Gentilly-2 Internal Initiating Events Frequencies Estimation for Level 1 PSA: The Need for Shared Generic CANDU Frequencies

K. Joober*[1], A. Bellil[1], J-F. Bolduc[1], T. Houasnia[2] and R. Vaillancourt[2]
[1] Énergie & analyses Énaq du Québec ltée
[2] Hydro-Québec Centrale nucléaire Gentilly-2 (G-2)
[*] Corresponding author: joober.khaled@sympatico.ca

Abstract

One of the most commonly used methods to derive site specific Initiating Events (IEs) frequencies is to obtain statistical estimates based on operating experience data (OPEX) as reported by the station. Likely, IEs occurrences are not that common to be able to estimate the IE frequency solely based on site data. Thus, a wider generic data source is usually needed and combined with the site data using a Bayesian approach in order to have statistically meaningful frequencies.

G-2 site specific events are analyzed from the station in-service date in 1983 to December 31st 2006. Generic events from the Canadian station are analyzed for five years (2002 to 2006) based on Licensee Event Report (LER) found in « COG OPEX database ». The IE frequency is then estimated using a Bayesian update process.

1. Introduction

An important task in the development of the Probabilistic Safety Assessment (PSA) is the initiating events analysis. The ASME PRA standard [1] defines an IE as "*any event either internal or external to the plant that perturbs the steady state operation of the plant, if operating, thereby initiating an abnormal event such as transient or LOCA within the plant. IEs trigger sequences of events that challenge plant control and safety systems whose failure could potentially lead to core damage or large early release*".

The general steps usually performed in the initiating events analysis task are as follows [2]:

1. Identification of IEs: This can be performed using engineering evaluation (e.g. FMEA) and/or deductive analysis approach (e.g. master logic diagram). It also should be complemented by a generic list and operational experience feedback.

2. Grouping of the IEs: The grouping of events considering for example the same success criteria on front-line systems, etc.

3. Assessment of IEs frequencies: The estimate of initiating events frequencies is generally first based on plant-specific data if sufficient data is available. However, since most of the EIs are relatively infrequent, there is a need to incorporate data from other plants (generic data) using a Bayesian update process. For rare events, fault tree

techniques can be used to augment generic data. For extremely rare initiating events, engineering judgment may be used.

This paper presents the IE analysis step 3 main results. The analysis was limited to events that occurred during full power operation. IEs with zero event count in the analysed specific and generic data windows are not presented in this paper.

2. Data Window

Most PSA assume that the frequency of IEs is constant over time, i.e. initiating events occur randomly according to a Poisson random process. One implication from this assumption is that the observation data window could be adapted to the frequency for a given IE. That is frequent events could have a shorter observation data window and less frequent events a broader data observation window.

Moreover, it is desirable to have the data reflect the most recent operating experience and the possible change in the number of events due to improvements in the design and operation of the plant [3].

2.1 G-2 Data Window

In order to give more weight to the G-2 events it was decided to consider all G-2 events from station in-service date in October 1st 1983 to December 31st 2006. This represents a period of 22.3 calendar years. This period is converted to effective full power years (EFPY) by multiplying the calendar years period by a gross capacity factor (GCF) of 79.5 % for G-2 up to the end of 2006 [3] which gives 17.7 EFPY.

2.2 Generic Data Window

The data from the most recent 5 years (from 2002 to 2006 at the start of this analysis) of the Canadian nuclear power plants (NPP) is first considered in this work. This is done to reflect the change in plant events over time (usually decrease in events) due to improvements in the design and operation of the plant. Moreover, licensee event reports (LER) data is in general of better quality for the most recent years. The analysis period for this generic data source corresponds to 62.9 EFPY as shown in Table 1.

In case there is no event in this time period the data window is augmented in the following order:

1. Canadian NPP from the in-service date to the end of 2001 which represents another 268.7 EFPY.

2. International experience, mainly from pressurised water reactors (PWRs) from reference [5].

The analysis results for these periods are not performed as part of this work and are not presented in this paper.

