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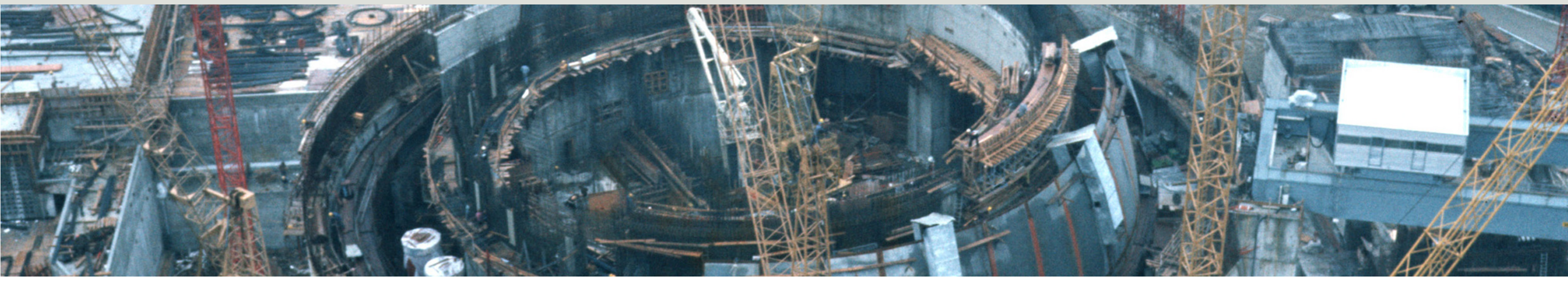

Probabilistic Safety Analysis (PSA): Applications

Guideline for Swiss Nuclear Installations

ENSI-A06/e

Edition March 2009

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Contents

1 Introduction

The Swiss Federal Nuclear Safety Inspectorate (ENSI) is the Regulatory Authority for nuclear safety and security of the nuclear installations in Switzerland. ENSI issues guidelines either in its capacity as a regulatory authority or based on a mandate in an ordinance. Guidelines are implementation support documents that formalize the implementation of legal requirements, and facilitate uniformity of the implementation practice. ENSI may allow deviations from the guidelines in individual cases provided that the suggested solution guarantees at least an equivalent level of nuclear safety or security.

2 Objective and Scope

This guideline formalizes the requirements for the application of Probabilistic Safety Analysis (PSA) for nuclear power plants. It presents the general principles, the requirements for maintenance and upgrade of the PSA, as well as the minimum required scope of PSA applications. The risk measure and evaluation criteria to be applied are defined for these PSA applications.

3 Regulatory Basis

Based on Article 4 Paragraph 3 of the Swiss Nuclear Energy Act (KEG) of 21 March 2003 (SR 732.1), licensees of nuclear installations are required to introduce all safety measures that are deemed necessary according to the experience and the state of the art, and in addition all safety measures that are reasonably achievable and contribute to further increase the safety. PSA is a tool to evaluate the necessity and adequacy of safety measures. In addition, this guideline is based on the following articles in the Nuclear Energy Ordinance (KEV) of 10 December 2004 (SR 732.11):

a. Article 33 Paragraph 1 a KEV (systematic safety evaluation: impact of plant modifications, events and findings on plant safety and in particular on the risk);

b. Article 8 Paragraph 5 KEV (requirements on measures for protection against accidents);

c. Article 10 Paragraph 1 k KEV (accident prevention to take priority over mitigation of consequences);

d. Article 24 Paragraph 1 b KEV (probabilistic requirements in order to get the construction permit of a new nuclear power plant);

e. Article 28 Paragraph 1 KEV (documents to be submitted with the application for an operating license, in particular the requirement for a current, plant-specific PSA according to Appendix 3);

f. Article 34 Paragraph 2 d KEV (Periodic Safety Review: PSR);

g. Article 35 Paragraph 1 KEV (Ageing surveillance);

h. Article 37 KEV in conjunction with Appendix 5 (periodic reporting: list of PSA-relevant plant modifications);

i. Article 40 Paragraph 1 c No. 4 and Paragraph 4 KEV (modifications requiring approval: Technical Specification);

j. Article 41 Paragraph 1 KEV (documentation, in particular a current, plant-specific PSA);

k. Article 82 KEV (transitional regulation).

