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Defining initiating events for purposes of probabilistic safety assessment

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FOREWORD

Safe operation of nuclear power plants is achieved by good design and prudent operational practice. Both these aspects should be considered in the safety assessment of a nuclear power plant. A plant specific probabilistic safety assessment (PSA) models a plant by addressing both design and operation and, in addition, takes account of operating experience.

PSA can be used to quantify the safety of a nuclear power plant and it has become a widely used tool for the safety assessment of nuclear plants. Although its methodology is fairly standard, the complexity of the full scope plant specific PSA is such that guidance is still needed to ensure its completeness. Within the framework of its programme on the international promotion of PSA, the IAEA has issued numerous documents on topics ranging from guidelines on how to perform PSAs to specific case studies and generic reliability databases.

The PSA model is structured so as to permit consideration of a spectrum of possible disturbances to a plant's normal operation (in PSA terminology, initiating events), the availability of systems designed to cope with a particular disturbance and operator actions in the course of the event. One of the areas where the level of completeness and the accuracy of the analysis could greatly influence a PSA model is the selection of the initiating events (IEs). The IAEA has developed this document in order to summarize and explain the different approaches to the selection of IEs. It is based on examples taken from several PSA studies.

This document is primarily directed towards technical staff involved in the performance or review of plant specific PSAs. It highlights different approaches and provides typical examples useful for defining the IEs. The document also includes the generic initiating event database, containing about 300 records taken from about 30 plant specific PSAs. In addition to its usefulness during the actual performance of a PSA, the generic IE database is of the utmost importance for peer reviews of PSAs, such as the IAEA's International Peer Review Service (IPERS) where reference to studies on similar NPPs is needed.

In the preparation of this document, the IAEA received support from several Member States in the form of material to be included. The main author of the document was D. Ilberg from Israel, assisted by B. Linquist from Sweden and J. Pereq from Israel. A. Bareith from Hungary wrote the section on WWER reactors. The document was reviewed by specialists from France, Germany, Hungary and the United Kingdom and by IAEA staff. The IAEA project officer responsible for this report was B. Tomic from the Safety Assessment Section of the Division of Nuclear Safety.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

Several reports have been prepared [1] within the framework of the IAEA's programme to provide Comprehensive Guidelines for Conducting Probabilistic Safety Assessment (PSA). The selection, grouping and frequency evaluation of initiating events (IEs) is one of the most important tasks to be accomplished during a PSA Level 1 study, and this document is intended to provide additional guidance on this issue.

The importance of the determination of IEs has been shown by performing uncertainty and sensitivity analyses of past PSAs and by some peer reviews of PSA studies. The inclusion of additional relevant IEs for completeness or revision of the estimates of their frequencies may change the results of a PSA study.

This document is intended to aid in the conduct and review of PSAs in Member States by providing reference information that can help the specialist in defining IEs. It provides guidance by describing available methods and providing tables that compare approaches used in various past PSAs and a database of IE lists and frequencies taken from many PSAs which are available in the open literature.

1.1. PURPOSE OF THE REPORT

The determination of initiating events is an important part of a PSA. IEs directly affect the core damage frequency in PSAs. They are also a class of operating occurrences reported at nuclear power plants worldwide. Processing of these occurrence reports on a plant specific basis provides information on the actual operating experience of that plant. Thus, the information provided in this document allows comparisons of plant specific operating experience with ranges and frequencies of IEs considered in PSA studies of similar types of plant.

This report is intended to provide the reader with a comprehensive overview of different approaches to defining IEs which have been taken in a number of PSA studies. It does not recommend the use of any specific approach. It is left to those performing the PSA to select the approach which is the most appropriate for their particular application, and then, if necessary, to acquire more information from relevant literature listed in the references.

1.1.1. Purpose of defining the initiating events

An IE is a postulated event that could occur in a nuclear power plant. It is an occurrence that creates a disturbance in a plant and has the potential to lead to core damage, depending on the successful operation or failure of the various mitigating systems in the plant.

The performance of PSAs to develop a comprehensive plant model requires as complete a list of IEs as possible. This list determines the points of departure of the accident sequences that would be studied in the search for the dominant sequences that may lead to core damage. Thus the frequency of IEs has a direct impact on the results for core damage frequency, as well as on the spectrum of importance of individual components or actions.

The consequences of ill defined IEs are various. A missing IE in a PSA means that the core damage frequency would be underestimated by the value of the IE frequency multiplied by the conditional probability of safety system failure given the occurrence of the

IE. A larger list of IEs than necessary (for example, due to inappropriate grouping) would result in waste of resources because of the analyses of additional unnecessary accident sequences. An IE list that is incomplete or is insufficiently precise in its frequency determination would generally result in an incorrect estimation of the core damage frequencies.

1.1.2. Using this document for the evaluation of operational occurrences

This document can assist in evaluating operational occurrences in three ways:

(a) It describes approaches used for identifying occurrences and assigning them to known or new basic initiating events, or broader categories of initiating events.

(b) It describes methods for evaluating IE frequencies from experiences of operating occurrences in a specific plant over a defined period of time and provides examples of the application of these methods. It can be used for selecting generic prior distribution for Bayesian updating of plant specific experience.

(c) It provides an extensive database of data from other power plants to help put the plant specific experience in perspective.

1.2. SCOPE OF THE REPORT

This report covers the topics of defining IEs for nuclear power plants with PWR and BWR type reactors (including WWERs). It does not cover CANDU type reactors or older types or types of relatively limited distribution.

The "Procedures for Conducting PSAs of Nuclear Power Plants" [1] classify the IEs into internal IEs and hazards[1] (internal and external hazards). Internal IEs are hardware failures in the plant or maloperation of plant hardware through human errors or due to man–machine interface problems.

External hazards (often called external events) are events originating outside the NPP that create extreme environments common to several plant systems. Internal hazards, which are originated within the station boundaries, create similar extreme environments, and include internal flooding, fire and missiles. Loss of connection to the grid (complete or partial) is considered here as an internal IE. The scope of the document is confined to internal events. However, an introductory section on hazards is given in Section 8.

1.3. STRUCTURE OF THE REPORT

The report is organized as follows:

Section 2 gives the historical background for defining IEs by considering various approaches that have been used in previous PSAs. It covers the methods that form the basis for approaches described in Sections 3–5.

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[1] 'Hazards' is the term used in this report for all types of 'external events' such as fires, floods (inside and outside the plant), earthquakes, etc.

Section 3 provides a review of selected approaches in some recent PSAs and describes the main methods used to ensure completeness in IE selection. A discussion of completeness of the IE list is given and tables of several lists that have been used in the past are provided.

Section 4 similarly deals with the approaches to IE grouping. A review of several grouping structures is provided by comparing several recent PSA approaches for BWRs and PWRs. Lists of IE groupings are combined from some of these PSAs to provide the PSA specialist with 'maximal' and 'minimal' lists to compare. The relative importance of some of the IE groups used in various PSAs is shown for BWRs and PWRs respectively.

Section 5 covers the determination of IE frequencies. The various methods are described and some examples are given. Several sources of data for the evaluation of IE frequencies are outlined.

Section 6 provides examples of derivations of the frequency of initiators taken from several PSAs. It includes examples of estimations of the frequency of:

- LOCA initiators;
- transients;
- special common cause initiators;
- ATWS initiators.

An example of the treatment of initiators for non-power operating modes is also provided in Section 6.

Section 7 is entirely devoted to WWER reactors. These reactors are treated separately because the level of development of definition, grouping and frequency determination is, for the time being, somewhat lower than for PWR reactors operated in other countries mentioned in this report.

A brief introduction to the treatment of internal and external hazards is provided in Section 8.

Section 9 covers the initiating event database. It describes the structure, the record format and the data sources used for the database.

The Appendix provides the printout of the database of initiating events compiled by the IAEA.

2. HISTORICAL BACKGROUND

The concept of initiating events (IEs) was introduced in the United States Nuclear Regulatory Commission's Reactor Safety Study (RSS) in 1975 [2] together with the event tree methodology. Twenty transient IE categories and four LOCA related categories were selected for BWRs. Twenty-three transient and six LOCA related categories were selected for PWRs. The basis for the selection of the LOCA IEs was the plant response (based on systems/trains which are required to work and/or estimated timing of the accident). Consistent with plant response three sizes of LOCAs were selected:

- Large LOCA: 6 inches (15.2 cm) up to double ended largest pipe diameter — denoted "A";
- Small LOCA: 2–6 inches (5.1–15.2 cm) equivalent diameter — denoted S1;
- Very small LOCA: 1/2-2 inches (1.2–5.1 cm) equivalent diameter — denoted S2.

In addition, reactor pressure vessel (RPV) rupture was considered for both PWRs and BWRs.

For PWRs two additional LOCA related IEs were selected:

- Interfacing LOCA (known as the RSS 'event V');
- Steam generator (SG) tube rupture.

The definition for the various LOCA categories was used in the event tree analysis without further grouping. Not so for the transients. A general division into anticipated and unanticipated transients was made. Almost all anticipated transients were modelled as one group in both PWR and BWR event trees. Similarly, unanticipated transients were modelled as a second group. An additional special initiating event was considered, namely the loss of off-site power for more than 30 minutes.

2.1. DEVELOPMENT OF DATA ON TRANSIENT INITIATING EVENTS

Following the use of operating experience in the RSS, the Electric Power Research Institute (EPRI) published in 1978 its first study of anticipated transients [3] referred to as NP-801. NP-801 includes a compilation of operational occurrences from 12 BWRs and 30 PWRs. For BWRs it reported 459 events in 37 selected categories of different transient IEs (compared to 20 in the RSS) and for PWRs it reported 1000 events categorized in 41 different IEs (an expansion of the 23 found in the RSS). The data in the NP-801 covered NPP experience up to 1978, which was the equivalent of 41 and 131 plant-years for BWRs and PWRs respectively. The NP-801 data was used extensively in probabilistic risk assessment (PRA) performed in the period 1978–1983 when the update of the NP-801 report, the NP-2230 [4], became available.

The NP-801 has many drawbacks that the NP-2230 has treated partially and the later update of transients event data by EG&G [5] further improved:

(a) NP-801 used an 'effective in-service date' as supplied by the utilities. NP-2230 uniformly used the first day of commercial operation as the starting point for reporting plant anticipated transients. Because of this change, 137 events of NP-801 were excluded from the NP-2230 update.

(b) NP-801 reports 191 events within 37 plant-years that occurred in the years subsequent to the first year of plant operation (less than half the total number of events). NP-2230 reports 647 events in 85.5 plant-years (70% of the total).

It is clear that the NP-801 included very early periods of plant operation, i.e. from criticality to commercial operation, whereas the NP-2230 included events that occurred only after commercial operation was initiated. In general this is about half a year later. Table 2.1 gives a comparison of the evaluation of selected initiator frequencies based on these two data sources [6].

The NP-2230 reported 903 events for BWRs and 2093 events for PWRs under the same IE categories as in NP-801. The number of plants covered increased to 16 BWRs and 36 PWRs with 101.5 plant-years and 213 plant-years for BWRs and PWRs respectively. This data was used in PRAs after 1983.

The EG&G study [5] updated further the transient IE database for BWR and PWR plants. It included for the same IE categories 251 BWR plant-years with 1832 events and 423 PWR plant-years with 3574 events. There most of the events come from the years following the first two years of plant operation. This database was selectively used in the NUREG-1150 [7] type PRAs performed in the period 1986–1989.

The first four columns of Table 2.1 show original BWR-PRA estimates, based on NP-801 [3]. The next four columns represent results obtained from applying the same methodology to the more recent data source (NP-2230). The two last columns present results using the updated source and the two stage Bayesian methodology [8]. It can be seen that most of the increase in the BNL initiator frequencies derives from the updated experience of BWR related events, rather than from the use of the Bayesian methodology.

The BWR-PRA [12] differentiated between the impact of failures during the first year of plant operation and failures occurring in later years. However, in the review [6] it was argued that the database used in NP-801 was not sufficiently refined for this purpose. The later update, given in NP-2230, showed that the impact of ignoring the first year of plant operating experience causes a reduction of about 20% in initiator frequencies (see last two columns of Table 2.1). The 'weighted average' approach utilized in the BWR-PRA weighted the data from the first year as (1/35) and the data from subsequent years as (34/35) (this approach does not consider ageing).

Table 2.1 shows that the number of shutdowns due to anticipated transients is higher than that experienced in recent years. This is apparently because the NP-2230 database extends to 1981 only. The updated review of the experimental data [6] published in 1985 may show a reduction in the frequencies of IEs. The trend of reduction in transient IE frequency continues today in many power plants, and therefore the latest data sources should be utilized whenever possible.

The EPRI reports were made available as part of the EPRI programme on anticipated transients without scram (ATWS). For this purpose they include a section for IEs that occur at a power level above 25% of full power. This is done because transients that occur at a lower power level do not challenge the reactor shutdown system. This is further discussed in an example in Section 5.1 which treats frequency determination for ATWS.

TABLE 2.1. COMPARISON OF SELECTED INITIATOR FREQUENCIES AND SOURCES OF DIFFERENCES

	SNPS-PRA [12] EPRI-NP-801 Data				BNL Review: [6] EPRI-NP-2230 Data				BNL Review: [6] Two-Stage Bayesian	
Transient	1st Year	Subseq. Years	All Years Average	SNPS-PRA Weighted Average*	1st Year	Subseq. Years	All Years Average	Weighted Average	Subseq. Years	All Years**
Loss of Condenser Vacuum (2,4,8)	1.6	0.38	0.67	0.41	1.0	0.38	0.47	0.40	0.40	0.50
Turbine Trip	16.9	4.14	7.3	4.46	13.4	6.39	7.39	6.59	6.85	7.89
MSIV Closure (5)	2.2	0.19	0.67	0.24	1.67	0.27	0.47	0.31	0.29	0.57
Loss of FW (22)	0.6	0.16	0.27	01.8	0.27	0.11	0.13	0.12	0.11	0.13
LOOP (31)	0.4	0.11	0.16	0.08+	0.13	0.12	0.12	0.08+	0.12	0.15++
IORV (11)	0.7	0.08	0.20	0.09	0.53	0.15	0.21	0.16	0.19	0.25
CRW (27,28)	0.1	0.03	0.04	0.03	0.13	0.10	0.11	0.10	0.11	0.12
TOTAL	22.5	5.09	9.3	5.49	17.1	7.52	8.9	7.76	8.07	9.65

Numbers in parentheses correspond to EPRI NP-801 [3] categories
\+ Based on SNPS grid data * Used in the PRA
++ Based on NSAC-80 report [10] ** Used in the BNL review

The validity of the NP-2230 [4] data was reviewed in two cases:

(a) In the Oconee PRA [9] the data of NP-2230 was checked against the licensee event report (LER) data of the Duke Power Co., the owner of the plant. It was found that in most of the cases a good agreement exists. Only in the division between 'partial loss of MFW' and 'turbine trips', significant differences were found. Many more events were categorized as turbine trips in the Oconee PRA than as the partial loss of MFW. This was based on the in-plant data records, which was considered more accurate than the EPRI data.

(b) A thorough comparison was performed in the EG&G study [5] of the NP-2230 database and the 'NRC Gray Book' [11] database for 11 plants which were selected for the comparison. For each plant selected, the events that occurred during the third and eighth year of operation were carefully compared. It was found that 66 (27%) events were categorized differently based on the event description in each of the two sources compared. However, about one third of the discrepancies were because the Gray Book event description contained less information than the NP-2230 description. The final conclusion was that on the whole the NP-2230 data was found to be valid and is indicative of US commercial NPP experience. This is because the deviations were small and, in general, did not cross 'borders' of the broad groups of transients used in the PRA studies. Another important conclusion was that a sufficient amount of details in the event descriptions, provided by the plants, is crucial for a correct categorization.

2.2. DEVELOPMENT OF DATA ON LOCA INITIATING EVENTS

Unlike transients, the categorization of LOCA IE categories has not much changed since the original RSS definitions. The main changes in definitions were the inclusion, in most of the newer PSAs, of the SG tube rupture (rather than the SG rupture in the RSS which was not further analysed there) and a group of very small LOCAs at various locations (rather than control rod drive (CRD) pump leakage in the RSS). Table 2.2 compares several frequencies used in PSA for the same LOCA IE category. It should be noted that break size definitions of various size LOCAs are not uniform in PSAs and different for PWRs and BWRs in particular.

The Reactor Safety Study (RSS) estimated LOCA frequencies by inference from generic data from pipe breaks in the non-nuclear industries. This is the basis for the mean values shown in Table 2.2 for the RSS. The reactor pressure vessel (RPV) rupture probability was also based on non-nuclear vessel experience. While the frequency of the latter did not change much in newer PRAs (most of them still use the RSS value), the LOCA frequencies have been reevaluated in the newer PRAs (e.g. the Midland PRA [13]).

Oconee [9] and Seabrook [14] PRAs used experiential data for the evaluation of part of the LOCA frequencies rather than the pipe break data used in the RSS. The Oconee PRA considered the following events in a population of 35 plants:

- Large LOCA (A): No event occurred;
- Small LOCA (S): One event that occurred at Zion Unit 1 in 1975;
- SG rupture (R): Three events of SG tube ruptures with leakage rates greater than 100 gpm occurred: Surry Unit 2 (Nov. 1972), Point Beach Unit 1 (Feb. 1975), and Prairie Island Unit 1 (Oct. 1979).

A two-stage Bayesian analysis was applied to the above generic data and to the Oconee plant specific experience which reflects none of the above events in any of the three units on-site. A review of the Oconee PRA [15] added another relevant event:

Very small LOCA (VS): One event that occurred at H.R. Robinson Unit 2 (May 1975).

This has added a frequency of 3×10^{-3} (see Table 2.2). The B/W owner group [17] based their estimate of 'VS' on the precursor study [55] which introduced the 'Robinson event' mentioned above.

To summarize, the development of frequency evaluation for RPV and large LOCA has not changed significantly since the RSS was made. On the other hand, very small LOCAs and interfacing LOCAs received additional attention and some new studies have been made which are discussed further in Sections 5.1.4 and 6.3 which provide examples of the treatment of the frequency of LOCA type IEs.

2.3. DEVELOPMENT OF DATA ON COMMON CAUSE INITIATORS

The RSS has already treated some IEs as special common cause initiators (CCIs). Two examples are the loss of off-site power for more than 30 minutes and the 'V' event. The PSAs that followed the RSS have added more CCIs, and in general the CCIs are of a plant specific nature. Some of the CCIs treated in PSAs are:

- Loss of instrument air;
- Loss of DC power bus;
- Loss of service water or component cooling water system;
- Loss of AC power bus(es);
- Steam line break;
- Reactor water level instrument line failure.

This section describes the treatment of one specific CCI common to all PSAs, namely the loss of off-site power initiator. Other approaches to CCI evaluation are covered in Sections 5.1.5 and 6.2.

Loss of off-site power (LOOP) experiential data have been reviewed in four studies since 1980:

(1) Scholl [24] reviewed the data received from licensees following a June 1980 NRC request to submit licensee experience with LOOP events. This review includes a list of 109 occurrences of LOOP events.

(2) The results of a LOOP study were summarized in EPRI-NP-2301 [25] which uses data collected from 47 nuclear power plant sites. The report presents frequency and duration of LOOPs based on 45 occurrences through April 1981, representing 375 plant-years of experience.

(3) A NSAC/ORNL study was reported in NSAC-80 [10] which covered 52 nuclear power plant sites, for the period prior to December 1983. It summarizes 47 LOOP events in 530 plant-years.

TABLE 2.2. A COMPARISON OF LOCA FREQUENCIES IN VARIOUS PRAs [16]

LOCA Type Initiator (*)	Very Small LOCA (<0.5 or 1")	Small LOCA (**) (0.5/1" to 2/3")	Medium LOCA (2/3" to 6")	Large LOCA (>6")	RPV Rupture	Interfacing LOCA	SG Tube Rupture
PRA	VS	S	M	A	RPV	ISLOCA	R
ARKANSAS IREP [18]	0.020	6.9E-4	1.6E-4	8.7E-5	—	—	—
MIDLAND PRA [13]	5.0E-3	3.3E-3	4.7E-4	2.0E-4	—	7.7E-7	0.014
B/W Owner Group [17]	8.3E-3	4.0E-4	—	—	—	—	0.017
OCONEE PRA [9]	—	3.0E-3	—	9.3E-4	1.1E-6	1.4E-7	8.6E-3
NUREG-4550 PWR [40]	0.020	1.0E-3	1.0E-3	5.0E-4	1.0E-8	1.0E-6	—
NUREG-4550 BWR [36]	0.030	3.0E-3	3.0E-4	1.0E-4	1.0E-8	<3.0E-7	—
LIMERICK PRA [20]	—	0.010	2.0E-3	4.0E-4	—	—	—
SHOREHAM PRA [12]	—	8.0E-3	3.0E-3	7.0E-4	3.0E-7	1.8E-7	—
BWR-6 [21]	—	1.2E-3	6.7E-4	2.1E-4	—	1.7E-7	—
SEABROOK PRA [14]	—	0.017	4.7-4	2.0E-4	2.7E-7	1.8E-6	0.014
RSS-PWR [2]	—	2.7E-3	8.1E-4	2.7E-4	1.0E-6	1.1E-5	—
RSS-BWR [2]	CRD Pump	2.7E-3	8.1E-4	2.7E-4	1.0E-6	—	—
PALUEL 1300 PSA [23]	—	2.0E-3	3.0E-4	1E-4	—		8.6E-3
German Risk Study [22]	2.8E-3	1.4E-4	7.5E-5	<E-7	—	<E-7	6.5E-3/1E-5

* VS: Very Small - less than 1.5 inch diameter break; S:Small LOCA - less than 3 inch diameter break; L:Large LOCA - Greater than 3 inch diameter if "Medium" is not considered, otherwise greater than 6 inch diameter size. It should be noted that the values of break sizes can vary from PWRs to BWRs and among different PSAs, and are given here for general indication only.

** Note: 2/3 should read 2 or 3" diameter break.

(4) An USNRC study [26] for the resolution of the 'Station Blackout' issue was reported in NUREG-1032. The study covered 52 NPPs (all the US NPP sites of December 1983 excluding three with one off-site power connection). It summarizes 55 events in 533 plant-years.

The review of the data sources has resulted in several findings:

(a) The Scholl database is rather conservative and needs additional evaluations prior to its utilization in PRAs.

(b) The NP-2301 database is more realistic. A few events are apparently missing from this source. Its recovery probability information is relatively conservative for use in PRAs.

(c) The NSAC-80 database appears to be suitable for realistic PRA analyses. It recommends exclusion of several total LOOP occurrences during shutdown which it judges to be 'impossible' during operation. It can however be assumed that these are inadvertent human errors that should be included in LOOP frequency evaluation for completeness. The later NUREG-1032 considered them in its statistics.

The above three studies reported the LOOP events by plant and per geographical regions having similar weather conditions and an interconnection agreement with respect to keeping a reliable electric supply in that region. Another approach was proposed by a later study:

(d) The NUREG-1032 data [26] are based on almost the same database as the NSAC-80. It includes the 'shutdown' events as well. The main improvement of this study is that it provides a breakdown of all the LOOP events into well defined causes which allows tailoring of the LOOP frequency of a new plant according to its design and also allows for evaluating the improvement that may be expected by a design change in an older plant (see Section 5.1.7 for more details on this approach).

All these data sources and approaches have been used by PSAs in the USA. The LOOP is considered a CCI when the conditions of no electric power are continuing for a long time period (no recovery of off-site power). When off-site power is recovered within a short time, it is considered a transient similar to loss of condenser, because condenser cooling would be lost in this case. Therefore, also data on recovery times of LOOP are part of the information needed for quantification of its CCI frequency. The data for recovery times in the USA have also been developed in parallel to the LOOP event data discussed above. NUREG/CR-5032 [27] is the more recent one for the USA.

One of the first works providing LOOP recovery data was the EPRI study [25]. The NSAC/ORNL study [10] provided more precise recovery information on events in the USA until December 1983. The NUREG-1032 [26] study divided the NSAC/ORNL [10] data (with small modifications) into three subgroups.

- Severe weather type LOOP;
- Grid related LOOP;
- Plant centred (hardware failure related) LOOP.

For each of these subgroups the study provides a frequency of occurrence versus duration plot. The plant centred LOOP occurrences are the most frequent cases but most of them are recovered within half an hour [27]. Therefore, they may be treated like internal events. The grid and weather related LOOP have lower frequency and larger duration and are therefore treated more like the special CCI group of IEs.

3. SELECTION OF INITIATORS

3.1. DEFINITION OF INITIATING EVENTS

An initiating event (IE) is one that creates a disturbance in the plant and has the potential to lead to core damage, considering successful operation or not of the various mitigating systems in the plant. The following definition of IE has been taken from Refs [1] and [28].

> "An initiating event is an incident that requires an automatic or operator initiated action to bring the plant into a safe and steady-state condition, where in the absence of such action the core damage states of concern can result in severe core damage. Initiating events are usually categorized in divisions of internal and external initiators reflecting the origin of the events".

This section deals only with internal events. Internal hazards (such as internal floods) and external hazards (such as seismic events) are covered in Section 7. The initiating events considered here are only those normally resulting in an automatic or manual scram and occurring above a certain power level (usually 5 to 25%).

3.2. MAIN CATEGORIES OF INITIATING EVENTS

The internal initiating events may be looked upon as consisting of three main categories:

(A) LOCAs;
(B) Transients;
(C) Special common cause initiating events (common cause initiators (CCIs)).

Some of the recent PSAs (for example the Paluel PSA [23]) devoted substantial effort to the events during plant shutdown. A brief introduction to these specific events is provided in Section 6.5.

(A) LOCAs

The loss of coolant accident (LOCA) initiators include primary system breaks resulting in loss of primary coolant. Pipe breaks and ruptures of different sizes, inadvertent opening and failures to re-close (stuck open) of valves are being considered in this category.

(B) Transients

The transient initiating events are those which introduce the disturbance in normal plant operation, without loss of primary coolant and which require an automatic or manual shutdown of the reactor. Typical examples of transient initiators include disturbance in feedwater flow, turbine/condenser, reactivity control, reactor recirculation, etc. Certain disturbances in some of the support systems will also fall into this category.

(C) Special common cause initiating events

Special initiating events are events which, in addition to requiring reactor shutdown, simultaneously disable one or more of the mitigating systems required to control the plant status following the initiator. Typically, they are unique to the plant being analysed.

Initiating events which are common to most plants and have a typical CCI character are events such as:

- Loss of off-site power;
- Loss of DC power;
- Spurious containment isolation;
- Loss of instrument air;
- Loss of component cooling.

Even though some of the LOCAs may have CCI character (damage to equipment due to pipe whip, environmental influence (e.g. high temperature or humidity) on safety systems), LOCAs are not normally looked upon as CCIs.

3.3 METHODOLOGY FOR THE IDENTIFICATION OF INITIATING EVENTS

3.3.1. Identification of LOCAs

Loss of coolant accidents (LOCAs) are usually identified by studying the primary and secondary system equipment, piping and valve arrangement. Due to the different sizes of piping and valves as well as different mitigating systems available, a number of LOCAs break sizes are normally assumed. The specific boundaries of the LOCA event categories for each plant are established on the bases of overall mitigating system performance requirements (or safety system success criteria) for various cases of LOCAs break sizes.

The mitigating systems usually provide protection for different sizes of steam versus liquid breaks. Therefore, different boundaries may be used for small and medium LOCAs of liquid and steam line breaks (for BWRs).

In general the categories used for LOCA initiators in PSA reflect small variation among past PSAs. There are in use between three to five basic size categories taken from the following list:

- Very very small LOCA (e.g. less than 1 inch equivalent inside diameter);
- Very small LOCA (e.g. 1–2 inches equivalent inside diameter);
- Small LOCA (e.g. 2–3 inches equivalent inside diameter);
- Medium LOCA (e.g. 3–6 inches equivalent inside diameter);
- Large LOCA (e.g. greater than 6 inches equivalent inside diameter).

Differentiation between hot leg and cold leg break locations are generally ignored in PSA studies.

In addition, the following LOCA initiators are also being considered in most PSAs:

- Reactor pressure vessel rupture;
- LOCA outside containment (loss of coolant accident via the interface of low pressure system with RCS. In this case, in addition to unrecoverable loss of coolant, the containment is bypassed);
- Steam generator tube rupture (PWRs only).

In several cases of steam generator tube rupture only one tube rupture is considered. Even though this is a very small LOCA the plant response is in general different from the very small LOCA case (due to filling of affected SG and eventually overpressurizing it) and, in addition, a path to bypass containment is created in this case, which makes this initiator unique.

Some PSAs consider specifically the following initiators:

- Reactor coolant pump seal leakage or failure (in PWRs);
- Control rod drive system leakage or failure (in BWRs).

In other PSAs they are a part of the 'very small LOCA' initiator presented above.

3.3.2. Identification of transients and special initiating events

In order to obtain a comprehensive list of transients and special IEs, a number of methods and approaches have been used in PSAs. An overview of the methods utilized is provided in Tables 3.1 (BWR) and 3.2 (PWR). The tables include the following main methods:

(A) Engineering evaluation or technical study of plant;
(B) Reference to previous PSAs;
(C) EPRI list of IEs (such as EPRI-NP-2230 or NUREG/CR 3862);
(D) Logical classification: MLD, energy balance, barrier analysis;
(E) Plant energy balance fault tree;
(F) Analysis of operating experience for actual plant;
(G) Failure mode and effect analysis;
(H) Other methods.

These methods are described below.

(A) Engineering evaluation or technical study of plant

In this approach, the plant systems (operational as well as safety) and major components are systematically reviewed to see whether any of the failure modes (e.g. failure to operate, spurious operation, disruption, collapse) could lead directly, or in combination with other failures, to significant disturbances of plant operation, requiring operation of mitigating systems. Partial failures of systems should also be considered since, although they are generally less severe than complete failure, they are of higher frequency and are often less readily detected. This is, in principle, similar to a classical engineering approach taken during the plant's design aimed towards providing safety systems to cope with the spectrum of design basis accidents. For PSA purposes, the initiators which are beyond design basis (for example, due to their low frequency of occurrence) are also taken into account.

In order to aid the engineering evaluation it may be helpful to structurize the initiating events by means of categorization trees, as has been done in the Swedish project SUPER-ASAR [29]. The basis for this structurization is the availability of main functions such as the primary system integrity, off-site power, feedwater and condensation in the main condenser.

This approach was at least partially utilized in many PSAs, and it is an indispensable method in identifying special, common cause IEs which are unique to a particular plant.

TABLE 3.1 APPROACHES USED FOR THE SELECTION OF IEs IN VARIOUS PSAs (BWRs)

NPP PSA	Method for Identifying IEs: Engineering Evaluation or Technical Study of Plant	Reference to previous PSAs	EPRI List of IEs (such as EPRI-NP-2230)	Logical Classification: MLD, Energy Balance, Barrier Analysis	Analysis of Operating Experience for Actual Plant	Other Identification Methods
Peach Bottom, [19] Unit 2(-89)	+*	+	+		+	GE-NEDO 24708A ASEP List
Grand Gulf, [36] Unit 1(-89)	+*	+	+		+	GE-NEDO 24708A ASEP List
Ringhals 1 [29]	+			+	+	
Barsebaeck 1 [29]	+	+	+		+	
Oskarshamn 1 [29]	+			+	+	
Oskarshamn 2 [29]	+					The Swedish Energy Comm. F3 - Study
Forsmark 3 [29]	+					
Forsmark 1/2 [29]	+			+		
Shoreham [12]	+	+			+	FSAR
Browns Ferry [30] 1(-82)			+		+	
Limerick 1 [20]			+		+	
Big Rock Point[46]			+			

* For CCI only

TABLE 3.2. APPROACHES USED FOR THE SELECTION OF IEs IN VARIOUS PSAs (PWRs)

NPP PSA / Method for Identifying IEs:	Engineering Evaluation or Technical Study of Plant	Reference to previous PSAs	EPRI List of IEs (such as EPRI-NP-2230)	Logical Classification: MLD, Energy Balance, Barrier Analysis	Analysis of Operating Experience for Actual Plant	Other Identification Methods
Ringhals 2 [29]	+		+			
Oconee NSAC [9]	+	+	+		+	
Seabrook (-83)[14]		+	+	+		FMEA FSAR
Midland [13]				+		
Surry, Unit 1 (-86) [40]		+	+		+	List of Subtle Interactions (SNL) ASEP List
Calvert Cliffs (-84) [41]			+			FMEA
Arkansas 1(-82)[18]			+		+	FMEA
Sizewell B(-82) [50]	+*					FSAR RESAR
German Risk [35] Study (Phase A)		+			+	
German Risk [22] Study (Phase B)						
Zion [44]	+	+		+	+	FSAR & Industry Experience
Maine Yankee [45]	+		+		+	FMEA
Yankee Rowe [31]	+		+	+	+	
Connecticut Yankee (56)	+	+			+	FMEA

* For CCI

(B) Reference to previous PSAs

A large number of PSAs as well as PSA reviews are available today. It is very useful to refer to the lists of IEs presented in these PSAs, especially those for similar reactors. Reference to available IE lists may serve as the starting point in compiling a list of plant specific IEs. This approach was utilized in many PSAs.

(C) EPRI list of IEs

The lists of IE categories for BWRs and PWRs provided in EPRI-NP-2230 [4] and NUREG/CR-3862 [5] was used as the starting point for IE selection in a large number of PSAs. The lists are derived by analyzing operating experience of a few hundred reactor-years in the USA. Therefore, the lists can be considered as one of the best sources for providing a generic IE list for PSA of a new plant of similar design and reasonably similar operating practice.

(D) Master logic diagram

The so-called master logic diagram (MLD) is similar to a fault tree. It presents a model of a plant in terms of individual events and their combinations. A typical top event may be the 'significant release of radioactive material'. It develops into a plant level logic structure whose basic input events are the initiating events. One example of an MLD is shown in Fig. 3.1 taken from Ref [7]. The initiating events are the lowest level of the tree.

The particular advantage of the MLD method is that the issue of completeness is put into a more tangible perspective compared to other methods. However, in the cumbersome task of listing the specific causes of initiating events, the possibility of incompleteness still exists. Examples of the use of MLD for selection of IEs can be found in Seabrook and Yankee Rowe PSAs [14, 31].

(E) Plant energy balance fault tree (EBFT)

A transient condition in a nuclear reactor implies an imbalance in the state of equilibrium in the transfer of thermal energy from the reactor core to the environment. Therefore, a fault tree analysis of the plant energy balance (sometimes referred to as heat balance) is a valuable tool for deriving initiating events. The energy imbalance fault tree analysis may be carried out for any degradation in thermal equilibrium including those from full power, partial power, hot standby and cold shutdown. The typical top event is 'imbalance in energy transfer causing a plant's IE to occur'. One example of an energy balance fault tree is shown in Fig. 3.2. As for the MLD, the basic input events derived from the analysis are the initiating events at the equipment level.

Relative to the MLD it is unnecessary to include additional system failures in this type of fault tree. There is no need to introduce associated simplifications in the logic and, hence, the deductive power of fault tree analysis need not to be diminished.

(F) Analysis of operating experience for actual plant

In this approach the operational history of the plant in question and of similar plants is reviewed to search for any events which should be added to the list of IEs. This approach

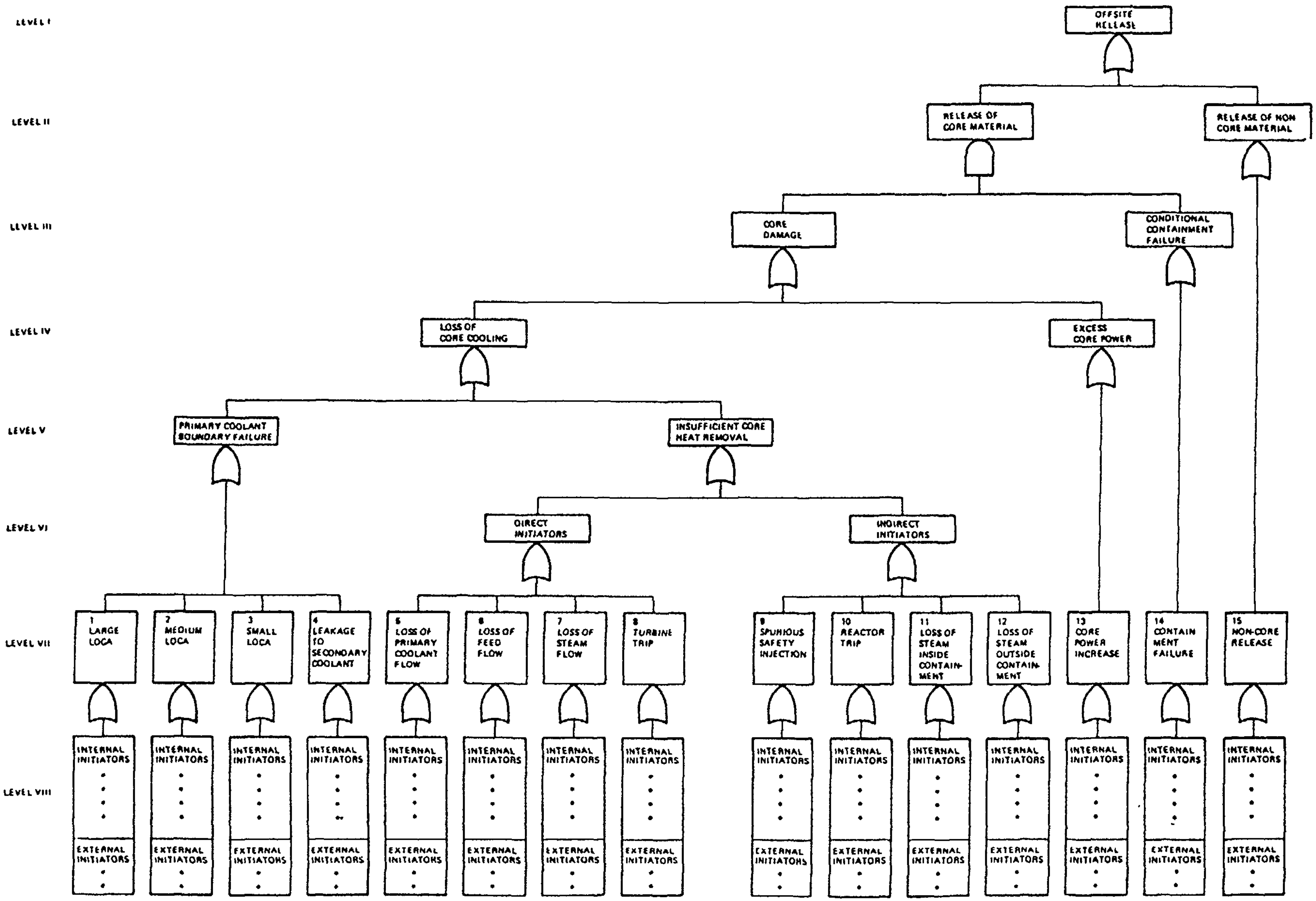

FIG. 3.1. Example of a master logic diagram (MLD).

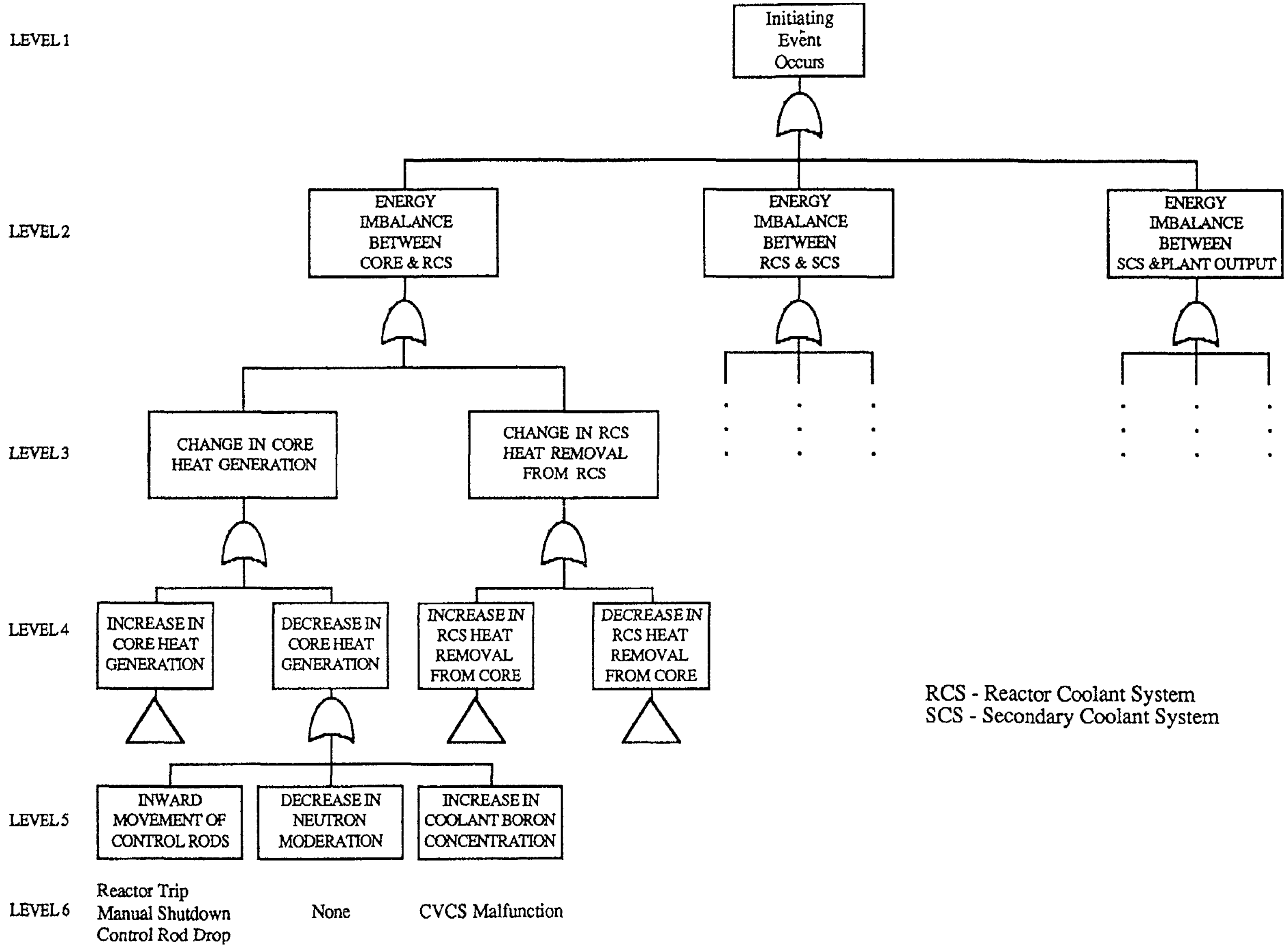

FIG. 3.2. Energy balance fault tree.

is to be looked upon as supplementary and would not, of course, be expected to reveal low frequency events but could show common cause initiating events of higher frequency. Such an approach is used also in precursor studies [32] to identify the IEs and to estimate the corresponding frequency.