Table 1 Generic Data Window

Unit	**In-Service Date**	**GCF % [4]**					**Period**	
		2006	**2005**	**2004**	**2003**	**2002**	**Years**	**EFPY**
Point Lepreau	February 1, 1983	79.1	79.2	77.5	85.4	68	5	3.89
Pickering-1	July 29, 1971	77.3	81.9				5	1.59
Pickering-2	December 20, 1971						5	0.00
Pickering-3	June 1, 1972						5	0.00
Pickering-4	June 17, 1973	66.2	66.4	72.2	70.3		5	2.75
Pickering-5	May 10, 1983	89.5	52.6	92.5	69.2	59.4	5	3.63
Pickering-6	February 1, 1984	86.5	63.3	61.5	73	89	5	3.73
Pickering-7	January 1, 1985	59.2	97.8	69.2	40	94.7	5	3.61
Pickering-8	February 28, 1986	64.9	93.6	55.5	87.6	80.4	5	3.82
Bruce-1	September 1, 1977						5	0.00
Bruce-2	September 1, 1977						5	0.00
Bruce-3	February 1, 1978	82.7	74.9	76.1			5	2.34
Bruce-4	January 18, 1979	80.6	83.3	81.9	93.8		5	3.40
Bruce-5	March 1, 1985	96.1	74	85	76.4	86.6	5	4.18
Bruce-6	September 14, 1984	98.8	79.2	75.7	97.6	51	5	4.02
Bruce-7	April 10, 1986	94.7	70.2	92.7	97.3	69.7	5	4.25
Bruce-8	May 22, 1987	76.2	99.4	82.3	71.5	96.4	5	4.26
Darlington-1	November 14, 1992	83.3	96.1	72.8	85.7	85.4	5	4.23
Darlington-2	October 9, 1990	98.5	79	92	79.4	94.9	5	4.44
Darlington-3	February 14, 1993	72.5	98.7	85.8	89.1	83.2	5	4.29
Darlington-4	June 14, 1993	97.1	85.6	95.1	70.7	97.2	5	4.46
Total								**62.89**

3. Event Count

Once the data window is fixed the analysis consisted of reviewing the LER to identify events considered as IE (i.e., according to the definition of an IE) and to map them to the corresponding grouping as defined in step 2 of EI analysis task.

For G-2 about 1700 LER were analysed from information provided by the station various reporting tools and documents such as:

- Significant events report (AES),
- Event report (RE),
- Detailed event report (RDE),
- Pre-event analysis (APE),

- Preliminary event report (REP), etc.

Among these 1700 G-2 events only about 700 events were reported in the database «COG OPEX database» [8]. From the G-2 events not reported in database «COG OPEX database» [8] 38 events were considered as IE.

For the Canadian NPP other than G-2 the information was analysed based solely on information provided in the database «COG OPEX database» [8]. The total events examined from the GOG OPEX database [5] are around 8000 events.

Events are considered as an IE if the following plant response due to the initial perturbation has been observed:

1. Plant manual or automatic trip using shutdown system # 1or #2 (SDS1 or SDS2).

2. Plant stepback or setback using the reactor regulating system (RRS).

A root cause analysis is performed for each event and associated to an EI group. We also considered low power events as valid initiating events if they can occur at full power as suggested by reference [3].

The event count for G-2 (N) as well as for the Canadian NPP (α_{prio}) for the analysed data windows are presented in table 2 for a selection of some IEs.

4. Frequency Estimation

4.1 Methodology

We assume that the frequency of IEs is constant over time, i.e. initiating events occur randomly according to a Poisson random process. The IEs frequency λ calculation is performed depending on the event count (N) at G-2:

1. If $N \geq 10$ at G-2, the maximum likelihood estimate (MLE) is used (arithmetic mean)

$$\lambda = N / T \quad (1)$$

Where

N : The observed number of events.

T : Cumulative observed time period (EFPY).

λ : IE frequency (events/EFPY).