4 General Principles

a. The use of the current, plant-specific PSA model that meets the requirements of the guideline ENSI-A05 is mandatory for PSA applications.

b. A justification is necessary if the full-scope PSA model in accordance with the guideline ENSI-A05 is not used.

c. Plant modifications and operational experience with impact on plant safety shall be evaluated by the licensee with relevant deterministic, operational and probabilistic arguments.

d. As part of the Periodic Safety Review (PSR), the licensee shall demonstrate that the sum of all plant modifications is either risk-neutral or risk-reducing.

e. The uncertainties quantified with the PSA as well as the model uncertainties shall be adequately considered in the application of PSA.

5 Maintenance and Upgrade of the PSA

Article 33 Paragraph 1 a and Article 41 Paragraph 1 KEV require a current, plant-specific PSA that shall be periodically maintained and upgraded based on the following principles:

For the Level 1 PSA:

a. A complete revision of the PSA shall at the latest be carried out in the course of the PSR. At this time, it shall be determined whether it is necessary to change the applied methods in order to reflect the state of the art (as far as not already described in ENSI-A05).

b. At least once every 5 years, plant-specific data shall be updated and plant modifications shall be incorporated into the PSA model and documented. The low-power and shutdown PSA shall be updated and submitted to ENSI at the latest one year after the update of the full-power PSA.

c. If the combined impact of the PSA-relevant plant modifications not yet incorporated in the PSA model is expected to result in more than about 10% change in Core Damage Frequency (*CDF*[1]) or Fuel Damage Frequency (*FDF*) respectively, these modifications shall be incorporated in the PSA model and documented within a year's time.

For the Level 2 PSA:

d. A complete revision of the PSA shall at the latest be carried out in the course of the PSR. At this time, it shall be determined whether it is necessary to change the applied methods in order to reflect the state of the art (as far as not already described in ENSI-A05).

e. The requirement of updating the Level 2 PSA outside the scope of PSR will be decided by ENSI on a case-by-case basis.

Changes to the PSA model shall be carried out according to a procedure that ensures that the PSA model represents the current state of the plant. The impact of the plant modifications not yet incorporated in the PSA model on *CDF*, *FDF* and Large Early Release Frequency (*LERF*) shall be quantitatively estimated (Article 37 KEV, Appendix 5 KEV) and summarized in a list. The reporting format and contents of the list are specified in Appendix 1.

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1 The risk measures *CDF*, *FDF* and *LERF* are defined in ENSI-A05.

6 Required Range of PSA Applications

In the following, those PSA applications are listed, which shall be carried out as a minimum requirement. Table 3 in Appendix 2 gives an overview of the context and the scope of the required PSA applications.

6.1 Probabilistic Evaluation of the Safety Level

According to Article 24 Paragraph 1 b and Article 28 Paragraph 1 d KEV, for the construction permit and the operation permit of a new nuclear power plant, it shall be demonstrated that the mean *CDF* of the plant is less than 10^{-5} per year.

Based on Article 22 Paragraph 2 g KEG as well as Article 33, Article 34 and Article 82 in connection with Article 8 Paragraph 5 KEV, the following risk measures and criteria shall be applied at existing operating plants for the probabilistic evaluation of the safety level and of the necessity of measures.

a. For the probabilistic evaluation of the safety level in full-power operation[2]:

   If the mean *CDF* (*LERF*) is greater than 10^{-5} per year (10^{-6} per year), measures to reduce the risk shall be identified and – to the extent appropriate – implemented.

b. For the probabilistic evaluation of the safety level in non-full-power operation:

   If the mean *FDF* is greater than 10^{-5} per year, measures to reduce the risk shall be identified and – to the extent appropriate – implemented.