(G) Failure mode and effect analysis

Failure mode and effect analysis (FMEA) is a powerful technique used extensively by the aerospace industry to reveal component failures that can impose critical effects on system performance. Dealing at the component level it becomes very detailed and laborious. Therefore, it is not commonly used in PSA.

Some PSAs have used this methodology to identify IEs within the plant control systems that have the potential to fail a mitigating system due to dependencies [31].

(H) Other methods

The most commonly used methods or approaches to selecting IEs were covered in the previous subsections. In special cases some other plant specific methods were used. One of these is the NUREG-1150 approach that has created a generic list of IEs in a special study and this IE list (ASEP list) was used as a generic list of IEs in all "1150" PSAs [33].

Another special approach is the Swedish SUPER-ASAR [29] project approach which is presented in Section 3.4.2.

3.4. EXAMPLES OF INITIATING EVENT LISTS

Section 3.3.2 presented several methods for identifying transient and special IEs. Two of the methods referred to give lists of IEs from other studies. This Section provides examples of the EPRI lists (see Section 3.3.2.C) and three specific lists that were used in previous PSAs (two for PWRs and one for BWRs):

For PWRs:

(A) The EPRI list (considers transients only);
(B) The Oconee list;
(C) The German Risk Study (Phase B) list.

For BWRs:

(A) The EPRI list (considers transients only);
(B) The Grand Gulf list.

As stated before, the EPRI lists for PWRs and BWRs are well known and are referred to by a number of PSAs, especially US studies.

The PSAs for Oconee and Grand Gulf provide IE lists that are partly based on the EPRI list, but in addition considerable work has been done using a few of the other methods discussed in Section 3.3.2 in order to find any additional IE which should be considered.

3.4.1. Initiating event lists for PWRs

(A) The EPRI list

The EPRI list of transient initiating events [4] is an expansion of the list presented in the RSS. The expansion is based on US reactor operating experience data. The data for PWRs includes 2023 events occurring over 213 plant-years at 36 plants. The EPRI PWR list is given in Table 3.3. The number associated with each initiator is referred to in Section 4.

(B) The Oconee list

The selection of the IEs used in the Oconee PSA [9] was performed in several steps. Initially, candidate IEs were proposed, and engineering evaluation performed for each IE. The IE list generated from those two activities was compared to IE lists available from EPRI, RSS, and the RSSMAP study for the Oconee NPP [34]. The LOCA events and their respective sizes were selected on the basis of mitigating system requirements. Finally, plant systems were reviewed to determine relevant special common cause initiators.

The resulting list of initiating events for Oconee is given in Tables 3.4–3.6. Table 3.6 shows the categorizations of the Oconee initiators and includes comments from the initiator selection process, which should clarify how a particular initiator has been treated. Table 3.6 indicates the final category (based on mitigation requirements) under which a particular initiator has been considered in an initiating event broad group that was further treated in an event tree analysis.

(C) The German Risk Study list

The German Risk Study (GRS) transient IEs [22, 35] are treated in two frequency classes: anticipated transients that are derived from operating experience and unlikely transients. The unlikely transients mainly include cases of steam line related events which have large effects on the plant systems. In comparison to these effects, the reactivity type transients inflict very small impact on the plant systems, and, therefore, they are not considered in phase B of the GRS. The list of transient initiators in the GRS are shown in Table 3.7. It should be noted that in many cases the anticipated transients considered in the GRS lead to the opening of the pressurizer relief valve and, in the case of its failure to close, a LOCA event occurs. These cases of LOCA are treated within the LOCA initiating events.

LOCA type IEs are covered in the GRS in great detail as shown in Table 3.8. It includes LOCAs of various sizes in the primary cooling system, in the pressurizer, LOCAs in the annulus between the circular containment liner and containment shielding (because they can impact on important additional systems by flooding) and steam generator tube ruptures.

For cases of failure of the reactor scram, ATWS events are also treated in the GRS.

Text cont. on p. 40.

TABLE 3.3. LIST OF PWR TRANSIENT INITIATING EVENTS

1. Loss of RCS flow (one loop)
2. Uncontrolled rod withdrawal
3. Problems with control-rod drive mechanism and/or rod drop
4. Leakage from control rods
5. Leakage in primary system
6. Low pressurizer pressure
7. Pressurizer leakage
8. High pressurizer pressure
9. Inadvertent safety injection signal
10. Containment pressure problems
11. CVCS malfunction–boron dilution
12. Pressure, temperature, power imbalance–rod-position error
13. Startup of inactive coolant pump
14. Total loss of RCS flow
15. Loss or reduction in feedwater flow (one loop)
16. Total loss of feedwater flow (all loops)
17. Full or partial closure of MSIV (one loop)
18. Closure of all MSIVs
19. Increase in feedwater flow (one loop)
20. Increase in feedwater flow (all loops)
21. Feedwater flow instability–operator error
22. Feedwater flow instability–miscellaneous mechanical causes
23. Loss of condensate pumps (one loop)
24. Loss of condensate pumps (all loops)
25. Loss of condenser vacuum
26. Steam-generator leakage
27. Condenser leakage
28. Miscellaneous leakage in secondary system
29. Sudden opening of steam relief valves
30. Loss of circulating water
31. Loss of component cooling
32. Loss of service-water system
33. Turbine trip, throttle valve closure, EHC problems
34. Generator trip or generator-caused faults
35. Loss of all off-site power
36. Pressurizer spray failure
37. Loss of power to necessary plant systems
38. Spurious trips–cause unknown
39. Automatic trip–no transient condition
40. Manual trip–no transient condition
41. Fire within plant

TABLE 3.4. LOCA INITIATORS INCLUDED IN THE OCONEE PRA

Event	Description
1. Small-break LOCA	A break or leak 1/2 to 4 inches in effective diameter. These are spontaneous events: induced LOCAs were treated directly.
2. Large LOCA	A break or rupture greater than 4 inches in effective diameter except those noted below.
3. Interfacing-system LOCA	A large loss of coolant through the valves acting as a boundary between high and low RCS pressure.
4. RPV rupture	A loss of reactor-vessel integrity precluding the ability to maintain coolant inventory.
5. Steam generator tube rupture	A rupture of a steam generator tube resulting in an RCS leak greater than 100 gpm.

TABLE 3.5. SPECIAL INITIATORS INCLUDED IN THE OCONEE PRA

Event	Description
1. Loss of instrument–air system	The system may cause a reactor trip and a failure of instrumentation and equipment that may be needed for a successful response to the trip.
2. Loss of service–water system	The system may fail by a pipe break or pump failure and in addition prevent other safety system operation that depend on service water cooling supply.
3. Loss of integrated and auxiliary control systems	The Integrated Central System (ICS) is controlling feedwater, pressurizer heaters etc., and may cause a transient with loss of a protecting system.
4. Loss of DC power system	Bus 3TC failure during power operation may result in failure to supply power to a number of pumps in train A supplying several mitigating systems.

TABLE 3.6. CATEGORIZATION OF THE OCONEE INITIATING EVENTS INTO TRANSIENTS AND LOCAs

Potential initiating event	Treatment in refinement process	Final category
1. Rod drop	If shutdown is required, the conditions after the trip are the same as those for the reactor trip category.	Reactor/turbine trip
2. Inadvertent rod withdrawal	It was determined that the reason for rod withdrawal would not affect the other reactor systems. ICS failures are treated elsewhere. Conditions after the trip are the same as for the reactor-trip category since the additional reactivity before the trip would not significantly change boundary conditions.	Reactor/turbine trip Event frequency judged to be very low. Consequences limited to brief RCS overpressure.
3. Rod ejection	Detailed analysis not performed. Probabilistic consideration is included in the frequency of non-mitigatable vessel ruptures.	Vessel rupture
4. Inadvertent boration or dilution	Boration would force the reactor toward shutdown. Credible dilutions would result in reactivity effects judged to be insignificant with respect to mitigation. Dilutions resulting in substantial reactivity were judged to be of small probability due to the time required and the amount of nonborated water required. Core melt due to a dilution accident was judged to be dominated by more frequent transients.	Reactor/turbine trip
5. Reactor trip		Reactor/turbine trip

TABLE 3.6. (cont.)

Potential initiating event	Treatment in refinement process	Final category
6. Cold-water addition	The conditions at shutdown are not significantly different from those of a reactor trip.	Reactor/turbine trip
7. RCP trip	Four-pump trip assumed to be dominated by the loss of off-site power or the loss of service water.	Loss of off-site power, loss of service water
8. RCP seizure	Only four-pump seizure was judged to be significant with respect to reactor conditions. Simultaneous four-pump seizure was judged to be of low probability and therefore would not result in dominant accident sequences.	
9. Flow-channel blockage	The potential for blockage was examined and found to be small due to the presence of internals at the bottom of the vessel which would prevent large objects from blocking flow channels. Smaller blockages were judged to not affect dominant core-melt risks.	
10. Loss of main feedwater	Due to its gross effects on heat removal through the secondary side, this was treated as an initiator.	Loss of main feedwater
11. Excess feedwater	Excessive feedwater that results in overcooling transients were treated as a transient category. Other excessive feedwater events that result in a loss of main feedwater as the cause of trip were included in the loss of main feedwater category.	Excessive feedwater

TABLE 3.6. (cont.)

Potential initiating event	Treatment in refinement process	Final category
12. Loss of condenser vacuum	Treated separately due to its long-term affects on main-feedwater availability and because it changes the availability of the condenser as a heat sink whereas the loss of main feedwater does not.	Loss of condenser vacuum
13. Inadequate main feedwater	Some trips result from feedwater instabilities that do not degrade the reliability of feedwater after the trip. These are included in the reactor-trip category. Another category was defined, partial loss of main feedwater, which describes the situation when the operability of the main feedwater system is degraded but not lost (e.g., one-pump trip or condensate-pump trip).	Reactor/turbine trip and partial loss of main feedwater
14. Feedwater or condensate line breaks	Treated as an initiating event due to the effects on the availability of condensate to the main and emergency feedwater systems.	Feedwater line break
15. Steam line breaks	Kept as a separate category due to its effects on both the primary and secondary systems including overcooling and loss of steam-dump capability. Sub-categories were used to differentiate the responses to breaks inside or outside the reactor building.	Steam line break

TABLE 3.6. (cont.)

Potential initiating event	Treatment in refinement process	Final category
16. Turbine and control valve malfunctions	No significant effects on responding plant systems.	Reactor/turbine trip
17. Turbine-bypass valve inadvertent opening	The inadvertent opening was included in the steamline break category.	Steam line break
18. Turbine malfunction	The only turbine malfunction singled out for special study was turbine-missile generation	Reactor/turbine trip
19. Loss of condenser circulating water	Effects of the transient were found to be included in the loss of condenser vacuum.	Loss of condenser vacuum
20. Small RCS pipe breaks	Grouped in one category based on an analysis of the response required. Breaks less than 4 inches in diameter were included in the small-LOCA-category	Small LOCA
21. Large RCS pipe breaks	All breaks greater than 4-inch diameter were grouped into one category based on required response and success criteria. Special locations for these LOCAs were reviewed, but the effects were judged not to be important when compared to the important mitigation system failure modes.	Large LOCA

TABLE 3.6. (cont.)

Potential initiating event	Treatment in refinement process	Final category
22. Inadvertent PORV or safety-valve opening	Included in the small LOCA category. It should be noted that PORV or safety-valve LOCAs resulting from other transients are modelled explicitly in the event tree.	Small LOCA
23. Reactor coolant pump seal failure	Included in the small LOCA category if the seal failure is a spontaneous initiating event. Seal failures resulting from inadequate protection after a trip are modelled explicitly.	Small LOCA
24. Control rod drive seal leakage	This event would have no significant differences from the small LOCA initiating event.	Small LOCA
25. Interfacing system loss of coolant	Treated in a special study due to the specificity of the event and its effects	Interfacing systems LOCA
26. Reactor vessel rupture	Treated as a separate initiating event. The separate category is defined to include ruptures that cannot be mitigated	Reactor vessel rupture
27. Steam generator tube leak/rupture	The event is defined as its own category due to unique operator actions and different radiological considerations from small LOCAs	Steam generator tube rupture
28. Charging exceeds letdown	Inadvertent HPI operation was judged to be the most restrictive case and is treated as an initiating-event category. Other possibilities are treated as subsets.	Spurious engineered safeguards signal

TABLE 3.6. (cont.)

Potential initiating event	Treatment in refinement process	Final category
29. Letdown exceeds charging	This transient was judged to be a subset of the small LOCA category.	Small LOCA
30. Inadvertent high pressure injection	Treated as a separate initiating events category due to different boundary conditions created by the event.	Spurious engineered safeguards signal
31. Failure on or off of pressurizer heaters	A review of the ICS identified the potential for failure causing heaters to fail on and the PORV to fail closed. This event was treated as a separate initiating event.	Spurious low pressurizer pressure signal
32. Failure on or off of pressurizer spray	Sprays failing off would result in a high pressure trip, but conditions after trip should not be greatly different from those of a reactor trip. Sprays failing on would have effects similar to those of event 26.	Reactor/turbine trip
33. Loss of off-site power	Treated as a category due to its effects on nearly all event-tree top events. The event was subdivided into switchyard faults and grid loss due to a difference in the availability of the Keowee overhead for the two cases and because of the different recovery potentials.	Loss of substation switchyard and loss of grid or feeders

TABLE 3.6. (cont.)

Potential initiating event	Treatment in refinement process	Final category
34. Loss of power to necessary systems	Because of the large number of potential initiating events in this category, a special study was conducted to determine the most significant events with respect to their effects on plant systems (see Table 3.5).	
35. Loss of power to control systems	These events were examined as subsets of event 29 and event 1 of Table 3.5.	
36. Loss of service water	This category was analysed for specific effects.	Loss of service water
37. Loss of component cooling	The service-water system at Oconee provides the most important needs. Component cooling requires service water for the heat exchangers. The loss of component cooling results from the loss of service water and was treated in that category.	Loss of service water
38. Loss of instrument air	Loss of the instrument air system affects a number of top events air significantly and also changes the availability of control-room indications to the operator. The air system was reviewed and the loss of system pressure was judged to be bounding case over less severe malfunctions.	Loss of instrument

TABLE 3.6. (cont.)

Potential initiating event	Treatment in refinement process	Final category
39. Integrated control system power	The ICS analysed in great detail both as a support system and an event initiator.	ICS initiators
40. Fires affecting necessary systems	See Section 6 on internal/external hazards.	
41. Internal flooding affecting necessary systems	The potential for internal plant flooding was recognized and reviewed. See Section 6 on internal/external hazards.	
42. Generator faults	Does not affect systems required for adequate response.	Reactor/turbine trip
43. Grid disturbances	Grid disturbances that result in generator trip do not change plant response. Other grid faults were assigned to the grid-failure initiating event.	Reactor/turbine trip
44. Administrative shutdowns	These shutdowns were included in the reactor-trip frequency if during the shutdown a trip was required. Shutdowns that proceed orderly and do not require a trip are considered successful responses.	Reactor/turbine trip
45. MSIV closure (1 or all)	Not relevant to the Oconee plants.	
46. ATWS	Treated separately (see Section 6.1).	

TABLE 3.7. TRANSIENT INITIATING EVENTS AND THEIR ESTIMATED FREQUENCY OF OCCURRENCE IN THE GERMAN RISK STUDY (PWR)

Transient initiating events	GRS phase B [22]	GRS phase A [35]
(D) Operational transients (anticipated transients):		
14. Loss of emergency power	0.13	0.1
15. Loss of main feedwater without loss of heat sink	0.15	0.8
16. Loss of main feedwater and loss of heat sink	0.29	0.3
17. Loss of heat sink with main feedwater on-line	0.36	–
(E) Steam line break transients (unlikely transients):		
18. Large leakage in steam line inside containment	1.6×10^{-4}	–
19. Large leakage in steam line outside containment	4.8×10^{-4}	–
20. Intermediate leakage in steam line inside containment	2.7×10^{-5}	–
21. Intermediate leakage in steam line outside containment	1.1×10^{-4}	
(F) ATWS		
22. ATWS by loss of main feedwater	4.7×10^{-6}	4×10^{-6}
23. ATWS by loss of emergency power	3.4×10^{-6}	
24. ATWS by loss of circulating water	7.5×10^{-6}	2.5×10^{-5}
25. ATWS by other transients	2.3×10^{-5}	

TABLE 3.8. LOCA INITIATING EVENTS AND THEIR ESTIMATED FREQUENCY OF OCCURRENCE IN THE GERMAN RISK STUDY (PWR)

LOCA initiating events		GRS phase B [22]	GRS phase A [35]
(A) Leakage in the primary cooling system piping:			
1a. Large LOCA	(>10")	$<10^{-7}$/year	2.7×10^{-4}/year
1b. Intermediate LOCA	(4-10")	$<10^{-7}$	
2. Small LOCA 1	(1.6-4")	9×10^{-5}	8×10^{-4}
3. Small LOCA 2	(1-1.6")	7.5×10^{-5}	
4. Small LOCA 3	(0.5-1")	7.5×10^{-5}	
5. Small LOCA 4	(0.25-0.5")	1.4×10^{-4}	2.7×10^{-3}
6. Small LOCA 5	(0.04-0.25")	2.8×10^{-3}	
(B) Pressurizer leakages following transients:			
7. Small leakage from pressurizer (0.4") following loss of main feedwater		3.2×10^{-5}	
8. Small leakage from pressurizer (0.4") following loss of heat sink		3.3×10^{-5}	
9. Small leakage from pressurizer (0.4") following other transients		1.2×10^{-4}	1.3×10^{-3}
10. Small leakage from pressurizer (0.8") by failure-open of safety valve		8.5×10^{-4}	
11. Leakage in the primary system in the round-space ('ring-space')		$<10^{-7}$	3×10^{-8}
(C) Steam generator tube leakage:			
12. Steam generator tube leakage (0.12–0.25")		1×10^{-5}	–
13. Steam generator tube leakage (0.02–0.12")		6.5×10^{-3}	–

3.4.2. Initiating event lists for BWRs

(A) The EPRI list

Since there is a considerable difference between IEs in PWR and BWR plants, EPRI developed a separate IE list based on operating experience of plants with BWRs. The data for BWRs include 903 events occurring over 101 reactor years. The EPRI BWR list is given in Table 3.9. As for the PWR list, the number associated with each initiator is given in Section 4.

(B) The Grand Gulf list [36]

This list was obtained by making use of the IE lists in about 10 available PSAs for US reactors. Generally, the initiators identified in these studies incorporate all events that had occurred in US nuclear power plants by 1980. A summary of these actual transient events is reported in EPRI-NP-801 [3] and its update EPRI-NP-2230 [4].

The above information was supplemented with actual plant trip data for Grand Gulf for a period of about 2 years of operation.

A review of the Grand Gulf design for special initiators was also undertaken.

The loss of coolant events were categorized into break size ranges, based on success criteria for the safety injection systems. The resulting list of initiating events for Grand Gulf is given in Tables 3.10 and 3.11.

3.5. COMPLETENESS OF IE LISTS

The first step in order to get a list of initiating events as complete as possible is to use several of the methods presented in Section 3.3. As an obvious minimum, one should include the following methods:

- Engineering evaluation;
- Reference to previous PSAs and other available IE lists (e.g. EPRI list);
- Reference to operating experience (if any relevant is available);
- Logical evaluation (MLD or EBFT).

Each approach examines potential IEs from a different perspective yielding a high degree of confidence that all risk significant IEs have been identified.

The more frequent IEs should, in general, be based on collected operating experience. This group of IEs is therefore considered to be relatively well covered.

The problem of completeness arises mainly with respect to the low frequency IEs. Low frequency IEs may be considered in three areas: pipe breaks or ruptures, other component failures and rare human errors. Pipe breaks and ruptures are considered to be relatively well covered if a proper engineering evaluation of the plant is performed. An important point is to examine piping outside the containment which has connections to the primary system, in order to reveal possible interfacing LOCA initiators. In that respect, the reliability of isolation valves has to be adequately considered.

Text cont. on p. 49.

TABLE 3.9. LIST OF BWR TRANSIENT INITIATING EVENTS

1. Electric load rejection
2. Electric load rejection with turbine bypass valve failure
3. Turbine trip
4. Turbine trip with turbine bypass valve failure
5. Main steam isolation valve (MSIV) closure
6. Inadvertent closure of one MSIV
7. Partial MSIV closure
8. Loss of normal condenser vacuum
9. Pressure regulator fails open
10. Pressure regulator fails closed
11. Inadvertent opening of a safety/relief valve (stuck)
12. Turbine bypass fails open
13. Turbine bypass or control valves cause increase in pressure (closed)
14. Recirculation control failure--increasing flow
15. Recirculation control failure--decreasing flow
16. Trip of one recirculation pump
17. Trip of all recirculation pumps
18. Abnormal startup of idle recirculation pump
19. Recirculation pump seizure
20. Feedwater--increasing flow at power
21. Loss of feedwater heater
22. Loss of all feedwater flow
23. Trip of one feedwater pump (or condensate pump)
24. Feedwater low flow
25. Low feedwater flow during startup or shutdown
26. High feedwater flow during startup or shutdown
27. Rod withdrawal at power
28. High flux due to rod withdrawal at startup
29. Inadvertent insertion of control rod or rods
30. Detected fault in reactor protection system
31. Loss of off-site power
32. Loss of auxiliary power (loss of auxiliary transformer)
33. Inadvertent startup of HPCI/HPCS
34. Scram due to plant occurrences
35. Spurious trip via instrumentation, RPS fault
36. Manual scram; no out-of-tolerance condition
37. Cause unknown

TABLE 3.10. LOCAs AND SPECIAL INITIATORS INCLUDED IN THE GRAND GULF PRA [36]

LOCA SIZE	
A (large LOCA)	Steam >0.4 sq.ft. (370 cm^2) Liquid >0.4 sq.ft.
S1 (intermediate LOCA)	Steam 0.13–0.4 sq.ft. Liquid 0.007–0.4 sq.ft.
S2 (small LOCA)	Steam <0.13 sq.ft. (100 cm^2) Liquid <0.007 sq.ft.(6.5cm^2)
S3	An S3 LOCA is a recirculation pump seal break and is isolable. If the operator does not recognize the break and fails to isolate, it is categorized as an S2 LOCA.
R	Vessel rupture.
V	Interfacing system LOCA.
SPECIAL INITIATOR	DESCRIPTION
Loss of instrument-air	The IE is included because a loss of instrument-air results in a plant trip and degrades one or more of the systems required to respond to the initiator.
Loss of emergency AC or DC vital bus	The IE is included because a loss of any AC or DC bus results in a plant trip and degrades one or more safety systems. In most BWRs, the drywell coolers are a non-safety system, but are powered by the safety buses. Loss of one safety bus may eventually lead to high drywell pressure trip and loss of a safety system powered by that bus.

TABLE 3.11. TRANSIENT INITIATORS IN THE GRAND GULF PRA

INITIATING EVENT GROUP	EPRI TRANSIENT CATEGORY	RATIONALE FOR INCLUSION
T1 (Transients that cause LOSP)	31	A loss of the offsite grid will result in a reactor scram and loss of normal AC power The on-site emergency diesel generators are required to start to supply AC loads.
	32	Loss of the normal and preferred station transformers result in a reactor scram and loss of normal AC power The on-site emergency diesel generators are required to start and supply AC loads.
T2 (Transients with loss of PCS*)	2	A generator load rejection results in fast closure of the turbine control valves which in turn scrams the reactor A subsequent failure of the turbine bypass valves to open results in complete isolation from the condenser.
	4	A turbine trip results in closure of the main turbine stop valves which in turn scrams the reactor. A subsequent failure of the turbine bypass valves to open results in complete isolation from the condenser.
	5	MSIV closure results in a reactor scram and complete isolation from the condenser.
	6	Closure of one MSIV may result in a high steam line flow signal that isolates the main steam lines from the condenser by closing the remaining MSIVs. MSIV closure scrams the reactor

Loss of PCS, that is, reactor isolation from main condenser

TABLE 3.11. (cont.)

INITIATING EVENT GROUP	EPRI TRANSIENT CATEGORY	RATIONALE FOR INCLUSION
T2 (Transients with loss of PCS*) (Concluded)	7	Partial closure of one MSIV may result in a high steam line flow s·gnal that isolates the main steam lines from the condenser by closing the remaining MSIVs. MSIV closure scrams the reactor.
	8	A loss of condenser vacuum causes a closure of the main turbine stop valves, MSIVs, and turbine bypass valves. The turbine trip initiates reactor scram.
	9	A pressure regulator failure in the open position will cause the main turbine control valves and bypass valves to completely open resulting in a low turbine inlet pressure isolation of the MSIVs. MSIV closure initiates reactor scram.
	10	A pressure regulator failure in the closed direction will result in closure of the main turbine control valves and inhibit opening of the turbine bypass valves. A high neutron flux signal will scram the reactor.
	12	An inadvertent or excessive opening of a turbine bypass valve will decrease the main steam line pressure resulting in a low turbine inlet pressure closure of the MSIVs. MSIV closure initiates reactor scram.
	13	Failure of the turbine control valves and bypass valves in the closed position will isolate the reactor from the main condenser. Closure of the turbine stop valves initiate reactor scram.
	37	All transients with unknown causes are assumed to result in isolation of the reactor vessel from the main condenser.

*Loss of PCS, that is, reactor isolation from main condenser.

TABLE 3.11. (cont.)

INITIATING EVENT GROUP	EPRI TRANSIENT CATEGORY	RATIONALE FOR INCLUSION
T3A (Transient with PCS available)	1	A generator load rejection causes a fast closure of the turbine control valves that in turn causes a reactor scram. The MSIVs will remain open and the turbine bypass valves will allow steam flow to the condenser.
	3	A turbine trip results in closure of the main turbine stop valves which in turn causes a reactor scram. The MSIVs will remain open and the turbine bypass valves will allow steam flow to the condenser.
	14	A failure of the recirculation flow control which increases recirculation flow results in a high neutron flux scram of the reactor. The turbine control valves will close upon decreasing turbine pressure. The MSIVs remain open and the turbine bypass valves will allow steam flow to the condenser
	15	A failure of the recirculation flow control which decreases recirculation flow will result in a vessel level swell to the Level 8 set point for scram and turbine trip. The MSIVs remain open and the turbine trip initiates bypass valve operation.
	16	The trip of on recirculation pump will not result in reactor scram. However if an additional failure occurs, it is assumed that the other recirculation pump trips. A level swell will occur causing a Level 8 reactor scram and turbine trip. The MSIVs remain open and the turbine trip initiate bypass valve operation.

TABLE 3.11. (cont.)

INITIATING EVENT GROUP	EPRI TRANSIENT CATEGORY	RATIONALE FOR INCLUSION
T3A (Transient with PCS available) (Continued)	17	A trip of all the recirculation pumps will result in a vessel level swell to the Level 8 setpoint for scram and turbine trip The MSIVs remain open and the turbine trip initiates bypass valve operation
	18	Attempts to startup a recirculation pump at high power levels will result in a reactor scram on high flux level The MSIVs remain open and the turbine bypass valves open following turbine trip
	19	A recirculation pump seizure will result in a level swell causing a Level 8 reactor scram and turbine trip The MSIVs remain open and the turbine trip initiates bypass valve operation
	20	A feedwater controller failure resulting in maximum flow to the vessel will result in a Level 8 scram and turbine trip The MSIVs remain open and the turbine trip will initiate bypass valve opening
	21	A loss of feedwater heating can result in a high neutron flux scram, of the reactor. The MSIVs remain open and the turbine bypass valves will allow steam flow to the condenser
	27	A rod withdrawal at power was shown in the FSAR not to result in a reactor scram since the rod withdrawal limiter mode of the Rod Control and Information System (RCIS) limits the withdrawal Even though the RCIS is single failure proof it is assumed that a control rod withdrawal will cause a high neutron flux scram of the reactor The MSIVs remain open following turbine trip

TABLE 3.11. (cont.)

INITIATING EVENT GROUP	EPRI TRANSIENT CATEGORY	RATIONALE FOR INCLUSION
T3A (Transient with PCS available) (Concluded)	29	Insertion of more than one control rod with failure of the RCIS assumed results in a turbine trip on low steam flow. The MSIVs remain open and the turbine bypass valves open.
	30	A fault in the reactor protection system can result in a reactor scram. The MSIVs remain open and the turbine bypass valves open following the turbine trip.
	33	An inadvertent startup of the HPCS can result in a reactor scram if the feedwater controller fails to respond correctly. A Level 3 or 8 signal is assumed to result in reactor scram. The MSIVs will remain open and the turbine trip initiates bypass valve operation.
	34	Scrams from plant occurrences are assumed not to cause MSIV closures or turbine bypass failure.
	35	A spurious trip via reactor protection system instrumentation does not cause MSIV closure or failure of the turbine bypass valves.
	36	A manual scram will not cause MSIV closure or failure of the turbine bypass valves.
T3B (Loss of FW transient)	22	A loss of all feedwater flow will result in a Level 3 trip of the reactor. The level will decrease to Level 2 actuating HPCS and RCIC. Operation of HPCS and RCIC will depressurize the plant to below the low pressure isolation of the MSIVs but closure of the MSIVs will not occur since the operator is expected to turn the reactor mode switch to SHUTDOWN thus inhibiting the low pressure MSIV closure signal.

TABLE 3.11. (cont.)

INITIATING EVENT GROUP	EPRI TRANSIENT CATEGORY	RATIONALE FOR INCLUSION
T3B (Loss of FW transient) (Concluded)	23	A trip of one feedwater or condensate pump may not result in a reactor scram if the feedwater control system compensates for the reduced flow. However an additional failure is assumed to occur resulting in a Level 3 reactor scram The MSIVs will remain open and the turbine bypass valves operate following turbine trip A complete loss of feedwater flow is consequently assumed.
	24	A transient resulting in low feedwater flow below the capabilities of the feedwater control system will result in a Level 3 reactor scram. The MSIVs will remain open and the turbine bypass valves operate following turbine trip. A complete loss of feedwater flow is conservatively assumed
T3C (IORV transient)	11	An inadvertent open Safety Relief Valve (SRV) will cause the suppression pool temperature to increase above 105°F if the SRV can not be closed (assumed in this category) A reactor scram is required by procedures. Note that suppression pool cooling will not prevent the suppression pool temperature from increasing during this type initiator.

Among the other two groups of initiators, it is possible that those IEs were not specifically considered in previous PSAs. Since those IEs, besides causing a disturbance in the reactor operation, also affect the function of some mitigating systems (low frequency CCIs), they should carefully be searched for. Naturally, such events that cause multiple effects are more dangerous than single failures events. In a number of cases such initiating events have been identified during the performance of other PSA tasks (e.g. fault tree analysis (FTA)). These initiators are most often plant specific and the engineering evaluation is the most suitable approach for their identification. Another support for the completeness of IE lists is obtained from the large size of the IE lists. Most full scope PSAs use a list of over 40 IEs that cover LOCAs of all sizes, transients and CCIs of all relevant support systems. These IEs are then grouped into broad categories which are therefore less sensitive to the failure to include a particular IE.

4. GROUPING OF INITIATING EVENTS

For each of the initiating events defined, an event tree, depicting the spectrum of plant responses (accidents and successes) should be made. Some of the initiating events would induce the same or a reasonably similar plant response. In that respect, the different IEs are grouped in order to decrease the amount of analyses required for PSA.

4.1. GENERAL PRINCIPLES FOR GROUPING

The grouping of initiating events shall be made in such a way that all events in the group impose essentially the same success criteria on the frontline systems as well as the same special conditions (challenges to plant operators, or to automatic plant responses) and shall result in the same core damage state. Thus the grouped events can be modelled using the same event/fault tree analysis.

Sometimes the amount of subsequent analysis needed may be further reduced by grouping together IEs that evoke the same type of plant response but for which the frontline system success criteria are not identical. The success criteria applied to that group of events would then be the most demanding (in terms of required systems) for any member of the group. The savings in effort must be weighed against the conservatism which this approach introduces.

Since there are some distinctive features for grouping each LOCA, transients or CCIs, each set is convened separately, for BWR and PWR type NPPs.

4.2. GROUPING OF INITIATING EVENTS FOR PWRs

4.2.1. LOCAs

As the plant response and success criteria is substantially different for LOCAs inside and outside containment, these two categories are separated. The inside containment division includes different break size LOCAs where the grouping is made according to a particular set of success criteria, i.e. different sets of mitigating systems are required for various pipe break sizes. The most common division of LOCAs caused by piping break, leakage or rupture is:

- Large LOCA (e.g. break size > 6 inch diameter, equivalent to > 300 cm^2 leak area);
- Medium LOCA (e.g. break size > 3 inch diameter, equivalent to 150–300 cm^2 leak area);
- Small LOCA (e.g. break size < 3, equivalent to < 150 cm^2 leak area).

These are very general divisions and they vary from one plant to the other. They are typical to liquid type LOCAs and may differ for steam side breaks.

In some cases, the small LOCA category is further subdivided. Sometimes plant response requires also a different set of equipment to mitigate the very small size LOCA, such as encountered when a RCP seal fails. Another reason to include a very small size LOCA is that it may affect the containment system response in another way than a small LOCA, e.g. a larger time period is needed for detection and other sensors are required to detect this kind of event.

In addition, the rupture or large leak of a reactor pressure vessel (RPV) is considered in most PSAs as a separate group. It is a case in which all available plant mitigating systems are considered ineffective.

The loss of coolant events 'outside containment' are interfacing LOCA and the steam generator tube rupture initiators. Although the coolant leaks for those events are comparable to some of the LOCAs inside containment, there are two principal differences between the two groups. A LOCA outside containment would, in addition to bypassing the containment and consequently creating the path for possible release of radioactivity to the environment, involve an unrecoverable loss of cooling water which becomes critical for the long term cooling.

Two examples of LOCA IE groups are given in Tables 3.4 and 3.8.

4.2.2. Transients

Grouping of transient initiators is not only plant specific, it also depends heavily on the purpose and scope of a PSA study. To illustrate possible approaches, Tables 4.1 and 4.2 compare the groupings chosen in several plant specific PSA studies with the EPRI list of transients for PWRs [4] grouped in 'Functions' as in NUREG/CR-3862 [5]. It can be seen that there are some substantial differences in groupings.

There are several reasons for these differences. The most important ones are:

- Different design operation features that affect plant response and success criteria;
- Different level of detail in PSA;
- Progress made in the PSA modelling in order to obtain more realistic results.

As a general rule, the amount of analysis (and cost) need to be weighted against the increase in modelling detail and in more realistic results. The approaches taken would vary from rather coarse grouping, where due to scoping analysis a lot of conservatism is involved, to precise specific grouping, in order to achieve best estimate results.

The minimal grouping would include:

- Transients with main feedwater (MFW) initially available (turbine/reactor trips);
- Transients with loss of MFW;
- Loss off-site power.

When more precise grouping is preferred, the following subgrouping would fit in the above indicated broad groups:

– Reactor trip	)	Transients with MFW
– Turbine trip	)	available
– Partial loss of MFW	)	
– Excessive FW flow	)	

– Total loss of MFW	)	
– Loss of condenser vacuum	)	Transients with
– Closure of one MSIV	)	loss of MFW
– Closure of all MSIVs	)	
– Core power excursion	)	
– Loss of primary flow	)	Loss of off-site power
– Loss of off-site power	)	

The increased details allow to take into account, in addition to the hardware configuration changes with respect to the mitigation of the sequence, the recovery times which may differ for each of the transient initiators (e.g. condenser recovery is longer than feedwater pumps recovery). Therefore, final sequences and associated probabilities are much more realistic.

4.2.3. Special common cause initiators

This broad group of initiators includes relatively low frequency IEs that in addition to initiating fault cause failures of mitigating systems. They are very much plant specific. The most common initiators of this group are:

- Loss of vital AC power bus;
- Loss of service water system;
- Loss of component cooling;
- Loss of a DC bus;
- Loss of instrument air;
- Loss of core level measuring instrument;
- Loss of ventilation system;
- Loss of room coolers;
- Steam line break in locations where it causes additional effects or containment isolation.

The loss of off-site power initiator is somewhat specific. Normally it is included in the transient group which implies that the diesel generators are available or the off-site power is being restored within 30 minutes. Otherwise it is called ‘plant blackout’ and becomes a more serious transient which has similar characteristics to a common cause initiator. Therefore it is sometimes treated under the CCI category. An example is the RSS [2] in which LOOP longer than 30 minutes were treated as a CCI. Several examples also exist of plant specific initiators such as the loss of a particular AC bus in a plant that causes a transient with loss of a part or severe degradation of the mitigating systems.

Since common cause initiators are rather specific as far as exact plant response is concerned, they are, in principle, treated individually, e.g. there is no grouping and an event tree is created for each initiator separately. In some cases, several such initiators, having similar plant response, would be grouped together (a typical example is loss of service water/loss of component cooling at some plants). An event tree will be constructed for the initiators with the most difficult plant response. The treatment of CCIs is in principle similar for BWR and PWR type plants.

TABLE 4.1. A COMPARISON OF EPRI PWR TRANSIENT CATEGORIES WITH SEVERAL PSA GROUPING SCHEMES (EXAMPLE 1)

Function (From [5])	IE Category (From [4])	OCONEE [9]	SEABROOK [14]	GERMAN RISK STUDY [35]	SURRY [40]	INDIAN POINT 3 [42]	ZION [44]
Main turbine/ generator	33 Turbine trip, throttle valve closure, EHC problems	1,2,3,6, 10,11,12,13 14,27,28,29 33,34,36,38	+		1,2,3,4,5,6,7 8,9,10,11,12, 13,14,15,19,20 23,33,34,36,	+	+
	34 Generator trip or generator caused faults	39	+	+	38,39,40	+	+
Condenser vacuum	25 Loss of condenser vacuum	25	+	+	+		
	27 Condenser leakage	*		18	17,18		
	30 Loss of circulating water	30			+		
Condensate	23 Loss of condensate pump (One loop)	*			*		
	24 Loss of condensate pumps (all loops)	*		+	+		
Feedwater	16 Total loss of feedwater flow (all loops)	16,24	+	+	+	+	+
	19 Increase in feedwater flow (1 loop)	19	+		*		
	20 Increase in feedwater flow (all loops)	20	+		*		
	15 Loss or reduction in feedwater flow (1 loop)		+		*	+	+
	21 Feedwater flow instability-operator error		+		+		
	22 Feedwater flow instability-miscellaneous mechanical causes	15,21, 22,23	+		+		
	28 Miscellaneous leakage in secondary system	*	+		+		

TABLE 4.1. (cont.)

Function	IE Category	OCONEE	SEABROOK	GERMAN RISK STUDY	SURRY	INDIAN POINT 3	ZION
Reactor system flow	1 Loss of RCS flow (1 loop)	*			*	+	+
	13 Startup of inactive coolant pump	*			*	+	+
	14 Total loss of RCS flow	*	+		*	+	+
Reactor system pressure control	6 Low pressurizer pressure	*			*		
	8 High pressurizer pressure				*		
	36 Pressurizer spray failure	*			*		
Reactivity control	2 Uncontrolled rod with-drawal	*	core power excursion		*		core power increase
	3 CRDM problems and/or rod drop	*			*		
	11 CVCS malfunction-boron dilutions	*			*		
	12 Pressure/temperature/power imbalance-rod position error	*			*		
Steam	17 Full or partial closure of MSIV (1 loop)		+		*	+	
	18 Closure of all MSIV		+	*	*	+	+
	29 Sudden opening of steam relief valves	*	+				
Safety injection	9 Inadvertent safety injection signal	+	+		*		+
	10 Containment pressure problems	*			*		
Electrical power	35 Loss of all offsite power	+	+	+	+	+	+
	37 Loss of power to necessary plant systems	+			Loss of AC Bus		

TABLE 4.1. (cont.)

Function	IE Category	OCONEE	SEABROOK	GERMAN RISK STUDY	SURRY	INDIAN POINT 3	ZION
Reactor integrity	4 Leakage from control rods	not included in PSA	Incl. in the LOCA categories	Incl. in the LOCA categories	*		
	5 Leakage in primary system				*		
	7 Pressurizer leakage				*		
	26 Steam generator leakage	*	*				
	– Rod ejection	+					
Miscellaneous	31 Loss of component cooling	+	+		+	+	+
	32 Loss of service water systems	+	+		+	+	+
	41 Fire within plant	*	*			+	
Spurious trips	38 Spurious trips – cause unknown	*			*		
	39 Auto trips – no transient condition	*			*		
	40 Manual trip – no transient condition (planned shutdown)				*		
Non EPRI List Categories	– Integrated Control System Failures	+					
	– Loss of Instrument Air	+					
	Spurious low Pressurizer Pressure signal	+					
Non EPRI List Categories	– Feedwater condensate line break	+					
	– Steam line break – large	+	+	+		+	+
	– medium			+			
	– Loss of DC Bus		+		+		

Notes: "+" taken into account in the same group
"*" taken into account but in a different group
"Blanks" indicate that the category was not considered in that PSA (in several cases the basis for the omission is provided in the PSAs)

TABLE 4.2. A COMPARISON OF EPRI PWR TRANSIENT CATEGORIES WITH SEVERAL PSA GROUPING SCHEMES (EXAMPLE 2)

Function (From [5])	IE Category (From [4])	SIZEWELL B [50]	RING-HALS-2 [29]	ANO-1 [18]	CALVERT CLIFFS 1 [41]
Main turbine/ generator	33 Turbine trip, throttle valve closure, EHC problems	FC4	CAT.2	+	+, 37
	34 Generator trip or generator caused faults	FC4	CAT.2	1,2,3,6,8-10, 13,14,15,17, 23,34	+
Condenser vacuum	25 Loss of condenser vacuum	FC4	CAT.2	*	*
	27 Condenser leakage	*	CAT.2		
	30 Loss of circulating water	FC4	CAT.2	*	*
Condensate	23 Loss of condensate pump (one loop)	FC4	CAT.2	*	*
	24 Loss of condensate pumps (all loops)	FC4	CAT.3A	*	*
Feedwater	16 Total loss of feedwater flow (all loops)	FC4	CAT.3A	+,24,25,29, 30,17,18	+, 18, 24,25 30
	19 Increase in feedwater flow (1 loop)	FC3	CAT.3B		*
	20 Increase in feedwater flow (all loops)	FC3	CAT.3B	+	*
	15 Loss or reduction in feedwater flow (1 loop)	FC4	CAT.2		*
	21 Feedwater flow instability-operator error	FC3	CAT.3B	+	+
	22 Feedwater flow instability-miscellaneous mechanical causes	FC3	CAT.3B	+	+
	28 Miscellaneous leakage in secondary system	*	CAT.2, 9,18,24, 32		*

TABLE 4.2. (cont.)