The 90 % confidence interval in this case is based on the chi-squared (χ^2) distribution [3] :

i) The lower limit

$$\lambda_{5\%} = \frac{\chi^2\left(\frac{\alpha}{2}, 2N\right)}{2 \cdot T} \tag{2}$$

ii) The upper limit

$$\lambda_{95\%} = \frac{\chi^2(0.95, 2N+2)}{2 \cdot T} \tag{3}$$

2. If $N < 10$ at G-2, a Bayesian update process is used based on Bayes theorem [3] :

$$f(\lambda \mid E) = \frac{f(\lambda)L(E \mid \lambda)}{\int L(E \mid \lambda) f(\lambda) d\lambda} \tag{4}$$

Where

$f(\lambda/E)$ = the probability of λ given evidence E (posterior distribution),

$f(\lambda)$ = the probability of λ prior to having evidence E (prior distribution), and

$L(E/\lambda)$ = the likelihood function (probability of the evidence given λ) .

When the evidence is in the form of N events generated by a Poisson process over an operational time t, as it is assumed in this work, the likelihood function is given by:

$$L(E \mid \lambda) = e^{-\lambda t} \frac{(\lambda t)^N}{N!} \tag{5}$$

For Bayesian estimation, a conjugate priori distribution is usually selected to determine the posteriori distribution for convenience reason [3], [5]. The conjugate family of prior distributions for Poisson data is the family of gamma distributions. The mean of the gamma distribution, also written as the expected value E (λ), is α/β where α is the shape parameter and β is the scale parameter.

The prior distribution is a gamma distribution with shape factor $\alpha_{prio.}$ and scale factor $\beta_{prio.}$ corresponding to :

- $\alpha_{prio.}$ Observed number of events for the Canadian NPP (generic data). In case of zero generic number of events the Jeffreys noninformative prior is used to convey little prior belief or information. With Poisson data the Jeffreys non-informative prior is obtained when $\alpha_{prio.} = 0.5$. This is the approach used in reference [5].

- $\beta_{prio.}$ Cumulative observed time period (EFPY) for the Canadian NPP.

The posterior distribution is a also a gamma distribution with shape factor $\alpha_{post.}$ and scale factor $\beta_{post.}$ equal to:

- $\alpha_{post.} = \alpha_{prio.} + N$ (N events count at G-2).

- $\beta_{post.} = \beta_{prio.} + T$ (T is the observed time period at G-2).

Therefore the posterior mean is given by:

$$E(\lambda) = \alpha_{post.} / \beta_{post.} \tag{6}$$

The confidence interval is given in this case by the $F_{5\%}$ and $F_{95\%}$ percentile of the gamma distribution cumulative distribution function F. An error factor (EF) is also determined based on the following equation:

$$EF = F_{95\%} / F_{50\%} \tag{7}$$

3. If no event is observed at G-2 and for the Canadian NPP for the observed data window, the data window is augmented as stated in section 2.2. For extremely rare IEs, engineering judgment may be used as proposed by ASME PRA standard [1]. The results for this part of the analysis are not included in this paper.

4.2 Results

The IE frequencies are presented in Table 2 and illustrated in Figure 1.

Table 2 illustrates that only one IE (GENT- general transient) has a count of more than 10. The frequency estimation in this case is performed based on Equation 1 and therefore no Bayesian update process is needed. In this case 14 events were observed and the MLE gives 7.9E-01 events per EFPY.

