takes suction from the plant's ultimate heat sink (e.g., the ocean, bay, lake, pond or cooling towers), removes heat from safety-related systems and components, and discharges the water back to the ultimate heat sink. A closed-cycle or intermediate SWS removes heat from safety-related equipment and discharges the heat through a heat exchanger to an open-cycle service water system.

For this report, the definition was specialized to include only emergency service water (ESW) systems. Therefore, the initiating event is Loss of Emergency Service Water (LOESW).

3.1.1.2 Data Collection and Review

Data for the LOESW baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for LOESW is 1988–2015. (There were no events.) The RADS database was used to collect the LOESW data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. These results also include the individual plant results for the same period. Table 3-1 summarizes the data obtained from RADS and used in the LOESW analysis.

Table 3-1. LOESW frequency data.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
0	2496	1988-2015	114	0.0%

3.1.1.3 Industry-Average Baselines

Table 3-2 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 3-2. Selected industry distribution of λ for LOESW.

Source	5%	Median	Mean	95%	Distribution		
					Type	α	β
JNID/IL	7.86E-07	9.10E-05	2.00E-04	7.68E-04	Gamma	0.500	2.500E+03

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

3.1.2 Partial Loss of Standby (Emergency) Service Water (PLOSWS)

3.1.2.1 Initiating Event Description

From Reference 3, the Partial Loss of Service Water System (PLOSWS) initiating event is a loss of one train of a multiple train system or partial loss of a single train system that impairs the ability of the system to perform its function. Examples include pump cavitation, strainer fouling, and piping rupture.

This category does not include loss of a redundant component in a SWS as long as the remaining, similar components provide the required level of performance. For example, a loss of a single SWS pump is not classified as a PLOSWS as long as the remaining operating or standby pumps can provide the required level of performance. A loss of service water to a single component in another system because of a blockage or incorrect line-up that does not affect the cooling to other components serviced by the train is not included under this category, but is instead classified as a failure of the system that the single component serves.

For this report, the definition was specialized to include only emergency service water (ESW) systems; therefore, the initiating event is Partial Loss of Emergency Service Water (PLOESW).

3.1.2.2 Data Collection and Review

Data for the PLOESW baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for PLOESW is 1988–2015. (With only four events, the entire period is chosen for the baseline.) Figure 3-1 shows the trend of the full PLOESW data set and the baseline period used in this analysis. The RADS database was used to collect the PLOESW data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. These results also include the individual plant results for the same period. Table 3-3 summarizes the data obtained from RADS and used in the PLOESW analysis.

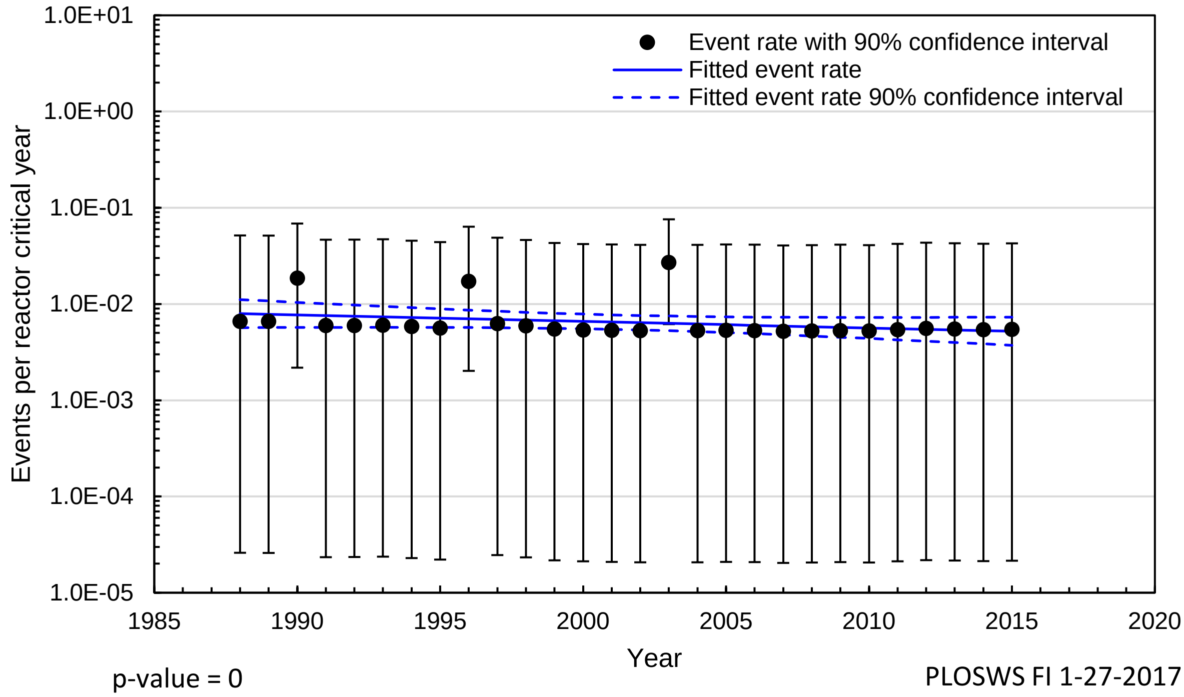

Figure 3-1. PLOSWS trend plot.