Function	IE Category	SIZEWELL B	RINGHALS-2	ANO-1	Calvert Cliffs-1
Reactor system flow	1 Loss of RCS flow (1 loop)	FC6	CAT.2	*	+,3,9,11,12
	13 Startup of inactive coolant pump		CAT.2		15,17,19,20,23,27
	14 Total loss of RCS flow	FC6	CAT.1	*	+28,29,38, 39,40
Reactor system pressure control	6 Low pressurizer pressure	FC-11	CAT.2	*	+
	8 High pressurizer pressure		CAT.1	*	+
	36 Pressurizer spray failure		CAT.1		+
Reactivity control	2 Uncontrolled rod with-drawal	FC-7	CAT.1	*	
	3 CRDM problems and/or rod drop	FC-7	CAT.2	*	*
	11 CVCS malfunction-boron dilutions	FC-7	CAT.1		*
	12 Pressure/temperature/power imbalance-rod position error	*	CAT.2		*
Steam	17 Full or partial closure of MSIV (1 loop)	FC-4	CAT.2	*	*
	18 Closure of all MSIV	FC-4	CAT.3B	*	*
	29 Sudden opening of steam relief valves	FC-3	CAT.2	*	*
Safety injection	9 Inadvertent safety injection signal	FC-8	CAT.3b	*	*
	10 Containment pressure problems		CAT.2	*	
Electrical power	35 Loss of all offsite power	FC-5	CAT.4	+	+
	37 Loss of power to necessary plant systems	FC-5	CAT/2	loss of AC Bus	*
Reactor integrity	4 Leakage from control rods	FC-9	CAT.3	incl.	
	5 Leakage in primary system	FC-9	CAT.2	in the	
	7 Pressurizer leakage	FC-9	CAT.2	LOCA	
	26 Steam generator leakage	FC-9	CAT.2	categories	
	- Rod ejection		*		

TABLE 4.2 (cont.)

Function	IE Category	SIZEWELL B	RING-HALS-2	ANO-1	Calvert Cliffs-1
Miscellaneous	31 Loss of component cooling	FC-11	CAT.2	+	+
	32 Loss of service water systems	FC-11	CAT.3C	+	+
	41 Fire within plant	FC-13			
Spurious trips	38 Spurious trips - cause unknown	FC-2	CAT.2	* *	*
	39 Auto trips - no transient condition		CAT.2		*
	40 Manual trip - no transient condition (planned shutdown)		CAT.2		*
Non-EPRI List Categories	- Integrated control system failure				
	- Spurious low pressurizer pressure signal				
	- Loss of Instrument Air				
	- Loss of core cooling during steam-gen. inspec.		CAT.0		
	- Feed water condensate line break	FC-4	CAT.3A		
	- Steam Line Break -Large -Medium	FC-3	CAT.3A		
	- Loss of DC Bus			+	+
	- Cold overpressurization	FC-8			

Notes: "+" taken into account in the same group
"*" taken into account but in a different PSA IE group
"Blanks" indicate that the category was not considered in that PSA (in several cases the basis for the omission is provided in the PSAs)
"FC"- Fault Category
CAT.0: Not regular (faulty) reactor shutdown conditions
CAT.1: Transients challenging RCS integrity
CAT.2: Transients that do not challenge RCS integrity or anxiliary core cooling systems
CAT.3: Transients challenging auxiliary core cooling systems with offsite power available
CAT.4: LOSP

4.3. GROUPING OF INITIATING EVENTS FOR BWRs

4.3.1. LOCAs

As mentioned in Section 4.2.1 LOCAs are grouped in different break size categories according to the different success criteria for the safety injection system. A typical division would correspond to the following [36]:

A	Large LOCA	Automatic depressurization of primary system is not required.
S_1	Medium LOCA	Automatic depressurization necessary in order to make it possible for low pressure ECCS injection
S_2	Small LOCA	Low pressure ECCS injection not required.

In addition to the above, some of the PSAs also define a small-small LOCA (S_3). The initiator behind that definition is a recirculation pump seal leak or pipe leak of similar size. Such leaks have occurred in power plants, primarily from the wearing out of these pump seals during normal operation. Sometimes the control rod drive (CRD) system leakage is considered in the small-scale LOCA category. As in the case of PWRs, interfacing LOCA is considered as a separate category. The same is true for reactor vessel rupture. In BWRs steam is generated directly in the reactor vessel, and LOCA events may be on steam and water piping. Since the loss of coolant rate will depend on rupture being either on the steam or water side, actual break sizes (or equivalent flow areas) for the steam and water side will be different for each of the LOCA groups. Again the need for depressurization and success criteria for low pressure safety injection will determine actual grouping. Table 3.9 provides an example of LOCA grouping used for the Grand Gulf PSA.

4.3.2. Transients

As for PWRs, grouping of initiating events depends on the plant response and scope of the PSA study. Table 4.3 compares grouping in PSA studies for Shoreham, Grand Gulf, Forsmark and Browns Ferry NPPs with EPRI Generic IE categories [4] and functions from NUREG/CR-3862 [5].

Table 4.3 shows that there are some major differences in grouping done in plant specific PSA study. One obvious reason is differences in design and logics of these NPPs. Another reason is that considerations regarding the severity of the individual initiators did influence the chosen grouping scheme. Decisions to utilize many IE groups mean that increased effort to obtain more realistic modelling was judged to be cost effective.

In the Reactor Safety Study (WASH-1400) [2] the BWR transients were originally grouped into three major categories:

T1: Transients involving loss of off-site power;
T2: Transients involving loss of the power conversion system (PCS) (MSIV closure, loss of condenser vacuum, etc.);
T3: Transients with the PCS initially available (turbine trip, etc.).

TABLE 4.3. A COMPARISON OF EPRI BWR TRANSIENT CATEGORIES WITH SEVERAL PSA GROUPING SCHEMES

Function	IE Category	SHOREHAM [12]	GRAND GULF [36]	FORSMARK 3 [29]	BROWNS FERRY [30]
Main turbine/ generator	1 Electric load rejection	1 35,36,37(T_t)	1 (T_3A), 14-21	1 T_s	T_2
	2 Electric load rejection with turbine bypass valve failure	* 32,33,34 27,28,29,30	2 (T_2), 4-10, 12, 13, 37	2 (T_t)	T_2
	3 Turbine trip	3 24,25,26(T_t)	3 (T_3A),27,29,30 33-36	3 (T_s), 6, 13	
	4 Turbine trip with turbine bypass valve failure	* 19,20,21,23 14,15,16,17 18,6,7,9,10 12,13 (T_t)	4 (T_2)	4(T_t) 12,5 7,8,9,10 39,40	T_2
Condenser vacuum	8 Loss of normal vacuum	8 (T_c)	*	*(T_t)	T_1
Condensate	–				
Feedwater	20 Feedwater-increasing flow at power	*	*	*	T_1
	21 Loss of feedwater heater	*	*	*	–
	22 Loss of all feedwater flow	22 (T_F)	22 (T_3B)	22 (T_f)	T_1
	23 Trip of one feedwater pump (or condensate pump)	*	23 (T_3B)	23 (T_f)	–
	24 Feedwater-low flow	*	24 (T_3B)	*	–
	25 Low feedwater flow during startup or shutdown	*	–	–	–
	26 High feedwater flow during startup or shutdown	*	–	–	–
Reactor system flow	14 Recirculation control failure-increasing flow	*	*	*	T_2
	15 Recirculation control failure-decreasing flow	*	*	*	–
	16 Trip of one recirculation pump	*	*	*	–
	17 Trip of all recirculation pumps	*	*	*	–
	18 Abnormal startup of idle recirculation pump	*	*	*	–
	19 Recirculation pump seizure	*	*	*	–

TABLE 4.3. (cont.)

Function	IE	Category	SHOREHAM	GRAND GULF	FORSMARK 3	BROWNS FERRY
Reactor system pressure control	9	Pressure regulator fails open	*	*	* (T_t)	T_2
	10	Pressure regulator fails closed	*	*	* (T_t)	-
	12	Turbine bypass fails open	*	*	* (T_t)	-
	13	Turbine bypass or control valves cause increased pressure (closed)	*	*	* (T_s)	T_2
Reactivity control	27	Rod withdrawal at power	*	*	*	-
	28	High flux due to rod with-drawal at startup	*	-	-	-
	29	Inadvertent insertion of rod or rods	*	*	*	-
Steam	5	Main steam isolation valve	5 (T_M)	*	* (T_t)	T_1
	6	Inadvertent closure of one MSIV (rest open)	*	*	* (T_s)	T_2
	7	Partial MSIV closure	*	*	*	-
	11	Inadvertent opening of a safety/relief valve (stuck)	11 (T_I)	11 (T_3C)	Small LOCA	-
Safety injection	33	Inadvertent startup of HPCI/HPCS	*	*	-	-
Electrical power	31	Loss of offsite power	31 (T_E)	31 (T_1)	31 (T_e)	T_3
	32	Loss of auxiliary power	*	32 (T_1)	32 (T_t)	T_1
Miscellaneous	34	Scram due to plant occurrences	*	*	* (T_s)	-
	30	Detected fault in reactor Protection system	*	*	30 16,17,18 19,27,29,20, 21,22,23,24, 14,15 (T_s)	-

TABLE 4.3. (cont.)

Function	IE Category	SHOREHAM	GRAND GULF	FORSMARK 3	BROWNS FERRY
Spurious trips	35 Spurious trip by way of instrumentation, RPS fault	*	*	35 (T_s)	-
	36 Manual scram-no out-of-tolerance condition	*	*	36 (T_s)	-
	37 Cause unknown	*	*	37,34 (T_s)	-
Not included in EPRI List	- Planned shutdown	+		+ (T_m)	-
	- Loss of DC bus	+		+	-
	- Loss of instrument air	+	+	-	-

Notes: "+" taken into account in the same group
"*" taken into account but in a different category
"Blanks" indicate that the category was not considered in that PSA (in several cases the basis for the omission is provided in the PSAs)
T_T - Turbine trip; T_C - Loss of condenser; T_E - LOSP;
T_M - MSIV closure; T_I - Inadvertent open relief valve;
T_F - Loss of feedwater flow.
(T_1, T_2, T_3 - same as in WASH-1400).

In some later PSAs, the number of categories was expanded. However, in the more recent PSAs for Peach Bottom and Grand Gulf, a review was conducted on the interim ASEP [33] IE list, in order to determine whether expansion of these categories was necessary. In addition, the actual operating history for Grand Gulf and Peach Bottom was reviewed (as given in post-trip analysis reports which summarize, among other things, the causes for plant shutdowns). This information was combined into the list of transient initiators. In general, it was found that transient events listed in EPRI NP-2230 [4] could remain grouped into the three main WASH-1400 [2] transient categories, but with a small modification: in the Peach Bottom and Grand Gulf PSAs, the T3 events category (transients with PCS available) is further grouped into three subcategories:

T3B: Loss of feedwater;
T3C: Inadvertent open relief valve;
T3A: All other T3 events.

The categorization used for Peach Bottom and Grand Gulf is also in general agreement with the categorization principles developed especially for the Swedish SUPER-ASAR project [29], where the categorization of transients is primarily based on the availability of the main functions: (1) off-site power, (2) feedwater, (3) the main condenser. The intention is that the IEs within each group should be similar with respect to affected process parameters and countermeasures. The following main groups are considered:

(1) Planned shutdown;
(2) Scram due to small disturbances;
(3) Loss of main condenser;
(4) Loss of feedwater;
(5) Loss of feedwater and main condenser;
(6) Loss of 400 KV outer grid;
(7) Pipe break of different sizes and locations.

The logic of this categorization is shown in Fig. 4.1.

The main groups are then further subdivided with respect to availability of (battery) supplied grid and the status of all systems covered by technical specifications (main group 2), availability of primary system decay heat removal (groups 3 and 5), partial or total loss of feedwater (group 4) and the availability of off-site grid and generator house load operation (group 6).

The Shoreham PSA [12] considers the following groups:

(1) Transients that result in turbine trip (T_T);
(2) Transients caused by main steam isolation valve (MSIV) closure which lead to isolation of the reactor vessel from the main condenser (T_M);
(3) Transients following loss of feed water flow (T_f);
(4) Transients resulting from loss of condenser (T_C);
(5) Transient resulting from loss of off-site power (T_E);
(6) Transients resulting from inadvertent open relief valve (IORV);
(7) Orderly and controlled manual shutdown (T_I).

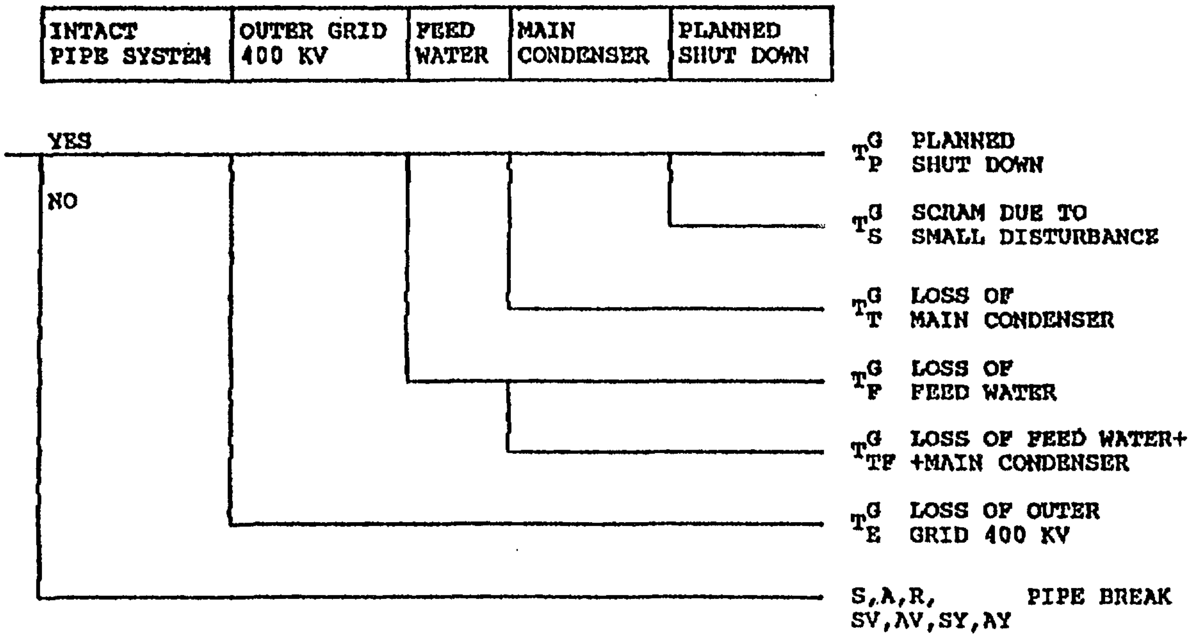

FIG. 4.1. Initiating event classification tree.

Compared to earlier PSAs, the Shoreham list has been expanded. The explanation is that the Shoreham PSA considers the severity of the different initiating events in more detail in order to achieve more realistic risk spectra. For example, event T_C (loss of condenser) and event T_M (closure of MSIV) are identical in the early stage of the event, and were treated as one group in some PSAs. Since the recovery time for loss of condenser is considered to be substantially larger than the necessary recovery time after MSIV closure, separating these initiators in two groups means getting more realistic results.

These examples are intended to give some guidance for grouping. But it has to be remembered that plant specific design or logics may necessitate different approaches. And, as mentioned earlier, the effects of the increased amount of work and the possible realism or conservatism in the result, have to be weighted against each other.

4.3.3. Special common cause initiators

Special common cause initiators such as loss of DC power, loss of instrument air or loss of component cooling influence the plant in different ways. For that reason they are normally not grouped. The discussion in Section 4.2.3 applies here too.

4.4. IMPACT OF VARIOUS TRANSIENTS ON CORE DAMAGE FREQUENCY

The impact of initiating events on core damage frequencies of the power plant can be measured by recording the conditional probability of core damage given a certain IE. Table 4.4 provides one example for a BWR plant taken from the Shoreham PSA [12] and its review [6]. Table 4.5 provides another example for a PWR plant taken from the German Risk Study, phase B [22].

TABLE 4.4. IMPACT OF TRANSIENT IEs ON THE SHOREHAM CDFs [12]

Transient initiator	Conditional probability of core damage	Initiating event frequency (per year)	Core damage frequency contribution (per year)
Turbine trip	5.5 E-6	6.85	3.8 E-5
Loss of condenser and MSIV	9.7 E-6	0.7	6.8 E-6
Loss of FW	4.4 E-6	0.1	4.4 E-7
Loss of off-site power	9.6 E-6	0.15	1.4 E-6
Inadvertent open of relief valve	4.8 E-6	0.2	9.6 E-7
TOTAL	–	8.0[a]	4.8 E-5[a]

[a] Does not include all contributions.

TABLE 4.5. IMPACT OF TRANSIENT IEs ON THE GERMAN RISK STUDY CDFs [22]

Transient initiator	Conditional probability of core damage	Initiating event frequency (per year)	Core damage frequency contribution (per year)
Loss of emergency power	1.7 E-5	0.13	2.2 E-6
Loss of FW without loss of heat sink	2.1 E-5	0.15	3.2 E-6
Loss of FW with loss of heat sink	2.3 E-5	0.29	6.7 E-6
Loss of heat sink without loss of FW	8.0 E-6	0.36	2.9 E-6
Large steam line break outside containment	2.1 E-3	5 × E-4	1.0 E-6
SG tube leakage (1–6 cm^2 area)	1.5 E-4	6.5 E-3	1.0 E-6
TOTAL	–	0.95[a]	1.7 E-5[a]

[a] Does not include all contributions.

5. DETERMINATION OF THE FREQUENCY OF INITIATORS

Several methods for the quantification of IE frequencies were employed in past PSAs. Each of the methods has its own advantages and disadvantages and is to be used in specific cases.

5.1. APPROACHES TO QUANTIFICATION

It is possible to distinguish between two extreme cases of PSA application:

(1) New plants for which no plant specific operating experience exists.
(2) Old plants for which a number of reactor-years of operating experience exist and occurrences have been experienced in a few IE categories. If the time period of plant operation is long enough, cases in which no occurrence has been recorded in a certain category may also be of significance for the quantification of the IE.

Different quantification approaches may be used for different groups of initiators. The example of division into IE groups for which different approaches may be used is as follows:

(1) General transients initiators;
(2) Large LOCAs, large steam line breaks (in containment), RPV failures;
(3) Small break LOCAs, small steam line breaks (in containment), interfacing LOCAs;
(4) SG tube rupture;
(5) Common cause initiators;
(6) ATWS initiators;
(7) Loss of off-site power.

The quantification approaches covered in this document are based on various PSAs and their reviews [9, 12, 14, 20, 29, 36]. These approaches are delineated in the following sections. For convenience they are listed here:

(1) One stage Bayesian methodology using generic experience data;
(2) Two stage Bayesian methodology using both generic and plant specific data;
(3) Mean frequencies from large operating experience data gathered over a long period;
(4) Expert opinion on rare events;
(5) Failure rates and mission time (valves, pumps, expansion joints);
(6) Failure rate per pipe length and for different pipe categories of quality (standard), diameter and environmental conditions;
(7) Fault tree analysis for special rare events such as common cause initiators. The top event being the occurrence frequency of this initiator;
(8) Use of similar plants' experience, other PSAs or generic studies such as the EPRI [4] and EG&G [5] studies;
(9) Special plant attributes and characteristics of the geographic location of the power plant.

Table 5.1 provides information on the initiator types for which the various approaches have been applied in the past.

TABLE 5.1. APPROACHES TO IE FREQUENCY DETERMINATION USED FOR VARIOUS TYPES OF IE CATEGORIES

	Quantification Approach								
IE	One Stage Bayesian	Two Stage Bayesian	Mean Frequ- encies	Failure rates & mission times	Failure rates & piping length	Fault Tree	Expert Opinion	Similar experience from other plants	Special plant and location attributes
Frequent Transient IE:									
new plants	+							+	
old plants		+	+					+	
Non Frequent Transients:									
old plants		+						+	
Large LOCAs									
Large SLB							+	+	
RPV failure									
Small LOCAs									
Small SLB									
Interfacing LOCAs				+	+			+	
SG Tube Rupture		+		+				+	
Common Cause Initiators	+			+		+			
ATWS Initiators:									
new plants	+		+					+	
old plants		+						+	
Loss of Offsite Powers:									
new plants	+								+
old plants		+	+						+

5.1.1. One and two stage Bayesian updating analyses

The Bayesian method has been used in several PSAs for evaluating the frequency of IEs. Two of the available references on the methodology are Appendix B of the Oconee PRA [9] and Chapter 5 of the PRA Procedures Guide [28]. When only generic data is available and plant specific information is not (e.g. a new plant) a one stage Bayesian analysis is performed. In this analysis the number of failures and the duration of time in which the failures occurred are the input evidence used to update a log-uniform or other type of prior distribution. The resulting posterior distribution is a generic distribution of the number of failures that can be expected for a population of plants of which the 'new plant' is a member.

When plant specific data on a number of failures and duration of time in which they were recorded is available ('old plant') a two stage Bayesian analysis can be made. The distribution of the population of plants obtained before is now taken to be the prior generic distributing and the plant specific data is the new evidence used to update the generic distribution in order to obtain a plant specific distribution of the failure frequencies. The Oconee PRA [9], the Indian Point-3 PRA [42] and other PRAs used this methodology to obtain plant specific IE frequencies of all events for which substantial plant specific occurrences were available.

5.1.2. Mean frequencies of frequent operational occurrences

This is a common approach used for quantification of anticipated transients and manual shutdowns. Limerick and Shoreham PSAs are examples were this approach was used rather than the previously described Bayesian approach. For the cases of large amount of data and when uncertainty bounds are not required, this approach may be effective.

In this approach the event records for a particular plant are searched for all relevant events that occurred during a given time period.

5.1.3. Expert opinion on rare events

Expert opinion may be used to obtain an estimate on the frequency of a rare event that has not been observed in nuclear power plant experience. A case in which expert opinion was used in the past is in the evaluation of seismic hazard to nuclear power plants in eastern USA. This was done because some of the data needed could not be obtained otherwise.

The frequency of RPV failure is also based in part on expert opinion. In this case the experts had to evaluate to what extent nuclear power plants are more reliable than the piping used by the non-nuclear industry, in which failure statistics could be obtained and evaluated. Similarly, the large LOCA frequency was obtained in the RSS.

The PRA Procedures Guide [28] provides some additional information on the treatment of expert opinion. A word of caution is always in place, that large uncertainties can be associated with expert opinion, so that the use of this approach requires careful planning and documentations to allow peer review.

5.1.4. Frequency estimation by evaluation of failure rates and mission times

This is an approach frequently used in PSA to evaluate various plant specific initiating events. The cases of valve and pipe breaks or leakages such as interfacing LOCA, a steam line break in a certain plant location and internal flooding [9, 15] are examples. In all these cases sources of valves and pipe failure rates are searched for the applicable values for the particular plant, based on piping/valve quality, type and size. Table 5.2 provides a comparison of the motor operated valve (MOV) failure rates taken from various sources, including PSA applications and other special studies.

For an interfacing LOCA frequency determination, additional information is needed on valve failure by rupture or leakage. Table 5.3 provides a comparison of several sources on valve failure, rupture and leakage rates as used in several PRAs and other special probabilistic studies. While the RSS used data from non-nuclear sources, NUREG/CR-1363 [37] uses LERs from many plants, most of them including small as well as very small leakages (technical specifications exceedance). The last two sources in Table 5.3 [38, 39] use more recent LERs. The data provided in the tables is used to estimate interfacing LOCA frequencies according to the number of valves available in every leak path identified in the particular plant under review, and on the basis of the testing intervals (e.g. relevant mission time to be applied to the failure rates) used on each leak path's through valves.

5.1.5. Fault tree analysis for special rare events

If a plant never experienced an initiating event, its frequency can be estimated using a fault tree approach. A fault tree is then constructed to include all equipment (and possibly human errors) contributing to an initiating event. Using plant design information and plant specific failure data, the frequency of IEs will be generated. A few examples taken from PSAs are given in Section 6.2.

5.1.6. Use of experience from other plants

This approach is used for plants having a similar design or vendor. In the case of new plants from the same vendor, the studies that are available can be applied to a group of plants of the same design. This group can be used as a source of data for the new plant. The approach is used in PSA performed for new plants of a design similar to those plants that have been in operation for some time.

Another utilization of this approach is for comparison or for obtaining a range of values to estimate the frequency of an initiating event. This is especially important when an IE is considered for exclusion from a specific PSA (for example, because the particular plant under consideration has several mitigating systems to render any accident sequence of this IE).

5.1.7. Special attributes of plant and location

For obtaining loss of off-site power (LOOP) frequencies a specific approach has been developed. An extensive study of factors contributing to the frequency of LOOP in nuclear power plants was performed in the USA with respect to the power plant blackout study. In the study the relation between LOOP frequency and plant specific station switchyard design features were investigated. Similarly, the correlation between LOOP frequency and number of off-site lines connected to the station switchyard were established. The impact of weather conditions in various locations on the LOOP frequency, as well as some additional factors were studied.

The NUREG-1032 report [26] is a manual to determine a frequency of LOOP for a certain plant out of plant specific information and its general attributes.

This approach is particularly suitable for new plants. For the older plants that have accumulated specific evidence of LOOP in their area, the two stage Bayesian approach using the NSAC-80 [10] could also be a suitable approach.

TABLE 5.2. MOTOR OPERATED VALVES FAILURE RATES

Source	Failure mode	Assessed range	Mean value
RSS [2]	Failure to operate (include command)	3E-4 to 3E-3/d	1.3E-3/d
NUREG/ CR-1363 (for BWRs) [37]	Failure to operate (include command)	—	8E-3/d
NUREG/ CR-1363 (for BWRs) [37]	Failure to operate (w/o command)	—	6E-3/d
Command failure of both MOVs (inboard and outboard)	Failure of inboard and outboard MOVs	—	2E-3/d
NUREG/ CR-4050 Generic data [6]	Failure to operate (include command)	1E-3 to 9E-3/d	3E-3/d
SNPS-PRA [12] App. A.2[a]	MOV spurious opening	—	1.6E-/h
CHU-1987 [39]	MOV Spurious opening	—	9.2E-8/h
CHU-1987 [39]	Failure to close	—	3.9E-6/d
CHU-1987 [39]	Inadvertent opening	—	1.2E-3/d

[a] Based on GE evaluation.

TABLE 5.3. CHECK AND MOTOR OPERATED VALVE RUPTURE OR EXCESSIVE LEAKAGE RATES

Source	Failure mode	Assessed range [h^{-1}]	Mean value [h^{-1}]
RSS [2]	Internal leakage (severe)	1E-6–1E-7	2.8 E-7
SEABROOK PRA [38]	Internal leakage (severe) (all sizes)	 5E-8–1E-10 5E-6–5E-8	1E.9 5E-7
NUREG/ CR/1363 [37]	Internal leakage (all sizes)	–	1E-6
CHU-1987 [39]	Internal leakage (all sizes)	–	3.4E-7
RSS [2]	*Rupture*	*1E-7–1E-9*	*2.7E-8*
NUREG/ CR-1363 [37]	External leakage/rupture	–	7E-8
SEABROOK PRA [38]	Rupture	–	<5E-9[a]
CHU-1987 [39]	Disk separation	–	1.4E-7

[a] Never occurred in more than 10 000 valve-years.

5.2. SOME EXAMPLES OF IE FREQUENCIES IN PSA

The Appendix provides a database printout which includes IE lists and their frequencies from a large number of published PSAs. In this section several examples of such IE lists with frequencies of occurrence are given and provide cross-reference to additional lists of this type given in previous sections.

Table 2.1 provides the Shoreham (BWR) transient IE frequencies.
Table 2.2 provides several PSA LOCA Frequencies for BWR and PWR plants.
Table 3.7 provides the German Risk Study (PWR) IE frequencies.
Table 5.4 provides the Grand Gulf (BWR) IE frequencies.
Table 5.5 provides the Big Rock Point (BWR) IE frequencies.
Table 6.1 provides average B/W (PWR) frequencies for ATWS IEs.

TABLE 5.4. GRAND GULF [36] INITIATING EVENT GROUPS[a] AND FREQUENCIES

Initiator nomenclature	Description	Mean Frequency (per year)
T1	Loss of off-site power (LOOP) transient	0.11
T2	Transients with loss of power Conversion system (PCS)	1.62
T3A	Transients with PCS initially available	4.51
T3B	Transients involving loss of feedwater (LOFW) but with the steam side of the PCS initially available	0.76
T3C	Transient caused by an inadvertent open relief valve (IORV) on the reactor vessel	0.14
TIAS	Transient caused by loss of instrument air	8.1E-4
A	Large loss of coolant accident (LOCA)	1.0E-4
S1	Intermediate LOCA	3.0E-4
S2	Small LOCA	3.0E-3
S3	Small-small LOCA (recirculation pump seal LOCA)	3.0E-2

[a] For the distribution of the individual transient initiating events over the groups of transients given above, see Section 3.4.2 (Table 3.11).

TABLE 5.5. INITIATING EVENTS FOR BWR PRA [46] FOR WHICH EVENT TREES WERE DEVELOPED

Initiating event	Frequency (per year)
Turbine trip	1.4
Loss of main condenser	6.0E-2
Spurious closure of MSIV	6.0E-2
Loss of feedwater	1.6E-1
Loss of off-site power	1.3E-1
Loss of instrument air	6.0E-2
Spurious opening of turbine Bypass valve	1.0E-1
Spurious opening of RDS Isolation valve	1.2E-3
Spurious closure of both recirculation line valves	1.7E-2
Stuck-open safety valve	2.6E-4
Interfacing LOCA	2.0E-3
High energy line break in recirculation pump room	3.9E-7
High energy line break in pipe tunnel	3.8E-6
Small LOCA	1.0E-3
Medium LOCA	1.0E-4
Large LOCA	1.0E-5
Small steam line break inside containment	1.0E-3
Medium steam line break inside containment	1.0E-4

6. EXAMPLES OF FREQUENCY DETERMINATION

This Section gives several selected examples of how some of the approaches discussed in Section 5 have been used in past PSAs and in the evaluation of operating experience.

6.1. EXAMPLE OF THE USE OF THE MEAN FREQUENCIES APPROACH

Anticipated transient without scram (ATWS) events are sometimes modelled as a separate class of initiating events [9, 12, 20, 22]. One of the reasons for their treatment as a separate IE (rather than branching out from transient IEs) is the fact that not all transients can lead to an ATWS. Transients that start at a low power level in the core would in most cases be controlled (because of negative reactivity coefficient feedback) at a power level which is within the cooling capabilities of the core cooling systems. Only transients starting at a high power level have a high chance of threatening the core integrity.

The EPRI and EG&G reports [4, 5] include separate tables on transients experienced at power levels of 25 to 110% of full power. Estimation of ATWS IE frequency can proceed using those tables as the source for operating experience. For Oconee (a PWR plant) 41 different IEs are reported in the EPRI tables. They should be reduced into a smaller group of transient initiators.

The Oconee PRA [9] starts with the data for B/W plants rather than the total PWR database in Ref [4]. The B/W plants have a relatively higher frequency of shutdowns and many more cases of loss of main feedwater than other types of PWR. The EPRI report includes 500 events in 48 plant-years for eight B/W plants. This data for 41 IEs was condensed into 12 transient categories. Out of these 12 categories, eight have been discarded for the following reasons:

– MSIV closure and RCS loop startup:	Not applicable to Oconee-3.
– Load increase and excessive cooldown:	Insignificant contributors because of their low frequency and comparable consequences to the other.
– Control rod withdrawal and boron dilution:	Bounded in their consequences by other more frequent transients.
– Loss of RCS flow and RCS depressurization:	Bounded by LOOP discussed later and small LOCA without scram. The latter is a less severe ATWS case.

The four ATWS categories retained were:	0–110%	25–110% power
– Loss of condenser vacuum	0.2/year	0.1/year
– Turbine trip	5.7/year	3.9/year
– Loss of main feedwater	0.7/year	0.2/year
– Loss of off-site Power	0.2/year	0.1/year
Total	6.8/year	4.3/year

In deriving the above frequencies the EPRI data was used and average frequencies were obtained from using the number of events that occurred in each category and the time duration in which these occurrences happened.

Table 6.1 is a summary of the collapse scheme of the 41 transients into the 12 transients considered first (four of them retained for the ATWS analysis). The table displays the frequencies evaluated in the Oconee PRA for the ATWS initiators based on data for B/W plants taken from the EPRI data [4].

6.2. EXAMPLES OF DETERMINATION OF THE FREQUENCY OF COMMON CAUSE INITIATORS

Common cause initiators (CCIs) are events which, in addition to plant trip, result in the degradation of one or more mitigation systems. A list of CCIs evaluated in various PSAs was given in Section 4.2.3. Here several examples of CCI frequency determination are summarized.

(a) Loss of 480 V bus (Surry-NPP [40])

Here the failure rate of the transformer is used (10^{-6}/h) with a mission time of 8760 hours to calculate a mean frequency of loss of 480 V bus which is al to 9×10^{-3}/year. The transformer failure rate was evaluated by reviewing several sources [9, 14, 41, 42, 44] to obtain a range of data. The median of this range was chosen. Since the transformer failure rate was significantly larger than the bus failure rate, the latter was neglected. Similar analysis was performed in the Oconee PRA where a value of 5.4×10^{-3}/year was found for IE 'loss of power to 4-KV switchgear 3TC'.

(b) Loss of charging pump cooling (Surry-NPP [40])

The charging pump cooling depends on component cooling/service water system operation. The frequency of the event was derived by comparing generic data from Ref. [5] with Surry plant specific data from an LER search.

(c) Loss of instrument air system (Oconee PRA [9])

In this case a fault tree for the top event 'instrument air system failure' was constructed. Failure rates for all basic events were generated and the top event frequency determined. Loss of instrument air event is a combination of several causes; the frequency is given below:

Contamination:	1.7×10^{-5}/h
Pipe rupture:	8.3×10^{-6}/h
Loss of station air and one instrument-air compressor:	1.1×10^{-6}/h
Other combination of station and instrument-air failures:	9.1×10^{-7}/h
Total:	2.7×10^{-5}/h or about 0.2/year.

TABLE 6.1. COLLAPSE OF 41 PWR IE CATEGORIES INTO 12 ATWS GROUPS USED IN OPRA [9, 15]

Transient Category Group	EPRI Initiating Event Categories Type (numbers correspond to events in table 4.1)	Number and Frequency of Occurrences in B/W Plants for Second Year and Later			
		0.110% power*	25-110% power**	Frequency 0-110%	Frequency 25-110%
Loss of Condenser Vacuum	25, 27, 30	8	3	0.20	0.10
Turbine Trip	3, 8, 12, 15, 19, 23, 28, 33, 34, 36, 38, 39, 40	314	104	7.90	3.61
Loss of main feedwater	16, 22, 24	30	7	0.75	0.24
Loss of offsite power	35	4	1	0.10	0.03
Load increase	21, 26, 29	5	0	0.13	<0.01
Loss of RCS Flow	1, 14	13	5	0.33	0.17
Control rod withdrawal	2	3	2	0.08	0.07
RCS depressurization	4, 5, 7	2	2	0.05	0.07
Boron dilution	11	0	0	<0.01	<0.01
Excessive cooldown	6, 9, 20	0	0	<0.01	<0.01
MSIV closure	17, 18	0	0	N.A.	N.A.
Inactive-RCS-loop startup	13	0	0	<0.01	<0.01
TOTAL		379	124	9.54	4.29

* Number of events (0 to 110% power) for the second year of operation and later on (until 1982 - 39.8 year of operation)
** Number of events (25 - 110% power) for the second year of operation and later on (until 1982 - 28.8 years of operation)

(d) Steam line break (Oconee PRA [9])

This event was evaluated on the basis of operating experience and two stage Bayesian updates: one event in 205 reactor-years was assumed as prior and no event in 19 reactor years as updating evidence. This resulted in an estimated frequency of 3×10^{-3}/years for this IE in Oconee. The same approach was used to determine the frequency of the 'steam generator tube rupture initiator'.

6.3. EXAMPLES OF DETERMINATION OF THE FREQUENCY OF LOCAs

Table 5.1 lists several approaches to the determination of LOCA frequencies. Section 5.2 discusses these approaches further. In the following we present several examples taken from actual studies:

6.3.1. Maine Yankee PSA [45]

The approach applied for the Maine Yankee PSA was to use similar experience from other plants. It used the WASH-1400 generic frequency for medium and large LOCAs (8.0E-4, 3.0E-4 respectively). For small LOCAs (three different subgroups) and for rupture of single SG tube, it utilized a combination of generic and plant specific data as follows:

(a) Small LOCAs can be the results of small pipe breaks from failure of reactor coolant pump seals [52] or can be induced by a transient event that challenges the PORV or other safety valves. Based on previous PSAs a generic value of 7×10^{-3} per year total was used. Based on plant specific information, a further breakdown was made as follows:

– Very, very small LOCA	5.10^{-3}/year;
– Very small LOCA	1×10^{-3}/year;
– Small LOCA	1×10^{-3}/year.

all of them have a range factor (uncertainty factor) of 5.

(b) Steam generator tube rupture is a very, very small LOCA that has the potential to bypass containment and is therefore treated separately. The frequency of this IE was calculated as follows:

Maine Yankee has 17 100 tubes of which 70 are plugged. A generic frequency of 1.4×10^{-6} per tube-gear was adopted from a previous PWR PSA which results in a SG tube rupture initiator probability of:

$1.4 \times 10^{-6} \times 17030 = 2.4 \times 10^{-2}$/year.

6.3.2. Interfacing LOCA frequency based on pipe and valve rupture/leakage failure rates

The interface (or intersystem) LOCA is defined as a leak from RCS due to rupture of low pressure piping connected to the primary system as a result of overpressurization which is due to loss of isolation between low and high pressure piping. An estimate of the frequency of interfacing LOCA may be determined based on the actual length of low pressure piping and the number of pipe sections between the valves. In such an evaluation the detail of the length of all piping paths from the RCS to the area outside containment is listed. For each path the pipe sections and associated valves are identified. Data on valve rupture/leakage rates such as in Tables 5.2 and 5.3 are used. Table 6.2 summarizes various paths to outside containment including their estimated frequencies (from Ref. [48]).

6.4. EXAMPLES OF DETERMINATION OF THE FREQUENCY OF TRANSIENTS

6.4.1. Maine Yankee PSA [45]

Maine Yankee is a PWR type plant which has been in operation since 1972. Therefore, the approach used for determining transient frequency is relying on the plant's own operating experience (i.e. 'old plant approach').

Three data sources were reviewed in the process of determining the transient frequencies:

(a) Outage and load reduction information (1/1/73 to 3/28/87);
(b) Scram database (for a similar period);
(c) The NUREG/CR-3862 [5] transient occurrences for Maine Yankee.

List (a) was used from the second year of operation (1/1/74). It correlates well with the other two sources in most cases. In general the in-plant records were the main source used in the frequency determination of the more frequent transient IEs. The list of plant trips was reviewed to identify events that occurred at a power level greater than 15% and were unscheduled. Over the thirteen years considered in the PSA, a relatively constant average frequency of nine events per year was observed.

Based on a review of plant response to transient initiators, it was concluded that three types of event should be considered in the Maine Yankee PSA:

(a) Normal plant trip (do not fail any additional equipment);
(b) Closure of all flow check valves or loss of condenser vacuum;
(c) Loss of all condensate pumps.

The frequency of IE (a) was determined for plant records as discussed above. The frequency of IE (b) was determined based on NUREG/CR-3862 [5] data for Maine Yankee which indicated a mean value of 0.04 per year for the closure of the check valves. The frequency that the NUREG/CR-3862 indicated for Maine Yankee on loss of condenser vacuum was too high compared to actual plant specific experience. Therefore the same frequency was used for both contributors in that category. It resulted in the total frequency of 0.08 per year for this class of IEs. The frequency of case (c) was determined to be 0.02 per year based on NUREG/CR-3862 data but modified to a somewhat higher value according to engineering judgement of the uncertainty involved in this event.

In addition to the above mentioned four categories of IEs additional special plant specific IEs considered in the Maine Yankee PSA include:

(a)	Loss of service water	2.0×10^{-4}/year;
(b)	Loss of secondary component cooling	2.3×10^{-2}/year;
(c)	Loss of primary component cooling	2.3×10^{-2}/year;
(d)	Loss of control air	3.2×10^{-2}/year.