For all other IEs a Bayesian update process is performed as described in previous section. As noted before in case of zero generic number of events the Jeffreys non-informative prior is used to convey little prior belief or information

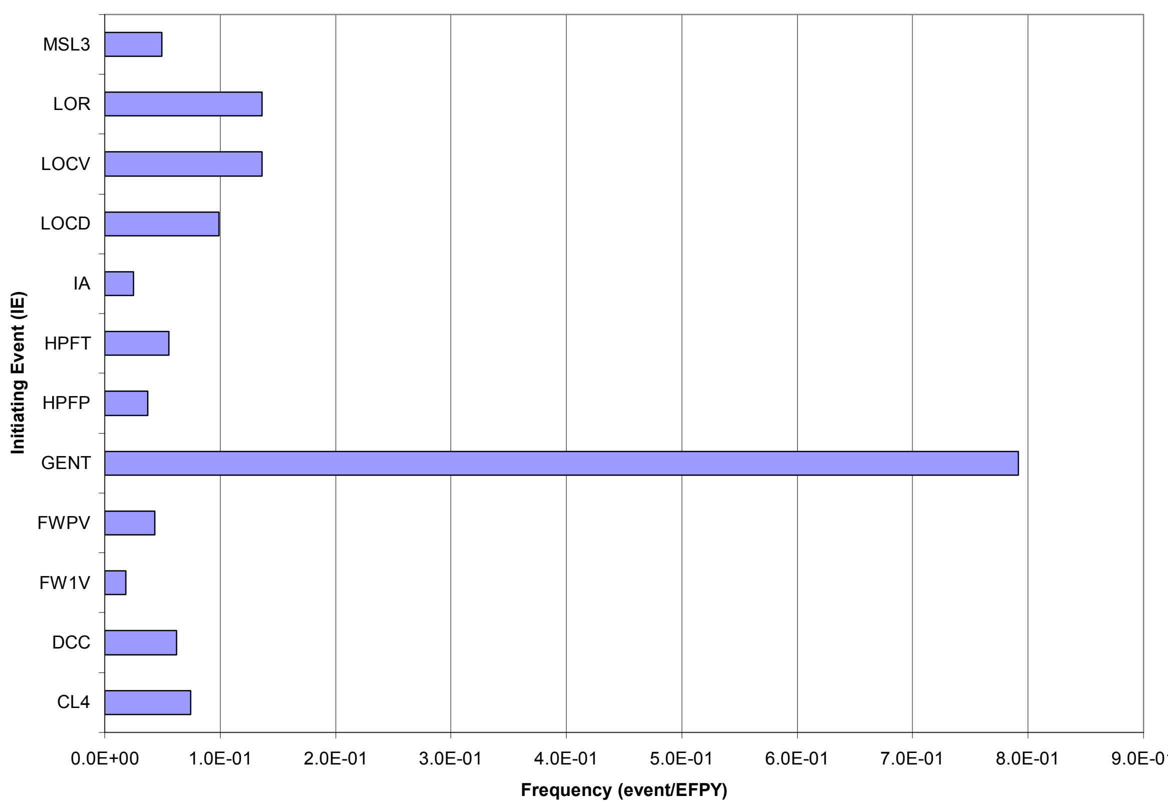

Figure 1 Sample of Initiating Events Frequencies at G-2

Table 2 Sample of Initiating Events Count and Frequencies at G-2

	Initiating Event	**T**	**N**	**Dist.**	**Prio.**	**Dist.**	**Post.**		**λ**	**(évé./**	**AEPP)**	**FE**
Label	**Description**	**EFPY**		α_{prio}	β_{prio}	α_{post}	β_{post}	$F_{5\%}$	$F_{50\%}$	**Mean**	$F_{95\%}$	
CL4	Total loss of Class IV power reactor operating. May include contribution from loss of grid & islanding failure.	17.69	4	2	62.89	6	80.58	3.2E-02	7.0E-02	**7.4E-02**	1.3E-01	1.9
DCC	Dual computer control failure	17.69	3	2	62.89	5	80.58	2.4E-02	5.8E-02	**6.2E-02**	1.1E-01	2.0
FW1V	Loss of feedwater flow to one SG	17.69	1	0.5	62.89	1.5	80.58	2.2E-03	1.5E-02	**1.9E-02**	4.8E-02	3.3
FWPV	Loss of main feedwater supply to SGs due to failure of pumps/valves	17.69	3	0.5	62.89	3.5	80.58	1.3E-02	3.9E-02	**4.3E-02**	8.7E-02	2.2
GENT	General Transients	17.69	14					4.8E-01	8.0E-01	**7.9E-01**	1.2E+00	1.5
HPFP	Partial loss of PHT pumped flow (PHT pump trip/bearing seizure/seal failure)	17.69	1	2	62.89	3	80.58	1.0E-02	3.3E-02	**3.7E-02**	7.8E-02	2.4
HPFT	Total loss of heat transport system (HTS) pumped flow	17.69	4	0.5	62.89	4.5	80.58	2.1E-02	5.2E-02	**5.6E-02**	1.0E-01	2.0
IA	Total loss of instrument air	17.69	1	1	62.89	2	80.58	4.4E-03	2.1E-02	**2.5E-02**	5.9E-02	2.8
LOCD	Loss of condensate flow to deaerator	17.69	5	3	62.89	8	80.58	4.9E-02	9.5E-02	**9.9E-02**	1.6E-01	1.7
LOCV	Loss of condenser vacuum	17.69	1	10	62.89	11	80.58	7.7E-02	1.3E-01	**1.4E-01**	2.1E-01	1.6
LOR	Loss of regulation - reactor operating full power	17.69	3	8	62.89	11	80.58	7.7E-02	1.3E-01	**1.4E-01**	2.1E-01	1.6
MSL3	Small steam line break inside turbine building (and Service Building)	17.69	3	1	62.89	4	80.58	1.7E-02	4.6E-02	**5.0E-02**	9.6E-02	2.1

5. Conclusion and Discussion

The analysis of G-2 events has shown that only for one IE the estimate of its frequency was based solely on plant-specific data. For all other events, data from Canadian NPP (generic data) was needed and incorporated using a Bayesian update process. The need for generic data is therefore paramount to the estimation of the IEs frequencies.