Table 3-3. PLOESW frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
4	2496	1988-2015	114	3.5%

3.1.2.3 Industry-Average Baselines

Table 3-4 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 3-4. Selected industry distribution of λ for PLOESW.

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	6.65E-04	1.80E-03	3.38E-03	Gamma	4.500	2.500E+03

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

3.1.3 Loss of Component Cooling Water (LOCCW)

3.1.3.1 Initiating Event Description

From Reference 3, the Loss of Component Cooling Water (LOCCW) initiating event is a complete loss of the component cooling water (CCW) system. CCW is a closed-cycle cooling water system that removes heat from safety-related equipment and discharges the heat through a heat exchanger to an open-cycle service water system.

3.1.3.2 Data Collection and Review

Data for LOCCW baselines were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for LOCCW is 1988–2015. (No events were identified, so the entire period was chosen for the baseline.) The RADS database was used to collect the LOCCW data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. These results also include the individual plant results for the same period. Table 3-5 summarizes the data obtained from RADS and used in the LOCCW analysis.

Table 3-5. LOCCW frequency data.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
0	2496	1988-2015	114	0.0%

3.1.3.3 Industry-Average Baselines

Table 3-6 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 3-6. Selected industry distribution of λ for LOCCW.

Source	5%	Median	Mean	95%	Distribution		
					Type	α	β
JNID/IL	7.86E-07	9.10E-05	2.00E-04	7.68E-04	Gamma	0.500	2.500E+03

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

3.1.4 Partial Loss of Component Cooling Water System (PLOCCW)

3.1.4.1 Initiating Event Description

From Reference 3, the Partial Loss of Component Cooling Water System (PLOCCW) initiating event is a loss of one train of a multiple train system or partial loss of a single train system that impairs the ability of the system to perform its function. Examples include pump cavitation, filter fouling, and piping rupture. The component cooling water (CCW) is a closed-cycle cooling water system that removes heat from safety-related equipment and discharges the heat through a heat exchanger to an open-cycle service water system.

These categories do not include a loss of a redundant component in a CCW as long as the remaining, similar components provide the required level of performance. For example, a loss of a single CCW pump is not classified as a partial loss of a CCW as long as the remaining operating or standby pumps can provide the required level of performance. A loss of CCW to a single component in another system because of a blockage or incorrect line-up that does not affect the cooling to other components serviced by the train is not included under this category, but is instead classified as a failure of the system that the single component serves.

3.1.4.2 Data Collection and Review

Data for the PLOCCW baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for PLOCCW is 1988–2015. (With only one event, the entire period is chosen for the baseline.) Figure 3-2 shows the trend of the full PLOCCW data set and the baseline period used in this analysis. The RADS database was used to collect the PLOCCW data for the baseline period. Results include total number of events and total reactor critical years (rcry’s) for the U.S. commercial nuclear power plant industry. These results also include the individual plant results for the same period. Table 3-7 summarizes the data obtained from RADS and used in the PLOCCW analysis.

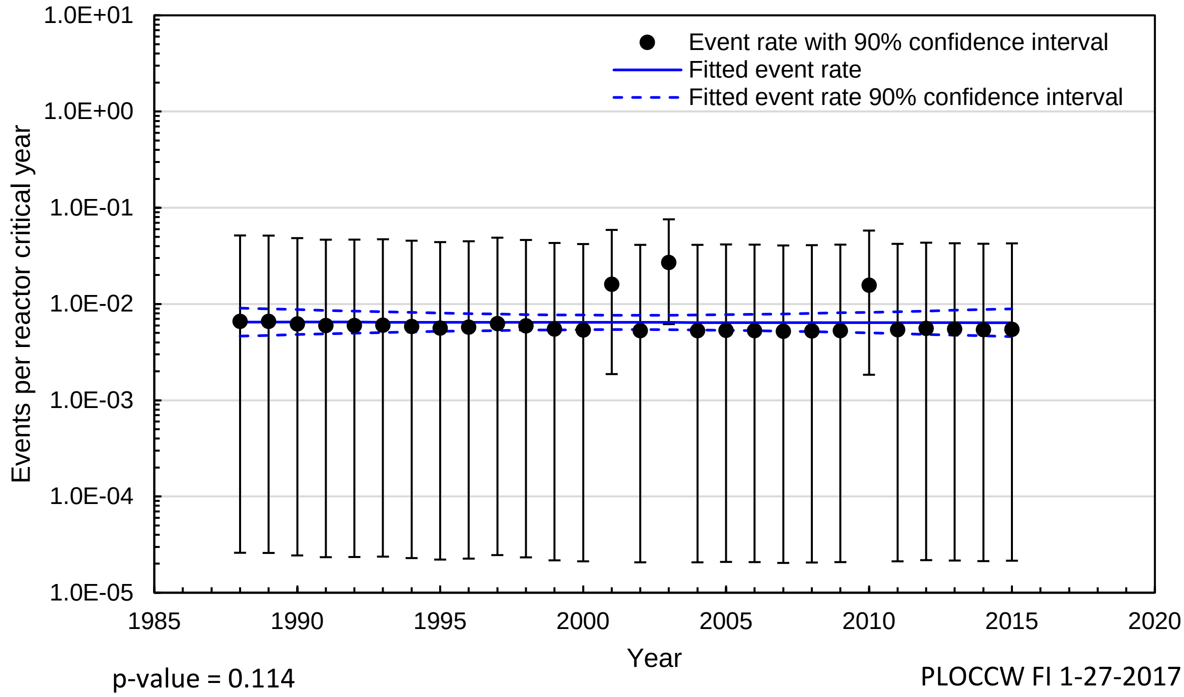

Figure 3-2. PLOCCW trend plot.