TABLE 6.2. ESTIMATED FREQUENCIES OF BREAKS OUTSIDE CONTAINMENT

BREAK LOCATION	CASE	BREAK SIZE	NUMBER OF: LINES	NUMBER OF: SECTIONS	NUMBER OF: VALVES (*)	ISOLATION VALVES: VALVES DESIGNATORS	ISOLATION VALVES: ASSUMED FAILURE PROBABILITY	INITIAL BREAK FLOW: STEAM OR LIQUID	ESTIMATED FREQUENCY OF BREAK OCCURRENCE	DESCRIPTION OF THE CASE ANALYZED
Reactor Core Isolation Cooling (RCIC) Steam Line	I	4"	1	1	1	IE51-MOV041	1.0	steam	2.1E-6	Break exclusion section and valve between Reactor Building penetration and the outboard MOV 042. (Elevation 87).
	II	3"	1	6	6	IE510MOV041 and MOV042	1.0	steam	5.8E-6	Non break exclusion sections and valves from outboard isolation valve up to RCIC turbine. Four openings per year of valve -042 are assumed (All elevation below elevation 87 down to elevation 8).
	III	1"	1	14	14	IE51-MOV048 and IE51-MOV047	1.0	steam	1.2E-3	Non break exclusion section and valves from Reactor Building Penetration up to the 1-1/2 HPCI/ RCIC drain line to condenser. Normally open (Elevation 87 down to Elevation 8).
RCIC/HPCI Steam Drain Line	I	1-1/2	1	1	0	IE51/IE41 AOV-081 or AOV-082	1.0	steam	5.E-5	Section between HPCI and RCIC drain lines connection and the penetration to the main steam tunnel. (Between elevation 11 and 70).
Reactor Water Cleanup System (RWCU) Supply Line	I	6"	1	1	1	MOV033 (F001) and MOVF102 and MOVF100 F106	1.0	Liquid	2.1E-6	First section in Reactor Building. It is break exclusion and normal operating. (Elevation 112)
	II	6"	1	1	0	The above and IG33-MOV034 or(F004)	1.0	Liquid	7.5E-6	Section from outboard isolation valve to the 6x3" reducer. Non break exclusion (Elevation 112).
	III	3"	2	3	3	same as above	1.0	Liquid	5.4E-4	Section and valves from reducer up to RWCU pumps. (Elevation 112)

The frequency of these events was estimated using special plant specific models such as fault trees.

6.4.2. German Risk Study (GRS) PSA phase B [22]

The German PSA (phase B) included two main types of transient initiators:

(a) Anticipated transient

- Loss of AC power	0.18/year
- Loss of main feedwater w/o loss of heat sink	0.1/year
- Loss of main feedwater with loss of heat sink	0.29/year
- Loss of main heat sink w/o loss of FW	0.36/year
Total:	0.93/year

(b) Unlikely transients

- Large leakage in steam line within containment	-1.6×10^{-4}/year
- Large leakage in steam line outside containment	-4.8×10^{-4}/year
- Medium leakage in steam line within containment	-2.7×10^{-5}/year
- Medium leakage in steam line outside containment	-1.1×10^{-4}/year
Total:	7.8×10^{-4}/year

(c) ATWS (various transient sources)

The frequency of case (a) were calculated using the two stage Bayesian updating with plant specific data. The plant data was based on the Biblis B experience since it was connected to the grid in 1977. Some modifications in the system design that were made in Biblis were taken into account. The prior distribution was assumed to be non-informative (uniform). The update was made using Gama distribution. The frequency of loss of AC power was calculated based on a database larger than the Biblis B database.

6.5. EVENTS IN SHUTDOWN

Traditionally, PSAs have been modelling plants assuming full power operation. Several incidents involving loss of decay heat removal capability during shutdown pointed out the vulnerability of the plant in the shutdown mode, and eventually prompted the performance of PSA studies for other operating modes than full power.

Two of the PSA studies for French 900 and 1300 MW(e) reactors performed respectively by the Commissariat à l'énergie atomique (CEA) and Electricité de France (EDF) indicated that a high percentage of the total risk is actually coming from the shutdown state. Similar events are currently being studied elsewhere.

Although loosing decay heat removal is the major event to be analysed in shutdown, if the analyses covers other non power states, additional events should be considered. In the CEA study of French 900 MW(e) reactors [49] the following non-power states were considered:

- period while the reactor is shut down, with temperature below 280°C and pressure below 133 bars, but above residual heat removal (RHR) system parameter;

- primary circuit full of water, RHR connected and in operation;
- primary circuit partially drained;
- refuelling, with reactor vessel head open and pond above reactor vessel full.

The list of initiators considered for some non-power states in this study includes:

- loss of primary coolant;
- SG tube rupture;
- LOCA outside containment;
- loss of electric power supply;
- dilution accident;
- loss of RHR.

In a recent study [47] performed at the Sandia National Laboratories, USA, initiating events for the operating modes other than full power operation for a BWR plant have been determined. The operating modes considered were:

- low power operation (less than 15% power);
- startup;
- hot shutdown;
- cold shutdown;
- refuelling.

The general criteria for determining the initiators were: (1) any disruption from normal operation requiring rapid shutdown and challenging safety systems to remove decay heat (for modes 1 and 2); and (2) any event which requires an automatic or manual action to prevent core damage (modes 3, 4 and 5).

The initiating events identified were divided into five major groups:

(1) Transients;
(2) LOCAs;
(3) Decay heat removal (DHR) challenge events;
(4) Special events;
(5) Hazards.

The first step in identification of initiating events was the evaluation of applicability of initiators considered in full power mode for each of the non-power modes. Additionally, initiators unique to modes other than full power were postulated. New types of initiators have been identified in shutdown modes (modes 3, 4 and 5) and were basically due to changing plant states (equipment out of service, etc.) and increased human activities in those modes. Table 6.3 summarizes potential initiating events identified for each of the non-power operating modes.

TABLE 6.3. POTENTIAL INITIATING EVENTS AND OPERATING MODES TO WHICH THEY APPLY

INITIATING EVENTS	MODE 1L	MODE 2	MODE 3	MODE 4	MODE 5
<u>TRANSIENTS</u>					
T1: Loss of Off-Site Power	X	X	X	X	X
T2: Loss of Power Conversion Sys	X	X			
T3A: Power Conversion Sys. Avail.	X	X			
T3B: Loss of Feedwater	X	X			
T3C: Inadvertent Open Relief Vlv.	X	X			
<u>LOSS OF COOLANT ACCIDENTS (LOCAs)</u>					
A: Large LOCA	X	X	X	X	X
S1: Intermediate LOCA	X	X	X	X	X
S2: Small LOCA	X	X	X	X	X
S3: Small-Small LOCA	X	X	X	X	X
H1: Diversion to Suppression Pool		X	X	X	X
H2: Diversion to Condenser		X	X	X	X
J: LOCA in Connected System		X	X	X	X
K: Test/Maintenance-Induced LOCA		X	X	X	X
<u>DHR CHALLENGE INITIATORS</u>					
E1A: Loss of Condensate (DHR)			X	X	
E1B: Loss of RHR-Shutdown Cooling		X	X	X	X
E1C: Loss of RWCU (DHR)					X
E1D: Loss of Alternate DHR System				X	X
E1E: Loss of RCIC		X	X		
E2A: Isolation of Condensate			X	X	
E2B: Iso. of RHR-Shutdown Cooling		X	X	X	X
E2C: Isolation of RWCU (DHR)					X
E2D: Iso. of Alternate DHR System				X	X
E2E: Isolation of RCIC		X	X		
<u>SPECIAL EVENTS</u>					
Criticality Events:					
T4A: Rod Withdrawal Error	X	X	X	X	X
T4B: Refueling Accident					X
Loss of Support Systems:					
T5A: Loss of Standby Serv. Water		X	X	X	X
T5B: Loss of Turb Bldg Cool Water		X	X	X	X
T5C: Loss of Plant Service Water				X	X
TIAS: Loss of Instrument Air Sys.	T2	X	X	X	X
TEP: Loss of electric power			X	X	X
Other Events:					
TORV: Inadvertent Open Relief Valve-Shutdown			X		
TIOP: Inadvertent Overpressurization		X	X	X	
<u>HAZARDS EVENTS</u>					
Internal Fire	X	X	X	X	X
Internal Flooding	X	X	X	X	X

T2 - Included Under T2 Transient

7. INITIATING EVENTS FOR WWER PLANTS

Most of the eastern European countries operating WWER units have now their own PSA programme. Two Level 1 PSA studies have been completed to date: the Greifswald NPP PSA, Germany, and the Loviisa NPP PSA, Finland. In addition to these analyses, a so-called 'Top Level Risk Study' has been performed recently in Bulgaria for the Kozloduy NPP, Units 1 to 4 [60], which involves limited scope PSA as well. The other ongoing PSA analyses for WWER plants show variety with respect to the status of completion: some are 'semi-finished' and some are in a preliminary state. While the other examples of PWR IEs appearing in this document are taken from completed studies, the information given in this Section represents the current status of selection, grouping and quantification of IEs for WWERs.

The collective term 'WWER' covers three types of PWR with different design characteristics:

(1) WWER-440/230;
(2) WWER-440/213;
(3) WWER-1000/320.

Due to the fact that at present PSA of WWER-440 type reactors is better developed than the WWER-1000 analyses, qualitative information and quantitative data provided in this Section primarily relate to WWER-440 reactor types, except for some references to WWER-1000 data, such as the accumulated operating experience for WWER-1000 plants in Section 7.3.

The 230 and 213 models represent two generations of the WWER-440 type reactors having distinctive safety features. The main differences in safety characteristics of the basic 230 and 213 models are briefly described below.

The maximum design basis accident is a LOCA with break size 100 mm for the 230 reactors and large LOCA with break size 500 mm (equivalent to the diameter of the main primary coolant pipeline) for the 213 reactors. The primary cooling circuit equipment and components of the 213 model are housed in a leaktight containment to cope with 0.15 MPa overpressure. No real containment but a pressure resistant confinement exists for the W230 type. The 213 type plants are equipped with core flooding hydroaccumulators as well as a bubbling condenser-vacuum system, while the 230 plants are not. The emergency core cooling system of the basic 230 model have a redundancy of 2 × 100%, designed to cope with the 100 mm LOCA. The emergency core cooling system of the 213 type consists of the high and low pressure systems, both with redundancy of 3 × 100% designed to cope with the 500 mm LOCA.

In the following sections the 213 and 230 types of WWER-440 reactors are treated together in order to cover the issue of IE definition for both models. Therefore the WWER initiating list (definition and grouping) and the associated frequencies should be handled with care to avoid misinterpretation of the information, e.g. use of an IE irrelevant to a given WWER type plant.

Within the framework of an IAEA project set up to help PSA activities of WWER owners and to promote the exchange of information among them, a specific workshop was held in Řež, Czechoslovakia, in February 1992. At that workshop a section was devoted to the definition, grouping and frequency determination of IEs for WWER type reactors. Sections 7.1 to 7.3 summarize the achievements of that common effort, reflecting also the contributions from the participating countries (Bulgaria, Czechoslovakia, Germany, Hungary, Poland and Russia). Regarding the determination of IE frequencies, some additional data is provided in Section 7.3 taken from the Kola-1 PSA study and from the 'Top Level Risk Study for Kozloduy' [60] besides the IE frequencies made available during the Řež workshop.

7.1. SELECTION OF WWER INITIATING EVENTS

The WWER operators had developed their own initiating event lists. At Řež the selection methodologies used by the participating teams were discussed in detail. The selection techniques followed by them are given below.

(A) Approach used for Kozloduy NPP, Bulgaria

In the identification of IEs for the Kozloduy NPP, the actual plant operating experience was used as much as possible. However, the major procedural steps were as follows:

(1) Generic IE lists and the experience gained throughout the world helped to form an initial cut of IEs and their sub-events.

(2) Plant specific information from the operating history was then used to modify this list.

(3) A list of IEs was derived as a result of steps (1) and (2) above.

(B) Approach used for Dukovany PSA, Czechoslovakia

(1) Selection of IEs (sources):

(a) Accident analysis in CSFR safety analysis reports.

(2) Selection criteria:

(a) IE must cause reactor scram; and/or
(b) A selected IE must cover several individual similar IEs.

(3) Based on the above methodology a list of 14 IEs was derived.

(C) Approach used in Greifswald PSA, Germany

(1) Reference to previous PSAs (especially to the German Risk Study (GRS) [22]).

(2) Use of the EPRI list of IEs [4].

(3) Use of the draft of German PSA guidelines.

(4) Engineering evaluation; technical study of the plant.

(5) Analysis of the plant specific operating experience.

(6) Performing a limited scope PSA, which excludes many of the initiating events.

(D) Approach used for Paks PSA, Hungary

(1) As a first step a generic list was gathered from publicly available PWR PSAs. This step considered the following sources:

(a) EPRI-NP-2230, Interim Report (1982) [4];
(b) Results of the German Risk Study, Phase B [22];
(c) Studvik SUPER-ASAR Report, 1987 [29];
(d) Oconee PRA-NSAC-60, 1984 [9];
(e) Zion PRA, 1981 [44];
(f) Seabrook station PRA, PLG-0300, 1983 [14];
(g) IAEA Initiating Events Database, 1988;
(h) A list of IEs especially made for WWER type reactors by the Soviet specialists for the former CMEA programmes.

(2) As a second step, the above sources were used to construct a comprehensive IE list of 96 IEs.

(3) The list was then discussed in an expert group consisting of specialists in PSA, thermohydraulic analysts, system analysts and experts from the operating personnel. As a result of this careful selection process a list of IEs was generated consisting of 66 different initiators.

(4) A further step was taken, of engineering evaluation, using master logic diagram (MLD) methodology, operating experience and comparison with results from other PSA studies.

(5) Further reduction of the IE list will be carried out on the basis of thermohydraulic analyses as they become available.

(E) Zarnowiec approach

(1) Several generic PSA studies, devoted to IEs, and several plant specific PSA reports were reviewed in order to identify candidates for WWER plant specific IEs. These sources included:

(a) EPRI categorization scheme as described in NUREG/CR-3862 [5];
(b) IAEA IE database (Status Dec. 1988);
(c) PRAs for NPPs with PWR type reactors such as Oconee PRA [9], Surry [40], Borselle;
(d) Available information on PSA results for WWER-440 power plants such as Loviisa, Kola, Dukovany.

(2) The information from all the above sources was carefully evaluated, and a generic type list of IEs was generated.

(F) Approach used for WWER IE selection, Russia

(1) In general, the approach is based on IAEA and NRC PSA procedure guides.

(2) As a first step a comprehensive list of possible IEs is composed by using:

(a) A detailed analysis of the plant;
(b) Operating experience from similar PSAs or NPPs;
(c) Environmental studies;
(d) Results from probabilistic assessment of equipment rupture possibilities;
(e) Generic databases (IAEA and others);
(f) The list of design basis accidents;
(g) Supervision organization (licensing authorities) requirement.

(3) All the above information is evaluated taking into account:

(a) Is the IE suggested a simple or a complex IE including several dependent or independent events?
(b) Where does the IE occur (reactor, spent fuel storage pool, etc.)?
(c) Duration of the IE. This applies to loss of off-site power and spurious opening of valves.
(d) The size of the failed device.
(e) The state of the NPP when the IE took place (power operation, refuelling, startup after refuelling, shutdown).
(f) Is the IE within the design basis? If it is, the plant response is known from safety analysis. If not a design basis IE then the plant response needs first of all to be understood, in order to facilitate evaluation of the suggested IE.
(g) Only IEs requiring a safety function performance are included in the Russian list of IEs.

As a result of the individual activities in the field of IE selection, the following sources were used to develop a generic IE list for Level 1 PSA of WWER-440 type reactors:

– List of IEs for Kozloduy PSA activities, Bulgaria;
– List of IEs selected for the preliminary NPP Dukovany PSA Study, Czechoslovakia;
– List of IEs used for Greifswald PSA, Germany;
– List of IEs for Level 1 PSA of Paks NPP, Unit 3, Hungary;
– Generic list of IEs, Poland;
– List of IEs for WWER-440 reactors, Russia.

The IE lists mentioned above were compared and evaluated in order to create as complete as possible a WWER-440 initiating event list. The IE list derived from this process is provided in Table 7.1 for LOCAs and in Table 7.2 for transients and CCIs, both taken from Ref. [59]. The list covers 80 internal initiators during full power operation. External and internal hazards as well as events during shutdown or low power operation are not included.

TABLE 7.1. CLASSIFICATION OF LOCAs FOR PSA OF WWER-440 PLANTS

Initiating event	Rationale for inclusion
1. GROSS REACTOR VESSEL RUPTURE	It causes an immediate loss of all cooling water in the reactor vessel. The mitigating systems are considered to be ineffective. Note: in case of smaller reactor vessel damage the low pressure injection can be effective.
2. LARGE LOSS OF COOLANT ACCIDENTS (OVER DN 200 mm)	
2.1. Large LOCA: loops 2, 3 and 5 cold leg	The break location affects the effectiveness of the high pressure emergency core cooling system due to the fact that the high pressure ECCS lines are connected to the cold leg of loops 2, 3 and 5 respectively. Both hydroaccumulators feeding to the downcomer are thought to be ineffective.
2.2. Large LOCA: loops 2, 3 and 5 hot leg	In this case the water fed by the hydroaccumulators connected to the upper plenum is lost. With respect to the core flooding system only the other two hydroaccumulators can mitigate the consequences of the initiating event. The effectiveness of high pressure injection is not affected as much as in the case of breaks on the cold leg of loops 2, 3, 5.
2.3. Large LOCA: loop 4 cold leg	One train of the low pressure ECCS is connected to loop 4, therefore this low pressure line is considered to be ineffective in addition to the hydro-accumulators feeding to the downcomer.

TABLE 7.1. (cont.)

Initiating event	Rationale for inclusion
2.4. Large LOCA: loop 4 hot leg	Similarly to the break on the cold leg of loop 4, one low pressure pump is almost lost in the case of this initiating event. On the other hand, the hydroaccumulators feeding to the upper plenum cannot be effective.
2.5. Large LOCA: loops 1 and 6 cold leg	No mitigating system line is connected to loops 1 and 6. Thus, this break location reduces the effectiveness of only the hydroaccumulators connected to the downcomer.
2.6. Large LOCA: loops 1 and 6 hot leg	The mitigation provided by the hydro-accumulators that are connected to the upper plenum is degraded by this initiating event. The effectiveness of the other mitigating systems are not affected by the location of the break itself.
2.7. Rupture of connection line of hydroaccumulator feeding down-comer	One low pressure ECCS line is connected to this hydroaccumulator line. Thus, one low pressure pump is lost in addition to the two 'lower' hydro-accumulators if this event occurs.
2.8. Rupture of connection line of hydroaccumulator feeding upper plenum	Concerning the effectiveness of the mitigating systems the initiating event implies the same consequences as the other hydroaccumulator line break (see IE 2.8). The only difference is that the other two hydroaccumulators are lost in this case.

TABLE 7.1. (cont.)

Initiating event	Rationale for inclusion
3. MEDIUM LOSS OF COOLANT ACCIDENTS (DN 32–200 mm)	
3.1. Medium LOCA not affecting emergency core coding system operation	No degradation is considered in the mitigating systems owing to the location of the break.
3.2. Medium LOCA affecting high pressure ECCS operation	Some portion of the cooling water fed by one of the high pressure pumps immediately leaks through the break, so that pump cannot perform its function.
3.3. Medium LOCA affecting low pressure ECCS operation	This initiating event causes degradation in the redundancy of the low pressure ECCS.
3.4. Medium LOCA affecting both high and low pressure ECCS operation	There are certain pipelines in the primary system the rupture of which reduces the effectiveness of both high and low pressure injection.
4. SMALL LOSS OF COOLANT ACCIDENTS	
4.1. Small LOCA initiating ECCS operation	Compared to IE No. 4.3 the break size in this case is such that it requires ECCS operation regardless of the power rate of the make-up water pumps
4.2. Inadvertent opening of pressurizer relief valve (DN<25mm)	The stuck open relief valve on the highest point of the primary circuit results in a specific non-isolable LOCA situation that should be handled as a separate IE.

TABLE 7.1. (cont.)

Initiating event	Rationale for inclusion
4.3. Small LOCA, that can be compensated by make-up water (DN<13mm)	The make-up water system is sufficient to keep the balance of coolant loss and coolant addition giving a wide time window to the operator to take corrective actions.
5. LOSS OF COOLANT ACCIDENTS OUTSIDE CONTAINMENT	
5.1.a. Steam generator tube rupture (ca. 5 cm^2)	The main difference why the single steam generator tube rupture is handled separately from other more extensive leakages between the primary and the secondary circuit is that in this case the operation of the ECCS is not required or can be prevented by correct operator actions.
5.1.b. Several steam generator tubes leakages	In addition to the ECCS operation this initiating event results in greater radioactivity release to the secondary circuit/environment than IE No. 5.1.a.
5.2. Steam generator collector rupture	The loss of coolant caused by this IE is considered to be equivalent to the loss in case of a medium LOCA in spite of the fact that the break size corresponds to a large LOCA (DN 400 mm), because the primary coolant goes directly to the secondary side and not to a low pressure space (i.e. the containment) in this case.

TABLE 7.1. (cont.)

Initiating event		Rationale for inclusion
5.3.	Interfacing-system LOCA towards high pressure ECCS	The basis for the distinction between initiating events Nos 5.3 and 5.4 are the different rates of flow bypassing the containment (DN 111 mm).
5.4.	Interfacing-system LOCA towards low pressure ECCS	The loss of primary system flow is much higher than in the case of IE 5.3. There is a factor of about 2 between the relevant nominal pipeline diameters for the two cases.

7.1.1. Selection of LOCAs

The LOCA type IEs on the WWER list are broken down into three main categories based on the break size. The categories are as follows:

- Large LOCA;
- Medium LOCA;
- Small LOCA.

It should be noted that the break sizes appearing in Table 7.1 reflect the state of classification as of February 1992. Verification and finalization of the various break size boundaries are still going on as discussed in Section 7.2.1.

The large LOCA events are further subdivided according to the location of the break. Hot leg and cold leg locations are treated separately in addition to other specific break locations. The reason for this detailed classification is the fact that breaks at different locations of the primary circuit can impact on the effectiveness of the mitigating systems differently. The medium size LOCAs are categorized likewise. The differentiation between IEs 4.1 and 4.3 within the small LOCA category does not necessarily indicate different break locations but implies different break sizes. The approximate small LOCA break sizes are given in Table 7.1.

In addition, three main LOCA categories outside the containment, including ruptures within the steam generator and other interfacing system LOCAs, are also covered in Table 7.1. Reactor vessel rupture is treated as a separate IE.

7.1.2. Selection of transients and common cause initiators

The WWER initiating event list of transients and CCIs (given in Table 7.2) is subdivided into nine major groups as follows:

1. Reduction in primary system coolant flow;
2. Loss or reduction in feedwater flow;
3. Reduction in steam flow;
4. Loss of steam;
5. Transients causing turbine(s) trip;
6. Failure in electric power supply and I&C systems;
7. Failure of auxiliary systems;
8. Unplanned reactor trip;
9. Reactivity transients.

The above classification follows the technological plant features. Thus, these categories do not agree totally with PSA needs and do not imply final grouping of transients and CCIs for WWER plants.

Group 1

Initiating events 1.1–1.4 do not require reactor shutdown if additional malfunctions do not occur, because the primary circuit is equipped with six loops. Even the simultaneous loss of three reactor coolant pumps — which is the strongest transient of those mentioned above — requires only the reduction of reactor power down to 50%. Due to the structure of the primary circuit these transients have moderate effects and are not likely to endanger core integrity. The simultaneous loss of all reactor coolant pumps is the only event in group 1 by which reactor shutdown is needed.

Group 2

The transients listed in group 2 include events causing partial or total loss of feedwater flow (see IEs 2.1 and 2.2), feedwater line ruptures (see IEs 2.2–2.10) and other events introducing disturbances in feedwater flow (see IEs 2.10–2.15). Loss of one feedwater pump initiating event does not require reactor shutdown. This initiating event usually creates little disturbance in the normal operation of the plant because the feedwater system has five feedwater pumps (with four of them normally operating).

The feedwater line ruptures are categorized according to break size and location. This grouping has been done by detailed investigation of the feedwater system piping arrangement. The primary basis for the classification is the degree of degradation in the secondary side cooling caused by the various feedwater line ruptures. This degradation strongly affects the availability of the main and/or emergency feedwater systems.

The initiating events 2.12 to 2.15 affect the feedwater availability only in the long term. However, these events to some extent affect the availability of the condenser(s) as a heat sink as well.

TABLE 7.2. LIST OF TRANSIENT AND COMMON CAUSE INITIATING EVENTS FOR LEVEL 1 PROBABILISTIC SAFETY ASSESSMENT OF WWER-440 PLANTS

1. REDUCTION IN PRIMARY SYSTEM COOLANT FLOW
   1.1. Trip of one or two reactor coolant pumps
   1.2. Simultaneous trip of three reactor coolant pumps (230)
   1.3. Inadvertent closure of main gate valve (primary circuit)
   1.4. Reactor coolant pump seizure
   1.5. Total loss of reactor coolant flow

2. LOSS OR REDUCTION IN FEEDWATER FLOW
   2.1. Loss of one feedwater pump
   2.2. Loss of all feedwater pumps (more than 50 %)
   2.3. Feedwater collector rupture
   2.4. Feedwater line rupture that can be isolated by separation of one steam generator and compensated by startup of reserve feedwater pump
   2.5. Feedwater line rupture that can be isolated by separation of one steam generator and cannot be compensated by startup of reserve feedwater pump
   2.6. Feedwater line rupture that can be isolated by separation of one half of feedwater collector and compensated by startup of reserve feedwater pump
   2.7. Feedwater line rupture that can be isolated by separation of one half of feedwater collector and cannot be compensated by startup of reserve feedwater pump
   2.8. Rupture of feedwater pump discharge line (before check valve)
   2.9. Rupture of feedwater pump suction line
   2.10. Feedwater line rupture inside containment
   2.11. Feedwater flow instability
   2.12. Loss of condenser pump (one)
   2.13. Loss of condenser pumps (all)
   2.14. Loss of condenser vacuum
   2.15. Loss of circulating water

3. REDUCTION IN STEAM FLOW
   3.1. Inadvertent closure of main steam isolation valve
   3.2. Miscellaneous leakages in secondary system

4. LOSS OF STEAM
   4.1. Inadvertent opening of steam generator relief valve
   4.2a. Inadvertent opening of steam dump valve (bypass)
   4.2b. Inadvertent opening of steam dump valve (to the atmosphere)
   4.3. Steam line rupture inside containment
   4.4. Steam line rupture outside containment
   4.5. Rupture of main steam collector
   4.6. Turbine control valve malfunction

TABLE 7.2. (cont.)

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5. TRANSIENTS CAUSING TURBINE IS TRIP
   - 5.1. Turbine trip (one turbine)
   - 5.2. Turbine tripe (both turbines)
   - 5.3. Total loss of electric load
   - 5.4. Generator faults/trip

6. FAILURES IN ELECTRIC POWER SUPPLY AND I&C SYSTEMS
   - 6.1. Loss of off-site power
   - 6.2. Loss of all 6 kV bus-bars
   - 6.3. Loss of one 6 kV bus-bar
   - 6.4. Loss of bus-bar EV (non-interruptible AC power)
   - 6.5. Spurious 'P_{box} > 1.1 bar' signal
   - 6.6. Spurious 'large LOCA' signal (213) or spurious 'low pressure in the primary circuit' signal (230)
   - 6.7. Spurious 'main steam collector rupture' signal (213) or spurious 'low pressure on the steam header' signal (230)
   - 6.8. Loss of normal grid connection

7. FAILURES OF AUXILIARY SYSTEMS
   - 7.1. Loss of service water system
   - 7.2. Loss of intermediate cooling to reactor coolant pumps
   - 7.3. Loss of intermediate cooling to control rods
   - 7.4. Loss of ventilation systems
   - 7.5. Loss of high pressure air
   - 7.6. Loss of room cooling in a vital instrumentation compartment
   - 7.7. Primary system purification system fault

8. UNPLANNED REACTOR TRIP
   - 8.1. Spurious reactor trip
   - 8.2. Reactor trip due to administrative procedure such as simultaneous unavailability of two safety systems

9. REACTIVITY TRANSIENTS
   - 9.1. Uncontrolled single control rod withdrawal
   - 9.2. Uncontrolled control rod group withdrawal
   - 9.3. Inadvertent boron dilution
   - 9.4. Control rod ejection with reactor vessel damage
   - 9.5. Anticipated transient without scram (ATWS)
   - 9.6. Control rod drop (single rod)
   - 9.7. Control rod ejection without reactor vessel damage
   - 9.8. Restart of isolated primary system loop (opening of main gate valve) with low boron concentration

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Groups 3 and 4

The initiating events grouped under categories 'reduction in steam flow' and 'loss of steam' are in accordance with IE groups used in most PSAs. These events include valve malfunctions and steam line breaks. The piping and valve arrangement on the steam side of the secondary circuit is necessarily reflected in groups 3 and 4. Steam line breaks inside and outside the containment are treated separately. In addition, the main steam line break is distinguished from the main steam collector rupture due to different sizes. Various valve openings leading to inadvertent steam dump/release are identified as separate IEs.

Group 5

Transients listed in group 5 are events resulting in turbine trip. A unique feature of this category is that it includes trip of one turbine and trip of two turbines. Trip of one turbine does not require reactor shutdown but reduced power (50%) operation.

Groups 6 and 7

The initiating events of groups 6 and 7 are mostly common cause initiating events and, apart from a few exceptions such as 'loss of off-site power', 'loss of normal grid connection' and 'loss of service water system' which are generic IEs, they are specific to WWERs.

As an example of how these events affect several systems/components simultaneously, the consequences of two CCIs resulting from bus-bar failures are described below.

(a) Loss of one 6 kV bus-bar

There are four vital 6 kV AC buses in the power supply system. Depending on which bus is lost, the initiating event can cause the loss of electric power supply to a make-up water pump, reactor coolant pump(s), feedwater pump(s) and condense pumps at the same time. In addition, the three safety bus-bars are also connected to the 6 kV buses.

(b) Loss of bus-bar EV (non-interruptible AC power)

The initiating event causes substantial loss of information in the control room. If it occurs, the state indicator lamps of the reactor coolant and feedwater pumps disappear together with the indicators of the 6 kV circuit breakers. Most of the controls cannot be operated from the control room. Some of the control circuits and interlocks on the secondary side are disabled too.

Group 6 also includes three initiators termed 'spurious signal'. These three events are also well supported by WWER operating experience. Loss of the component cooling system is broken down to two events, namely the loss of intermediate cooling to reactor coolant pumps and the loss of intermediate cooling to control rods. The failures of other auxiliary systems such as the ventilation systems are not further subdivided in the list.

Group 8

Two reactor trip categories are defined. The first category, the spurious reactor trip, includes both manual and automatic trips. The other category covers reactor trips required by the technical specifications.

Group 9

The last group of the WWER initiating event list covers the reactivity transients. Most of the initiating events in this category result in core power distribution asymmetry through reactor power increase (IEs 9.1, 9.2, 9.3, 9.4, 9.7, 9.8). The control rod drop IE has just opposite effects on the core. The control rod ejection with reactor vessel damage initiating event (9.4) causes substantial loss of primary coolant, i.e., a LOCA situation in addition to the reactivity transient. Anticipated transient without scram (ATWS) events are also included in group 9. Those transients by which the ATWS events should be taken into account are not identified in the list.

7.2. GROUPING OF WWER INITIATING EVENTS

7.2.1. Grouping of LOCAs

In the WWER initiating event list the division of LOCAs is based on the following break size boundaries:

- Large LOCA: break size > 200 mm DN;
- Medium LOCA: 32 mm DN ≤ break size ≤ 200 mm DN;
- Small LOCA: break size < 32 mm DN.

As regards large LOCAs, some additional thermalhydraulic analyses are still under way to investigate the break size limits with respect to plant response and mitigating system requirements. Grouping of large LOCAs based on break location, e.g. breaks on hot and cold legs, etc. are not fully justified by thermal hydraulic calculations either.

The medium LOCAs may have to be further subdivided. The latest calculations suggest that the broad spectrum of 32–200 mm should be split into 2–3 subcategories. However, these results are not reflected in the current WWER list because some more plant response analyses are needed to finalize the break size boundaries. The location of the medium LOCAs affects the availability of the emergency core cooling systems; they are therefore differentiated according to break location as well.

Within the small LOCA category a very small LOCA with break size < 13 mm DN is identified besides the pressurizer valve malfunction. The latter is a specific LOCA situation (see Table 7.1).

The break size boundaries related to the subcategories in the small LOCA group should also be verified by plant response analysis.

The loss of coolant accidents outside containment includes three steam generator failures (tube(s) or collector rupture) and two additional interfacing system LOCAs. These events are handled as separate groups from other LOCAs because they require specific operator actions. Moreover, the unrecoverable primary coolant released to the environment can have substantial radiological effects in the case of steam generator failures.

Rupture of a reactor vessel is treated as a separate group on the WWER IE list as described in Table 7.1.

To demonstrate the efforts being made in this area, Table 7.3 shows the results of some plant response analyses for WWER-440/213 type reactors. The categorization of LOCAs given in Table 7.3 is based on thermalhydraulic analyses performed in Czechoslovakia, the Russian Federation, Poland and on engineering judgement. The rationale of this classification and the mitigating system requirements for the various LOCA categories are briefly described below (based on Ref. [63]).

TABLE 7.3. SYSTEM SUCCESS CRITERIA FOR LOCAs CLASSIFIED ACCORDING TO BREAK SIZE RANGE (BASED ON REF. [63])

LOCA category	Break size range (mm)	System Success Criteria						
		HPI	HPR	LPI	LPR	CFS	EFS	SCS
L1	10 ≤ D ≤ 20	1			1		1	PM
		1	1					PM
				1	1	0–4[a]	1	HR
L2	20 ≤ D ≤ (50–70)	1	1					
		1			1			
				1	1	0–4[a]	1	HR
L3	(50–70) ≤ D ≤ (120–150)	1	1					
		1			1			
L4	(120–150) ≤ D ≤ 200	1	1					
		1			1			
				1	1	1+1[b]		
L5	200 ≤ D ≤ 300	1			1			
				1	1	1+1[b]		
L6	300 ≤ D ≤ 500			1	1	1+1[b]		

HPI – High pressure injection system.
HPR – High pressure recirculation system.
LPI – Low pressure injection system.
LPR – Low pressure recirculation system.
CFS – Core flooding system (hydroaccumulator).
EFS – Emergency/auxiliary feedwater system
SCS – Secondary pressure control system.
PM – Pressure maintenance mode.
HR – Heat removal mode (30 K/h).

a Availability of CFS affects timing requirements for secondary side cooling initiation and related probability of operator error.

b 1 out of 2 hydroaccumulators connected to upper plenum and 1 out of 2 hydroaccumulators connected to downcomer are required to operate.

- LOCA group L1

  Breaks of size 10–20 mm DN are considered as the smallest LOCA category. The lower boundary is estimated as the highest break size that can be compensated by normal makeup water system. For this category energy removed by coolant flow through the break is insufficient to ensure depressurization if secondary side cooling is not provided.

  Regarding the mitigating system requirements the high pressure injection/recirculation system (HPI/HPR) is required to operate for the initial phase of the accident and the low pressure recirculation system (LPR) can provide long term cooling. If LPR fails HPR can be used for long term cooling too. In both cases secondary side cooling is necessary besides ECCS.

  If HPS is operable but secondary side cooling is not provided, primary feed-and-bleed is the only way of accident mitigation.

  If HPS fails in the early phase of the accident, secondary feed-and-bleed and LPS should provide RCS depressurization and long term cooling respectively. In this case the availability of core flooding hydroaccumulators (CFS) has substantial effect on the time window for the operator to initiate depressurization.

- LOCA group L2

  The lower break size boundary (~20 mm DN) is based on requirements against secondary side cooling. To mitigate LOCAs of size larger than this lower limit only HPS is required even if secondary side cooling is not provided.

  Upper boundary (~50–70 mm DN) of this category is considered to be the break size by which sufficient time margin is available for correct plant state diagnosis and for manual alignment of secondary side cooling. Accident mitigation related to this category is similar to that of group L1 except that secondary side cooling is not needed if HPS does not fail.

- LOCA group L3

  The estimation of upper boundary for this category (~120–150 mm DN) is based on engineering judgement.

  As regards the mitigation system requirements HPS is the only system that can establish stable core cooling in the initial phase and RCS depressurization down to LPS operational pressure limit. Secondary feed-and-bleed is considered to be unachievable because of timing requirements. Accident mitigation related to this category is similar to that of group L2 except that in the case of HPS failure secondary feed-and-bleed with subsequent use of LPS is not taken into account.

- LOCA group L4

  The upper break size boundary (200 mm DN) is estimated on the basis of existing thermal hydraulic evidence.

For LOCAs of this category the operation of CFS and LPS provides an alternative way of successful accident mitigation besides the mitigation measures described for group L3.

– LOCA group L5

The upper break size boundary (300 mm DN) is estimated on the basis of existing thermal hydraulic evidence.

For LOCAs of this category the operation of CFS and LPS is needed in the early phase of the accident while LPR is required for long term cooling. In contrast to group L4 in this case the HPS alone is insufficient for successful accident mitigation.

– LOCA group L6

This category covers all break sizes larger than 300 mm DN.

For LOCAs of this category the operation of CFS and LPS is the only measure that can provide successful accident mitigation

7.2.2. Grouping of transients and common cause initiators

The breakdown of 69 WWER transients and CCIs to nine main categories represents a classification primarily based on engineering judgement. A master logic diagram type scheme was the basis for the categorization.

All the disturbances that could potentially lead to loss of core cooling without primary coolant boundary failure or to an increase in core power were taken into account by identifying their main causes. These causes and the investigation of plant specific potentials for CCIs then formed the basis for the nine groups listed in Table 7.2. Within each category those events requiring different plant response i.e. different event tree models according to the plant response analyses and engineering knowledge, were treated separately.

This approach appeared to be an effective means for both establishing a comprehensive list of WWER IEs and performing an initial grouping of transients and CCIs. Nevertheless, since the transient list is rather complex, comprising a large number of initiators, the current grouping on a judgemental basis can only be regarded as a preliminary categorization that needs improvement/finalization to be supported by thermal hydraulic analysis and event tree/fault tree modelling as well. The refinement of the initial grouping is in a rather different state in the individual WWER PSAs, therefore no further categorization of transients and CCIs is given here in addition to the one suggested by Table 7.2.

7.3. DETERMINATION OF THE FREQUENCY OF INITIATORS FOR WWER PLANTS

At present the IE frequencies used in the individual PSAs in countries operating WWER type units are mainly based on their own operating experience and, for rare events, on generic data not specifically developed for WWER plants. However, for the majority of

IEs listed in Table 7.2 and for some events of Table 7.1 a substantial amount of operating experience has been accumulated in the last twenty years, i.e.:

(A) 190 reactor-years of type WWER-440 (both 230 and 213) and of type WWER-1000 in the former USSR and in its successor countries;

(B) 50 reactor-years of type WWER-440 (230) in Kozloduy, Bulgaria;

(C) 50 reactor-years of type WWER-440 (230) in Greifswald, Germany;

(D) About 30 reactor-years of type WWER-440 (213) in Paks, Hungary;

(E) Similar experience of type WWER-440 (both 230 and 213) and of type WWER-1000, in Czechoslovakia.

The ongoing elaboration of this experience and the exchange of information among WWER operators aim to facilitate the development of a generic WWER IE frequency list. Most of the transients ('likely' transients) and some LOCAs, such as very small LOCAs or PORVs stuck open, can also be quantified on the basis of the existing operating experience.

A list of IE frequencies prepared in the manner described above massively supports the use of a variety of approaches, mainly the Bayesian updating techniques and the mean frequency concept to determination of IE intensities for the individual plants.

Concerning rare events, especially LOCAs, the frequencies that have been used for PSAs of WWERs are mostly based on generic data. However, some fracture mechanic studies have recently been carried out in Russia to determine LOCA frequencies. According to these calculations, the LOCA frequencies are as follows:

– Large LOCA (break size > 150 mm DN)	4.0×10^{-6}/year;
– Medium LOCA (50mm DN < break size < 150 mm DN)	4.2×10^{-4}/year;
– Small LOCA (break size < 50 mm DN)	3.0×10^{-3}/year.

In addition to fracture mechanics, engineering judgement is needed to assess to what extent generic nuclear IE data or even non-nuclear data can be used for WWER plants. The results of these analyses provide frequency of rare events figuring in the WWER IE list.

The fault tree analysis is undertaken in parallel with the event tree/fault tree developments in the individual PSAs for those plant specific CCIs that occur due to system/subsystem failures (e.g. loss of intermediate cooling to reactor coolant pumps) and for which the observed failure data is scarce.

Table 7.4 summarizes the IE event data made available at the Řež workshop (data columns 1 to 5), supplemented by the IE frequencies used for the Level 1 PSA of Kola-1 NPP and also by the IE frequencies used in the top level risk study for Kozloduy Units 1 to 4 [60] (last two data columns of Table 7.4). The first two data columns of the table contain the IE that was used in the two completed WWER PSAs, in the Greifswald and Loviisa PSAs respectively.

TABLE 7.4. FREQUENCY OF INITIATING EVENTS FOR PSA OF WWER REACTORS

	Initiating Event	Data Source							Comments
		Greifswald PSA (1/year)	Loviisa PSA (1/year)	Bulgarian Oper. exp. and Gen. Data (1/year)	Polish Generic Data (1/year)	Russian Data (1/year)	Kola-1 PSA(8) (1/year)	"Top Level Risk Study for Kozloduy"(10) (1/year)	
	Reactor pressure vessel rupture	1.0×10^{-5}(1)	5.5×10^{-6}(6)	1.0×10^{-4}(5)		$< 1.0\times10^{-5}$			
	Large LOCA	1.0×10^{-6}(2)	5.3×10^{-4}	1.0×10^{-4}(5)	2.0×10^{-4}(5)	4.0×10^{-6}(2)		1.0×10^{-4}	
	Medium LOCA	1.2×10^{-4}	5.3×10^{-4}	1.0×10^{-4}(5)	7.5×10^{-4}(5)	4.2×10^{-4}(2)		1.0×10^{-4}	
	Small LOCA	5.0×10^{-4}	4.0×10^{-3}		3.6×10^{-3}(5)	3.0×10^{-3}(2)	3.7×10^{-3} (< 32 mm)	$1.0\times10^{-2}/1.0\times10^{-3}$(9)	
							2.8×10^{-4} (32 to 60 mm)		
							1.3×10^{-4} (60 to 100 mm)		
LOCAS	Very small LOCA	2.0×10^{-3}(3)		3.5×10^{-1}(4)	2.4×10^{-2}			3.4×10^{-1}	
	Hydroaccumulator LOCA		3.1×10^{-5}						
	Single SG tube rupture	1.0×10^{-2}(3)	1.8×10^{-3}	1.0×10^{-2}			1.0×10^{-2}	5.0×10^{-1} (leakage)	
								1.0×10^{-2} (rupture)	
	Several SG tubes leakages	1.0×10^{-3}	5.3×10^{-4}	(7)					
	SG collector break	3.0×10^{-4}							
	SG header cover rupture						7.1×10^{-3}		
	LOCA outside containment	1.0×10^{-3}	2.3×10^{-6}						
	Inadvertent opening of PORV/pressurizer valve leakage		3.4×10^{-3}	3.0×10^{-2}			1.6×10^{-2}		
	Steam line rupture inside confinement						4.8×10^{-4}		

Comments:

(1) Based on embrittlement

(2) Based on fracture mechanics

(3) Based on operating experience

(4) 17 events occurred in 50 r. yrs.