In the case where several measures reduce *LERF* by an equal amount, the principle to follow is that specified in Article 10 Paragraph 1 k KEV: preference is to be given to measures that not only reduce *LERF* but also reduce *CDF*.

The assessment of the safety for operating nuclear power plants shall be carried out during the annual systematic safety evaluation as part of the report on probabilistic evaluation of operational experience (see Appendix 3) and as part of the PSR.

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2 The terms full-power operation and non-full-power operation for PSA purposes are defined in ENSI-A05.

6.2 Evaluation of the Balance of the Risk Contributors

Based on Article 33 Paragraph 1 a and Article 34 Paragraph 2 d KEV, the balance among the contributors to risk shall be investigated as follows:

a. The balance among the risk contributions from accident sequences, components and human actions shall be evaluated. If any of the accident sequences, components or human actions are found by PSA to have a remarkably high contribution, measures to reduce the risk shall be identified and – to the extent appropriate – implemented.

b. If an initiating event category contributes more than 60% to the mean *CDF* and its contribution is more than $6 \cdot 10^{-6}$ per year, measures to reduce the risk shall be identified and – to the extent appropriate – implemented.

c. If the ratio of the mean *CDF* to the $CDF_{Baseline}$ (see determination of the $CDF_{Baseline}$ in Appendix 3) is greater than 1.2, measures to reduce the risk due to planned or unplanned maintenance shall be identified and – to the extent appropriate – implemented.

The evaluation of the balance of the risk contributions shall at least be carried out in the course of the PSR.

6.3 Probabilistic Evaluation of the Technical Specifications

Based on Article 24 Paragraph 2 a, Article 28 Paragraph 1 b, Article 33 Paragraph 1 a, Article 34, and Article 40 Paragraph 1 c No. 4 and Paragraph 4 KEV, the Technical Specifications shall be evaluated as follows:

6.3.1 Probabilistic Evaluation of the Completeness and the Balance of the Allowed Outage Times

In defining the allowed outage times, it shall be ensured that components shown to be significant to safety from the PSA point of view (see Chapter 6.5) are

a. considered in the Technical Specifications (completeness), and

b. assigned to correspondingly short allowed outage time categories (balance).

Based on the risk measures *CDF* and *LERF*, a review of the completeness and the balance of the allowed outage times shall be carried out in the course of the PSR.

6.3.2 Probabilistic Evaluation of Component Maintenance during Full-Power Operation

In addition to the deterministic requirements for the maintenance of components, the following probabilistic requirements shall be satisfied during power operation:

a. Maintenance work shall be planned in such a way that

- no component unavailability configuration *i* resulting from maintenance will result in a Conditional Core Damage Frequency ($CCDF_i$; for computation see Appendix 3) greater than $1 \cdot 10^{-4}$ per year, and
- the total cumulative maintenance time for components shall be limited such that the portion of the Incremental Cumulative Core Damage Probability ($IC_{um}CDP$, see Appendix 3) resulting from maintenance is less than $5 \cdot 10^{-7}$.

b. Compliance with the above mentioned requirements shall be demonstrated either by a previous enveloping analysis along with an additional probabilistic evaluation of operational experience or assessed with the help of a risk monitor. Any deviations from the requirements on maintenance planning mentioned under Letter a shall be justified.

6.3.3 Probabilistic Evaluation of Changes to Technical Specifications

The risk impact of a change to the Technical Specifications shall be evaluated.

a. This applies to all PSA-relevant changes to the Technical Specifications.

b. A change to the Technical Specifications resulting in an increase in risk is tolerable, if

- the impact of the change on the mean *CDF*, *FDF* and *LERF* is insignificant[3], and
- the *CDF* calculated considering the change remains below 10^{-5} per year.

c. If the interval between functional tests is extended, it shall be shown additionally that

- the plant-specific failure rates of the associated components are not greater than the corresponding generic failure rates, and

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3 Insignificant means: $\Delta CDF < 10^{-7}$ per year, $\Delta FDF < 10^{-7}$ per year, $\Delta LERF < 10^{-8}$ per year

- the increase of the *CDF* does not exceed 1% when considering the requested change and assuming failure rates of the affected components that are proportionally increased to the increase of the test interval.

d. Even if the above mentioned requirements are met, measures shall be identified and – to the extent appropriate – implemented in order to compensate for or to minimize the risk increase resulting from the plant modification.