Table 3-7. PLOCCW frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
4	2496	1988-2015	114	3.5%

3.1.4.3 Industry-Average Baselines

Table 3-8 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 3-8. Selected industry distribution of λ for PLOCCW.

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	6.65E-04	1.80E-03	3.38E-03	Gamma	4.500	2.500E+03

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

3.2 Loss of Instrument Control Air

3.2.1 Loss of Instrument Air at Boiling Water Reactors (LOIA (BWR))

3.2.1.1 Initiating Event Description

From Reference 3, the Loss of Instrument Air at Boiling Water Reactors (LOIA (BWR)) initiating event is a total or partial loss of an instrument or control air system that leads to a reactor trip or occurs shortly after the reactor trip. Examples include ruptured air headers, damaged air compressors with insufficient backup capability, losses of power to air compressors, line fitting failures, improper system line-ups, and undesired operations of pneumatic devices in other systems caused by low air header pressure.

3.2.1.2 Data Collection and Review

Data for the LOIA (BWR) baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for LOIA (BWR) is 1991–2015. Figure 3-3 shows the trend of the full LOIA (BWR) data set and the baseline period used in this analysis. The RADS database was used to collect the LOIA (BWR) data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. These results also include the individual plant results for the same period. Table 3-9 summarizes the data obtained from RADS and used in the LOIA (BWR) analysis.

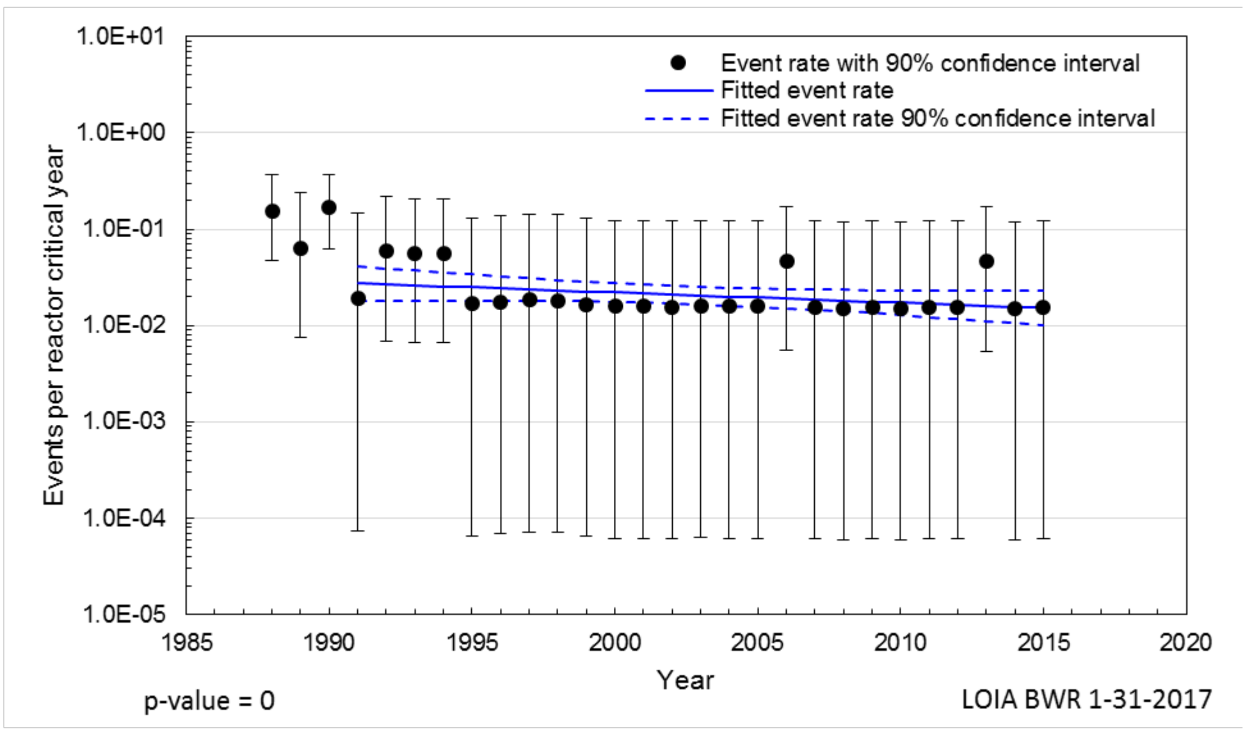

Figure 3-3. LOIA (BWR) trend plot.

Table 3-9. LOIA (BWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
5	761	1991-2015	37	13.5%

3.2.1.3 Industry-Average Baselines

Table 3-10 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 3-10. Selected industry distribution of λ for LOIA (BWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	3.01E-03	7.23E-03	1.29E-02	Gamma	5.500	7.610E+02

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

3.2.2 Loss of Instrument Air at Pressurized Water Reactors (LOIA (PWR))

3.2.2.1 Initiating Event Description

From Reference 3, the Loss of Instrument Air at Pressurized Water Reactors (LOIA (PWR)) initiating event is a total or partial loss of an instrument or control air system that leads to a reactor trip or occurs shortly after the reactor trip. Examples include ruptured air headers, damaged air compressors with insufficient backup capability, losses of power to air compressors, line fitting failures, improper system line-ups, and undesired operations of pneumatic devices in other systems caused by low air header pressure.