(5) Generic information (not from WWERS)

(6) Includes PTS considerations

(7) Experienced but not quantified yet

(8) Based on operating experience and Bayesian updating

(9) Units 1, 2/3, 4

(10) Based on operating experience and engineering judgement

TABLE 7.4. (cont.)

	Initiating Event	Data Source							Comments
		Greifswald PSA (1/year)	Loviisa PSA (1/year)	Bulgarian Oper. exp. and Gen. Data (1/year)	Polish Generic Data (1/year)	Russian Data (1/year)	Kola-1 PSA(8) (1/year)	"Top Level Risk Study for Kozloduy"(10) (1/year)	
TRANSIENTS & CCIS	Total loss of RCS flow							1.0×10^{-3}	(1) Based on embrittlement (2) Based on fracture mechanics (3) Based on operating experience (4) 17 events occurred in 50 r. yrs. (5) Generic information (not from WWERS) (6) Includes PTS considerations (7) Experienced but not quantified yet (8) Based on operating experience and Bayesian updating (9) Units 1, 2/3, 4 (10) Based on operating experience and engineering judgement
	Partial loss of RCS flow							5.0×10^{-1}	
	RCP failure (incl. RCP shaft break)						8.8×10^{-3}		
	Core flow blockage							$1.0\times10^{-4}/1.0\times10^{-1}$(9)	
	Total loss of main feedwater	1.5×10^{-3}	2.8×10^{-3}	1.0×10^{-3}-1.0×10^{-4}(5)				1.0×10^{-2}	
	Partial loss of main feedwater		2.3×10^{-2}	1.7×10^{-1}				1.0×10^{-1}	
	Feedwater line rupture						7.1×10^{-3}	5.0×10^{-3}	
	Feedwater header rupture						7.1×10^{-3}	5.0×10^{-3}	
	Main steam line break	1.0×10^{-3}		1.0×10^{-3}-1.0×10^{-4}(5)	2.5×10^{-3}(5)		4.4×10^{-4}	1.0×10^{-3}	
	Main steam header break							1.0×10^{-3}	
	Inadvertent opening of SG relief valve						1.6×10^{-2}	1.0×10^{-2}	
	Inadvertent opening of steam dump valve to atmosphere						7.5×10^{-3}	5.0×10^{-2}	
	Inadvertent opening of steam dump valve to condenser							5.0×10^{-2}	
	Turbine trip (one turbine)		1.7×10^{-1}	8.5×10^{-1}				1.0	
	Turbine trip (both turbines)			3.0×10^{-1}				1.0×10^{-1}	
	Generator trip			8.5×10^{-1}					
	Reactor trip						2.1	1.0 (spurious trip)	

TABLE 7.4. (cont.)

	Initiating Event	Data Source							Comments
		Greifswald PSA (1/year)	Loviisa PSA (1/year)	Bulgarian Oper. exp. and Gen. Data (1/year)	Polish Generic Data (1/year)	Russian Data (1/year)	Kola-1 PSA(8) (1/year)	'Top Level Risk Study for Kozloduy'(10) (1/year)	
TRANSIENTS & CCIS	Control rod withdrawal (bank)							1.0×10^{-2}	(1) Based on embrittlement (2) Based on fracture mechanics (3) Based on operating experience (4) 17 events occurred in 50 r. yrs. (5) Generic information (not from WWERS) (6) Includes PTS considerations (7) Experienced but not quantified yet (8) Based on operating experience and Bayesian updating (9) Units 1, 2/3, 4 (10) Based on operating experience and engineering judgement
	Control rod withdrawal (ejection)							$1.0\times10^{1}/1.0\times10^{-5(9)}$	
	Inadvertent boron dilution							1.0×10^{-2}	
	Single control rod drop or insertion			6.0×10^{2}				1.0×10^{-1}	
	ATWS		8.0×10^{-8}						
	Reconnection of idle loop							1.0×10^{-4}	
	Inadvertent high pressure safety injection							1.0×10^{-2}	
	Loss of off-site power for more than 2 Hrs	1.0×10^{-2}	2.5×10^{-2}	3.0×10^{-2}			8.0×10^{2}	1.0×10^{-2}	
	Loss of normal grid connection							1.0×10^{-1}	
	Total loss of DC							$1.0\times10^{-2}/1.0\times10^{-4(9)}$	
	Loss of AC/DC bus	1.0×10^{-6}	7.0×10^{-7}	9.0×10^{-2}					
	Loss of component/service water cooling	1.2×10^{-4}	3.2×10^{-5}	9.0×10^{2}					
	Loss of instrumentation room cooling		4.0×10^{-4}						
	RCS purification fault							5.0×10^{-3}	

Data provided by the Bulgarian team for Kozloduy is a mixture of generic IE frequencies and data derived from the quantification of the operating experience. The latter is mainly related to 'likely transients' while LOCA frequencies are based on generic data except very small LOCA.

The Polish data originates from a collection of generic IE frequencies for LOCAs.

Frequencies given in the Russian data represent the results of the fracture mechanic studies for rare LOCAs. The vast operating experience is not represented in that data column. However, the Kola PSA data is mainly based on operating experience from the WWER type of NPPs.

In the Kola PSA initiating event frequencies were estimated by the use of Bayesian updating. For the calculations both WWER and non-WWER sources (primarily the German Risk Study, Phase B [22]) were taken as prior distributions. The updating was done on the basis of in-country operating experience mainly for WWER-440 and in some cases for WWER-1000 type plants. Feedwater line ruptures (two of them are analysed in the study), however, were estimated by using Loviisa NPP data.

The top level risk study data for Kozloduy was determined by considering generic western PWR frequencies and by modification of those using applicable site specific operating experience provided by the plant and engineering judgement as well. Engineering judgement was also considered essential to determine the applicability of western PWR frequencies.

8. HAZARDS

When estimating the risks of nuclear power plants, contributions from both internal initiating events and external initiators (or hazards) need to be considered. While early PSAs had mainly been addressing internal initiators, the current practice worldwide is to include hazards into full scope PSAs. Some of the PSA studies which include hazards have shown that these have a larger potential risk for the environment than internal initiators [53, 54].

8.1. MAIN CATEGORIES OF HAZARDS

Hazards are usually classified under two main categories: internal and external hazards.

The list of hazards is quite established and most PSAs include a sublist of the following hazards:

Internal hazards:

- Fires;
- Internal floods;
- Turbine missiles.

External hazards:

- Forest fires;
- External floods, high waters;
- Airplane crash;
- Seismic events;
- Extreme winds and/or tornados;
- Release of chemical or toxic gas.

Some of the hazards have subcategories or an entire magnitude frequency distribution of initiating events.

Internal and external hazards may also be divided according to their origin:

- Natural (floods, seismic, tornados, etc.);
- Human (turbine missiles, airplane crashes, chemical release, etc.).

8.2. EVALUATION OF HAZARD FREQUENCIES

8.2.1. Seismic analysis

Unlike internal IEs, the occurrence of the seismic initiator is completely independent from the plant. The seismic hazard at a given power plant is characterized by a hazard function which gives the probability of exceedance (per year) of a ground motion parameter, such as peak ground acceleration. Figure 8.1 shows a representative hazard curve for a plant site (nominal with upper and lower uncertainty bounds). The curves are derived from a

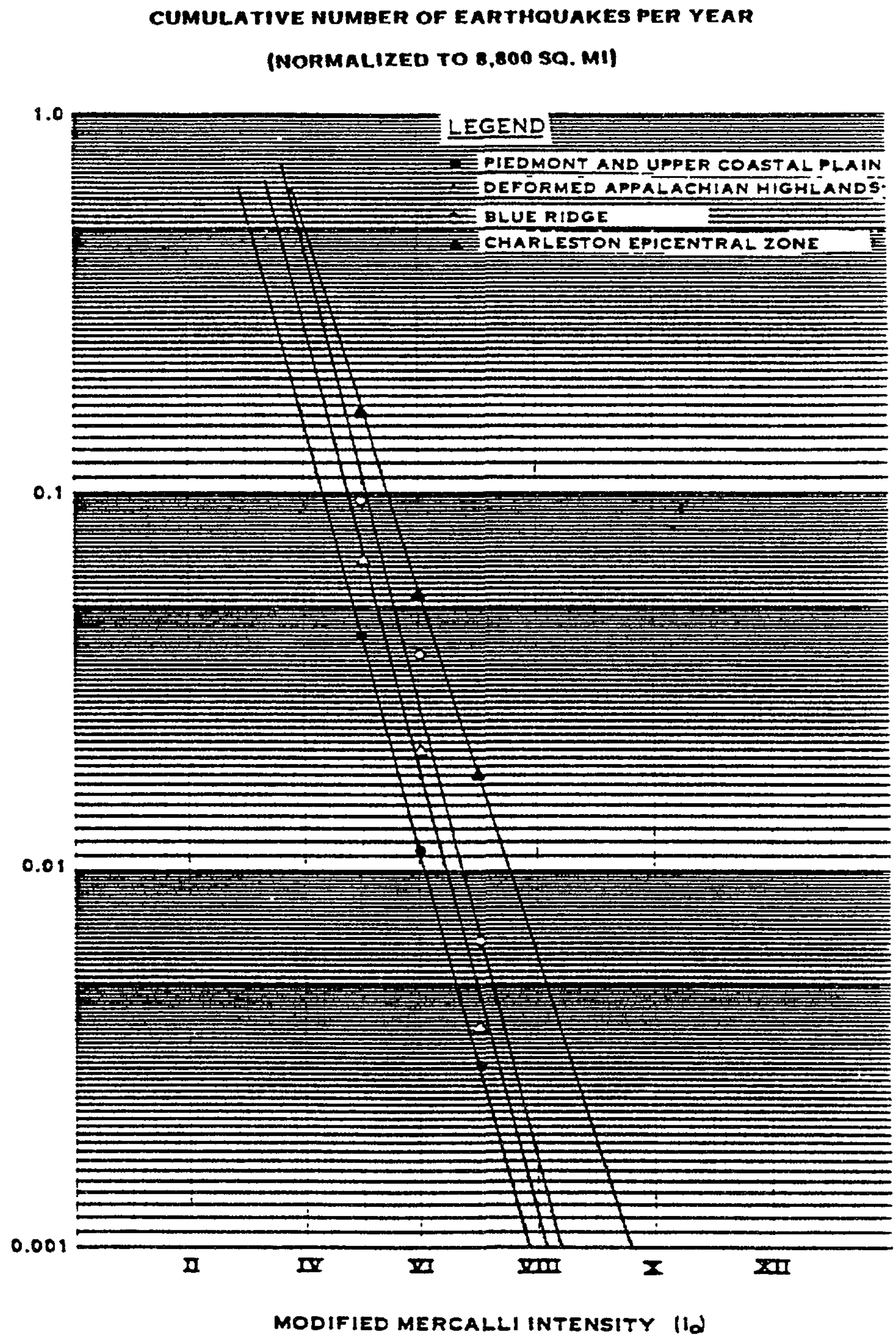

FIG. 8 1 Example of seismic hazard curve

combination of recorded seismic data, estimates of the magnitudes of the known events, geological information and use of expert opinions.

In addition, the frequency characteristics of the earthquakes is required in order to generate artificial acceleration time histories for two horizontal and one vertical component.

8.2.2. Fire analysis

Fire analysis consists of the following main steps:

- Identification of critical locations susceptible to fire;
- Estimations of the potential frequency of fire in these locations;
- Analysis of fire propagation, taking into account mitigating systems like fire alarms and fire suppression;
- Identification of the potential for causing LOCAs or transients and the continuation from that point to core damage using the common PSA methodologies.

When fires are treated as initiating events, then the first two steps are performed. Fire analysis is different from seismic hazard analysis but similar to internal flood analysis (see Section 8.2.3). The main fire locations that were considered in past PSAs are given in Table 8.1. For these locations it is also possible to use historical data from generic nuclear power plant fire experience (LERs).

The determination of the frequency of fires in a chosen location can proceed in two ways:

(a) Using the generic operating experience in nuclear power plants. This is a relative rough estimation of the frequency of fires, that does not include plant specific features.

(b) Considering the plant specific amount of combustible material in each location and using probability of fire ignition based on the amount of combustible material and the design of the area with respect to existing fire ignition sources.

Table 8.1 provides examples of fire frequencies used in several PSA studies.

8.2.3. Flood analysis

Flood analysis consists of the following main steps (for internal floods):

- Identification of critical locations susceptible to floods;

- Estimation of the potential frequency of floods in the critical location, from internal floods;

- Identification of floods flow rates in these locations based on pipe and value size characteristics;

- Determination of the critical water level heights at which plant safety systems will fail to perform their functions;

- Computation of time required to reach the critical levels;

- Determination of corrective action and time available to their successful performance (and success probability based on this time periods);

TABLE 8.1. HAZARD FREQUENCIES IN SEVERAL PSAs

Hazard	Frequency of Occurrence (per year):							
	Oconee [9]	Zion [43]	German Risk Study [22]	Midland [13]	Seabrook [14]	Big Rock Point [46]	Indian Point [42]	Limerick [20]
Earthquake:								
- Intensity 1			7.0E-4	1.5E-4	1.0E-6	1.0E-3		
- Intensity 2			9.5E-5	2.0E-5	8.2E-7	1.0E-4		
- Intensity 3			5.0E-6	6.6E-7	3.6E-4	1.0E-5		
- Intensity 4				3.9E-8	1.1E-4			
- Intensity 5					4.3E-5			
- Intensity 6					2.0E-5			
- Intensity 7					2.0E-5			
- Intensity 8					2.5E-6			
High Wind/Tornado	3.5E-3	1.0E-3	---		7.8E-5	---	1.0E-4	2.3E-4
Airplane Crash:								
- On containment			6.3E-7		1.2E-8			
- On Control room					1.4E-7			
- On Aux. Building					2.0E.7			
External Flood:								
- In Building	2.3E-5		---	---	---	---	3.0E-3	
- Loss of Heat Sink					1.6E-6			
Internal Flood:								
(Containment)			9.0E-7	---	---			
			4.0E-6					
(Turbine Building)			---	---	3.2E-4			

TABLE 8.1. (cont.)

Hazard	Frequency of Occurrence (per year):							
	Oconee [9]	Zion [43]	German Risk Study [22]	Midland [13]	Seabrook [14]	Big Rock Point [46]	Indian Point [42]	Limerick [20]
Fires:								
- Vital Bus or Panel					---			
- Cable Spreading Room					5.0E-4	4.2E-3		
- Turbine Building					6.4E-4	---		
- Control Room					1.3E-5	1.0E-4		
- Containment Penetration						1.8E-3		
- Aux. Building				4.0E-3	---	---		
Turbine Missiles				---	7.4E-8			
Hazardous Chemicals			---	6.0E-3	---	---		

- Performing flood event tree analyses of the various flood sequences that may lead to core damage;

- Identification of the critical sequences, their consequence in term of core damage category and their probability. Summarizing impact on core damage frequency.

The frequency of the flood IEs may be determined by calculating the frequency of pipe break and maintenance errors probabilities (e.g. improper isolation following maintenance). This is done for any building (turbine, auxiliary or containment building). Table 8.2 provides an example of flood frequencies reported in the Oconee PSA review [15, 51].

TABLE 8.2. FLOOD INITIATOR FREQUENCIES AND THEIR CONTRIBUTION TO CORE DAMADE FREQUENCY [15, 51]

Flood Initiators	Flood Flow Rate (1000 gpm)	Type of Isolation Possible	Total Conditional Probability of Core Damage	Contribution of Flood Initiator to Core Damage Frequency (Yy-1)	Frequency of Flood Initiator (Yy-1)
FVL1N	350	non isolable	0.24	2.4E-6	1.0E-5
FVL1II	350	isolable inlet	0.014	2.5E-6	1.8E-4
FVL1IO	350	isolable outlet	0.012	2.3E-6	1.9E-4
FVL2N	170-349	non isolable	0.19	3.4E-5	1.8E-4
FVL2II	170-349	isolable inlet	0.010	1.8E-7	1.8E-5
FVL2IO	170-349	isolable outlet			
FL1N	120-169	non isolable	0.080	3.0E-6	3.8E-5
FL1II	120-169	isolable inlet	8.3E-4	4.3E-7	5.3E-4
FL1IO	120-169	isolable outlet	1.6E-3	8.8E-7	5.5E-4
FL2N	60-119	non isolable	0.016	6.6E-6	4.1E-4
FL2II	60-119	isolable inlet	1.2E-3	6.5E-6	5.4E-3
FL2I0	60-119	isolable outlet	3.3E-4	1.7E-6	5.1E-3
FMN	12-59	non isolable	4.4E-3	1.1E-5	2.7E-3
FMII	12-59	isolable inlet	6.6E-4	5.3E-6	8.0E-3
FMIO	12-59	isolable outlet	1.0E-4	1.1E-6	1.1E-2
TOTAL			2.4E-3	7.8E-5	3.4E-2

9. DESCRIPTION OF THE DATABASE OF INITIATING EVENTS

9.1. GENERAL DEFINITIONS

For the purpose of providing examples of IE selection, grouping and frequency determination, the Appendix summarizes IE data from a number of PSAs. This IE data allows the user of the report to implement the methodology of the use of past experience in determining the IEs for his study. To ease the use of the database, two printouts have been provided: (1) database sorted in accordance to data source, and (2) database sorted in accordance to initiating event category.

The format of the data and the contents of each of the database field in the Appendix is as follows:

No:	The sequential number of a record. More than 270 entries are given for 25 different studies.
PLANT TYPE:	Reactor type (BWR or PWR only).
IE SET:	Basic general category to which the IE belongs (LOCA, transient or common cause initiator).
IE CATEGORY:	Specific category where an initiator belongs. It provides the grouping of the IE which was used in the PSA study (the individual event tree is, in principle, developed for specific IE category listed here).
INITIATORS:	Basic initiating event considered before the grouping (for transients) and the defined break size range (for LOCAs).
FREQ DESC:	Frequency descriptor or units used (mean event per reactor-year, or point estimate events per reactor-year, or median events per reactor-year.
FREQ:	Frequency of the initiating event or of a group of initiators.
RANGE VAR:	Range of variation in IE frequency. An error factor (EF 10 means error factor of 10) or a range of variation (5% to 95% confidence level) is provided.
UPPER BND:	Upper bound of the frequency distribution. In most cases this is the 95% upper bound. In case EF is used, this field remains blank.
LOWER BND:	Lower bound of the frequency distribution. In most cases this is the 5% lower bound. In case EF is used, this field remains blank.
SOURCE:	Provides the name of the data source. It also provides in some cases the exact table in which the data appears in the source.

ULT SOURCE: Ultimate source that was used to generate the specific data for a particular PSA study. The different cases include:

- Generic data: data originating elsewhere (calculation of operating experience, expert opinion) and generally representative of a spectrum of different plants;
- Generic data updated with plant operating experience: generic data used as a prior, and plant specific data used in Bayesian updating, to generate IE frequency;
- Plant operating experience: plant data gathered from events reported during plant life;
- Engineering evaluation: expert opinion of individuals familiar with plant design and operation;
- System analysis using fault tree model: one of the methodologies described in this manual for determination of CCI frequency.

The references of the various sources are as follows (in the order of appearance in the Appendix):

	Reference
German Risk Study	[35]
IREP ANO-1	[18]
Limeric PRA	[20]
NUREG/CR 4550	[7, 19]
Oconee PRA	[9]
Sequoyah NPP RSSMAP	[34]
Shoreham PRA	[12]
WASH-1400	[2]
IREP Calvert Cliffs 1	[41]
IREP Browns Ferry 1	[30]
Barseback 1 & 2	[29]
Forsmark 3	[29]
Oskarshamn 1, 2 & 3	[29]
Ringhals 1 & 2	[29]
Zion NPP PSS	[44]
Angra NPP PSA	[58]
Caorso NPP PSS	[57]

9.2. SUMMARY OF DATABASE STATUS

INTERFACING LOCA

Number of records: 11
Highest value: 4.6E-6/year
Lowest value: 1.0E-8/year

LARGE LOCA

Number of records: 26
Highest value: 9.4E-4/year
Lowest value: 1.0E-7/year

MEDIUM LOCA

Number of records: 24
Highest value: 3.0E-3/year
Lowest value: 1.0E-6/year

REACTOR VESSEL RUPTURE

Number of records: 7
Highest value: 1.1E-6/year
Lowest value: 1.0E-7/year

SMALL LOCA

Number of records: 26
Highest value: 2.3E-2/year
Lowest value: 1.8E-3/year

SMALL SMALL LOCA

Number of records: 7
Highest value: 3.0E-2/year
Lowest value: 1.3E-3/year

STEAM GENERATOR TUBE RUPTURE

Number of records: 4
Highest value: 2.4E-2/year
Lowest value: 8.2E-3/year

TRANSIENT CAUSED BY INADVERTENT OPENING OF RELIEF VALVE

Number of records: 5
Highest value: 1.4E-1/year
Lowest value: 1.3E-2/year

TRANSIENT CAUSED BY INADVERTENT SAFETY INJECTION

Number of records: 2
Highest value: 1.2E-0/year
Lowest value: 1.0E-2/year

TRANSIENT CAUSED BY ISOLATION EVENTS

Number of records: 13
Highest value: 2.3E-0/year
Lowest value: 1.5E-1/year

TRANSIENT CAUSED BY LOSS OF AC BUS

Number of records: 6
Highest value: 3.5E-2/year
Lowest value: 9.0E-4/year

TRANSIENT CAUSED BY LOSS OF DC BUS

Number of records: 7
Highest value: 3.2E-1/year
Lowest value: 9.0E-4/year

TRANSIENT CAUSED BY LOSS OF FEEDWATER

Number of records: 15
Highest value: 5.2E-0/year
Lowest value: 1.7E-1/year

TRANSIENT CAUSED BY LOSS OF OFFSITE POWER

Number of records: 23
Highest value: 3.2E-1/year
Lowest value: 2.0E-2/year

TRANSIENT CAUSED BY LOSS OF POWER CONVERSION SYSTEM

Number of records: 7
Highest value: 1.5E-0/year
Lowest value: 1.3E-1/year

TRANSIENT CAUSED BY LOSS OF SERVICE WATER

Number of records: 4
Highest value: 4.0E-3/year
Lowest value: 2.4E-5/year

TRANSIENT CAUSED BY STEAM LINE BREAK

Number of records: 8
Highest value: 3.9E-2/year
Lowest value: 3.8E-8/year

TRANSIENT CAUSED BY TURBINE OR REACTOR TRIP OR WITH FRONT LINE SYSTEMS AVAILABLE

Number of records: 23
Highest value: 10.0E-0/year
Lowest value: 9.2E-1/year

Appendix

PRINTOUT OF THE DATABASE ON GENERIC INITIATING EVENTS

	PLANT TYPE	IE SET	IE CATEGORY	INITIATORS	FREQ DESC	FREQ
1	PWR	LOCA	large LOCA	break with equ.area greater than 400 sq.cm	mean event/r.yr.	2.7E-4/yr
2	PWR	LOCA	medium LOCA	break with equ.area 80 to 400 sq.cm	mean event/r.yr.	8.0E-4/yr
3	PWR	LOCA	small LOCA	break with equ.area 2 to 80 sq.cm	mean event/r.yr.	2.7E-3/yr
4	PWR	LOCA	small leak on pressurizer	stuck open relief or safety valve	mean event/r.yr.	1.3E-3/yr
5	PWR	LOCA	large LOCA	break with equ.size greater than 13.5 inches	mean event/r.yr.	7.5E-5/yr
6	PWR	LOCA	large LOCA	break with equ.size 10 to 13.5 inches	mean event/r.yr.	1.2E-5/yr
7	PWR	LOCA	medium LOCA	break with equ.size 4 to 10 inches	mean event/r.yr.	1.6E-4/yr
8	PWR	LOCA	small LOCA	break with equ.size 1.66 to 4 inches	mean event/r.yr.	3.8E-4/yr
9	PWR	LOCA	small LOCA	break with equ.size 1.2 to 1.66 inches stuck open prsrz safety valve	mean event/r.yr.	3.1E-4/yr
10	PWR	LOCA	small small LOCA	break with equivalent size .38 to 1.2 inches stuck open relief valve RCP seal failure	mean event/r.yr.	2.0E-2/yr
11	BWR	LOCA	large LOCA	break with equ.size greater than 4 inches	mean event/r.yr.	4.0E-4/yr
12	BWR	LOCA	medium LOCA	break with equ.size 1 to 4 inches	mean event/r.yr.	2.0E-3/yr
13	BWR	LOCA	small LOCA	break with equ.size up to 1 inch	mean event/r.yr.	1.0E-2/yr
14	PWR	LOCA	large LOCA	break with equivalent size > 6 inches	mean event/r.yr.	5.0E-4/yr
15	BWR	LOCA	large LOCA	break with equivalent area > .1-.3 sq ft.	mean event/r.yr.	1.0E-4/yr
16	PWR	LOCA	medium LOCA	break with equivalent size 2 to 6 inches	mean event/r.yr.	1.0E-3/yr
17	BWR	LOCA	medium LOCA	break with equivalent area .004-.3 sqft(liq) .1-.3sqft(steam)	mean event/r.yr.	3.0E-4/yr
18	PWR	LOCA	small LOCA	break with equivalent size .5-2 inches	mean event/r.yr.	1.0E-3/yr
19	BWR	LOCA	small LOCA	break with equivalent area <.005 sqft(liq) or <.1 sqft(steam)	mean event/r.yr.	3.0E-3/yr
20	PWR	LOCA	small small LOCA	break with equ.size >.5 inch or flow 50-100 gpm	mean event/r.yr.	2.0E-2/yr
21	BWR	LOCA	small small LOCA	break with flow 50 to 100 gpm	mean event/r.yr.	3.0E-2/yr
22	PWR	LOCA	large LOCA	break with equivalent size 6 to 29 inches	mean event/r.yr.	5.0E-4/yr
23	PWR	LOCA	medium LOCA	break with equivalent size 2 to 6 inches	mean event/r.yr.	1.0E-3/yr
24	PWR	LOCA	small LOCA	break with equivalent size 1/2 to 2 inches	mean event/r.yr.	1.0E-3/yr
25	PWR	LOCA	small small LOCA	break with equivalent size <.5 inches	mean event/r.yr.	2.0E-2/yr
26	PWR	LOCA	intersystem LOCA		mean event/r.yr.	1.0E-6/yr
27	BWR	LOCA	large LOCA	break with equivalent area >.1 sqft	mean event/r.yr.	2.7E-4/yr
28	BWR	LOCA	medium LOCA	break with equivalent area .004 -.1 sqft	mean event/r.yr.	8.0E-4/yr
29	BWR	LOCA	small LOCA	break with equivalent area < .004 sqft(liq) or.01 sqft(steam)	mean event/r.yr.	2.7E-3/yr
30	BWR	LOCA	small small LOCA	leaks with 50-100 gpm flow (recirculation pump seal)	mean event/r.yr.	2.7E-2/yr
31	BWR	LOCA	intersystem LOCA		mean event/r.yr.	1.0E-8/yr
32	BWR	LOCA	large LOCA	break with equivalent size >.3 sq ft	mean event/r.yr.	3.0E-4/yr
33	BWR	LOCA	medium LOCA	break with equivalent area .005-.3 sqft(liq) and .1-.3sqft(steam)	mean event/r.yr.	8.0E-4/yr
34	BWR	LOCA	small LOCA	break with equivalent area <.005 sqft(liq) and <.1 sqft(steam)	mean event/r.yr.	3.0E-3/yr
35	BWR	LOCA	small small LOCA	leaks with 50 to 100 gpm flow (recirculation pump seal)	mean event/r.yr.	3.0E-2/yr
36	BWR	LOCA	intersystem LOCA		mean event/r.yr.	1.0E-8/yr
37	PWR	LOCA	large LOCA	break with effective diameter > 4 inches	mean event/r.yr.	9.3E-4/yr
38	PWR	LOCA	reactor vessel rupture		mean event/r.yr.	1.1E-6/yr
39	PWR	LOCA	small LOCA	break with equivalent size 1/2 to 4 inches inadvertant PORV or safety valve opening RCP seal failure control rod drive seal leakage	mean event/r.yr.	3.0E-3/yr
40	PWR	LOCA	steam generator tube rupture	tube rupture with leak greater than 100 gpm	mean event/r.yr.	8.6E-3/yr
41	PWR	LOCA	large LOCA	break with equivalent size >6 inches	mean event/r.yr.	4.7E-5/yr
42	PWR	LOCA	medium LOCA	break with equivalent size 2-6 inches	mean event/r.yr.	9.8E-4/yr
43	PWR	LOCA	small LOCA	break with equivalent size .5-2 inches	mean event/r.yr.	1.8E-3/yr
44	PWR	LOCA	intersystem LOCA		mean event/r.yr.	4.6E-6/yr
45	PWR	LOCA	reactor vessel rupture		mean event/r.yr.	1.0E-7/yr
46	BWR	LOCA	large LOCA	break with equivalent size >4 inches	point est. event/r.yr.	7.0E-4/yr
47	BWR	LOCA	medium LOCA	break with equivalent size 1-4 inches	point est. event/r.yr.	3.0E-3/yr
48	BWR	LOCA	small LOCA	break with equivalent size <1 inch	point est. event/r.yr.	8.0E-3/yr
49	BWR	LOCA	reactor vessel rupture		point est. event/r.yr.	3.0E-7/yr
50	BWR	LOCA	intersystem LOCA		point est. event/r.yr.	1.2E-7/yr
51	PWR & BWR	LOCA	reactor vessel rupture		median, event/reactor yr.	1.0E-7/yr
52	PWR & BWR	LOCA	large LOCA	break with equivalent size > 6 inches	median, event/reactor yr.	1.0E-4/yr
53	BWR	LOCA	medium LOCA	break with equivalent size 2.5-8.5 in (liq) and 4.7-6 in (steam)	median, events/reactor yr	3.0E-4/yr
54	PWR	LOCA	medium LOCA	break with equivalent size 2-6 inches	median, events/reactor yr	3.0E-4/yr
55	BWR	LOCA	small LOCA	break with equivalent size 0.6-2.6 in (liq) and 1.0-4.7in (steam)	median, events/reactor yr	1.0E-3/yr
56	PWR	LOCA	small LOCA	break with equivalent size 1/2-2 inches	median, events/reactor yr	1.0E-3/yr
57	PWR	LOCA	intersystem LOCA		median, event/reactor yr.	4.0E-6/yr
58	PWR	LOCA	large LOCA	break with equivalent size >4.3 inches	mean event/r.yr.	2.3E-4/yr
59	PWR	LOCA	medium LOCA	break with equivalent size 1.9 to 4.3 inches	mean event/r.yr.	2.4E-4/yr
60	PWR	LOCA	small LOCA	break with equivalent size .3 to 1.9 inches reactor coolant pump seal rupture	mean event/r.yr.	2.1E-2/yr
61	BWR	LOCA	large LOCA (1)	break with equivalent area .3 to 4.3 sq ft liquid suction side	mean event/r.yr.	9.9E-6/yr
62	BWR	LOCA	large LOCA (2)	break with equivalent area .3 to 4.3 sq ft liquid discharge side	mean event/r.yr.	3.9E-5/yr
63	BWR	LOCA	large LOCA (3)	break with equivalent area 1.4 to 4.1 sq ft steam	mean event/r.yr.	5.2E-5/yr
64	BWR	LOCA	medium LOCA (1)	break with equivalent size .12 to .3 sq ft liquid	mean event/r.yr.	9.0E-5/yr
65	BWR	LOCA	medium LOCA (2)	break with equivalent area .12 to 1.4 sq ft steam	mean event/r.yr.	2.1E-4/yr
66	BWR	LOCA	small LOCA	break with equivalent area <.12 sq ft	mean event/r.yr.	1.0E-3/yr
67	PWR	LOCA	large LOCA		mean events/r.yr.	2.0E-4/yr
68	PWR	LOCA	medium LOCA		mean events/r.yr.	4.6E-4/yr
69	PWR	LOCA	small LOCA (nonisolable)		mean events/r.yr.	4.0E-3/yr
70	PWR	LOCA	Small LOCA (isolable)		mean events/r.yr.	2.3E-2/yr
71	PWR	LOCA	steam generator tube rupture		mean events/r.yr.	8.2E-3/yr
72	BWR	LOCA	large LOCA	Break with equivalent flow >2000 kg/s	mean event/r.yr.	3.0E-4/yr
73	BWR	LOCA	medium LOCA	Break with equivalent flow 30 to 2000 kg/s	mean event/r.yr.	9.0E-4/yr
74	BWR	LOCA	small LOCA	break with equivalent flow 10 to 30 kg/s	mean event/r.yr.	3.0E-3/yr
75	BWR	LOCA	reactor vessel rupture		mean event/r.yr.	2.7E-7/yr
76	BWR	LOCA	large LOCA	break with equivalent area >450 sq cm	mean event/r.yr.	1.0E-4/yr
77	BWR	LOCA	medium LOCA	break with equivalent area 80 to 450 sq cm	mean event/r.yr.	3.0E-4/yr
78	BWR	LOCA	small LOCA	break with equivalent area < 80 sq cm	mean event/r.ry.	1.0E-3/yr

RANGE VAR	UPPER BND	LOWER BND	SOURCE		ELT SOURCE
EF 10	n/a	n/a	German Risk Study, table F1, 4-5		generic data sources
EF 10	n/a	n/a	German Risk Study table F1, 4-5		generic data sources
EF 10	n/a	n/a	German Risk Study, table F1, 4-5		generic data sources
EF 6	n/a	n/a	German Risk Study, table F1, 4-5		operating experience and engineering analysis
n/a	n/a	n/a	IREP-ANO1 table 4-7		RSS data combined with ANO-1 break ranges
n/a	n/.a	n/a	IREP-ANO1 table 4-7		RSS data combined with ANO-1 break ranges
n/a	n/a	n/a	IREP-ANO1 table 4-7		RSS data combined with ANO-1 break ranges
n/a	n/a	n/a	IREP-ANO1 table 4-7		RSS data combined with ANO-1 break ranges
n/a	n/a	n/a	IREP-ANO1 table 4-7		RSS data combined with ANO-1 break ranges
n/a	n/a	n/a	IREP-ANO1 table 4-7		RSS data combined with ANO-1 break ranges
n/a	n/a	n/a	Limeric PRA, table A.1 6		engineering evaluation and different data sources
n/a	n/a	n/a	Limeric PRA, table A.1 6		engineering evaluation and different data sources
n/a	n/a	n/a	Limeric PRA, table A.1 6		engineering evaluation and different data sources
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology...	ta le VIII 1-1	average of past PRAs
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology...	ta le VIII 1-1	RSS value
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology..	ta le VIII 1-1	average o past PRAs
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology...	ta le VIII 1-1	assesed from nuclear, industrial and other data sources
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology...	ta le VIII.1-1	average of past PRAs
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology...	ta le VIII.1-1	assesed form nuclear, industrial and other data Sources
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology...	ta le VIII.1-1	NPP operating experience
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology...	ta le VIII.1-1	review of past PRAs
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...,	ab a IV.3-1	Based on NUREG/CR - 4550 Vol. 1
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...,	ab a IV.3-1	Based on NUREG/CR - 4550 Vol 1
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...,	ab a IV.3-1	Based on NUREG/CR - 4550 Vol 1
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...,	ab a IV.3-1	Based on NUREG/CR - 4550 Vol 1
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...,	ab a IV.3-1	RSS data
n/a	n/a	n/a	NUREG/CR-4550, Vol.4, Peach Bottom...,	ble IV.3-1	past PRAs and other sources
n/a	n/a	n/a	NUREG/CR-4550, Vol.4, Peach Bottom...,	ble IV.3-1	past PRAs and other sources
n/a	n/a	n/a	NUREG/CR-4550, Vol.4, Peach Bottom...,	ble IV.3-1	past PRAs and other sources
n/a	n/a	n/a	NUREG/CR-4550, Vol.4, Peach Bottom...,	ble IV.3-1	past PRAs and other sources
n/a	n/a	n/a	NUREG/CR-4550, Vol.4, Peach Bottom...,	ble IV.3-1	analysis of the system interfaces using generic failure data
n/a	n/a	n/a	NUREG/CR-4550, Vol.6, Grand Gulf 1...	ta le IV.3-1	
n/a	n/a	n/a	NUREG/CR-4550, Vol.6, Grand Gulf 1...	ta le IV.3-1	
n/a	n/a	n/a	NUREG/CR-4550, Vol.6, Grand Gulf 1...	ta le IV.3-1	
n/a	n/a	n/a	NUREG/CR-4550, Vol.6, Grand Gulf 1...	ta le IV.3-1	
n/a	n/a	n/a	NUREG/CR-4550, Vol.6, Grand Gulf 1...	ta le IV.3-1	Analysis of the system interfaces using generic failure data
95%, 5%	2.8E-3/yr	9.4E-7/yr	Oconee PRA, table 5 9		generic prior updated with plant specific operating experience
95%, 5%	4.1E-6/yr	6.0E-8/yr	Oconee PRA, table 5.9		various sources
95%, 5%	1.2E-2/yr	1.0E-6/yr	Oconee PRA, table 5.9		update of generic prior
95%, 5%	2.7E-2/yr	2.6E-5/yr	Oconee PRA, table 5 9		generic data updated with plant specific operating experience
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4		engineering evaluation using generic data
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4		engineering evaluation using generic data
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4		engineering evaluation using generic data
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4		engineering evaluation using generic data
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4		generic data
EF 10	n/a	n/a	Shoreham PRA, Appendix A.1		generic data based on collection of operating experiences
EF 3	n/a	n/a	Shoreham PRA, Appendix A.1		generic data based on actual reactor operating experience
EF 3	n/a	n/a	Shoreham PRA, Appendix A.1		generic data based on reactor operating experience
EF 10	n/a	n/a	Shoreham PRA, Appendix A.1		generic data sources
n/a	n/a	n/a	Shoreham PRA, Appendix A.1		system evaluation
90% lognormal distr	1.0E-6/yr	1.0E-8/yr	WASH 1400, Reactor Safety Study, App	ad x V, chapter 4 5	based on non-nuclear experience
90% lognormal distr	1.0E-3/yr	1.0E-5/yr	WASH-1400, Reactor Safety Study, App	nd x III, table III 6-9	based on number of nuclear,industrial and other data sources
90% lognormal dist.	3.0E-3/yr	3.0E-5/yr	WASH-1400, Reactor Safety Study, App	nd x III, table III 6-9	assesment based on nuclear, industrial and other data sources
90% lognormal distr	3.0E-3/yr	3.0E-5/yr	WASH-1400, Reactor Safety Study, App	nd x III, table III 6-9	assesment based on nuclear industrial and other data sources
90% lognormal distr	1.0E-2/yr	1.0E-4/yr	WASH 1400, Reactor Safety Study, App	nd x III, table III 6-9	assesment based on nuclear, industrial and other data sources
90% lognormal distr	1.0E-2/yr	1.0E-4/yr	WASH-1400, Reactor Safety Study, App	nd x III, table III 6-9	assesed based on nuclear industrial and other data sources
EF 10			WASH-1400, Reactor Safety Study, App	nd x V, chapter 4 4	analysis of the system interface using generic failure rates
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.2		generic data
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.2		generic data
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.2		generic data
n/a	n/a	n/a	Browns Ferry Unit 1 IREP, table 5		generic data
n/a	n/a	n/a	Browns Ferry Unit 1 IREP, table 5		generic data
n/a	n/a	n/a	Browns Ferry Unit 1 IREP, table 5		generic data
n/a	n/a	n/a	Browns Ferry Unit 1 IREP, table 5		generic data
n/a	n/a	n/a	Browns Ferry Unit 1 IREP, table 5		generic data
n/a	n/a	n/a	Browns Ferry Unit 1 IREP, table 5		generic data
95%; 5% of distribution	5.2E-4/yr	7.6E-6/yr	Old PWR		Generic data updated with plant specific operating experience
95%, 5% of distribution	1.2E-3/yr	2.3E-5/yr	Old PWR		Generic data updated with plant specific operating experience
95%, 5% of distribution	1.4E-2/yr	1.2E-4/yr	Old PWR		Generic data updated with plant specific operating experience
95%, 5% of distribution	5.0E-2/yr	3.3E-3/yr	Old PWR		Generic data updated with plant specific operating experience
95%, 5% of distribution	2.1E-2/yr	3.1E-4/yr	Old PWR		Generic data updated with plant specific operating experience
n/a	n/a	n/a	Barseback 1 & 2 NPP		generic sources
n/a	n/a	n/a	Barseback 1 & 2 NPP		literature sources
n/a	n/a	n/a	Barseback 1 & 2 NPP		literature data
n/a	n/a	n/a	Forsmark 3 NPP		literature sources
n/a	n/a	n/a	Forsmark 3 NPP		literature sources
n/a	n/a	n/a	Forsmark 3 NPP		literature sources
n/a	n/a	n/a	Forsmark 3 NPP		literature sources