6.4 Probabilistic Evaluation of Changes to Structures and Systems

Based on Article 24 Paragraph 2 a, Article 28 Paragraph 1 b, Article 33 Paragraph 1 a as well as Article 40 Paragraph 1 a and Paragraph 2 KEV, the impact of structural and system-related plant modifications on the risk shall be assessed.

a. This applies to all PSA-relevant structural or system-related plant modifications.

b. A structural or system-related plant modification associated with a risk increase is admissible if

   - the impact of the modification on the mean *CDF*, *FDF* and *LERF* is insignificant, and
   - the *CDF* calculated considering the modification remains below 10^{-5} per year.

c. Even if the above mentioned requirements are met, measures shall be identified and – to the extent appropriate – implemented in order to compensate for or to minimize the risk increase resulting from to the plant modification.

6.5 Risk Significance of Components

Based on Article 35 Paragraph 1 and Article 40 Paragraph 1 a KEV, the following criteria shall be used for the evaluation of the risk significance of components:

a. A component is regarded as significant to safety from the PSA point of view if the following – in terms of *CDF* or *FDF* or *LERF* – applies (selection criterion):

$$FV \geq 10^{-3} \text{ or } RAW \geq 2$$

The Fussell-Vesely (*FV*) and Risk Achievement Worth (*RAW*) importance measures for components shall be determined according to Appendix 4.

b. Components, which are regarded as significant to safety from the PSA point of view, shall be included in a list with the above mentioned importance measures. This list is an integral part of the operating documents.

The list shall be updated at the time of the PSR.

6.6 Probabilistic Evaluation of Operational Experience

Based on Article 33 Paragraph 1 a and 1 b and Article 37 Paragraph 1 KEV, the operational experience shall be evaluated with the PSA as follows.

6.6.1 Annual Evaluation of Operational Experience

a. The effects of PSA-relevant plant modifications carried out during the year shall be assessed as specified in Appendix 1.

b. The following probabilistic safety indicators shall be determined and assessed as specified in Appendix 3:

   - the maximum annual risk peak ($CCDF_{i,\, max}$), and
   - the incremental cumulative core damage probability ($IC_{um}CDP$).

c. The trend of these safety indicators shall be assessed.

d. The contributions to $IC_{um}CDP$ shall be reported in terms of the four categories of “maintenance“, “repair“, “test“ and “reactor trip“. The maintenance contribution to $IC_{um}CDP$ shall be assessed regarding compliance with the criterion described in Chapter 6.3.

e. The dominant contributions to $IC_{um}CDP$ shall be identified and evaluated for both events and susceptibility to component or system failure.

f. If methodological changes are made in the PSA and have significant impact on the *CDF*, the probabilistic safety indicators (Appendix 3) shall be updated retrospectively such that a current assessment of these indicators is available for a minimum of 5 calendar years.

g. The probabilistic evaluation of operational experience shall be documented in accordance with Appendix 3.

6.6.2 Probabilistic Rating of Reportable Events

a. Reportable events that affect PSA-relevant structures, systems, components or operator actions shall be evaluated by means of PSA.

b. The probabilistic rating of events shall be established in accordance with Table 1.