3.2.2.2 Data Collection and Review

Data for the LOIA (PWR) baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for LOIA (PWR) is 1997–2015. Figure 3-4 shows the trend of the full LOIA (PWR) data set and the baseline period used in this analysis. The RADS database was used to collect the LOIA (PWR) data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. These results also include the individual plant results for the same period. Table 3-11 summarizes the data obtained from RADS and used in the LOIA (PWR) analysis.

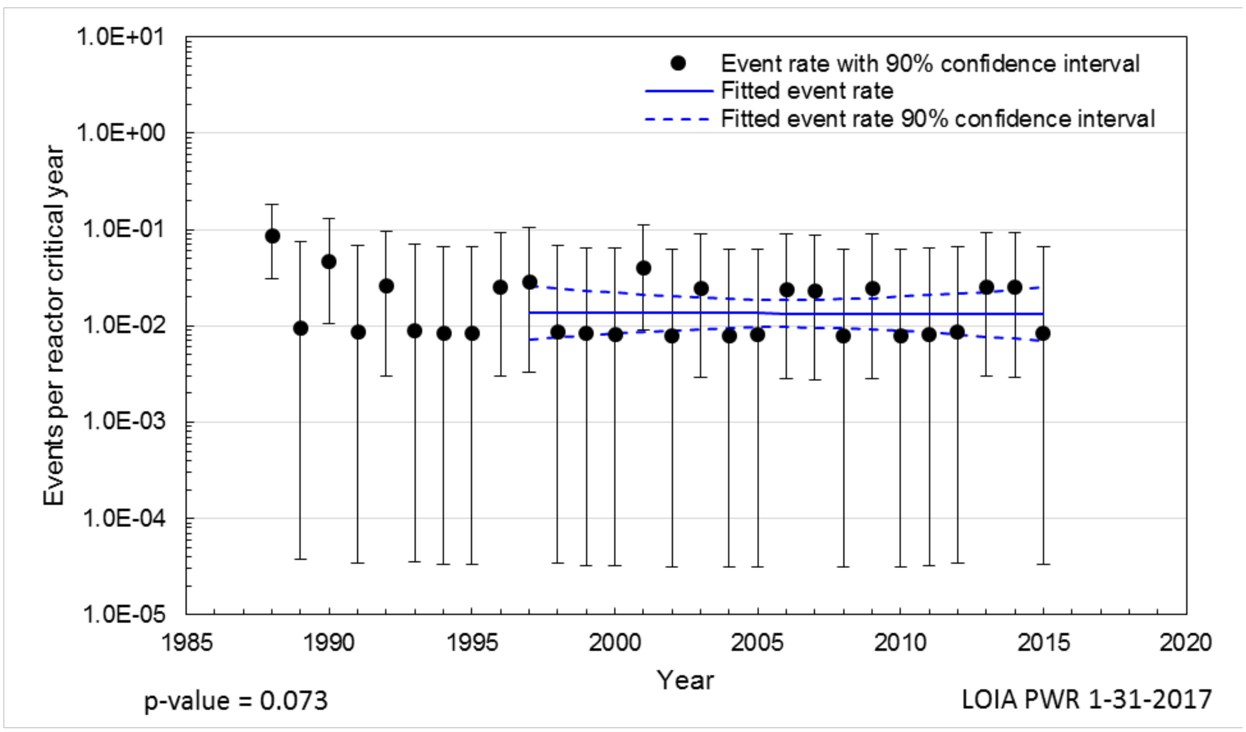

Figure 3-4. LOIA (PWR) trend plot.

Table 3-11. LOIA (PWR) frequency data for baseline period.

Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
Events	Reactor Critical Years (rcry)			
9	1154	1997-2015	70	10.0%

3.2.2.3 Industry-Average Baselines

Table 3-12 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 3-12. Selected industry distribution of λ for LOIA (PWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	4.40E-03	8.24E-03	1.31E-02	Gamma	9.500	1.150E+03

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

4 Loss of Offsite Power

4.1 Loss of Offsite Power, Power Operations

4.1.1 Initiating Event Description

From Reference 3, the Loss of Offsite Power, Power Operations (LOOP.PO) initiating event is a simultaneous loss of electrical power to all safety-related buses that causes emergency power generators to start and supply power to the safety-related buses. The offsite power boundary extends from the offsite electrical power grid to the output breaker (inclusive) of the step-down transformer that feeds the first safety-related bus with an emergency power generator. The plant switchyard and service-type transformers are included within the offsite power boundary. This category includes the momentary or prolonged degradation of grid voltage that causes all emergency power generators to start (if operable) and load onto their associated safety-related buses (if available).

This category does not include a LOOP event that occurs while the plant is shutdown. In addition, it does not include any momentary undervoltage event that results in the automatic start of all emergency power generators, but in which the generators do not tie on to their respective buses due to the short duration of the undervoltage.

4.1.2 Data Collection and Review

Data for the LOOP.PO baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for LOOP.PO is 1997–2015. Table 4-1 summarizes the data used in the LOOP.PO analysis. Figure 4-1 shows the trend of the full LOOP.PO data set and the baseline period used in this analysis. The RADS database was used to collect the LOOP.PO data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. These results also include the individual plant results for the same period. Table 4-1 summarizes the data obtained from RADS and used in the LOOP.PO analysis.