	PLANT TYPE	IE SET	IE CATEGORY	INITIATORS	FREQ DESC	FREQ
79	BWR	LOCA	large LOCA	break with equivalent flow 600 to 2000 kg/s	mean event/r.yr.	1.0E-7/yr
80	BWR	LOCA	medium LOCA	break with equivalent flow 35 to 600 kg/s	mean event/r.yr.	1.1E-5/yr
81	BWR	LOCA	small LOCA	break with equivalent flow <35 kg/s	mean event/r.yr.	1.1E-2/yr
82	BWR	LOCA	Intersystem LOCA		mean event/r.yr.	1.0E-7/yr
83	BWR	LOCA	large LOCA	break with equivalent flow > 2000 kg/s	mean event/r.yr.	1.0E-7/yr
84	BWR	LOCA	medium LOCA	break with equivalent flow 30 to 2000 kg/s	mean event/r.yr.	1.0E-6/yr
85	BWR	LOCA	small LOCA	break with equivalent flow 10 to 30 kg/s	mean event/r.yr.	1.0E-2/yr
86	BWR	LOCA	large LOCA	break with equivalent flow >3000 kg/s	mean event/r.yr.	1.0E-4/yr
87	BWR	LOCA	medium LOCA		mean event/r.yr.	5.0E-4/yr
88	BWR	LOCA	small LOCA		mean event/r.yr.	1.0E-3/yr
89	BWR	LOCA	large LOCA	break with equivalent flow >1200 kg/s	mean event/r.yr.	3.0E-4/yr
90	BWR	LOCA	medium LOCA	break with equivalent flow 35 to 1200 kg/s	mean event/r.yr.	9.0E-4/yr
91	BWR	LOCA	small LOCA	break with equivalent flow 5 to 35 kg/s	mean event/r.yr.	3.0E-3/yr
92	BWR	LOCA	Intersystem LOCA		mean event/r.yr.	1.9E-7/yr
93	BWR	LOCA	pressure vessel rupture		mean event/r.yr.	2.7E-7/yr
94	PWR	LOCA	large LOCA	break with equivalent diameter greater than 15 cm	mean event/r.yr.	4.0E-4/yr
95	PWR	LOCA	medium LOCA	break with equivalent diameter between 5 nad 15 cm	mean event/r.yr.	8.1E-4/yr
96	PWR	LOCA	small LOCA		mean event/r.yr.	1.1E-2/yr
97	PWR	LOCA	reactor vessel rupture		mean event/r.yr.	2.7E-7/yr
98	PWR	LOCA	Intersystem LOCA		mean event/r.yr.	4.2E-8/yr
99	PWR	LOCA	steam generator tube rupture		mean event/r.yr.	9.7E-3/yr
100	PWR	LOCA	large LOCA	Break with equivalent diameter greater than 6 inches RPV failure	mean events/r.yr.	9.4E-4/yr
101	PWR	LOCA	medium LOCA	Break with equivalent sizes between 2 and 6 inches Multiple pressurizer safety and relief valve failure	mean event/r.yr.	9.4E-4/yr
102	PWR	LOCA	small LOCA	Break with equivalent diameter smaller than 2 inches Pressurizer safety and relief valve failure CRDM failures RCP seal failure	mean event/r.yr.	3.5E-2/yr
103	PWR	LOCA	steam generator tube rupture		mean event/r.yr.	2.4E-2/yr
104	PWR	LOCA	small LOCA		mean event/r.yr.	3.5E-2/yr
105	PWR	LOCA	steam generator tube rupture		mean event/r.yr.	3.7E-2/yr
106	PWR	LOCA	medium LOCA		mean event/r.yr.	9.4E-4/yr
107	PWR	LOCA	large LOCA		mean event/r.yr.	9.4E-4/yr
108	PWR	LOCA	intersystem LOCA		mean event/r.yr.	4.0E-7/yr
109	BWR	LOCA	large LOCA inside drywell	break size greater than .3 sq ft	mean event/r.yr.	2.7E-4/yr
110	BWR	LOCA	large LOCA outside drywell	break size greater than .3 sq ft	mean event/r.yr.	1.0E-4/yr
111	BWR	LOCA	medium LOCA	break sizes between .1 and .3 sq ft(steam)and .004 and .2(liquid)	mean event/r.yr.	2.7E-3/yr
112	BWR	LOCA	small LOCA	break size up to .1sq ft (steam) or .004 sq ft (liquid)	mean event/r.yr.	2.7E-2/yr
113	BWR	LOCA	intersystem LOCA-LPCI break		mean event/r.yr.	7.3E-8/yr
114	BWR	LOCA	intersystem LOCA-CS break		mean event/r.yr.	2.3E-7/yr
115	BWR	LOCA	ECCS breaks		mean event/r.yr.	1.3E-5/yr
116	BWR	LOCA	reactor vessel rupture		mean event/r.yr.	3.0E-7/yr
117	PWR	transient	loss of feedwater(FW)	component failures in FW system	mean event/r.yr.	8.0E-1/yr
118	PWR	transient	loss of off site power(LOOP)	loss of voltage on more than 1 power bus	mean event/r.yr.	1.0E-1/yr
119	PWR	transient	loss of off site power		mean event/r.yr.	3.2E-1/yr
120	PWR	transient	loss of power conversion system (PCS)	Total loss of FW flow Full or partial closure of MSIV (1 loop) Closure of all MISV-s Increase in FW flow FW flow instability (operator error) FW flow instability (miscelaneous mechanical causes) Loss of condensate pumps (all) Loss of condenser vacuum (total) Opening of steam relief valves Loss of circulating water (CW)	mean event/r.yr.	1.0E+0/yr
121	PWR	transient	other than loss of PCS (turbine trip)	Loss of RCP flow (1 loop) Uncontrolled rod withdrawal CRDM problems and/or rod drop High or low przr pressure Containment pressure problems Full or partial closure of MSIV Turbine trip, throttle valve closure, EHC problems Generator trip and generator caused problems Spurious auto trip Manual trip due to false signals	mean event/r.yr.	7.1E+0/yr
122	BWR	transient	isolation of reactor from condenser	Closure of all MSIV Turbine trip without bypass Loss of condenser	mean event/r.yr.	1.1E+0/yr
123	BWR	transient	turbine trip	Partial closure of MSIV Turbine trip with bypass Startup of idle recirc.loop Pressure regulation failure Inadvertant opening of bypass Rod withdrawal Disturbance of FW Electric load rejection	mean event/r.yr.	4.0E+0/yr
124	BWR	transient	inadvertant open relief valve		mean event/r.yr.	6.0E-2/yr
125	BWR	transient	loss of feedwater	total loss of feedwater	mean event/r.yr.	7.0E-1/yr
126	BWR	transient	loss of off site power		mean event/r.yr.	5.3E-2/yr

RANGE/VAR	UPPER BND	LOWER BND	SOURCE		ULT SOURCE
n/a	n/a	n/a	Oskarshamn 1 NPP		literature sources (application of LBB criteria)
n/a	n/a	n/a	Oskarshamn 1 NPP		literature sources
n/a	n/a	n/a	Oskarshamn 1 NPP		literature sources
n/a	n/a	n/a	Oskarshamn 1 NPP		literature sources
n/a	n/a	n/a	Oskarshamn 2 NPP		literature sources
n/a	n/a	n/a	Oskarshamn 2 NPP		literature sources
n/a	n/a	n/a	Oskarshamn 2 NPP		literature sources
n/a	n/a	n/a	Oskarshamn 3 NPP		literature sources
n/a	n/a	n/a	Oskarshamn 3 NPP		literature sources
n/a	n/a	n/a	Oskarshamn 3 NPP		literature sources
n/a	n/a	n/a	Ringhals 1 NPP		literature sources
n/a	n/a	n/a	Ringhals 1 NPP		literature sources
n/a	n/a	n/a	Ringhals 1 NPP		literature sources
n/a	n/a	n/a	Ringhals 1 NPP		literature sources
n/a	n/a	n/a	Ringhals 1 NPP		literature data
95%	1.2E-3/yr	n/a	Ringhals 2 NPP		literature sources
95%	3.0E-3/yr	n/a	Ringhals 2 NPP		literature sources
95%	2.5E-2/yr	n/a	Ringhals 2 NPP		literature sources
95%	1.0E-6/yr	n/a	Ringhals 2 NPP		literature sources
n/a	n/a	n/a	Ringhals 2 NPP		
95%	2.0E-2/yr	n/a	Ringhals 2 NPP		literature sources
95%, 5%	3.6E-3/yr	3.3E-5/yr	Zion NPP PSS table 1 1.1-2		literature sources and plant operating experience
95%, 5%	3.6E-3/yr	3.3E-5/yr	Zion NPP PSS table 1 1.1-2		literature sources and plant operating experience
95%, 5%	7.2E-2/yr	1.3E-2/yr	Zion NPP PSS table 1 1.1-2		literature sources and plant operating experience
95%, 5%	7.7E-2/yr	2.8E-3/yr	Zion NPP PSS table 1.1.1-2		literature sources and plant operating experience
n/a	n/a	n/a	Angra NPP PSA, Summary report		literature sources
n/a	n/a	n/a	Angra NPP PSA, Summary report		literature sources
n/a	n/a	n/a	Angra NPP PSA, Summary report		literature sources
n/a	n/a	n/a	Angra NPP PSA, Summary report		literature sources
n/a	n/a	n/a	Angra NPP PSA, Summary report		literature sources
n/a	n/a	n/a	Caorso NPP PSS table A-2		literature sources
n/a	n/a	n/a	Caorso NPP PSS table A-2		literature sources
n/a	n/a	n/a	Caorso NPP PSS table A-2		literature sources
n/a	n/a	n/a	Caorso NPP PSS table A-2		literature sources
n/a	n/a	n/a	Caorso NPP PSS table A-2		literature data
n/a	n/a	n/a	Caorso NPP PSS table A-2		literature sources
n/a	n/a	n/a	Caorso NPP PSS table A-2		literature sources
n/a	n/a	n/a	Caorso NPP PSS table 3.5		literature sources
EF 3	n/a	n/a	German Risk Study, table F1, 4-5		german operating experience
EF 3	n/a	n/a	German Risk Study, table F1, 4-5		system analysis
n/a	n/a	n/a	IREP-ANO1 table 4-7		EPRI NP-801 and data from the utility
n/a	n/a	n/a	IREP-ANO1 table 4-7		EPRI NP-801 and plant specific info
n/a	n/a	n/a	IREP-ANO1 table 4-7		EPRI NP-801 and plant specific info
n/a	n/a	n/a	Limeric PRA, table A.1.3		BWR operating experience
n/a	n/a	n/a	Limeric PRA, table A.1.3		BWR operating experience
n/a	n/a	n/a	Limeric PRA, table A.1.3		BWR operating experience
n/a	n/a	n/a	Limeric PRA, table A.1.3		BWR operating experience
n/a	n/a	n/a	Limeric PRA, table A.6.2		regional grid data

	PLANT TYPE	IE SET	IE CATEGORY	INITIATORS	FREQ DESC	FREQ
127	PWR	transient	loss of PCS	Total loss of FW flow (listed in DESCR.field) FW flow instability-operator error;mechanical causes Full or partial closure of MSIV(one loop) Closure of all MSIV-s Loss of condensate pumps (all loops) Loss of condenser vacuum Condenser leakage SG leakage Opening of steam relief valve Miscelaneous leakages on secondary side Loss of CW Loss of CC Loss of SW	mean event/r yr	1 4E+0/yr
128	BWR	transient	loss of power conversion system	Electric load rejection with bypass failure Turbine trip with bypass failure Full MSIV closure Partial MSIV closure Loss of condenser vacuum Pressure regulator fails open Pressure regulator fails closed Turbine bypass fails open Turbine bypass fails close Turbine bypass or control valve increase pressure	mean event/r yr	1 6E+0/yr
129	PWR & BWR	transient	loss of a DC bus		mean event/r yr	5 0E-3/yr
130	PWR	transient	other than loss of PCS	Loss of RCS flow(one loop); Total loss of RCS flow Uncontroled rod withdrawal;CVCS problems-boron dilution CRDM mechanical problems or rod drop Leakage from: control rod, primary system, pressurizer Pressurizer pressure: low or high; Pressurizer spray failure Inadvertant SI signal Containment pressure problem Spurious trip; auto or manual trip(no transient conditions) Pressure-temperature-power imbalance rod position error Startup of inactive RCS loop Loss or reduction of FW flow(one loop) Increase in FW flow:one loop;all loops Loss of condensate pumps:one loop; all loops Turbine trip,generator trip;throttle valve closure;EHC problems	mean event/r yr	6.8E+0/yr
131	BWR	transient	loss of FW (steam side PCS available)	loss of all FW flow FW low flow	mean event/r yr	5 6E-1/yr
132	BWR	transient	other than loss of PCS	Turbine trip ,electric load rejection Recirculation control failure: decreasing flow; increasing flow Recirculation pump trip: one, all Startup of idle recirculation pump Recirculation pump seizure FW increasing flow; loss of FW heater Trip of one FW or condensate pump Rod withdrawal,inadvertant insertion of rods Inadvertant startup of HPCI/HPCS Spurious trip via instrumentation RPS fault SCRAM due to other plant occurances Manual SCRAM-no transient condition SCRAM cause unknown	mean event/r yr	4 8E+0/yr
133	PWR	transient	loss of off site power	loss of all offsite power loss of power to neccessary plant system	mean event/r yr	1 0E-1/yr
134	BWR	transient	loss of off site power	loss of off site power loss of auxiliary power (transformer)	mean event/r yr	1.0E-1/yr
135	BWR	transient	inadvertant opening of relief valve		mean event/r yr	1 4E-1/yr
136	PWR	transient	turbine trip with MFW available	turbine trip; reactor trip loss of load MSIV closure loss of turbine control	mean event/r yr	7.3E+0/yr
137	PWR	transient	loss of main feedwater	failure of main FW high SG water level inadvertant SI signal	mean event/r yr	9 4E-1/yr
138	PWR	transient	loss of charging pump cooling	loss of charging pump CC loss of charging pump SW	mean event/r yr	3 0E-2/yr
139	PWR	transient	loss of off site power	failure of offsite power grid loss of station reserve power loss of power to the switchyard	mean event/r yr	7 0E-2/yr
140	BWR	transient	loss of PCS		mean event/r yr	1 5E+0/yr
141	BWR	transient	other than loss of PCS	inadvertant open relief valve in primary system loss of FW but steam side of PCS initially available other initiators with same consequences	mean event/r yr	2 6E+0/yr
142	BWR	transient	loss of off site power		mean event/r yr	7 0E-2/yr
143	BWR	transient	loss of PCS		mean event/r yr	1 5E+0/yr
144	BWR	transient	loss of FW and steam side PCS available		mean event/r yr	7 0E-2/yr
145	BWR	transient	other than loss of PCS		mean event/r yr	5 4E+0/yr

RANGE VAR	UPPER BND	LOWER BND	SOURCE	ULT SOURCE
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology..., table VIII 1-1	Based on NUREG/CR 3862
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology..., table VIII 1-1	Based on NUREG/CR 3862
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology..., table VIII 1-1	DC power supply study, NUREG 0666
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology..., table VIII 1-1	Based on NUREG/CR 3862
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology..., table VIII 1 1	Based on NUREG/CR 3862
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology..., table VIII.1-1	
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology..., table VIII 1-1	plant blackout analysis NUREG 1032
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology..., table VIII.1-1	Plant blackout analysis NUREG 1032
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology..., table VIII.1 1	Based on NUREG/CR 3862
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1..., table IV.3-1	plant specific operating experience
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1 ..., table IV 3-1	plant specific operating experience
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1..., table IV.3-1	Comparison of NUREG/CR 3862 and operating experience
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1..., table IV 3-1	NUREG 1032
n/a	n/a	n/a	NUREG/CR-4550, Vol.4, Peach Bottom.. , table IV 3-1	nuclear sources and plant operating experience
n/a	n/a	n/a	NUREG/CR-4550, Vol.4, Peach Bottom...., table IV 3-1	nuclear sources and plant operating experience
n/a	n/a	n/a	NUREG/CR-4550, Vol 4 Peach Bottom...., table IV 3-1	nuclear sources and plant operating experience
n/a	n/a	n/a	NUREG/CR-4550, Vol 6, Grand Gulf 1 , table IV 3 1	different nuclear sources and plant operating experience
n/a	n/a	n/a	NUREG/CR-4550, Vol.6, Grand Gulf 1 , table IV 3 1	different nuclear sources and plant operating experience
n/a	n/a	n/a	NUREG/CR-4550, Vol 6, Grand Gulf 1 table IV 3 1	different nuclear sources and plant operating experience

PLANT TYPE		IE SET	IE CATEGORY	INITIATORS	FREQ DESC	FREQ
146	BWR	transient	loss of off site power		mean event/r.yr.	7.0E-2/yr
147	BWR	transient	inadvertant opening of relief valve		mean event/r.yr.	1.4E-1/yr
148	PWR	transient	reactor/turbine trip	Rod drop Inadvertant rod withdrawal Inadvertant boration or dilution Reactor trip Cold water addition Inadequate main FW Turbine and control valve malfunction Pressurizer spray failure Turbine trip Generator faults Grid distrurbances Administrative shutdowns	mean event/r.yr.	4.9E+0/yr
149	PWR	transient	spurious SI signal	HPI flow	mean event/r.yr.	1.0E-2/yr
150	PWR	transient	steam line break	steam line rupture turbine bypass valve inadvertant opening	mean event/r.yr.	3.0E-3/yr
151	PWR	transient	loss of condenser	loss of condenser vacuum loss of condenser circulating water	mean event/r.yr.	2.1E-1/yr
152	PWR	transient	FW line break (large)		mean event/r.yr.	9.3E-4/yr
153	PWR	transient	loss of main FW		mean event/r.yr.	6.4E-1/yr
154	PWR	transient	partial loss of main FW		mean event/r.yr.	6.9E-1/yr
155	PWR	transient	loss of off site power	failure of grid failure of feeders	mean event/r.yr.	4.0E-2/yr
156	PWR	transient	loss of off site power (substation faults)		mean event/r.yr.	1.3E-1/yr
157	PWR	transient	excessive feedwater		mean event/r.yr.	9.2E-2/yr
158	PWR	transient	spurious low pressurizer signal		mean event/r.yr.	4.4E-2/yr
159	PWR	transient	loss of off site power		mean event/r.yr.	2.0E-1/yr
160	PWR	transient	loss of main FW		mean event/r.yr.	3.0E+0/yr
161	PWR	transient	transient with main FW available		mean event/r.yr.	4.0E+0/yr
162	BWR	transient	loss of condenser	loss of normal condenser vacuum turbine trip with bypass valve failure electric load rejection with bypass valve failure	mean event/r.yr.	4.1E-1/yr
163	BWR	transient	MSIV closure		mean event/r.yr.	2.4E-1/yr
164	BWR	transient	loss of feedwater		mean event/r.yr.	1.8E-1/yr
165	BWR	transient	loss of off site power		mean event/r.yr.	8.0E-2/yr
166	BWR	transient	inadvertant opening of relief valve		mean event/r.yr.	9.0E-2/yr
167	BWR	transient	control rod withdrawal		mean event/r.yr.	3.0E-2/yr
168	BWR	transient	turbine trip	Electric load rejection Turbine trip Inadvertant closure of one MSIV; partial MSIV closure Pressure regulator fails: open; closed Turbine bypass fails open Turbine bypass or control valve cause increase pressure Recirculation flow failure:decreasing;increasing Trip of recirculation pump: one; all Abnormal startup of idle recirculation pump Recirculation pump seizure Loss of FW heater Trip of one FW or condensate pump	mean event/r.yr.	4.5E+0/yr
169	BWR	transient	main steam line break		point est. event/r.yr.	3.8E-8/yr
170	BWR	transient	feedwater line break		point est. event/r.yr.	8.7E-9/yr
171	BWR	transient	HPCI/RCIC line breaks		point est. event/r.yr.	1.4E-8/yr
172	BWR	transient	rapid shutdown	Rod withdrawal at power FW controler failure; loss of FW flow Recirculation flow control failure:decreasing;increasing Startup of idle recirculation pump Loss of FW heating Inadvertant HPCI start Loss of auxiliary power Turbine trip(turbine valve closure);load rejection(stop valve cl) MSIV closure Recirculation pump trip (one pump) Recircualtion pump seizure:one pump; two pumps T-G pressure regulator failure-rapid opening Rod ejection; rod drop accident Startup of idle recirc.pump with simultaneous turbine trip	median, event/reactor yr.	1.0E+1/yr
173	PWR	transient	rapid shutdown	Turbine trip; loss of main generator; LOOP Loss of condenser vacuum Inadvertant MSIV closure Loss of main FW; loss of main CW; loss of condensate pumps Inadvertant openig of SG PORV-s Increase in MFW flow Opening of all bypass valves(steam dump) Uncontroled rod withdrawal; control assembly drop Boron dilution (CVCS malfunction) Startup of inactive RCS loop Opening of pressurizer SRV or RV Loss of RCS coolant flow; siezure of all RCP-s Rupture of FW piping or main steam lines; rupture of SG Rupture of CRDM housing	median, events/reactor yı	1.0E+1/yr

RANGE VAR.	UPPER BND	LOWER BND	SOURCE	UET SOURCE
n/a	n/a	n/a	NUREG/CR-4550, Vol.6, Grand Gulf 1 , t ble IV.3-1	assesment of different nuclear sources
n/a	n/a	n/a	NUREG/CR-4550, Vol.6, Grand Gulf 1 , t ble IV 3-1	different nuclear sources and plant operating experience
95%, 5%	5.7E+0/yr	4 1E+0/yr	Oconee PRA, table 5 9	generic data updated with plant specific operating experience
95%, 5%	4.3E-2/yr	7 8E-6/yr	Oconee PRA, table 5 9	generic data updated with plant specific operating experience
95%, 5%	1.2E-2/yr	1 0E-6/yr	Oconee PRA, table 5.9	generic data updated with plant specific operating experience
95%, 5%	3.8E+0/yr 3 8E-1/yr	8.3E-2/yr	Oconee PRA, table 5.9 BNL Review, table 4.3	generic data updated with plant specific operating experience recalculation made by the reviewers *
95%, 5%	2.8E-3/yr	6.9E-7/yr	Oconee PRA, table 5.9	generic data updated with plant specific operating experience
95%, 5%	9.2E-1/yr	3 6E-1/yr	Oconee PRA, table 5.9	generic data updated with plant specific operating experience
95%, 5%	9.7E-1/yr	4.0E-1/yr	Oconee PRA, table 5.9	generic data updated with plant specific operating experience
95%, 5%	9.7E-2/yr	7.1E-3/yr	Oconee PRA, table 5.9	generic data updated with plant specific operating experience
95%, 5%	3.0E-1/yr	2.2E-2/yr	Oconee PRA, table 5.9	generic data updated with plant specific operating experience
95%, 5%	2.1E-1/yr	1.8E-2/yr	Oconee PRA, table 5 9	generic data updated with plant specific operating exeperience
95%, 5%	1 7E-1/yr	1.7E-3/yr	Oconee PRA, table 5.9	system analysis
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4	
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4	
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4	
EF 3	n/a	n/a	Shoreham PRA, Appendix A, table A.1-'	generic data based on nuclear operating expeirence taken from EPRI NP-801
EF 3	n/a	n/a	Shoreham PRA, Appendix A, table A.1-'	generic data based on nuclear operating expeirence etc.
EF 3	n/a	n/a	Shoreham PRA, Appendix A, table A 1-	generic data based on nuclear operating expeirence etc.
EF 3	n/a	n/a	Shoreham PRA, Appendix A, table A 1-	utility specific data
EF 3	n/a	n/a	Shoreham PRA, Appendix A, table A 1-	generic data and engineering evaluation
n/a	n/a	n/a	Shoreham PRA, Appendix A, table A.1-	generic data based on nuclear operating expeirence etc.
EF 3	n/a	n/a	Shoreham PRA, Appendix A, table A.1-	generic data based on nuclear operating expeirence etc.
n/a	n/a	n/a	Shoreham PRA, Appendix A.1	engineering evaluation
n/a	n/a	n/a	Shoreham PRA, Appendix A.1	engineering evaluation
n/a	n/a	n/a	Shoreham PRA, Appendix A.1	engineering evaluation
EF 2			WASH-1400, Reactor Safety Study, Apr en ix V, chapter 4.3 & Appendix I, Table I - 4.12	NPP operating experience, engineering estimate
EF 2			WASH-1400, Reactor Safety Study, Apr er ix V, chapter 4 3 & Appendix I, Table I - 4 9	NPP operating experience, engineering estimate

PLANT TYPE		IE SET	IE CATEGORY	INITIATORS	FREQ DESC	FREQ
174	PWR	transient	loss of off site power		mean event/r yr	1 4E-1/yr
175	PWR	transient	total interuption of PCS	Total loss of FW flow Closure of all MSIVs FW flow instability Loss of all condensate pumps Loss of condenser vacuum Loss of CW	mean event/r yr	8.0E-1/yr
176	PWR	transient	transient requiring RCS pressure relief	turbine trip or throttle valve closure generator trip or generator caused faults loss of power to neccessary plant systems	mean event/r yr	1 9E+0/yr
177	PWR	transient	transient which do not affect front line systems	Loss of RCS flow (one loop); total loss of RCS flow CRDM problems, rod drop Pressurizer pressure: low;high Inadvertant SI signal CVCS malfunction- boron dilution Pressure/temprerature/power imbalance Loss or reduction in main FW (1 loop) Full or partial closure of one MSIV Increase in FW flow: one loop; all loops Loss of condensate pump Condenser leakage; Lekage in secondary system Sudden opening of relief valve Pressurizer spray failure Spurious trip; Manual trip;auto trip-No transient conditions	mean event/r yr	6.8E+0/yr
178	BWR	transient	loss of PCS	MSIV closure Loss of normal condenser vacuum Pressure regulator fails open Loss of feedwater flow Loss of off site power Loss of auxiliary power Increased flow at power	mean event/r yr	1 7E+0/yr
179	BWR	transient	other than loss of PCS	Electric load rejection Electric load rejection with bypass failure Turbine trip Turbine trip with bypass failure Inadvertant opening of MSIV Pressure regulator fails closed Bypass/control valve causing pressure increase Recirculation control fails causing increased flow	mean event/r yr.	1.7E+0/yr
180	PWR	transient	turbine trip		mean events/r yr.	1 0E-0/yr
181	PWR	transient	loss of main FW (total loss)		mean events/r yr	3.3E-1/yr
182	PWR	transient	loss of main FW (partial)		mean events/r yr	4.9E-1/yr
183	PWR	transient	excessive FW flow		mean events/r yr	3.1E-1/yr
184	PWR	transient	loss of condenser vacuum		mean events/r yr.	8.1E-2/yr
185	PWR	transient	closure of one MSIV		mean events/r yr	7.2E-2/yr
186	PWR	transient	closure of all MSIVs		mean events/r yr	2.0E-3/yr
187	PWR	transient	core power excursion		mean events/r yr	1.5E-2/yr
188	PWR	transient	steam line break (inside containment)		mean events/r yr.	4.6E-4/yr
189	PWR	transient	steam line break (outside containment)		mean events/r yr.	4.2E-3/yr
190	PWR	transient	main steam relief valve openings		mean events/r.yr.	1 6E-2/yr
191	PWR	transient	inadvertant SI		mean events/r yr.	6.3E-2/yr
192	PWR	transient	total loss of reactor coolant flow		mean events/r yr	6.9E-2/yr
193	PWR	transient	loss of off-site power		mean events/r yr.	4 9E-2/yr
194	BWR	transient	transient with front line systems available		mean event/r yr	2.0E+0/yr
195	BWR	transient	loss of condenser		mean event/r yr	3.3E-1/yr
196	BWR	transient	loss of FW (total loss)		mean event/r.yr	1.6E-1/yr
197	BWR	transient	loss of FW and condenser		mean event/r yr.	3.3E-1/yr
198	BWR	transient	loss of off-site power		mean event/r yr.	1 6E-1/yr
199	BWR	transient	transient with front line systems available		mean event/r yr.	1 7E-1/yr
200	BWR	transient	loss of condenser		mean event/r.yr.	2.3E+0/yr
201	BWR	transient	loss of FW		mean event/r yr	1 7E-1/yr
202	BWR	transient	loss of FW and condenser		mean event/r yr	1 7E-1/yr
203	BWR	transient	loss of off-site power		mean event/r yr	1 8E-1/yr
204	BWR	transient	transient with front line systems available		mean event/r yr	1.9E+0/yr
205	BWR	transient	loss of condenser		mean event/r yr.	3.2E-1/yr
206	BWR	transient	loss of FW and condenser		mean event/r yr	3.2E-1/yr
207	BWR	transient	loss of off-site power		mean event/r yr	3.2E-1/yr
208	BWR	transient	transient with front line systems available		mean event/r yr	1 6E+0/yr
209	BWR	transient	loss of condensor		mean event/r yr	7.8E-1/yr
210	BWR	transient	loss of FW		mean event/r yr	7.8E-1/yr
211	BWR	transient	loss of off-site power		mean event/r yr.	2.0E-2/yr
212	BWR	transient	transient with front line systems available		mean event/r yr	9 8E-1/yr
213	BWR	transient	loss of condenser		mean event/r yr	1 6E-1/yr
214	BWR	transient	loss of FW and condenser		mean event/r yr	9.8E-1/yr
215	BWR	transient	loss of off site power		mean event/r yr	1.6E-1/yr
216	BWR	transient	transient with front line systems available		mean event/r.yr.	9.2E-1/yr
217	BWR	transient	loss of condenser		mean event/r yr	2.0E+0/yr
218	BWR	transient	loss of FW & condenser		mean event/r yr	3.7E-1/yr
219	BWR	transient	loss of off site power		mean event/r yr	1 8E-1/yr
220	PWR	transient	transient with PCS isolation		mean event/r yr	1.3E-1/yr
221	PWR	transient	transient with front line systems available		mean event/r.yr	6 0E+0/yr
222	PWR	transient	loss of off site power		mean event/r yr	7 0E-1/yr
223	PWR	transient	transient requiring RCS pressure relief		mean event/r yr	4 0E-1/yr
224	PWR	transient	inadvertant safety injection		mean event/r yr	1 2E+0/yr
225	PWR	transient	steam line break		mean event/r yr	4 4E-4/yr

RANGE VAR	UPPER BND	LOWER BND	SOURCE	ULT SOURCE
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.8	EPRI NP-2230
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.8	EPRI NP-2230
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.8	EPRI NP-2230
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.8	EPRI NP-2230
n/a	n/a	n/a	Browns Ferry Unit 1 IREP, table 6	plant specific data
n/a	n/a	n/a	Browns Ferry Unit 1 IREP, table 6	plant specific data
95%; 5% of distribution	1.4E-0/yr	7.1E-1/yr	Old PWR	Generic data updated with plant specific operating experience
95%; 5% of distribution	4.9E-1/yr	1.7E-1/yr	Old PWR	Generic data updated with plant specific operating experience
95%; 5% of distribution	6.5E-1/yr	2.9E-1/yr	Old PWR	Generic data updated with plant specific operating experience
95%; 5% of distribution	4.9E-1/yr	1.4E-1/yr	Old PWR	Generic data updated with plant specific operating experience
95%; 5% of distribution	1.5E-1/yr	1.8E-2/yr	Old PWR	Generic data updated with plant specific operating experience
95%; 5% of distribution	1.5E-1/yr	1.8E-2/yr	Old PWR	Generic data updated with plant specific operating experience
95%; 5% of distribution	5.7E-3/yr	3.9E-5/yr	Old PWR	Generic data updated with plant specific operating experience
95%; 5% of distribution	3.6E-2/yr	6.7E-4/yr	Old PWR	Generic data updated with plant specific operating experience
95%; 5% of distribution	1.2E-3/yr	2.3E-5/yr	Old PWR	Generic data updated with plant specific operating experience
95%; 5% of distribution	1.3E-2/yr	1.1E-4/yr	Old PWR	Generic data updated with plant specific operating experience
95%; 5% of distribution	4.4E-2/yr	5.7E-4/yr	Old PWR	Generic data updated with plant specific operating experience
95%; 5% of distribution	1.2E-1/yr	1.4E-2/yr	Old PWR	Generic data updated with plant specific operating experience
95%; 5% of distribution	1.5E-1/yr	1.8E-2/yr	Old PWR	Generic data updated with plant specific operating experience
95%; 5% of distribution	1.1E-1/yr	7.1E-3/yr	Old PWR	Generic data updated with plant specific operating experience
n/a	n/a	n/a	Barseback 1 NPP	plant operating experience (8.11 years)
n/a	n/a	n/a	Barseback 1 NPP	plant specific operating experience (8.11 years)
n/a	n/a	n/a	Barseback 1 NPP	plant specific operating experience (8.11 years)
n/a	n/a	n/a	Barseback 1 NPP	plant specific operating experience (8.11 years)
n/a	n/a	n/a	Barseback 1 NPP	plant specific operating experience
n/a	n/a	n/a	Oskarshamn 1 NPP	plant operating experience (5.7 years)
n/a	n/a	n/a	Oskarshamn 1 NPP	plant operating experience (5.7 years)
n/a	n/a	n/a	Oskarshamn 1 NPP	plant specific operating experience (5.7 years)
n/a	n/a	n/a	Oskarshamn 1 NPP	plant specific operating experience(5.7 years)
n/a	n/a	n/a	Oskarshamn 1 NPP	plant specific operating experience (5.7 years)
n/a	n/a	n/a	Oskarshamn 2 NPP	plant operating experience (6.2 years)
n/a	n/a	n/a	Oskarshamn 2 NPP	plant specific operating experience (6.2 years)
n/a	n/a	n/a	Oskarshamn 2 NPP	plant specific operating experience (6.2 years)
n/a	n/a	n/a	Oskarshamn 2 NPP	plant specific experience
n/a	n/a	n/a	Oskarshamn 3 NPP	plant specific operating experience (6.4 years)
n/a	n/a	n/a	Oskarshamn 3 NPP	plant specific operating experience (6.4 years)
n/a	n/a	n/a	Oskarshamn 3 NPP	plant specific operating experience (6.4 years)
n/a	n/a	n/a	Oskarshamn 3 NPP	literature sources
n/a	n/a	n/a	Barseback 2 NPP	plant specific operating experience
n/a	n/a	n/a	Barseback 2 NPP	plant specific operating experience
n/a	n/a	n/a	Barseback 2 NPP	plant specific operating experience
n/a	n/a	n/a	Barseback 2 NPP	plant specific operating experience
n/a	n/a	n/a	Ringhals 1 NPP	plant specific operating experience (5.44 years)
n/a	n/a	n/a	Ringhals 1 NPP	plant specific operating experience (5.44 years)
n/a	n/a	n/a	Ringhals 1 NPP	plant specific operating experience
n/a	n/a	n/a	Ringhals 1 NPP	plant specific operating experience (5.44 years)
95%	4.0E-1/yr	n/a	Ringhals 2 NPP	plant specific operating experience
95%	9.6E+0/yr	n/a	Ringhals 2 NPP	plant specific operating experience
95%	1.2E+0/yr	n/a	Ringhals 2 NPP	plant specific operating experience
95%	8.6E-1/yr	n/a	Ringhals 2 NPP	plant specific operating experience
95%	2.0E+0/yr	n/a	Ringhals 2 NPP	plant specific operating experience
95%	1.2E-3/yr	n/a	Ringhals 2 NPP	Literature sources

	PLANT TYPE	IE SET	IE CATEGORY	INITIATORS	FREQ DESC	FREQ
226	PWR	transient	transient occurring after shutdown		mean event/r yr	1 0E+0/yr
227	PWR	transient	loss of RCS flow		mean event/r yr	3 6E-1/yr
228	PWR	transient	loss of feedwater flow	FW pipe rupture outside containment Loss / reduction of FW flow in one SG Loss of FW flow to all SG FW flow instability - operator error FW flow instability - mechanical causes Loss of one condensate pump Loss of all condensate pumps Condenser leakage Other secondary leakage	mean event/r yr	5.2E+0/yr
229	PWR	transient	partial loss of steam flow	MSIV Closure Full closure of MSIV Partial closure of MSIV Other losses of steam flow	mean event/r yr	2.5E-1/yr
230	PWR	transient	turbine trip	Closure of all MSIV-s Increase of FW flow in one SG Loss of condenser vacuum Loss of CW Throttle valve closure/Electrohydraulic control problem Generator trip / generator caused faults Turbine trip due to over speed Other turbine trips	mean event/r yr	3 7E+0/yr
231	PWR	transient	loss of off site power		mean event/r yr	5.8E-2/yr
232	PWR	transient	spurious safety injection		mean event/r yr	6.4E-1/yr
233	PWR	transient	reactor trip	CRDM problem / rod drops High and low pressurizer pressure High pressurizer level Spurious automatic trip - no transient condition Automatic/manual trip - operator error Manual trip due to false signals Spurious trip - cause unknown Primary system pressure temperature or power imbalance	mean event/r yr	3.8E+0/yr
234	PWR	transient	loss of steam inside containment	Steam pipe rupture inside containment FW pipe rupture inside containment Steam relief or safety valve open inadvertantly (leak upstream of MSIV) Other steam losses inside containment	mean event/r yr	9 4E-4/yr
235	PWR	transient	loss of steam outside containment	Steam pipe rupture outside containment Throttle valve/Electrohydraulic control problems Steam and dump valve fail open Other steam losses outside containment	mean event/r yr	9 4E-4/yr
236	PWR	transient	core power increase	Uncontrolled rod withdrawal Boron dilution - CVCS malfunction Core inlet temperature drop Other positive reactivity additions	mean event/r yr	2.3E-2/yr
237	PWR	transient	main steam line break		mean event/r yr	3.9E-2/yr
238	PWR	transient	loss of off-site power		mean event/r.yr.	1.5E-1/yr
239	PWR	transient	loss of main feedwater		mean event/r yr	3.2E+0/yr
240	PWR	transient	reactor trip		mean event/r yr	3 1E+0/yr
241	PWR	transient	turbine trip		mean event/r yr	3 0E+0/yr
242	BWR	transient	manual shutdown or spurious trip	Spurious trip via instrumentation Scram due to plant occurences Detected fault in RPS Inadvertant insertion of rod or rods Manual shutdown	mean event/r yr	4.5E+0/yr
243	BWR	transient	turbine trip	Turbine trip Generator trip High FW during startup or shutdown Inadvertant startup of HPCI Recirculation control failure increasing flow Turbine bypass or control valve failures Trip of one or all recirculation pumps Loss of FW heater Recirculation pump seizure Pressure regulator fails open Turbine bypass fails open FW flow increasing at power FW controller maximum demand	mean event/r yr	2.7E+0/yr
244	BWR	transient	MSIV closure	MSIV closure Partial MSIV closure Inadvertant closure of one MSIV Loss of condenser vacuum Turbine trip with bypass failure Generator trip with bypass failure Pressure regulator fails closed	mean event/r yr	4.1E-1/yr
245	BWR	transient	inadvertant opening of relief valve		mean event/r yr	1 3E-2/yr
246	BWR	transient	loss of feedwater	Loss of all FW flow Trip of one of FW pumps FW low Flow Low FW flow during startup or shutdown	mean event/r yr	2.1E-1/yr
247	BWR	transient	loss of off site power		mean event/r yr	6.7E-2/yr

RANGE VAR	UPPER BND	LOWER BND	SOURCE		DET SOURCE
95%	1 7E+0/yr	n/a	Ringhals 2 NPP		plant specific operating experience
95%,5%	6.0E-1/yr	1 9E-1/yr	Zion NPP PSS table 1 1.1-2		literature sources and plant operating experience
95%,5%	6 4E+0/yr	4 1E+0/yr	Zion NPP PSS table 1 1 1 2		literature sources (prior) and plant operating experience
95%,5%	5.2E-1/yr	9 6E-2/yr	Zion NPP PSS table 1 1 1-2		literature sources (prior) and plant operating experience
95%,5%	4 7E+0/yr	2.8E+0/yr	Zion NPP PSS table 1 1 1-2		literature sources (prior) and plant operating experience
95%,5%	1 7E-1/yr	8.3E-3/yr	Zion NPP PSS table 1.1 1-2		literature sources and plant operating experience
95%,5%	1.1E+0/yr	3.3E-1/yr	Zion NPP PSS table 1.1 1-2		literature sources and plant operating experience
95%,5%	4 7E+0/yr	2.9E+0/yr	Zion NPP PSS table 1 1 1 2		literature sources and plant operating experience
95%,5%	3.6E-3/yr	3.3E-5/yr	Zion NPP PSS table 1 1 1-2		literature sources and plant operating experience
95%,5%	3.6E-3/yr	3.3E-5/yr	Zion NPP PSS table 1 1 1-2		literature sources and plant operating experience
95%,5%	6.1E-2/yr	4 6E-3/yr	Zion NPP PSS table 1 1 1-2		literature sources and plant operating experience
n/a	n/a	n/a	Angra NPP PSA, Summary report		literature sources
n/a	n/a	n/a	Angra NPP PSA, Summary report		frequency based on specific study
n/a	n/a	n/a	Angra NPP PSA, Summary report		literature sources
n/a	n/a	n/a	Angra NPP PSA, Summary report		literature sources
n/a	n/a	n/a	Angra NPP PSA, Summary report		literature sources
n/a	n/a	n/a	Caorso NPP PSS table A-2		plant specific operating experience
n/a	n/a	n/a	Caorso NPP PSS table A-2		plant specific operating expereince
n/a	n/a	n/a	Caorso NPP PSS table A-2		plant specific operating experience
n/a	n/a	n/a	Caorso NPP PSS table A-2		study for the plant with similar valves (Alto Lazio)
n/a	n/a	n/a	Caorso NPP PSS table A-2		plant specific operating experience
n/a	n/a	n/a	Caorso NPP PSS table A-2		Historical data for north Italy