The first source of the uncertainty is related to the analysis of LER and its association to the appropriate EI group. This will be achieved first by establishing precise definitions of IEs with quantifiable criteria when applicable and as much details as to what should and should not be included in each IE.

In order to reduce the source of uncertainty, we believe that a concerted effort to establish a common generic database for IEs is very important. This is currently the practice in the US where the NRC provides with a publicly available database (reference [5] to [6]) used by utilities to update their site specific data. An open database shared by, for example, all COG members will have the following advantages:

1. Reduces the analysis errors: the database will be consulted and reviewed by all members.
2. Completeness an exhaustiveness of the data: The comparison between the OPEX information and that compiled at G-2 and LER found in «COG OPEX database» [8] showed that there are less events reported in this latter database. For example this analysis considered 38 IE that are not reported in «COG OPEX database» [8]. Therefore information in the COG database is not exhaustive and some of the EIs could be missed if site specific data is not consulted. Therefore, if all plants share their site data, a common database will be more exhaustive and representative of the historic return of experience data.
3. Reduces the cost: The analysis process is time consuming and expensive and performing the analysis once and shared by all members is surely the most economic way.
4. Facilitates the PSA periodic update of the database as required by reference [9].

6. References

[1] ASME, "ADDENDA to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications", RA-Sa-2003, December 2003.

[2] Safety Series No. 50-P-4, "Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 1)", IAEA, Vienna, 1992.

[3] NUREG/CR-6823 SAND2003-3348P, "Handbook of Parameter Estimation for Probabilistic Risk Assessment", U.S. Nuclear Regulatory Commission, September 2003.

[4] GOG, "SUMMARY OF CANDU PERFORMANCE 2000 to 2006".

[5] NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants", U.S. Nuclear Regulatory Commission, February 2007.

[6] NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995", U.S. Nuclear Regulatory Commission, February 1999.

[7] "Rates of Initiating Events at U.S. Nuclear Power Plants: 1988-2006", U.S. Nuclear Regulatory Commission, update to NUREG/CR-5750 data, December 2007.

[8] COG, "COG OPEX database", Candu Owner Group Web Acces Database.

[9] CNSC, "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants", S-294, April 2005.