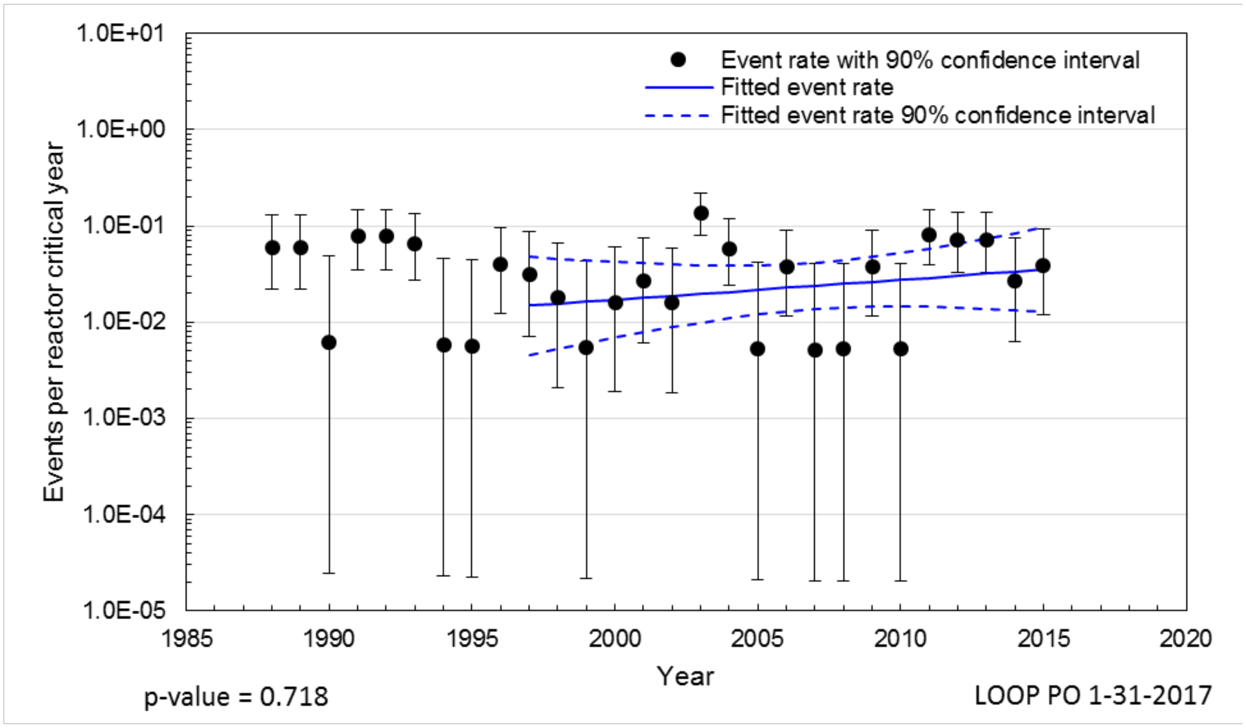

Figure 4-1. LOOP.PO (Power Operations) trend plot.

Table 4-1. LOOP frequency data for baseline period.

LOOP Category	Data After Review		Baseline Period	Counts Number of Plants	Percent of Plants with Events
	Events	Reactor Critical Years (rcry)			
PO.LOOP	54	1752	1997-2015	106	44.3%
PO.LOOP-GR	18	1752	1997-2015	10	50.0%
PO.LOOP-PC	3	1752	1997-2015	106	2.8%
PO.LOOP-SC	23	1752	1997-2015	106	21.7%
PO.LOOP-WR	13	2567	1986-2015	114	10.5%

4.1.3 Industry-Average Baselines

Table 4-2 lists the industry-average frequency distributions for the four LOOP categories and total LOOP. These industry-average frequencies do not account for any recovery.

Table 4-2. Selected industry distributions of λ for LOOP.

Event	Source	5%	Mean	95%	Distribution		
					Type	α	β
PO.LOOP	JNID/IL	2.45E-02	3.11E-02	3.84E-02	Gamma	54.50	1.750E+03
PO.LOOP-GR	EB/PP/KS	1.10E-04	1.10E-02	3.94E-02	Gamma	0.61	5.530E+01
PO.LOOP-PC	JNID/IL	6.19E-04	2.00E-03	4.02E-03	Gamma	3.50	1.750E+03
PO.LOOP-SC	JNID/IL	9.22E-03	1.34E-02	1.83E-02	Gamma	23.50	1.750E+03
PO.LOOP-WR	EB/PL/KS	7.86E-04	5.08E-03	1.25E-02	Gamma	1.80	3.540E+02

Note – EB/PL/KS is an empirical Bayes analysis at the plant level with the Kass-Steffey adjustment. JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

4.2 Loss of Offsite Power, Shutdown Operations

4.2.1 Initiating Event Description

From Reference 3, the Loss of Offsite Power, Shutdown Operations (LOOP.SD) initiating event is a simultaneous loss of electrical power to all safety-related buses that causes emergency power generators to start and supply power to the safety-related buses. The offsite power boundary extends from the offsite electrical power grid to the output breaker (inclusive) of the step-down transformer that feeds the first safety-related bus with an emergency power generator. The plant switchyard and service-type transformers are included within the offsite power boundary. This category includes the momentary or prolonged degradation of grid voltage that causes all emergency power generators to start (if operable) and load onto their associated safety-related buses (if available).