	PLANT TYPE	IE SET	IE CATEGORY	INITIATORS	FREQ DESC	FREQ
248	PWR	CC Initiator	loss of an AC bus	short to ground on AC bus	mean event/r.yr	3.5E-2/yr
249	PWR	CC Initiator	loss of a DC bus		mean event/r yr	1.8E-2/yr
250	PWR	CC Initiator	loss of service water	failure of normally open discharge MOV	mean event/r yr	2.6E-3/yr
251	PWR	CC Initiator	loss of a DC bus		mean event/r yr	9.0E-4/yr
252	PWR	CC Initiator	loss of an AC bus	short on 4160 V bus short on 480 V bus failure of 4160/480 V transformer	mean event/r yr	9.0E-3/yr
253	BWR	CC Initiator	loss of a DC bus		mean event/r yr	9.0E-4/yr
254	BWR	CC Initiator	loss of an AC bus		mean event/r.yr	9.0E-4/yr
255	PWR	CC Initiator	loss of service water		mean event/r.yr	4.0E-3/yr
256	PWR	CC Initiator	loss of power bus KI(ICS supply)		mean event/r.yr	2.0E-2/yr
257	PWR	CC Initiator	loss of an AC bus		mean event/r.yr	5.4E-3/yr
258	PWR	CC Initiator	loss of instrument air		mean event/r.yr.	1.7E-1/yr
259	PWR	CC Initiator	loss of 4 kV switchgear		mean event/r yr	5.4E-3/yr
260	BWR	CC Initiator	reactor water level instrumentation failure		point est. event/r.yr	3.6E-2/yr
261	BWR	CC Initiator	loss of an AC bus		point est. event/r yr	3.5E-2/yr
262	BWR	CC Initiator	loss of a DC bus		point est. event/r ye.	3.0E-3/yr
263	PWR & BWR	CC Initiator	Loss of offsite power >30 minute			4.0E-2/yr
264	PWR	CC Initiator	loss of service water		mean event/r yr	1.8E-3/yr
265	PWR	CC Initiator	failure of DC bus		mean event/r yr	3.6E-2/yr
266	BWR	CC initiator	loss of dedicated DC power		mean event/r yr.	3.2E-1/yr
267	PWR	CC Initiator	loss of service water system		mean event/r yr	2.4E-5/yr
268	PWR	CC Initiator	loss of service water		mean event/r yr.	9.4E-4/yr
269	PWR	CC Initiator	loss of component cooling		mean event/r yr	9.4E-4/yr
270	BWR	CC Initiator	loss of TBCCW system	Loss of cooling water to TBCCW system Loss of condensate pumps Loss of service water	mean event/r yr.	3.2E-2/yr
271	BWR	CC Initiator	loss of instrument air	Loss of instrument air Loss of RBCCW system	mean event/r.yr.	1.0E-3/yr
272	BWR	CC Initiator	loss of 4kV emergency bus		mean event/r.yr.	1.1E-4/yr

RANGE VAR	UPPER BND	LOWER BND	SOURCE	DLT SOURCE
n/a	n/a	n/a	IREP-ANO1 table 4-7	OCONEE PRA RSSMAP data base
n/a	n/a	n/a	IREP-ANO1 table 4-7	data taken from other PSA
n/a	n/a	n/a	IREP-ANO1 table 4-7	engineering evaluation using generic reliability data
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1..., table IV.3-1	Derived from generic component failure data
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1..., table IV.3-1	Derived from generic component failure data
n/a	n/a	n/a	NUREG/CR-4550, Vol.4, Peach Bottom..., table IV 3-1	nuclear sources and plant operating experience
n/a	n/a	n/a	NUREG/CR-4550, Vol.4, Peach Bottom..., table IV.3-1	nuclear sources and plant operating experience
n/a	n/a	n/a	Oconee PRA, table 5.9	system analysis using fault tree model
95%, 5%	7.5E-2/yr	7.5E-4/yr	Oconee PRA, table 5.9	analysis of plants systems
95%, 5%	1.3E-2/yr	2.7E-5/yr	Oconee PRA, table 5.9	plant operating experience
n/a	n/a	n/a	Oconee PRA, table 5.9	derived from system analysis using fault tree model
95%, 5%	1.3E-2/yr	2.7E-5/yr	Oconee PRA, table 5.9	system analysis-basic equipment fault
EF 3	n/a	n/a	Shoreham PRA, Appendix A.1	estimated from nuclear operating experience
EF 10	n/a	n/a	Shoreham PRA, Appendix A.1	nuclear operating experience and engineering evaluation
EF 3	n/a	n/a	Shoreham PRA, Appendix A.1	nuclear operating experience
			WASH-1400, Reactor Safety Study, Appendix I, Fig.I-4 11 & 4 12	
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.8	system evaluation using generic failure data
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.8	engineering evaluation and nuclear experience data
n/a	n/a	n/a	Oskarshamn 2 NPP	plant specific operating experience (6.2 years)
n/a	n/a	n/a	Ringhals 2 NPP	literature sources
95%, 5%	3.6E-3/yr	3.3E-5/yr	Zion NPP PSS table 1.1.1-2	literature sources (prior) and plant operating experience
95%, 5%	3.6E-3/yr	3.3E-5/yr	Zion NPP PSS table 1.1.1-2	literature sources (prior) and plant operating experience
n/a	n/a	n/a	Caorso NPP PSS table A-2	analysis using fault tree method
n/a	n/a	n/a	Caorso NPP PSS table A-2	analysis of instrument air system
n/a	n/a	n/a	Caorso NPP PSA table A-2	analysis of emergency power system by fault tree

	PLANT TYPE	IE SET	IE CATEGORY	INITIATORS	FREQ DESC	FREQ
1	PWR	LOCA	large LOCA	break with equ.area greater than 400 sq cm	mean event/r yr	2.7E-4/yr
2	PWR	LOCA	medium LOCA	break with equ area 80 to 400 sq cm	mean event/r yr	8.0E-4/yr
3	PWR	LOCA	small LOCA	break with equ.area 2 to 80 sq cm	mean event/r yr	2.7E-3/yr
4	PWR	LOCA	small leak on pressurizer	stuck open relief or safety valve	mean event/r yr	1.3E-3/yr
5	PWR	transient	loss of feedwater(FW)	component failures in FW system	mean event/r yr	8.0E-1/yr
6	PWR	transient	loss of off site power(LOOP)	loss of voltage on more than 1 power bus	mean event/r yr	1.0E-1/yr
7	PWR	LOCA	large LOCA	break with equ.size greater than 13.5 inches	mean event/r yr	7.5E-5/yr
8	PWR	LOCA	large LOCA	break with equ.size 10 to 13.5 inches	mean event/r yr	1.2E-5/yr
9	PWR	LOCA	medium LOCA	break with equ.size 4 to 10 inches	mean event/r yr	1 6E-4/yr
10	PWR	LOCA	small LOCA	break with equ.size 1 66 to 4 inches	mean event/r yr	3.8E-4/yr
11	PWR	LOCA	small LOCA	break with equ.size 1.2 to 1 66 inches stuck open prsrz safety valve	mean event/r yr	3.1E-4/yr
12	PWR	LOCA	small small LOCA	break with equivalent size .38 to 1.2 inches stuck open relief valve RCP seal failure	mean event/r yr	2.0E-2/yr
13	PWR	transient	loss of off site power		mean event/r yr	3.2E-1/yr
14	PWR	transient	loss of power conversion system (PCS)	Total loss of FW flow Full or partial closure of MSIV (1 loop) Closure of all MISV-s Increase in FW flow FW flow instability (operator error) FW flow instability (miscelaneous mechanical causes) Loss of condensate pumps (all) Loss of condenser vacuum (total) Opening of steam relief valves Loss of circulating water (CW)	mean event/r yr	1.0E+0/yr
15	PWR	transient	other than loss of PCS (turbine trip)	Loss of RCP flow (1 loop) Uncontrolled rod withdrawal CRDM problems and/or rod drop High or low prszr pressure Containment pressure problems Full or partial closure of MSIV Turbine trip, throttle valve closure, EHC problems Generator trip and generator caused problems Spurious auto trip Manual trip due to false signals	mean event/r yr	7 1E+0/yr
16	PWR	CC Initiator	loss of an AC bus	short to ground on AC bus	mean event/r yr	3 5E-2/yr
17	PWR	CC Initiator	loss of a DC bus		mean event/r yr	1.8E-2/yr
18	PWR	CC Initiator	loss of service water	failure of normally open discharge MOV	mean event/r yr	2.6E-3/yr
19	BWR	transient	isolation of reactor from condenser	Closure of all MSIV Turbine trip without bypass Loss of condenser	mean event/r yr	1 1E+0/yr
20	BWR	transient	turbine trip	Partial closure of MSIV Turbine trip with bypass Startup of idle recirc loop Pressure regulation failure Inadvertant opening of bypass Rod withdrawal Disturbance of FW Electric load rejection	mean event/r yr	4.0E+0/yr
21	BWR	transient	inadvertant open relief valve		mean event/r yr	6.0E-2/yr
22	BWR	transient	loss of feedwater	total loss of feedwater	mean event/r yr	7.0E-1/yr
23	BWR	transient	loss of off site power		mean event/r yr	5.3E-2/yr
24	BWR	LOCA	large LOCA	break with equ.size greater than 4 inches	mean event/r yr	4.0E-4/yr
25	BWR	LOCA	medium LOCA	break with equ.size 1 to 4 inches	mean event/r yr	2.0E-3/yr
26	BWR	LOCA	small LOCA	break with equ.size up to 1 inch	mean event/r yr	1 0E-2/yr
27	PWR	LOCA	large LOCA	break with equivalent size > 6 inches	mean event/r yr	5.0E-4/yr
28	BWR	LOCA	large LOCA	break with equivalent area > 1-.3 sq ft.	mean event/r yr	1.0E-4/yr
29	PWR	LOCA	medium LOCA	break with equivalent size 2 to 6 inches	mean event/r yr	1 0E-3/yr
30	BWR	LOCA	medium LOCA	break with equivalent area .004-.3 sqft(liq) 1-.3sqft(steam)	mean event/r yr	3.0E-4/yr
31	PWR	LOCA	small LOCA	break with equivalent size .5-2 inches	mean event/r yr	1.0E-3/yr
32	BWR	LOCA	small LOCA	break with equivalent area <.005 sqft(liq) or <.1 sqft(steam)	mean event/r yr	3.0E-3/yr
33	PWR	LOCA	small small LOCA	break with equ.size >.5 inch or flow 50-100 gpm	mean event/r yr	2.0E-2/yr
34	BWR	LOCA	small small LOCA	break with flow 50 to 100 gpm	mean event/r yr	3 0E-2/yr
35	PWR	transient	loss of PCS	Total loss of FW flow (listed in DESCR.field) FW flow instability-operator error,mechanical causes Full or partial closure of MSIV(one loop) Closure of all MSIV-s Loss of condensate pumps (all loops) Loss of condenser vacuum Condenser leakage SG leakage Opening of steam relief valve Miscelaneous leakages on secondary side Loss of CW Loss of CC Loss of SW	mean event/r yr	1 4E+0/yr

RANGE VAR	UPPER BND	LOWER BND	SOURCE	ULT SOURCE
EF 10	n/a	n/a	German Risk Study, table F1, 4-5	generic data sources
EF 10	n/a	n/a	German Risk Study, table F1, 4-5	generic data sources
EF 10	n/a	n/a	German Risk Study, table F1, 4-5.	generic data sources
EF 6	n/a	n/a	German Risk Study, table F1, 4-5.	operating experience and engineering analysis
EF 3	n/a	n/a	German Risk Study, table F1, 4-5.	german operating experience
EF 3	n/a	n/a	German Risk Study, table F1, 4-5	system analysis
n/a	n/a	n/a	IREP-ANO1 table 4-7	RSS data combined with ANO-1 break ranges
n/a	n/a	n/a	IREP-ANO1 table 4-7	RSS data combined with ANO-1 break ranges
n/a	n/a	n/a	IREP-ANO1 table 4-7	RSS data combined with ANO-1 break ranges
n/a	n/a	n/a	IREP-ANO1 table 4-7	RSS data combined with ANO-1 break ranges
n/a	n/a	n/a	IREP-ANO1 table 4-7	RSS data combined with ANO-1 break ranges
n/a	n/a	n/a	IREP-ANO1 table 4-7	RSS data combined with ANO-1 break ranges
n/a	n/a	n/a	IREP-ANO1 table 4-7	EPRI NP-801 and data from the utility
n/a	n/a	n/a	IREP-ANO1 table 4-7	EPRI NP-801 and plant specific info
n/a	n/a	n/a	IREP-ANO1 table 4-7	EPRI NP-801 and plant specific info
n/a	n/a	n/a	IREP-ANO1 table 4-7	OCONEE PRA RSSMAP data base
n/a	n/a	n/a	IREP-ANO1 table 4-7	data taken from other PSA
n/a	n/a	n/a	IREP-ANO1 table 4-7	engineering evaluation using generic reliability data
n/a	n/a	n/a	Limeric PRA, table A.1.3	BWR operating experience
n/a	n/a	n/a	Limeric PRA, table A.1.3	BWR operating experience
n/a	n/a	n/a	Limeric PRA, table A.1.3	BWR operating experience
n/a	n/a	n/a	Limeric PRA, table A.1.3	BWR operating experience
n/a	n/a	n/a	Limeric PRA, table A 6.2	regional grid data
n/a	n/a	n/a	Limeric PRA, table A.1.6	engineering evaluation and different data sources
n/a	n/a	n/a	Limeric PRA, table A.1 6	engineering evaluation and different data sources
n/a	n/a	n/a	Limeric PRA, table A.1.6	engineering evaluation and different data sources
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology., table VIII.1-1	average of past PRAs
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology., table VIII.1-1	RSS value
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology., table VIII.1-1	average of past PRAs
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology., table VIII.1-1	assesed from nuclear, industrial and other data sources
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology., table VIII.1-1	average of past PRAs
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology., table VIII.1-1	assesed form nuclear, industrial and other data sources
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology., table VIII.1-1	NPP operating experience
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology., table VIII 1-1	review of past PRAs
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology., table VIII 1 1	Based on NUREG/CR 3862

	PLANT TYPE	IE SET	IE CATEGORY	INITIATORS	FREQ DESC	FREQ
36	BWR	transient	loss of power conversion system	Electric load rejection with bypass failure Turbine trip with bypass failure Full MSIV closure Partial MSIV closure Loss of condenser vacuum Pressure regulator fails open Pressure regulator fails closed Turbine bypass fails open Turbine bypass fails close Turbine bypass or control valve increase pressure	mean event/r.yr.	1.6E+0/yr
37	PWR & BWR	transient	loss of a DC bus		mean event/r.yr.	5.0E-3/yr
38	PWR	transient	other than loss of PCS	Loss of RCS flow(one loop); Total loss of RCS flow Uncontroled rod withdrawal,CVCS problems-boron dilution CRDM mechanical problems or rod drop Leakage from: control rod, primary system, pressurizer Pressurizer pressure: low or high; Pressurizer spray failure Inadvertant SI signal Containment pressure problem Spurious trip; auto or manual trip(no transient conditions) Pressure-temperature-power imbalance rod position error Startup of inactive RCS loop Loss or reduction of FW flow(one loop) Increase in FW flow:one loop;all loops Loss of condensate pumps:one loop; all loops Turbine trip;generator trip;throttle valve closure;EHC problems	mean event/r.yr.	6.8E+0/yr
39	BWR	transient	loss of FW (steam side PCS available)	loss of all FW flow FW low flow	mean event/r.yr.	5.6E-1/yr
40	BWR	transient	other than loss of PCS	Turbine trip ;electric load rejection Recirculation control failure: decreasing flow; increasing flow Recirculation pump trip: one; all Startup of idle recirculation pump Recirculation pump seizure FW increasing flow; loss of FW heater Trip of one FW or condensate pump Rod withdrawal;inadvertant insertion of rods Inadvertant startup of HPCI/HPCS Spurious trip via instrumentation RPS fault SCRAM due to other plant occurances Manual SCRAM-no transient condition SCRAM cause unknown	mean event/r.yr.	4.8E+0/yr
41	PWR	transient	loss of off site power	loss of all offsite power loss of power to neccessary plant system	mean event/r.yr.	1.0E-1/yr
42	BWR	transient	loss of off site power	loss of off site power loss of auxiliary power (transformer)	mean event/r.yr.	1.0E-1/yr
43	BWR	transient	inadvertant opening of relief valve		mean event/r.yr.	1.4E-1/yr
44	PWR	LOCA	large LOCA	break with equivalent size 6 to 29 inches	mean event/r.yr.	5.0E-4/yr
45	PWR	LOCA	medium LOCA	break with equivalent size 2 to 6 inches	mean event/r.yr.	1.0E-3/yr
46	PWR	LOCA	small LOCA	break with equivalent size 1/2 to 2 inches	mean event/r.yr.	1.0E-3/yr
47	PWR	LOCA	small small LOCA	break with equivalent size <.5 inches	mean event/r.yr.	2.0E-2/yr
48	PWR	LOCA	intersystem LOCA		mean event/r.yr.	1.0E-6/yr
49	PWR	transient	turbine trip with MFW available	turbine trip; reactor trip loss of load MSIV closure loss of turbine control	mean event/r.yr.	7.3E+0/yr
50	PWR	transient	loss of main feedwater	failure of main FW high SG water level inadvertant SI signal	mean event/r.yr.	9.4E-1/yr
51	PWR	transient	loss of charging pump cooling	loss of charging pump CC loss of charging pump SW	mean event/r.yr.	3.0E-2/yr
52	PWR	transient	loss of off site power	failure of offsite power grid loss of station reserve power loss of power to the switchyard	mean event/r.yr.	7.0E-2/yr
53	PWR	CC Initiator	loss of a DC bus		mean event/r.yr.	9.0E-4/yr
54	PWR	CC Initiator	loss of an AC bus	short on 4160 V bus short on 480 V bus failure of 4160/480 V transformer	mean event/r.yr.	9.0E-3/yr
55	BWR	LOCA	large LOCA	break with equivalent area > 1 sqft	mean event/r.yr.	2.7E-4/yr
56	BWR	LOCA	medium LOCA	break with equivalent area .004 - 1 sqft	mean event/r.yr.	8.0E-4/yr
57	BWR	LOCA	small LOCA	break with equivalent area < .004 sqft(liq) or.01 sqft(steam)	mean event/r.yr.	2.7E-3/yr
58	BWR	LOCA	small small LOCA	leaks with 50-100 gpm flow (recirculation pump seal)	mean event/r.yr.	2.7E-2/yr
59	BWR	LOCA	intersystem LOCA		mean event/r.yr.	1.0E-8/yr
60	BWR	transient	loss of PCS		mean event/r.yr.	1.5E+0/yr
61	BWR	transient	other than loss of PCS	inadvertant open relief valve in primary system loss of FW but steam side of PCS initially available other initiators with same conseqences	mean event/r.yr.	2.6E+0/yr
62	BWR	transient	loss of off site power		mean event/r.yr.	7.0E-2/yr
63	BWR	CC Initiator	loss of a DC bus		mean event/r.yr.	9.0E-4/yr
64	BWR	CC Initiator	loss of an AC bus		mean event/r.yr.	[illegible]0E-4/yr

RANGE VAR	UPPER BND	LOWER BND	SOURCE		ULT SOURCE
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology...	table VIII.1-1	Based on NUREG/CR 3862
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology...	table VIII 1-1	DC power supply study, NUREG 0666
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology...	table VIII.1-1	Based on NUREG/CR 3862
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology ...	table VIII.1-1	Based on NUREG/CR 3862
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology ...	table VIII.1-1	
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology ...	table VIII 1-1	plant blackout analysis NUREG 1032
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology ...	table VIII.1-1	Plant blackout analysis NUREG 1032
EF 3	n/a	n/a	NUREG/CR 4550, Vol.1, Methodology ...	table VIII 1-1	Based on NUREG/CR 3862
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...	table IV.3-1	Based on NUREG/CR 4550 Vol. 1
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...	table IV.3-1	Based on NUREG/CR 4550 Vol. 1
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...	table IV.3-1	Based on NUREG/CR 4550 Vol. 1
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...	table IV.3-1	Based on NUREG/CR 4550 Vol. 1
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...	table IV.3-1	RSS data
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...	table IV.3-1	plant specific operating experience
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...	table IV.3-1	plant specific operating experience
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...	table IV.3-1	Comparison of NUREG/CR 3862 and operating experience
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...	table IV.3-1	NUREG 1032
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...	table IV 3-1	Derived from generic component failure data
n/a	n/a	n/a	NUREG/CR 4550, Vol.3, Surry Unit 1...	table IV.3-1	Derived from generic component failure data
n/a	n/a	n/a	NUREG/CR-4550, Vol.4, Peach Bottom...	table IV 3-1	past PRAs and other sources
n/a	n/a	n/a	NUREG/CR-4550, Vol.4, Peach Bottom...	table IV.3-1	past PRAs and other sources
n/a	n/a	n/a	NUREG/CR-4550, Vol.4, Peach Bottom...	table IV 3-1	past PRAs and other sources
n/a	n/a	n/a	NUREG/CR-4550, Vol.4, Peach Bottom...	table IV.3-1	past PRAs and other sources
n/a	n/a	n/a	NUREG/CR-4550, Vol 4, Peach Bottom...	table IV 3-1	analysis of the system interfaces using generic failure data
n/a	n/a	n/a	NUREG/CR-4550, Vol.4, Peach Bottom...	table IV 3-1	nuclear sources and plant operating experience
n/a	n/a	n/a	NUREG/CR-4550, Vol.4, Peach Bottom...	table IV 3-1	nuclear sources and plant operating experience
n/a	n/a	n/a	NUREG/CR-4550 Vol 4, Peach Bottom...	table IV 3-1	nuclear sources and plant operating experience
n/a	n/a	n/a	NUREG/CR-4550 Vol 4 Peach Bottom...	table IV 3-1	nuclear sources and plant operating experience
n/a	n/a	n/a	NUREG/CR-4550 Vol 4 Peach Bottom...	table IV 3-1	nuclear sources and plant operating experience

	PLANT TYPE	IE SET	IE CATEGORY	INITIATORS	FREQ DESC	FREQ
65	BWR	LOCA	large LOCA	break with equivalent size > 3 sq ft	mean event/r yr	3.0E-4/yr
66	BWR	LOCA	medium LOCA	break with equivalent area .005-.3 sqft(liq) and .1-.3sqft(steam)	mean event/r yr	9.0E-4/yr
67	BWR	LOCA	small LOCA	break with equivalent area <.005 sqft(liq) and <.1 sqft(steam)	mean event/r yr	3.0E-3/yr
68	BWR	LOCA	small small LOCA	leaks with 50 to 100 gpm flow (recirculation pump seal)	mean event/r yr	3.0E-2/yr
69	BWR	LOCA	intersystem LOCA		mean event/r yr	1.0E-8/yr
70	BWR	transient	loss of PCS		mean event/r yr	1.5E+0/yr
71	BWR	transient	loss of FW and steam side PCS available		mean event/r yr	7.0E-2/yr
72	BWR	transient	other than loss of PCS		mean event/r yr	5.4E+0/yr
73	BWR	transient	loss of off site power		mean event/r yr	7.0E-2/yr
74	BWR	transient	inadvertant opening of relief valve		mean event/r yr	1.4E-1/yr
75	PWR	LOCA	large LOCA	break with effective diameter > 4 inches	mean event/r yr	9.3E-4/yr
76	PWR	LOCA	reactor vessel rupture		mean event/r yr	1.1E-6/yr
77	PWR	LOCA	small LOCA	break with equivalent size 1/2 to 4 inches inadvertant PORV or safety valve opening RCP seal failure control rod drive seal leakage	mean event/r yr	3.0E-3/yr
78	PWR	LOCA	steam generator tube rupture	tube rupture with leak greater than 100 gpm	mean event/r yr	8.6E-3/yr
79	PWR	transient	reactor/turbine trip	Rod drop Inadvertant rod withdrawal Inadvertant boration or dilution Reactor trip Cold water addition Inadequate main FW Turbine and control valve malfunction Pressurizer spray failure Turbine trip Generator faults Grid distrurbances Administrative shutdowns	mean event/r yr	4.9E+0/yr
80	PWR	transient	spurious SI signal	HPI flow	mean event/r yr	1.0E-2/yr
81	PWR	transient	steam line break	steam line rupture turbine bypass valve inadvertant opening	mean event/r yr	3.0E-3/yr
82	PWR	transient	loss of condenser	loss of condenser vacuum loss of condenser circulating water	mean event/r yr	2.1E-1/yr
83	PWR	transient	FW line break (large)		mean event/r yr	9.3E-4/yr
84	PWR	transient	loss of main FW		mean event/r yr	6.4E-1/yr
85	PWR	transient	partial loss of main FW		mean event/r yr	6.9E-1/yr
86	PWR	transient	loss of off site power	failure of grid failure of feeders	mean event/r yr	4.0E-2/yr
87	PWR	transient	loss of off site power (substation faults)		mean event/r yr	1.3E-1/yr
88	PWR	transient	excessive feedwater		mean event/r yr	9.2E-2/yr
89	PWR	transient	spurious low pressurizer signal		mean event/r yr	4.4E-2/yr
90	PWR	CC Initiator	loss of service water		mean event/r yr	4.0E-3/yr
91	PWR	CC Initiator	loss of power bus KI(ICS supply)		mean event/r yr	2.0E-2/yr
92	PWR	CC Initiator	loss of an AC bus		mean event/r yr	5.4E-3/yr
93	PWR	CC Initiator	loss of instrument air		mean event/r yr	1.7E-1/yr
94	PWR	CC Initiator	loss of 4 kV switchgear		mean event/r yr	5.4E-3/yr
95	PWR	LOCA	large LOCA	break with equivalent size >6 inches	mean event/r yr	4.7E-5/yr
96	PWR	LOCA	medium LOCA	break with equivalent size 2-6 inches	mean event/r yr	9.8E-4/yr
97	PWR	LOCA	small LOCA	break with equivalent size .5-2 inches	mean event/r yr	1.8E-3/yr
98	PWR	LOCA	intersystem LOCA		mean event/r yr	4.6E-6/yr
99	PWR	LOCA	reactor vessel rupture		mean event/r.yr.	1.0E-7/yr
100	PWR	transient	loss of off site power		mean event/r yr	2.0E-1/yr
101	PWR	transient	loss of main FW		mean event/r.yr.	3.0E+0/yr
102	PWR	transient	transient with main FW available		mean event/r yr	4.0E+0/yr
103	BWR	LOCA	large LOCA	break with equivalent size >4 inches	point est. event/r yr	7.0E-4/yr
104	BWR	LOCA	medium LOCA	break with equivalent size 1-4 inches	point est. event/r yr	3.0E-3/yr
105	BWR	LOCA	small LOCA	break with equivalent size <1 inch	point est. event/r yr	8.0E-3/yr
106	BWR	LOCA	reactor vessel rupture		point est. event/r yr	3.0E-7/yr
107	BWR	LOCA	intersystem LOCA		point est. event/r yr	1.2E-7/yr
108	BWR	transient	loss of condenser	loss of normal condenser vacuum turbine trip with bypass valve failure electric load rejection with bypass valve failure	mean event/r.yr	4.1E-1/yr
109	BWR	transient	MSIV closure		mean event/r yr	2.4E-1/yr
110	BWR	transient	loss of feedwater		mean event/r yr	1.8E-1/yr
111	BWR	transient	loss of off site power		mean event/r yr	8.0E-2/yr
112	BWR	transient	inadvertant opening of relief valve		mean event/r yr	9.0E-2/yr
113	BWR	transient	control rod withdrawal		mean event/r yr	3.0E-2/yr
114	BWR	transient	turbine trip	Electric load rejection Turbine trip Inadvertant closure of one MSIV, partial MSIV closure Pressure regulator fails: open; closed Turbine bypass fails-open Turbine bypass or control valve cause increase pressure Recirculation flow failure.decreasing;increasing Trip of recirculation pump: one; all Abnormal startup of idle recirculation pump Recirculation pump seizure Loss of FW heater Trip of one FW or condensate pump	mean event/r yr	4.5E+0/yr
115	BWR	transient	main steam line break		point est. event/r yr	3.8E-8/yr
116	BWR	transient	feedwater line break		point est. event/r yr	8.7E-9/yr

RANGE VAR	UPPER BND	LOWER BND	SOURCE	[illegible]	ULT SOURCE
n/a	n/a	n/a	NUREG/CR-4550, Vol 6, Grand Gulf 1...	ble IV 3-1	
n/a	n/a	n/a	NUREG/CR-4550 Vol 6, Grand Gulf 1	ble IV 3-1	
n/a	n/a	n/a	NUREG/CR-4550, Vol 6, Grand Gulf 1	ble IV 3-1	
n/a	n/a	n/a	NUREG/CR-4550, Vol 6, Grand Gulf 1...	ble IV 3 1	
n/a	n/a	n/a	NUREG/CR-4550, Vol 6, Grand Gulf 1...	ble IV 3-1	Analysis of the system interfaces using generic failure data
n/a	n/a	n/a	NUREG/CR-4550, Vol.6, Grand Gulf 1...	ble IV.3-1	different nuclear sources and plant operating experience
n/a	n/a	n/a	NUREG/CR-4550, Vol 6, Grand Gulf 1...	ble IV.3-1	different nuclear sources and plant operating experience
n/a	n/a	n/a	NUREG/CR-4550, Vol.6, Grand Gulf 1...	ble IV 3-1	different nuclear sources and plant operating experience
n/a	n/a	n/a	NUREG/CR-4550, Vol.6, Grand Gulf 1...	ble IV 3-1	assesment of different nuclear sources
n/a	n/a	n/a	NUREG/CR-4550, Vol.6, Grand Gulf 1...	ble IV.3-1	different nuclear sources and plant operating experience
95% 5%	2.8E-3/yr	9 4E-7/yr	Oconee PRA, table 5 9		generic prior updated with plant specific operating experience
95%, 5%	4 1E-6/yr	6.0E-8/yr	Oconee PRA, table 5 9		various sources
95%, 5%	1.2E-2/yr	1 0E-6/yr	Oconee PRA, table 5 9		update of generic prior
95% 5%	2.7E-2/yr	2.6E-5/yr	Oconee PRA table 5 9		generic data updated with plant specific operating experience
95%, 5%	5.7E+0/yr	4 1E+0/yr	Oconee PRA, table 5.9		generic data updated with plant specific operating experience
95% 5%	4.3E-2/yr	7.8E-6/yr	Oconee PRA, table 5 9		generic data updated with plant specific operating experience
95%, 5%	1.2E-2/yr	1.0E-6/yr	Oconee PRA, table 5 9		generic data updated with plant specific operating experience
95%, 5%	3.8E+0/yr	8.3E-2/yr	Oconee PRA, table 5.9		generic data updated with plant specific operating experience
	3 8E-1/yr		BNL Review, table 4.3		recalculation made by the reviewers
95%, 5%	2.8E-3/yr	6.9E-7/yr	Oconee PRA, table 5 9		generic data updated with plant specific operating experience
95%, 5%	9.2E-1/yr	3 6E-1/yr	Oconee PRA, table 5 9		generic data updated with plant specific operating experience
95%, 5%	9 7E-1/yr	4.0E-1/yr	Oconee PRA, table 5 9		generic data updated with plant specific operating experience
95%, 5%	9 7E-2/yr	7 1E-3/yr	Oconee PRA, table 5 9		generic data updated with plant specific operating experience
95%, 5%	3 0E-1/yr	2.2E-2/yr	Oconee PRA, table 5 9		generic data updated with plant specific operating experience
95%, 5%	2.1E-1/yr	1.8E-2/yr	Oconee PRA, table 5 9		generic data updated with plant specific operating exeperience
95%, 5%	1 7E-1/yr	1 7E-3/yr	Oconee PRA, table 5 9		system analysis
n/a	n/a	n/a	Oconee PRA, table 5 9		system analysis using fault tree model
95%, 5%	7.5E-2/yr	7.5E-4/yr	Oconee PRA, table 5 9		analysis of plants systems
95% 5%	1.3E-2/yr	2.7E-5/yr	Oconee PRA, table 5 9		plant operating experience
n/a	n/a	n/a	Oconee PRA, table 5 9		derived from system analysis using fault tree model
95%, 5%	1.3E-2/yr	2.7E-5/yr	Oconee PRA, table 5 9		system analysis-basic equipment fault
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4		engineering evaluation using generic data
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4		engineering evaluation using generic data
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4		engineering evaluation using generic data
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4		engineering evaluation using generic data
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4		generic data
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4		
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4		
n/a	n/a	n/a	Sequoyah NPP RSSMAP, table 7-4		
EF 10	n/a	n/a	Shoreham PRA, Appendix A.1		generic data based on collection of operating experiences
EF 3	n/a	n/a	Shoreham PRA, Appendix A.1		generic data based on actual reactor operating experience
EF 3	n/a	n/a	Shoreham PRA, Appendix A.1		generic data based on reactor operating experience
EF 10	n/a	n/a	Shoreham PRA, Appendix A 1		generic data sources
n/a	n/a	n/a	Shoreham PRA, Appendix A.1		system evaluation
EF 3	n/a	n/a	Shoreham PRA, Appendix A, table A.1-		generic data based on nuclear operating expeirence taken from EPRI NP-801
EF 3	n/a	n/a	Shoreham PRA, Appendix A, table A.1		generic data based on nuclear operating expeirence etc.
EF 3	n/a	n/a	Shoreham PRA, Appendix A, table A 1-		generic data based on nuclear operating expeirence etc.
EF 3	n/a	n/a	Shoreham PRA, Appendix A, table A.1-		utility specific data
EF 3	n/a	n/a	Shoreham PRA, Appendix A, table A.1		generic data and engineering evaluation
n/a	n/a	n/a	Shoreham PRA, Appendix A, table A.1-		generic data based on nuclear operating expeirence etc.
EF 3	n/a	n/a	Shoreham PRA, Appendix A, table A.1		generic data based on nuclear operating expeirence etc.
n/a	n/a	n/a	Shoreham PRA, Appendix A 1		engineering evaluation
n/a	n/a	n/a	Shoreham PRA, Appendix A 1		engineering evaluation

	PLANT TYPE	IE SET	IE CATEGORY	INITIATORS	FREQ DESC	FREQ
117	BWR	transient	HPCI/RCIC line breaks		point est. event/r yr	1 4E-8/yr
118	BWR	CC Initiator	reactor water level instrumentation failure		point est. event/r yr	3 6E-2/yr
119	BWR	CC Initiator	loss of an AC bus		point est. event/r yr	3 5E-2/yr
120	BWR	CC Initiator	loss of a DC bus		point est. event/r ye.	3 0E-3/yr
121	PWR & BWR	LOCA	reactor vessel rupture		median, event/reactor yr	1 0E-7/yr
122	PWR & BWR	LOCA	large LOCA	break with equivalent size > 6 inches	median, event/reactor yr	1 0E-4/yr
123	BWR	LOCA	medium LOCA	break with equivalent size 2.5-8.5 in (liq) and 4 7-6 in (steam)	median, events/reactor yr	3 0E-4/yr
124	PWR	LOCA	medium LOCA	break with equivalent size 2-6 inches	median, events/reactor yr	3 0E-4/yr
125	BWR	LOCA	small LOCA	break with equivalent size 0.6-2.6 in (liq) and 1.0-4 7in (steam)	median, events/reactor yr	1 0E-3/yr
126	PWR	LOCA	small LOCA	break with equivalent size 1/2-2 inches	median, events/reactor yr	1 0E-3/yr
127	PWR	LOCA	intersystem LOCA		median, event/reactor yr	4 0E-6/yr
128	BWR	transient	rapid shutdown	Rod withdrawal at power	median, event/reactor yr	1 0E+1/yr
				FW controler failure; loss of FW flow		
				Recirculation flow control failure:decreasing;increasing		
				Startup of idle recirculation pump		
				Loss of FW heating		
				Inadvertant HPCI start		
				Loss of auxiliary power		
				Turbine trip(turbine valve closure);load rejection(stop valve cl)		
				MSIV closure		
				Recirculation pump trip (one pump)		
				Recirculation pump seizure:one pump; two pumps		
				T-G pressure regulator failure-rapid opening		
				Rod ejection, rod drop accident		
				Startup of idle recirc.pump with simultaneous turbine trip		
129	PWR	transient	rapid shutdown	Turbine trip; loss of main generator; LOOP	median, events/reactor yr	1 0E+1/yr
				Loss of condenser vacuum		
				Inadvertant MSIV closure		
				Loss of main FW, loss of main CW, loss of condensate pumps		
				Inadvertant openig of SG PORV-s		
				Increase in MFW flow		
				Opening of all bypass valves(steam dump)		
				Uncontroled rod withdrawal; control assembly drop		
				Boron dilution (CVCS malfunction)		
				Startup of inactive RCS loop		
				Opening of pressurizer SRV or RV		
				Loss of RCS coolant flow; siezure of all RCP-s		
				Rupture of FW piping or main steam lines; rupture of SG		
				Rupture of CRDM housing		
130	PWR & BWR	CC Initiator	Loss of offsite power >30 minute			4.0E-2/yr
131	PWR	LOCA	large LOCA	break with equivalent size >4.3 inches	mean event/r yr	2.3E-4/yr
132	PWR	LOCA	medium LOCA	break with equivalent size 1 9 to 4.3 inches	mean event/r yr	2.4E-4/yr
133	PWR	LOCA	small LOCA	break with equivalent size .3 to 1.9 inches	mean event/r yr	2.1E-2/yr
				reactor coolant pump seal rupture		
134	PWR	transient	loss of off site power		mean event/r yr	1 4E-1/yr
135	PWR	transient	total interuption of PCS	Total loss of FW flow	mean event/r yr	8.0E-1/yr
				Closure of all MSIVs		
				FW flow instability		
				Loss of all condensate pumps		
				Loss of condenser vacuum		
				Loss of CW		
136	PWR	transient	transient requiring RCS pressure relief	turbine trip or throttle valve closure	mean event/r yr	1 9E+0/yr
				generator trip or generator caused faults		
				loss of power to neccessary plant systems		
137	PWR	transient	transient which do not affect front line systems	Loss of RCS flow (one loop); total loss of RCS flow	mean event/r yr	6.8E+0/yr
				CRDM problems, rod drop		
				Pressurizer pressure: low;high		
				Inadvertant SI signal		
				CVCS malfunction- boron dilution		
				Pressure/temprerature/power imbalance		
				Loss or reduction in main FW (1 loop)		
				Full or partial closure of one MSIV		
				Increase in FW flow: one loop; all loops		
				Loss of condensate pump		
				Condenser leakage; Lekage in secondary system		
				Sudden opening of relief valve		
				Pressurizer spray failure		
				Spurious trip; Manual trip;auto trip-No transient conditions		
138	PWR	CC Initiator	loss of service water		mean event/r yr	1 8E-3/yr
139	PWR	CC Initiator	failure of DC bus		mean event/r yr	3 6E-2/yr
140	BWR	LOCA	large LOCA (1)	break with equivalent area .3 to 4.3 sq ft liquid suction side	mean event/r yr	9 9E-6/yr
141	BWR	LOCA	large LOCA (2)	break with equivalent area .5 to 4.3 sq ft liquid discharge side	mean event/r yr	3 9E-5/yr
142	BWR	LOCA	large LOCA (3)	break with equivalent area 1 4 to 4 1 sq ft steam	mean event/r yr	5.2E-5/yr
143	BWR	LOCA	medium LOCA (1)	break with equivalent size .12 to .3 sq ft liquid	mean event/r yr	9 0E-5/yr
144	BWR	LOCA	medium LOCA (2)	break with equivalent area .12 to 1 4 sq ft steam	mean event/r yr	2.1E-4/yr
145	BWR	LOCA	small LOCA	break with equivalent area < .12 sq ft	mean event/r yr	0E-3/yr

RANGE VAR	UPPER BND	LOWER BND	SOURCE		ULT SOURCE
n/a	n/a	n/a	Shoreham PRA, Appendix A 1		engineering evaluation
EF 3	n/a	n/a	Shoreham PRA, Appendix A 1		estimated from nuclear operating experience
EF 10	n/a	n/a	Shoreham PRA, Appendix A 1		nuclear operating experience and engineering evaluation
EF 3	n/a	n/a	Shoreham PRA, Appendix A 1		nuclear operating experience
90% lognormal distr	1 0E-6/yr	1 0E-8/yr	WASH 1400 Reactor Safety Study, Append	x V, chapter 4.5	based on non-nuclear experience
90% lognormal distr	1 0E-3/yr	1 0E-5/yr	WASH-1400, Reactor Safety Study, Appenc	x III, table III 6-9	based on number of nuclear,industrial and other data sources
90% lognormal dist.	3.0E-3/yr	3.0E-5/yr	WASH-1400, Reactor Safety Study, Append	x III, table III 6-9	assesment based on nuclear, industrial and other data sources
90% lognormal distr	3.0E-3/yr	3.0E-5/yr	WASH-1400, Reactor Safety Study, Append	x III, table III 6-9	assesment based on nuclear industrial and other data sources
90% lognormal distr.	1.0E-2/yr	1 0E-4/yr	WASH 1400, Reactor Safety Study, Appe id	x III, table III 6-9	assessment based on nuclear, industrial and other data sources
90% lognormal distr	1 0E-2/yr	1 0E-4/yr	WASH-1400, Reactor Safety Study, Appe id	x III, table III 6-9	assesed based on nuclear industrial and other data sources
EF 10			WASH-1400, Reactor Safety Study, Appe id	x V, chapter 4 4	analysis of the system interface using generic failure rates
EF 2			WASH-1400, Reactor Safety Study, Appe id & Appendix I, Table I - 4 12	x V, chapter 4.3	NPP operating experience, engineering estimate
EF 2			WASH-1400, Reactor Safety Study, Append & Appendix I, Table I - 4 9	x V, chapter 4.3	NPP operating experience, engineering estimate
			WASH-1400, Reactor Safety Study, Append	x I, Fig.I-4 11 & 4 12	
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.2		generic data
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.2		generic data
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.2		generic data
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.8		EPRI NP-2230
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.8		EPRI NP-2230
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.8		EPRI NP-2230
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.8		EPRI NP-2230
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.8		system evaluation using generic failure data
n/a	n/a	n/a	Calvert Cliffs Unit 1 IREP, table 4.8		engineering evaluation and nuclear experience data
n/a	n/a	n/a	Browns Ferry Unit 1 IREP, table 5		generic data
n/a	n/a	n/a	Browns Ferry Unit 1 IREP, table 5		generic data
n/a	n/a	n/a	Browns Ferry Unit 1 IREP, table 5		generic data
n/a	n/a	n/a	Browns Ferry Unit 1 IREP, table 5		generic data
n/a	n/a	n/a	Browns Ferry Unit 1 IREP, table 5		generic data
n/a	n/a	n/a	Browns Ferry Unit 1 IREP, table 5		generic data