This category does not include a LOOP event that occurs while the plant is at power. In addition, it does not include any momentary under-voltage event that results in the automatic start of all emergency power generators, but in which the generators do not tie on to their respective buses due to the short duration of the under-voltage.

4.2.2 Data Collection and Review

Data for the LOOP.SD baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for LOOP.SD is 1997–2015. Table 4-3 summarizes the data used in the LOOP.SD analysis. Figure 4-2 shows the trend of the full LOOP.SD data set and the baseline period used in this analysis. The RADS database was used to collect the LOOP.SD data for the baseline period. Results include total number of events and total reactor shutdown years for the U.S. commercial nuclear power plant industry. These results also include the individual plant results for the same period. Table 4-4 summarizes the data obtained from RADS and used in the LOOP.SD analysis.

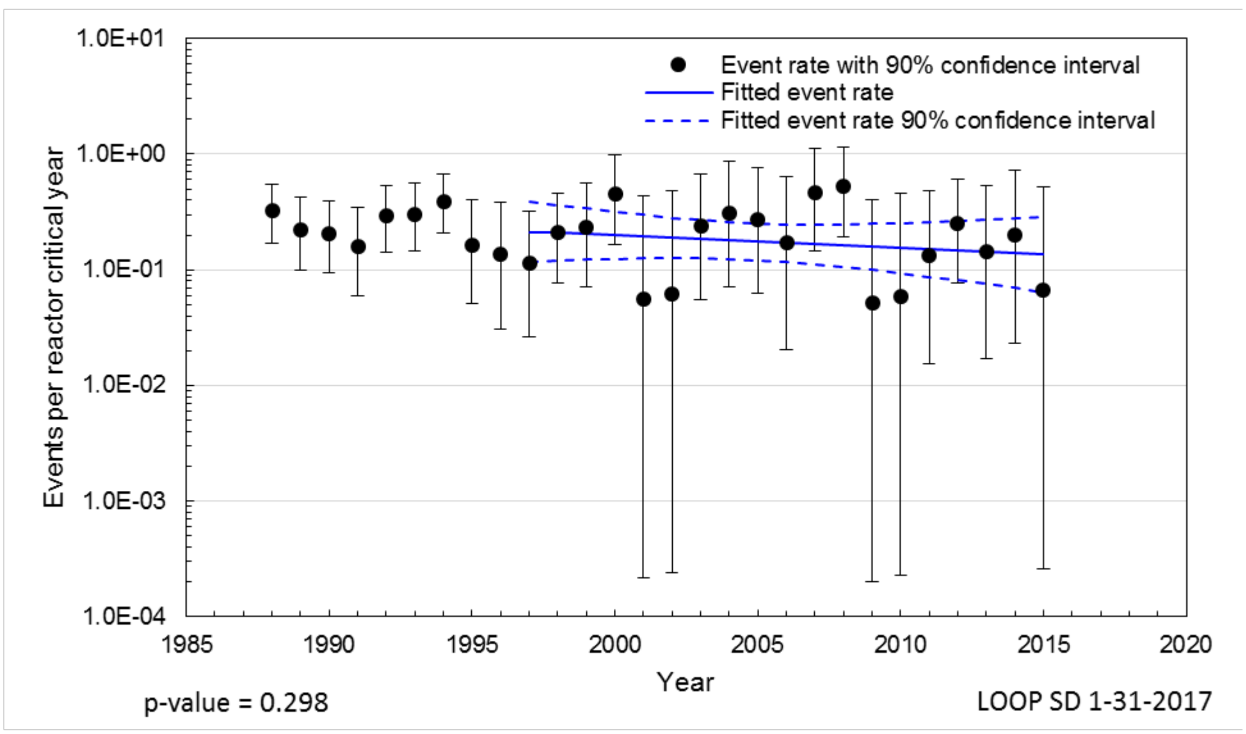

Figure 4-2. LOOP.SD (Shutdown Operations) trend plot.

Table 4-3. LOOP.SD frequency data for baseline period.

LOOP Category	Data After Review		Baseline Period	Counts Number of Plants	Percent of Plants with Events
	Events	Reactor Shutdown Years			
SD.LOOP	36	214	1997-2015	110	26.4%
SD.LOOP-GR	6	468	1986-2015	10	40.0%
SD.LOOP-PC	23	468	1986-2015	114	15.8%
SD.LOOP-SC	17	214	1997-2015	110	14.5%
SD.LOOP-WR	16	468	1986-2015	114	11.4%

4.2.3 Industry-Average Baselines

Table 4-4 lists the industry-average frequency distributions for the four LOOP.SD categories and total LOOP.SD. These industry-average frequencies do not account for any recovery.

Table 4-4. Selected industry distributions of λ for LOOP.SD.

Event	Source	5%	Mean	95%	Distribution		
					Type	α	β
SD.LOOP	EB/PL/KS	5.12E-02	1.69E-01	3.43E-01	Gamma	3.40	2.010E+01
SD.LOOP-GR	JNID/IL	6.29E-03	1.39E-02	2.39E-02	Gamma	6.50	4.680E+02
SD.LOOP-PC	EB/PL/KS	2.04E-03	4.80E-02	1.48E-01	Gamma	0.93	1.930E+01
SD.LOOP-SC	JNID/IL	5.27E-02	8.20E-02	1.17E-01	Gamma	17.50	2.130E+02
SD.LOOP-WR	EB/PL/KS	2.60E-04	3.39E-02	1.24E-01	Gamma	0.57	1.690E+01

Note – EB/PL/KS is an empirical Bayes analysis at the plant level with the Kass-Steffey adjustment. JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

5 Electrical Power

5.1 Loss of Safety-Related AC Bus

5.1.1 Loss of Vital AC Bus (LOAC)

5.1.1.1 Initiating Event Description

From Reference 3, the Loss of Vital AC Bus (LOAC) initiating event is any sustained de-energization of a safety-related bus due to the inability to connect to any of the normal or alternative electrical power supplies. The bus must be damaged or its power source unavailable for reasons beyond an open, remotely-operated feeder-breaker from a live power source. Examples include supply cable grounds, failed insulators, damaged disconnects, transformer deluge actuations, and improper uses of grounding devices.

5.1.1.2 Data Collection and Review

Data for the LOAC baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for LOAC is 1992–2015. Figure 5-1 shows the trend of the full LOAC data set and the baseline period used in this analysis. The RADS database was used to collect the LOAC data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. These results also include the individual plant results for the same period. Table 5-1 summarizes the baseline data obtained from RADS and used in the LOAC analysis.

The LOAC results shown here in Table 5-1 and Table 5-2 include a calculated value to adjust the LOAC frequency to use in PRA models where the LOAC initiator can be caused by more than a single AC bus. The calculated value (LOAC2) consists of dividing the mean by two and recalculating the uncertainty using an alpha parameter of 0.3.

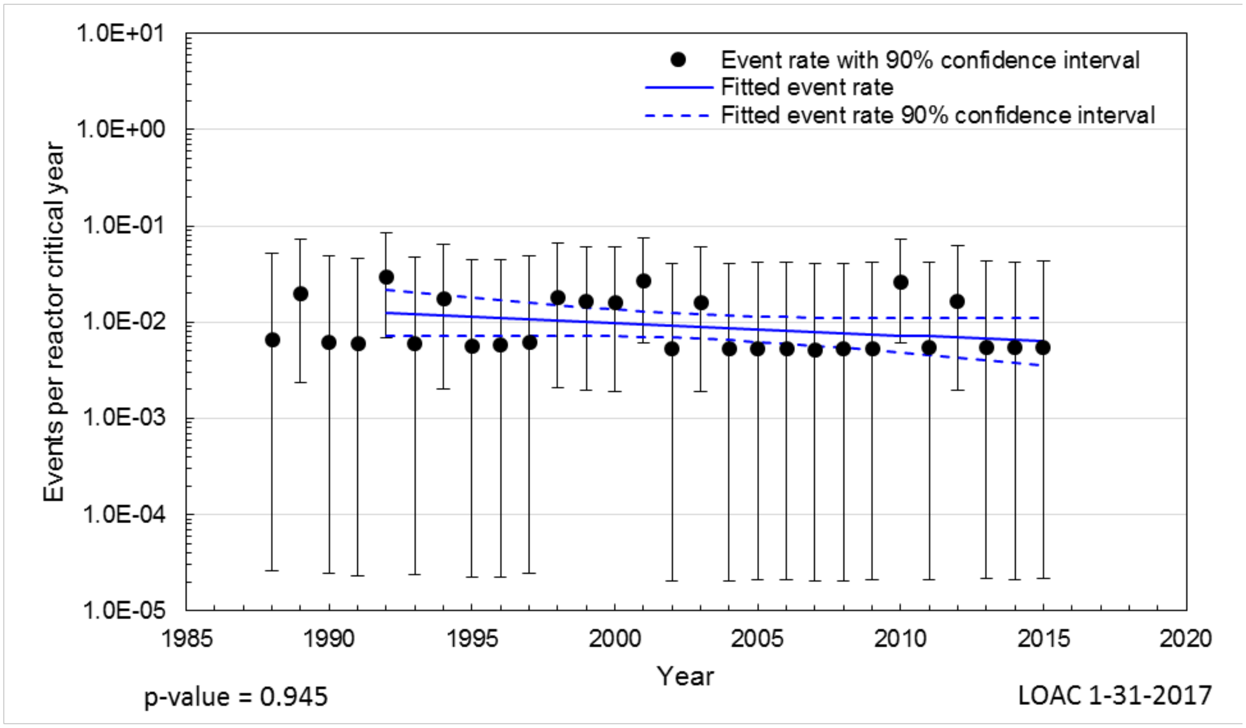

Figure 5-1. LOAC trend plot.

Table 5-1. LOAC frequency data for baseline period.

IE	Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
	Events	Reactor Critical Years (rcry)			
LOAC	12	2180	1992-2015	112	10.7%
LOAC 4160V FI	7	2180	1992-2015	112	6.3%
LOAC LOWV FI	5	2180	1992-2015	112	4.5%
LOACB2	12	2180	1992-2015	112	10.7%

5.1.1.3 Industry-Average Baselines

Table 5-2 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 5-2. Selected industry distribution of λ for LOAC.