	PLANT TYPE	IE SET	IE CATEGORY	INITIATORS	FREQ DESC	FREQ
146	BWR	transient	loss of PCS	MSIV closure Loss of normal condenser vacuum Pressure regulator fails open Loss of feedwater flow Loss of off site power Loss of auxiliary power Increased flow at power	mean event/r yr	1 7E+0/yr
147	BWR	transient	other than loss of PCS	Electric load rejection Electric load rejection with bypass failure Turbine trip Turbine trip with bypass failure Inadvertant opening of MSIV Pressure regualtor fails closed Bypass/control valve causing pressure increase Recirculation control fails causing increased flow	mean event/r.yr	1.7E+0/yr
148	PWR	LOCA	large LOCA		mean events/r.yr	2.0E-4/yr
149	PWR	LOCA	medium LOCA		mean events/r yr	4 6E-4/yr
150	PWR	LOCA	small LOCA (nonisolable)		mean events/r.yr	4 0E-3/yr
151	PWR	LOCA	Small LOCA (isolable)		mean events/r yr	2.3E-2/yr
152	PWR	LOCA	steam generator tube rupture		mean events/r yr	8.2E-3/yr
153	PWR	transient	turbine trip		mean events/r.yr.	1 0E-0/yr
154	PWR	transient	loss of main FW (total loss)		mean events/r yr	3.3E-1/yr
155	PWR	transient	loss of main FW (partial)		mean events/r.yr.	4.9E-1/yr
156	PWR	transient	excessive FW flow		mean events/r.yr	3 1E-1/yr
157	PWR	transient	loss of condenser vacuum		mean events/r.yr.	8.1E-2/yr
158	PWR	transient	closure of one MSIV		mean events/r yr	7.2E-2/yr
159	PWR	transient	closure of all MSIVs		mean events/r.yr.	2.0E-3/yr
160	PWR	transient	core power excursion		mean events/r.yr	1.5E-2/yr
161	PWR	transient	steam line break (inside containment)		mean events/r.yr.	4.6E-4/yr
162	PWR	transient	steam line break (outside containment)		mean events/r.yr	4.2E-3/yr
163	PWR	transient	main steam relief valve openings		mean events/r.yr	1.6E-2/yr
164	PWR	transient	inadvertant SI		mean events/r yr	6.3E-2/yr
165	PWR	transient	total loss of reactor coolant flow		mean events/r.yr	6.9E-2/yr
166	PWR	transient	loss of off-site power		mean events/r yr	4 9E-2/yr
167	BWR	LOCA	large LOCA	Break with equivalent flow >2000 kg/s	mean event/r yr	3 0E-4/yr
168	BWR	LOCA	medium LOCA	Break with equivalent flow 30 to 2000 kg/s	mean event/r yr	9 0E-4/yr
169	BWR	LOCA	small LOCA	break with equivalent flow 10 to 30 kg/s	mean event/r yr	3 0E-3/yr
170	BWR	transient	transient with front line systems available		mean event/r.yr.	2.0E+0/yr
171	BWR	transient	loss of condenser		mean event/r yr	3 3E-1/yr
172	BWR	transient	loss of FW (total loss)		mean event/r.yr.	1 6E-1/yr
173	BWR	transient	loss of FW and condenser		mean event/r.yr	3 3E-1/yr
174	BWR	transient	loss of off-site power		mean event/r.yr.	1.6E-1/yr
175	BWR	LOCA	reactor vessel rupture		mean event/r yr	2.7E-7/yr
176	BWR	LOCA	large LOCA	break with equivalent area >450 sq cm	mean event/r.yr	1 0E-4/yr
177	BWR	LOCA	medium LOCA	break with equivalent area 80 to 450 sq cm	mean event/r yr	3 0E-4/yr
178	BWR	LOCA	small LOCA	break with equivalent area < 80 sq cm	mean event/r.yr.	1 0E-3/yr
179	BWR	LOCA	large LOCA	break with equivalent flow 600 to 2000 kg/s	mean event/r yr	1.0E-7/yr
180	BWR	LOCA	medium LOCA	break with equivalent flow 35 to 600 kg/s	mean event/r.yr.	1.1E-5/yr
181	BWR	LOCA	small LOCA	break with equivalent flow <35 kg/s	mean event/r yr	1 1E-2/yr
182	BWR	LOCA	Intersystem LOCA		mean event/r.yr.	1 0E-7/yr
183	BWR	transient	transient with front line systems available		mean event/r.yr	1.7E-1/yr
184	BWR	transient	loss of condenser		mean event/r.yr.	2.3E+0/yr
185	BWR	transient	loss of FW		mean event/r.yr	1 7E-1/yr
186	BWR	transient	loss of FW and condenser		mean event/r.yr	1 7E-1/yr
187	BWR	transient	loss of off-site power		mean event/r.yr.	1 8E-1/yr
188	BWR	LOCA	large LOCA	break with equivalent flow > 2000 kg/s	mean event/r.yr.	1.0E-7/yr
189	BWR	LOCA	medium LOCA	break with equivalent flow 30 to 2000 kg/s	mean event/r.yr.	1 0E-6/yr
190	BWR	LOCA	small LOCA	break with equivalent flow 10 to 30 kg/s	mean event/r.yr.	1.0E-2/yr
191	BWR	transient	transient with front line systems available		mean event/r yr.	1.9E+0/yr
192	BWR	transient	loss of condenser		mean event/r.yr.	3.2E-1/yr
193	BWR	transient	loss of FW and condenser		mean event/r yr	3.2E-1/yr
194	BWR	transient	loss of off-site power		mean event/r.yr	3.2E-1/yr
195	BWR	CC Initiator	loss of dedicated DC power		mean event/r.yr	3.2E-1/yr
196	BWR	LOCA	large LOCA	break with equivalent flow >3000 kg/s	mean event/r.yr	1 0E-4/yr
197	BWR	LOCA	medium LOCA		mean event/r yr	5 0E-4/yr
198	BWR	LOCA	small LOCA		mean event/r.yr	1 0E-3/yr
199	BWR	transient	transient with front line systems available		mean event/r.yr	1.6E+0/yr
200	BWR	transient	loss of condensor		mean event/r.yr	7 8E-1/yr
201	BWR	transient	loss of FW		mean event/r yr.	7 8E-1/yr
202	BWR	transient	loss of off-site power		mean event/r.yr.	2.0E-2/yr
203	BWR	transient	transient with front line systems available		mean event/r yr	9 8E-1/yr
204	BWR	transient	loss of condenser		mean event/r.yr	1 6E-1/yr
205	BWR	transient	loss of FW and condenser		mean event/r.yr.	9.8E-1/yr
206	BWR	transient	loss of off site power		mean event/r.yr	1 6E-1/yr
207	BWR	LOCA	large LOCA	break with equivalent flow >1200 kg/s	mean event/r.yr.	3 0E-4/yr
208	BWR	LOCA	medium LOCA	break with equivalent flow 35 to 1200 kg/s	mean event/r yr	9 0E-4/yr
209	BWR	LOCA	small LOCA	break with equivalent flow 5 to 35 kg/s	mean event/r yr	3 0E-3/yr
210	BWR	LOCA	Intersystem LOCA		mean event/r yr	1 9E-7/yr
211	BWR	LOCA	pressure vessel rupture		mean event/r yr	2.7E-7/yr

RANGE VAR	UPPER BND	LOWER BND	SOURCE	UET SOURCE
n/a	n/a	n/a	Browns Ferry Unit 1 IREP table 6	plant specific data
n/a	n/a	n/a	Browns Ferry Unit 1 IREP, table 6	plant specific data
95% 5% of distribution	5.2E-4/yr	7 6E-6/yr	Old PWR	Generic data updated with plant specific operating experience
95% 5% of distribution	1.2E-3/yr	2.3E-5/yr	Old PWR	Generic data updated with plant specific operating experience
95% 5% of distribution	1 4E-2/yr	1.2E-4/yr	Old PWR	Generic data updated with plant specific operating experience
95%, 5% of distribution	5 0E-2/yr	3.3E-3/yr	Old PWR	Generic data updated with plant specific operating experience
95%, 5% of distribution	2.1E-2/yr	3.1E-4/yr	Old PWR	Generic data updated with plant specific operating experience
95%, 5% of distribution	1 4E-0/yr	7 1E-1/yr	Old PWR	Generic data updated with plant specific operating experience
95%, 5% of distribution	4.9E-1/yr	1 7E-1/yr	Old PWR	Generic data updated with plant specific operating experience
95%, 5% of distribution	6.5E-1/yr	2.9E-1/yr	Old PWR	Generic data updated with plant specific operating experience
95%, 5% of distribution	4 9E-1/yr	1 4E-1/yr	Old PWR	Generic data updated with plant specific operating experience
95% 5% of distribution	1.5E-1/yr	1.8E-2/yr	Old PWR	Generic data updated with plant specific operating experience
95%, 5% of distribution	1.5E-1/yr	1 8E-2/yr	Old PWR	Generic data updated with plant specific operating experience
95%, 5% of distribution	5 7E-3/yr	3 9E-5/yr	Old PWR	Generic data updated with plant specific operating experience
95%, 5% of distribution	3.6E-2/yr	6.7E-4/yr	Old PWR	Generic data updated with plant specific operating experience
95%, 5% of distribution	1.2E-3/yr	2.3E-5/yr	Old PWR	Generic data updated with plant specific operating experience
95%, 5% of distribution	1.3E-2/yr	1 1E-4/yr	Old PWR	Generic data updated with plant specific operating experience
95% 5% of distribution	4.4E-2/yr	5.7E-4/yr	Old PWR	Generic data updated with plant specific operating experience
95%, 5% of distribution	1.2E-1/yr	1.4E-2/yr	Old PWR	Generic data updated with plant specific operating experience
95%, 5% of distribution	1.5E-1/yr	1.8E-2/yr	Old PWR	Generic data updated with plant specific operating experience
95%, 5% of distribution	1 1E-1/yr	7 1E-3/yr	Old PWR	Generic data updated with plant specific operating experience
n/a	n/a	n/a	Barseback 1 & 2 NPP	generic sources
n/a	n/a	n/a	Barseback 1 & 2 NPP	literature sources
n/a	n/a	n/a	Barseback 1 & 2 NPP	literature data
n/a	n/a	n/a	Barseback 1 NPP	plant operating experience (8 11 years)
n/a	n/a	n/a	Barseback 1 NPP	plant specific operating experience (8 11 years)
n/a	n/a	n/a	Barseback 1 NPP	plant specific operating experience (8 11 years)
n/a	n/a	n/a	Barseback 1 NPP	plant specific operating experience (8 11 years)
n/a	n/a	n/a	Barseback 1 NPP	plant specific operating experience
n/a	n/a	n/a	Forsmark 3 NPP	literature sources
n/a	n/a	n/a	Forsmark 3 NPP	literature sources
n/a	n/a	n/a	Forsmark 3 NPP	literature sources
n/a	n/a	n/a	Forsmark 3 NPP	literature sources
n/a	n/a	n/a	Oskarshamn 1 NPP	litearure sources (application of LBB criteria)
n/a	n/a	n/a	Oskarshamn 1 NPP	literature sources
n/a	n/a	n/a	Oskarshamn 1 NPP	litearture sources
n/a	n/a	n/a	Oskarshamn 1 NPP	literature sources
n/a	n/a	n/a	Oskarshamn 1 NPP	plant operating experience (5 7 years)
n/a	n/a	n/a	Oskarshamn 1 NPP	plant operating experience (5.7 years)
n/a	n/a	n/a	Oskarshamn 1 NPP	plant specific operating experience (5 7 years)
n/a	n/a	n/a	Oskarshamn 1 NPP	plant specific operating experience(5 7 years)
n/a	n/a	n/a	Oskarshamn 1 NPP	plant specific operating experience (5 7 years)
n/a	n/a	n/a	Oskarshamn 2 NPP	literature sources
n/a	n/a	n/a	Oskarshamn 2 NPP	literature sources
n/a	n/a	n/a	Oskarshamn 2 NPP	literature sources
n/a	n/a	n/a	Oskarshamn 2 NPP	plant operating experience (6.2 years)
n/a	n/a	n/a	Oskarshamn 2 NPP	plant specific operating experience (6.2 years)
n/a	n/a	n/a	Oskarshamn 2 NPP	plant specific operating experience (6.2 years)
n/a	n/a	n/a	Oskarshamn 2 NPP	plant specific experience
n/a	n/a	n/a	Oskarshamn 2 NPP	plant specific operating experience (6.2 years)
n/a	n/a	n/a	Oskarshamn 3 NPP	literature sources
n/a	n/a	n/a	Oskarshamn 3 NPP	literature sources
n/a	n/a	n/a	Oskarshamn 3 NPP	literature sources
n/a	n/a	n/a	Oskarshamn 3 NPP	plant specific operating experience (6 4 years)
n/a	n/a	n/a	Oskarshamn 3 NPP	plant specific operating experience (6.4 years)
n/a	n/a	n/a	Oskarshamn 3 NPP	plant specific operating experience (6 4 years)
n/a	n/a	n/a	Oskarshamn 3 NPP	literature sources
n/a	n/a	n/a	Barseback 2 NPP	plant specific operating experience
n/a	n/a	n/a	Barseback 2 NPP	plant specific operating experience
n/a	n/a	n/a	Barseback 2 NPP	plant specific operating experience
n/a	n/a	n/a	Barseback 2 NPP	plant specific operating experience
n/a	n/a	n/a	Ringhals 1 NPP	litearture sources
n/a	n/a	n/a	Ringhals 1 NPP	literature sources
n/a	n/a	n/a	Ringhals 1 NPP	literature sources
n/a	n/a	n/a	Ringhals 1 NPP	literature sources
n/a	n/a	n/a	Ringhals 1 NPP	literature data

	PLANT TYPE	IE SET	IE CATEGORY	INITIATORS	FREQ DESC.		FREQ
212	BWR	transient	transient with front line systems available		mean	event/r.yr	9.2E-1/yr
213	BWR	transient	loss of condenser		mean	event/r.yr	2.0E+0/yr
214	BWR	transient	loss of FW & condenser		mean	event/r.yr	3.7E-1/yr
215	BWR	transient	loss of off site power		mean	event/r.yr	1.8E-1/yr
216	PWR	LOCA	large LOCA	break with equivalent diameter greater than 15 cm	mean	event/r.yr	4.0E-4/yr
217	PWR	LOCA	medium LOCA	break with equivalent diameter between 5 nad 15 cm	mean	event/r.yr	8.1E-4/yr
218	PWR	LOCA	small LOCA		mean	event/r.yr	1.1E-2/yr
219	PWR	LOCA	reactor vessel rupture		mean	event/r yr.	2.7E-7/yr
220	PWR	LOCA	Intersystem LOCA		mean	event/r.yr.	4.2E-8/yr
221	PWR	LOCA	steam generator tube rupture		mean	event/r.yr.	9.7E-3/yr
222	PWR	transient	transient with PCS isolation		mean	event/r.yr.	1.3E-1/yr
223	PWR	transient	transient with front line systems available		mean	event/r.yr	6.0E+0/yr
224	PWR	transient	loss of off site power		mean	event/r.yr.	7.0E-1/yr
225	PWR	transient	transient requiring RCS pressure relief		mean	event/r.yr	4.0E-1/yr
226	PWR	transient	inadvertant safety injection		mean	event/r.yr.	1.2E+0/yr
227	PWR	transient	steam line break		mean	event/r.yr.	4.4E-4/yr
228	PWR	transient	transient occurring after shutdown		mean	event/r.yr	1.0E+0/yr
229	PWR	CC Initiator	loss of service water system		mean	event/r.yr.	2.4E-5/yr
230	PWR	LOCA	large LOCA	Break with equivalent diameter greater than 6 inches RPV failure	mean	events/r.yr	9.4E-4/yr
231	PWR	LOCA	medium LOCA	Break with equivalent sizes between 2 and 6 inches Multiple pressurizer safety and relief valve failure	mean	event/r yr.	9.4E-4/yr
232	PWR	LOCA	small LOCA	Break with equivalent diameter smaller than 2 inches Pressurizer safety and relief valve failure CRDM failures RCP seal failure	mean	event/r.yr	3.5E-2/yr
233	PWR	LOCA	steam generator tube rupture		mean	event/r.yr	2.4E-2/yr
234	PWR	transient	loss of RCS flow		mean	event/r.yr.	3.6E-1/yr
235	PWR	transient	loss of feedwater flow	FW pipe rupture outside containment Loss / reduction of FW flow in one SG Loss of FW flow to all SG FW flow instability - operator error FW flow instability - mechanical causes Loss of one condensate pump Loss of all condensate pumps Condenser leakage Other secondary leakage	mean	event/r.yr.	5.2E+0/yr
236	PWR	transient	partial loss of steam flow	MSIV Closure Full closure of MSIV Partial closure of MSIV Other losses of steam flow	mean	event/r.yr.	2.5E-1/yr
237	PWR	transient	turbine trip	Closure of all MSIV-s Increase of FW flow in one SG Loss of condenser vacuum Loss of CW Throttle valve closure/Electrohydraulic control problem Generator trip / generator caused faults Turbine trip due to over speed Other turbine trips	mean	event/r yr.	3.7E+0/yr
238	PWR	transient	loss of off site power		mean	event/r yr	5.8E-2/yr
239	PWR	transient	spurious safety injection		mean	event/r yr	6.4E-1/yr
240	PWR	transient	reactor trip	CRDM problem / rod drops High and low pressurizer pressure High pressurizer level Spurious automatic trip - no transient condition Automatic/manual trip - operator error Manual trip due to false signals Spurious trip - cause unknown Primary system pressure temperature or power unbalance	mean	event/r.yr	3.8E+0/yr
241	PWR	transient	loss of steam inside containment	Steam pipe rupture inside containment FW pipe rupture inside containment Steam relief or safety valve open inadvertantly (leak upstream of MSIV) Other steam losses inside containment	mean	event/r yr.	9.4E-4/yr
242	PWR	transient	loss of steam outside containment	Steam pipe rupture outside containment Throttle valve/Electrohydraulic control problems Steam and dump valve fail open Other steam losses outside containment	mean	event/r yr	9.4E-4/yr
243	PWR	transient	core power increase	Uncontrolled rod withdrawal Boron dilution - CVCS malfunction Core inlet temperature drop Other positive reactivity additions	mean	event/r yr.	2.3E-2/yr
244	PWR	CC Initiator	loss of component cooling		mean	event/r.yr.	9.4E-4/yr
245	PWR	CC Initiator	loss of service water		mean	event/r.yr	9.4E-4/yr
246	PWR	LOCA	small LOCA		mean	event/r yr	3.5E-2/yr
247	PWR	LOCA	steam generator tube rupture		mean	event/r yr.	3.7E-2/yr
248	PWR	LOCA	medium LOCA		mean	event/r.yr.	9.4E-4/yr
249	PWR	LOCA	large LOCA		mean	event/r yr	9.4E-4/yr
250	PWR	LOCA	intersystem LOCA		mean	event/r yr	4.0E-7/yr
251	PWR	transient	main steam line break		mean	event/r yr	3.9E-2/yr

RANGE VAR	UPPER BND	LOWER BND	SOURCE	ULT SOURCE
n/a	n/a	n/a	Ringhals 1 NPP	plant specific operating experience (5.44 years)
n/a	n/a	n/a	Ringhals 1 NPP	plant specific operating experience (5.44 years)
n/a	n/a	n/a	Ringhals 1 NPP	plant specific operating experience
n/a	n/a	n/a	Ringhals 1 NPP	plant specific operating experience (5.44 years)
95%	1.2E-3/yr	n/a	Ringhals 2 NPP	literature sources
95%	3.0E-3/yr	n/a	Ringhals 2 NPP	literature sources
95%	2.5E-2/yr	n/a	Ringhals 2 NPP	literature sources
95%	1.0E-6/yr	n/a	Ringhals 2 NPP	literature sources
n/a	n/a	n/a	Ringhals 2 NPP	
95%	2.0E-2/yr	n/a	Ringhals 2 NPP	litearture sources
95%	4.0E-1/yr	n/a	Ringhals 2 NPP	plant specific operating experience
95%	9.6E+0/yr	n/a	Ringhals 2 NPP	plant specific operating experience
95%	1.2E+0/yr	n/a	Ringhals 2 NPP	plant specific operating experience
95%	8.6E-1/yr	n/a	Ringhals 2 NPP	plant specific operating experience
95%	2.0E+0/yr	n/a	Ringhals 2 NPP	plant specific operating experience
95%	1.2E-3/yr	n/a	Ringhals 2 NPP	Literature sources
95%	1.7E+0/yr	n/a	Ringhals 2 NPP	plant specific operating experience
n/a	n/a	n/a	Ringhals 2 NPP	literature sources
95%, 5%	3.6E-3/yr	3.3E-5/yr	Zion NPP PSS table 1.1.1-2	literature sources and plant operating experience
95%, 5%	3.6E-3/yr	3.3E-5/yr	Zion NPP PSS table 1.1.1-2	literature sources and plant operating experience
95%, 5%	7.2E-2/yr	1.3E-2/yr	Zion NPP PSS table 1.1.1-2	literature sources and plant operating experience
95%, 5%	7.7E-2/yr	2.8E-3/yr	Zion NPP PSS table 1.1.1-2	literature sources and plant operating experience
95%, 5%	6.0E-1/yr	1.9E-1/yr	Zion NPP PSS table 1.1.1-2	literature sources and plant operating experience
95%, 5%	6.4E+0/yr	4.1E+0/yr	Zion NPP PSS table 1.1.1-2	literature sources (prior) and plant operating experience
95%, 5%	5.2E-1/yr	9.6E-2/yr	Zion NPP PSS table 1.1.1-2	literature sources (prior) and plant operating experience
95%, 5%	4.7E+0/yr	2.8E+0/yr	Zion NPP PSS table 1.1.1-2	literature sources (prior) and plant operating experience
95%, 5%	1.7E-1/yr	8.3E-3/yr	Zion NPP PSS table 1.1.1-2	literature sources and plant operating experience
95%, 5%	1.1E+0/yr	3.3E-1/yr	Zion NPP PSS table 1.1.1-2	literature sources and plant operating experience
95%, 5%	4.7E+0/yr	2.9E+0/yr	Zion NPP PSS table 1.1.1-2	literature sources and plant operating experience
95%, 5%	3.6E-3/yr	3.3E-5/yr	Zion NPP PSS table 1.1.1-2	literature sources and plant operating experience
95%, 5%	3.6E-3/yr	3.3E-5/yr	Zion NPP PSS table 1.1.1-2	literature sources and plant operating experience
95%, 5%	6.1E-2/yr	4.6E-3/yr	Zion NPP PSS table 1.1.1-2	literature sources and plant operating experience
95%, 5%	3.6E-3/yr	3.3E-5/yr	Zion NPP PSS table 1.1.1-2	literature sources (prior) and plant operating experience
95%, 5%	3.6E-3/yr	3.3E-5/yr	Zion NPP PSS table 1.1.1-2	literature sources (prior) and plant operating experience
n/a	n/a	n/a	Angra NPP PSA, Summary report	literature sources
n/a	n/a	n/a	Angra NPP PSA, Summary report	literature sources
n/a	n/a	n/a	Angra NPP PSA, Summary report	literature sources
n/a	n/a	n/a	Angra NPP PSA, Summary report	literature sources
n/a	n/a	n/a	Angra NPP PSA, Summary report	literature sources
n/a	n/a	n/a	Angra NPP PSA, Summary report	literature sources

	PLANT TYPE	IE SET	IE CATEGORY	INITIATORS	FREQ DESC	FREQ
252	PWR	transient	loss of off-site power		mean event/r yr	[illegible].5E-1/yr
253	PWR	transient	loss of main feedwater		mean event/r yr	1.2E+0/yr
254	PWR	transient	reactor trip		mean event/r yr	3.1E+0/yr
255	PWR	transient	turbine trip		mean event/r yr	3.0E+0/yr
256	BWR	LOCA	large LOCA inside drywell	break size greater than 3 sq ft	mean event/r yr	2.7E-4/yr
257	BWR	LOCA	large LOCA outside drywell	break size greater than 3 sq ft	mean event/r yr	1.0E-4/yr
258	BWR	LOCA	medium LOCA	break sizes between .1 and .3 sq ft(steam)and .004 and .2(liquid)	mean event/r yr	2.7E-3/yr
259	BWR	LOCA	small LOCA	break size up to .1sq ft (steam) or .004 sq ft (liquid)	mean event/r yr	2.7E-2/yr
260	BWR	LOCA	intersystem LOCA-LPCI break		mean event/r yr	7.3E-8/yr
261	BWR	LOCA	intersystem LOCA-CS break		mean event/r yr	2.3E-7/yr
262	BWR	LOCA	ECCS breaks		mean event/r yr	1.3E-5/yr
263	BWR	LOCA	reactor vessel rupture		mean event/r yr	3.0E-7/yr
264	BWR	transient	manual shutdown or spurious trip	Spurious trip via instrumentation Scram due to plant occurences Detected fault in RPS Inadvertant insertion of rod or rods Manual shutdown	mean event/r yr	4.5E+0/yr
265	BWR	transient	turbine trip	Turbine trip Generator trip High FW during startup or shutdown Inadvertant startup of HPCI Recirculation control failure increasing flow Turbine bypass or control valve failures Trip of one or all recirculation pumps Loss of FW heater Recirculation pump seizure Pressure regulator fails open Turbine bypass fails open FW flow increasing at power FW controller maximum demand	mean event/r yr.	2.7E+0/yr
266	BWR	transient	MSIV closure	MSIV closure Partial MSIV closure Inadvertant closure of one MSIV Loss of condenser vacuum Turbine trip with bypass failure Generator trip with bypass failure Pressure regulator fails closed	mean event/r yr	4.1E-1/yr
267	BWR	transient	inadvertant opening of relief valve		mean event/r yr	1.3E-2/yr
268	BWR	transient	loss of feedwater	Loss of all FW flow Trip of one of FW pumps FW low Flow Low FW flow during startup or shutdown	mean event/r yr	2.1E-1/yr
269	BWR	transient	loss of off site power		mean event/r yr	6.7E-2/yr
270	BWR	CC Initiator	loss of TBCCW system	Loss of cooling water to TBCCW system Loss of condensate pumps Loss of service water	mean event/r yr.	3.2E-2/yr
271	BWR	CC Initiator	loss of instrument air	Loss of instrument air Loss of RBCCW system	mean event/r yr	1.0E-3/yr
272	BWR	CC Initiator	loss of 6kV emergency bus		mean event/r yr	1.1E-4/yr

RANGE VAR	UPPER BND	LOWER BND	SOURCE	ULT SOURCE
n/a	n/a	n/a	Angra NPP PSA, Summary report	frequency based on specific study
n/a	n/a	n/a	Angra NPP PSA, Summary report	literature sources
n/a	n/a	n/a	Angra NPP PSA, Summary report	literature sources
n/a	n/a	n/a	Angra NPP PSA, Summary report	literature sources
n/a	n/a	n/a	Caorso NPP PSS table A-2	literature sources
n/a	n/a	n/a	Caorso NPP PSS table A-2	literature sources
n/a	n/a	n/a	Caorso NPP PSS table A-2	literature sources
n/a	n/a	n/a	Caorso NPP PSS table A-2	literature sources
n/a	n/a	n/a	Caorso NPP PSS table A-2	literature data
n/a	n/a	n/a	Caorso NPP PSS table A-2	literature sources
n/a	n/a	n/a	Caorso NPP PSS table A-2	literature sources
n/a	n/a	n/a	Caorso NPP PSS table 3 5	literature sources
n/a	n/a	n/a	Caorso NPP PSS table A-2	plant specific operating experience
n/a	n/a	n/a	Caorso NPP PSS table A-2	plant specific operating experience
n/a	n/a	n/a	Caorso NPP PSS table A-2	plant specific operating experience
n/a	n/a	n/a	Caorso NPP PSS table A-2	study for the plant with similar valves (Alto Lazio)
n/a	n/a	n/a	Caorso NPP PSS table A 2	plant specific operating experience
n/a	n/a	n/a	Caorso NPP PSS table A-2	Historical data for north Italy
n/a	n/a	n/a	Caorso NPP PSS table A-2	analysis using fault tree method
n/a	n/a	n/a	Caorso NPP PSS table A-2	analysis of instrument air system
n/a	n/a	n/a	Caorso NPP PSA table A-2	analysis of emergency power system by fault tree

REFERENCES

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

[2] NUCLEAR REGULATORY COMMISSION, Reactor Safety Study: An Assessment of Accident Risks in US Commercial Nuclear Power Plants, Rep. WASH-1400-MR (NUREG/75-014), Washington, DC (1975).

[3] LEVERENZ, F.L., et al., ATWS: A Reappraisal, Part 3: Frequency of Anticipated Transients, Rep. EPRI-NP-801, Palo Alto, CA (1978).

[4] McCLYMONT, A.S., POEHLMAN, B.W., ATWS: A Reappraisal, Part 3: Frequency of Anticipated Transients, Rep. EPRI-NP-2230, Palo Alto, CA (1982).

[5] MACKOWIAK, D.P., et al., Development of Treatment of Initiating Event Frequencies for Use in Probabilistic Risk Assessments, Rep. NUREG/CR-3862, EG&G 2323, Idaho National Engineering Laboratories, Idaho Falls, ID (1985).

[6] ILBERG, D., et al., A Review of the Shoreham Nuclear Power Station Probabilistic Risk Assessment, Rep. NUREG/CR-4050, USNRC, Washington, DC (1985).

[7] NUCLEAR REGULATORY COMMISSION, Severe Accident Risks: An Assessment for Five US Nuclear Power Plants, Final Summary Report, Rep. NUREG-1150, 3 volumes, Washington, DC (1990).

[8] KAPLAN, S., On a Two Stage Bayesian Procedure for Determining Failure Rates from the Experimental Data, Rep. PLG-0191, Pickard, Lowe and Garrick, Irvine, CA (1981).

[9] ELECTRIC POWER RESEARCH INSTITUTE, A Probabilistic Risk Assessment of Oconee Unit 3, Rep. NSAC-60, EPRI, Palo Alto, CA (1984).

[10] WYCKOFF, H., Losses of Off-site Power at US Nuclear Power Plants through 1983, Rep. NSAC-80, EPRI, Palo Alto, CA (1984).

[11] NUCLEAR REGULATORY COMMISSION, Operating Units Status Reports (referred to as 'Gray Books'), published quarterly as USNRC NUREG-0200.

[12] SCIENCE APPLICATIONS, INC., Probabilistic Risk Assessment: Shoreham Nuclear Power Station, Rep. SAI372-83-PA01, SAIC, NY (1983).

[13] CONSUMERS POWER COMPANY AND PICKARD, LOWE AND GARRICK, Midland Nuclear Plant - Probabilistic Risk Assessment, Irvine, CA (1984).

[14] PICKARD, LOWE AND GARRICK, Seabrook Station Probabilistic Safety Assessment, Rep. PLG-0300, Irvine, CA (1983).

[15] HANAN, N., et al., A Review of the Oconee-3 PRA, Rep. NUREG/CR-4374, USNRC, Washington, DC (1986).

[16] ILBERG, D., A Review of initiating events selection and frequencies for PSA Studies, Evaluation of Reliability Data Sources, IAEA-TECDOC-504, IAEA, Vienna (1989).

[17] BABCOK AND WILCOX, Owners Group Probabilistic Evaluation of Pressurized Thermal Shock Phase 1, Internal Report, Rep. BAW-1791 (1983).

[18] KOLB, G.J., et al., Interim Reliability Evaluation Program: Analysis of the ANO-Unit 1 Nuclear Power Plant, Rep. NUREG/CR-2787, Sandia National Laboratories, Albuquerque, NM (1982).

[19] KOLACZKOWSKI, M., et al., Analysis of Core Damage Frequency From Internal Events: Peach Bottom, Unit 2, Rep. NUREG/CR-4550, Vol.4, SAND86-2084, Sandia National Laboratories, Albuquerque, NM, and Livermore, CA (1986).

[20] PHILADELPHIA ELECTRIC COMPANY, Probabilistic Risk Assessment: Limerick Generating Station, Revision 4, Philadelphia, PA (1982).

[21] GENERAL ELECTRIC COMPANY, BWR-6 GESSAR PRA (1982).
[22] GESELLSCHAFT FÜR ANLAGEN- UND-REAKTORSICHERHEIT (GRS) mbH, German Risk Study Phase B, GRS-72, Verlag TÜV, Cologne (1989).
GESELLSCHAFT FÜR ANLAGEN- UND-REAKTORSICHERHEIT (GRS) mbH, German Risk Study Phase B, Vol. 1 -- GRS-A-1600, Verlag, TÜV, Cologne (1989).
[23] ELECTRICITE DE FRANCE, Paluel NPP PSA, EDF, Paris (1990).
[24] SCHOLL, R.F., Loss of Off-site Power Survey Status Report, Revision 3, Report of the Systematic Evaluation Program Branch, Division of Licensing, USNRC, Washington, DC (1980).
[25] McCLYMOND, A.S., POEHLMAN, B.W., Loss of Off-site Power at Nuclear Power Plants: Data and Analysis, Rep. EPRI-NP-2301, Palo Alto, CA (1982).
[26] BARANOWSKY, P.W., Evaluation of Station Blackout Accidents at Nuclear Power Plants, Rep. NUREG-1032, US Nuclear Regulatory Commission, Washington, DC (1988).
[27] IMAN, R.L., HORA, S.C., Modelling Time to Recovery and Initiating Event Frequency for Loss of Off-Site Power Incidents at Nuclear Power Plants, Rep. NUREG/CR-5032, USNRC, Washington, DC (1988).
[28] NUCLEAR REGULATORY COMMISSION, Probabilistic Risk Analysis: Procedures Guide, Rep. NUREG-2300, Washington, DC (1983).
[29] BLOMQUIST, R., PÖRN, K., SUPER-ASAR: Initiating Events in Swedish Nuclear Power Plants, Classification and Qualitative Comparisons, Rep. Studsvik/NP-87/103, Nyköping (1987).
[30] MAYS, S.E., et al., Interim Reliability Evaluation Program: Analysis of the Browns Ferry, Unit 1, Nuclear Plant, Rep. NUREG/CR-2802, EG&G, Idaho National Engineering Laboratory (EG&G, Idaho, Inc.), Idaho Falls, ID, and Energy Incorporated, Seattle, WA (1982).
[31] YANKEE NUCLEAR SERVICES DIVISION, Yankee Nuclear Power Station Probabilistic Safety Study, Framingham, MA (1980).
[32] HOERTNER, H., KAFKA, P., REICHART, G., The German Precursor Study — Methodology and Insights, Reliab. Eng. Syst. Saf. **27** (1990) 53.
[33] DROUIN, M.T., HARPER, F.T., CAMP, A.L., Analysis of Core Damage Frequency from Internal Events: Methodology Guidelines, Rep. NUREG/CR-4550, Vol. 1, SAND86-2084, Sandia National Laboratories, Albuquerque, NM (1987).
[34] HATCH, S., et al., Reactor Safety Study Methodology Applications Program: Oconee Unit 1, PWR Power Plant, Rep. NUREG/CR-1659, SAND80-1897, Sandia National Laboratories, Albuquerque, NM (1981).
[35] DEUTSCHE RISIKOSTUDIE KERNKRAFTWERKE GRS, Verlag TÜV Rheinland, Cologne (1980).
[36] DROUIN, M.T., et al., Analysis of Core Damage Frequency From Internal Events: Grand Gulf, Unit 1, Rep. NUREG/CR-4550, Vol. 6, SAND86-2084, Sandia National Laboratories, Albuquerque, NM, and Livermore, CA (1987).
[37] HUBBLE, W.H., MILLER, C.F., Data Summaries of LERs on Valves at US Commercial Nuclear Power Plants, Reps NUREG/CR-1363 (1980) and NUREG/CR-1963 (1985), USNRC, Washington, DC.
[38] FLEMING, K.N., Realistic Assessment of Interfacing Systems LOCA Scenarios in PWR Plants, Proc. Int. Topical Meeting on Probabilistic Safety Assessment and Risk Management, Zurich (1987) and Verlag TÜV, Rheinland, Cologne (1987).
[39] CHU, T.L., et al., Interfacing Systems LOCAs at Boiling Water Reactors, Proc. Int. Topical Meeting on Probabilistic Safety Assessment and Risk Management, Zurich (1987) and Verlag TÜV, Rheinland, Cologne (1987).

[40] BERTUCIO, C., et al., Analysis of Core Damage Frequency From Internal Events: Surry, Unit 1, Rep. NUREG/CR-4550, Vol.3, SAND 86-2084, Sandia National Laboratories, Albuquerque, NM, and Livermore, CA (1986).
[41] PAYNE, A.C., et al., Interim Reliability Evaluation Program: Analysis of the Calvert Cliffs Unit 1 Nuclear Power Plant, Rep. NUREG/CR-3511, Vols. 1 and 2, Sandia National Laboratories, Albuquerque, NM and Livermore, CA (1984).
[42] CONSOLIDATED EDISON, Indian Point Probabilistic Safety Study, Power Authority of the State of New York and Consolidated Edison, NY (1982).
[43] WHEELER, T.A., Analysis of Core Damage Frequency From Internal Events: Zion Unit 1, Rep. NUREG/CR-4550, Vol. 7, Sandia National Laboratories, Albuquerque, NM (1986).
[44] COMMONWEALTH EDISON COMPANY, Zion Probabilistic Safety Study, Chicago (1981).
[45] NORTHEAST UTILITIES SERVICE, Maine Yankee Probabilistic Safety Assessment, Hartford (1989).
[46] CONSUMERS POWER COMPANY, Big Rock Point Nuclear Power Plant Probabilistic Risk Assessment (1981).
[47] HAKE, T.M., WHITEHEAD, D.W., Initiating Event Analysis for a BWR Low Power and Shutdown Accident Frequency Assessment presented at the PSAM '91, Los Angeles (1991).
[48] ILBERG, D., HANAN, N., An Evaluation of Non-isolated LOCA Outside the Drywell in a BWR, Brookhaven National Laboratory, Technical Report A-3740, Upton, NY (1985).
[49] LANORE, J.M., CHAMP, M., Probabilistic Safety Assessment of French PWRs: Importance of Non-Power States, Use of Probabilistic Safety Assessment for Operational Safety (Proc. Symp. Vienna, 1991), IAEA, Vienna (1992).
[50] SHEPHERD, C.H., Sizewell-B PWR Preconstruction Safety Report, Reference Report — Initiating Fault Frequencies, National Nuclear Corporation, Whetston (1982).
[51] ILBERG, D., HANAN, N., A Review of the Turbine-Building Flooding Study of the Oconee PRA, Proc. Int. ANS/ENS Topical Meeting on Thermal Reactor Safety, San Diego (1986).
[52] BOARDMAN, T., Leak Rate Analysis of the Westinghouse Reactor Coolant Pump, Rep. NUREG/CR-4294, Washington, DC (1985).
[53] KIMURA, G.Y., et al., Evaluation of External Hazards to Nuclear Power Plants in the United States, Rep. NUREG/CR-5042, Lawrence Livermore National Laboratory, CA (1987).
[54] Treatment of External Hazards in PSA, International Atomic Energy Agency, Vienna, Austria (in preparation).
[55] MINARICK, J.N., KUKIELKA, C.A., Precursors to Potential Severe Core Damage Accidents: 1969–1979 — A Status Report, Rep. NUREG/CR-2497, Washington, DC (1982).
[56] Connecticut Yankee (Haddam Neck Plant) Probabilistic Safety Study, Northeast Utilities Service Company Report, NUSCO-149, Hartford, CA (1986).
[57] ENTE NAZIONALE PER L'ENERGIA ELETTRICA, Probabilistic Safety Study Caorso NPP, Rep. ENEL DCO 401.V040,VR001, ENEL-DCO, Rome (1988).
[58] ARRIETA, L.A.I., ANGRA-1 Probabilistic Safety Study, Final Report, CNEN, Rio de Janeiro (1988).
[59] IAEA Regional Programme on PSA for WWER Reactors, Report of a Workshop in Řež, (1992).
[60] BEQE, Top Level Risk Study for Kozloduy Units 1 to 4, Report No. 58-01-R-001 (1992).

LIST OF ABBREVIATIONS

AC	Alternating current
AFW	Auxiliary feedwater
ATWS	Anticipated transient without scram
BWR	Boiling water reactor
B/W	Babcock and Wilcox Co.
CC	Component cooling
CCI	Common cause initiator
CCW	Component cooling water
CDF	Core damage frequency
CFS	Containment fan system
CW	Circulating water
CRD	Control rod drive
CRDM	Control rod drive mechanism
CRW	Control rod withdrawal
DC	Direct current
DHR	Decay heat removal
EBFT	Energy balance fault tree
EHC	Electro-hydraulic control
FMEA	Failure mode and effect analysis
FSAR	Failure safety analysis report
FTA	Fault tree analysis
FW	Feedwater
GPM	Gallon per minute
GRS	German Risk Study
HPCI	High pressure core injection
HPCS	High pressure core spray
HPI	High pressure injection
ICS	Integrated control circuit
IE	Initiating event
IORV	Inadvertent open relief valve
ISLOCA	Interfacing system LOCA
LER	Licensee event report
LOCA	Loss of coolant accident
LOFW	Loss of feedwater
LOOP	Loss of off-site power/loss of station power
MFW	Main feedwater
MSIV	Main steam isolation valve
MLD	Master logic diagram
MOV	Motor operated valve
NPP	Nuclear power plant
PCS	Power conversion system
PORV	Power operated relief valve
PWR	Pressurized water reactor
PRA	Probabilistic risk assessment
PSA	Probabilistic safety assessment
RCS	Reactor coolant system
RCP	Reactor coolant pump

RHR	Residual heat removal
RPS	Reactor protection system
RPV	Reactor pressure vessel
RV	Relief valve
SG	Steam generator
SIS	Safety injection signal
SRV	Safety relief valve
WWER	Soviet designed PWR (water moderated, water cooled reactor)