IE	Source	5%	Mean	95%	Distribution		
					Type	α	β
LOAC	JNID/IL	3.35E-03	5.73E-03	8.64E-03	Gamma	12.500	2.180E+03
LOAC 4160V FI	JNID/IL	1.67E-03	3.44E-03	5.73E-03	Gamma	7.500	2.180E+03
LOAC LOWV FI	JNID/IL	1.05E-03	2.52E-03	4.51E-03	Gamma	5.500	2.180E+03
LOACB2	JNID/IL	3.07E-07	2.87E-03	1.31E-02	Gamma	0.300	1.047E+02

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

5.1.2 Loss of Vital DC Bus (LODC)

5.1.2.1 Initiating Event Description

From Reference 3, the Loss of Vital DC Bus (LODC) initiating event is any sustained de-energization of a safety-related bus due to the inability to connect to any of the normal or alternative electrical power supplies. The bus must be damaged or its power source unavailable for reasons beyond an open, remotely-operated feeder-breaker from a live power source. Examples include supply cable grounds, failed insulators, damaged disconnects, transformer deluge actuations, and improper uses of grounding devices.

5.1.2.2 Data Collection and Review

Data for the LODC baseline were obtained from the IEDB, as accessed using RADS. Using the process outlined in Section D.1.2 of Reference 6, the optimized baseline period for LODC is 1988–2015. (With only one event, the entire period is used for the baseline.) Figure 5-2 shows the trend of the full LODC data set and the baseline period used in this analysis. The RADS database was used to collect the LODC data for the baseline period. Results include total number of events and total reactor critical years (rcry's) for the U.S. commercial nuclear power plant industry. These results also include the individual plant results for the same period. Table 5-3 summarizes the data obtained from RADS and used in the LODC analysis.

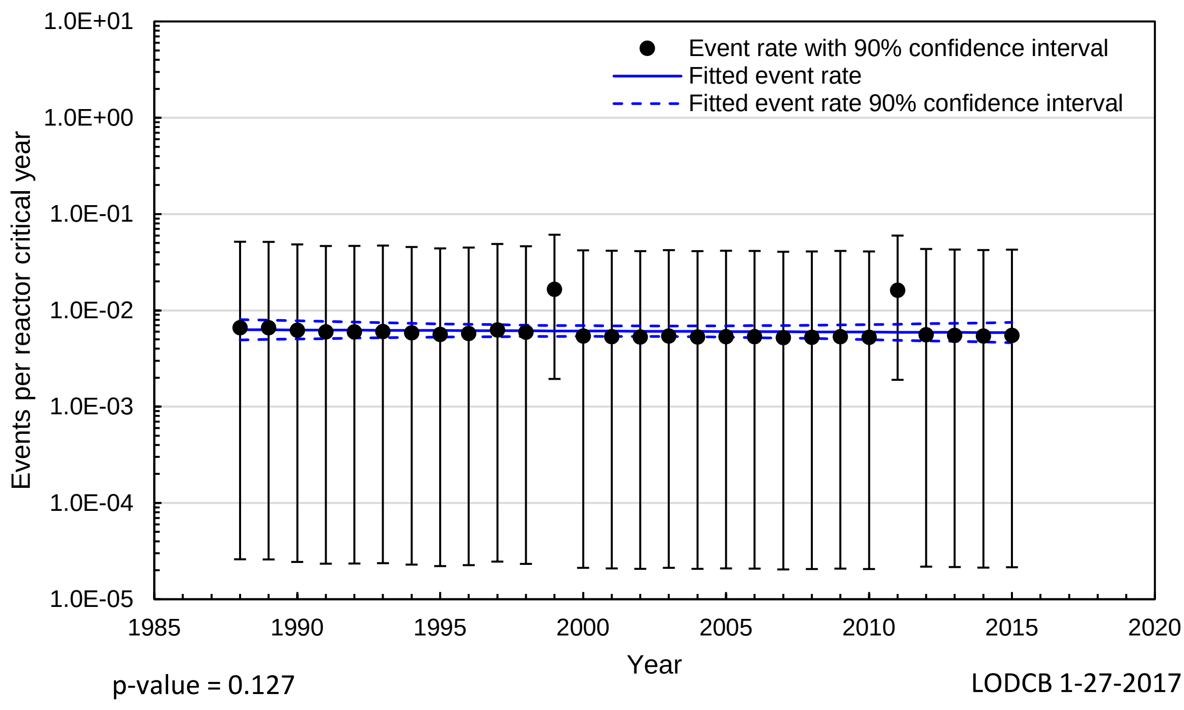

Figure 5-2. LODC trend plot.

The LODC results shown here in Table 5-2 and Table 5-4 include a calculated value to adjust the LODC frequency to use in PRA models where the LODC initiator can be caused by more than a single DC bus. The calculated value (LODC2) consists of dividing the mean by two and recalculating the uncertainty using an alpha parameter of 0.3.

Table 5-3. LODC frequency data for baseline period.

IE	Data After Review		Baseline Period	Number of Plants	Percent of Plants with Events
	Events	Reactor Critical Years (rcry)			
LODC	2	2496	1988-2015	114	1.8%
LODCB2	2	2496	1988-2015	114	1.8%

5.1.2.3 Industry-Average Baselines

Table 5-4 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 5-4. Selected industry distribution of λ for LODC.

IE	Source	5%	Mean	95%	Distribution		
					Type	α	β
LODC	JNID/IL	2.29E-04	1.00E-03	2.21E-03	Gamma	2.500	2.500E+03
LODCB2	JNID/IL	5.35E-08	5.00E-04	2.29E-03	Gamma	0.300	6.000E+02

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

6 References

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