Table 1: Relationship between $ICCDP_{Event}$ and INES-Scale

$ICCDP_{Event}$	INES
$1 > ICCDP_{Event} \geq 1 \cdot 10^{-2}$	3
$1 \cdot 10^{-2} > ICCDP_{Event} \geq 1 \cdot 10^{-4}$	2
$1 \cdot 10^{-4} > ICCDP_{Event} \geq 1 \cdot 10^{-6}$	1
$1 \cdot 10^{-6} > ICCDP_{Event} \geq 1 \cdot 10^{-8}$	0

c. $ICCDP_{Event}$ shall be determined as specified in Appendix 3.

7 Transitional Regulation

The application of *LERF* for the systematic assessment of the plant modifications specified in the Chapters 6.3.3 and 6.4 shall be implemented no later than the 1 January 2010.

This guideline was approved by ENSI on 1 May 2008.

Director of ENSI: signed U. Schmocker

Appendix 1 List of PSA-Relevant Plant Modifications

The list of PSA-relevant plant modifications required in the Chapters 5 and 6.6 of this guideline shall be documented in accordance with Table 2:

Table 2: List of PSA-Relevant Plant Modifications

No. of modification request	Description of Modification	Date of Implementation	Incorporated in PSA model	Impact			
				Comments	Quantitative Estimate		
					Δ*CDF*	Δ*FDF*	Δ*LERF*
Total effect of all plant modifications							
Percentage effect of plant modifications not incorporated in model							

Appendix 2 Overview of the Required PSA Applications

The following table shows what scope of analysis is required in what context.

Table 3: Context and Scope of the Required PSA Applications

Context	Application		
	Evaluations	**Risk Measures**	**References**
Construction permit and operation permit	o Probabilistic evaluation of the safety level	*CDF, FDF, LERF*	Chapter 6.1
	o Evaluation of the balance of risk contributions	*CDF*	Chapter 6.2
	o Balance and completeness of Technical Specifications (only for operation permit)	*CDF, LERF*	Chapter 6.3
	o Probabilistic evaluation of structural and system-related plant modifications requiring approval	*CDF, FDF, LERF*	Chapter 6.4
	o Identification of components that are significant to safety from the PSA point of view	*CDF, FDF, LERF*	Chapter 6.5
PSR	o Evaluation of the safety level as well as the impacts of plant modifications	*CDF, FDF, LERF*	Chapter 5, 6.1
	o Evaluation of the balance of the risk contributions	*CDF*	Chapter 6.2
	o Balance and completeness of Technical Specifications	*CDF, LERF*	Chapter 6.3
	o Identification of components that are significant to safety from the PSA point of view	*CDF, FDF, LERF*	Chapter 6.5
Systematic safety evaluation	o Report on probabilistic evaluation of operational experience	*CDF, CCDF,* $IC_{um}CDP$, $CDF_{Baseline}$	Chapter 6.1, 6.6, and Appendix 3
Plant modifications	o Changes to Technical Specifications o Structural and system-related plant modifications	*CDF, FDF, LERF*	Chapter 6.3 Chapter 6.4
Event	o Probabilistic evaluation of events	$ICCDP_{Event}$	Chapter 6.6, Appendix 3

Appendix 3 Procedure for Probabilistic Evaluation of Operational Experience

A3.1 Risk Measures for Evaluation of Operational Experience

This section describes the procedure for the determination of risk measures for the probabilistic evaluation of operational experience.

a. A so-called **zero maintenance model** shall be constructed based on the current plant-specific PSA model, by setting as available all basic events representing mean component unavailabilities due to planned maintenance, repair, or tests. Besides internal events, the zero maintenance model shall also comprise area events and external events. The *CDF* obtained using the zero maintenance model is called $CDF_{Baseline}$.

b. When calculating the duration of component unavailability, a distinction is made between the following three scenarios:

   - In case of a component failure, the duration of the resulting component unavailability is the component maintenance down time plus the unavailability duration resulting from latent failure[4].
   - In case of maintenance, the duration of maintenance (maintenance down time) shall be taken as the component unavailability duration.
   - In case of a test during which the considered component is unavailable, the duration of the component unavailability is assumed to be the test duration.

c. A **component unavailability configuration** is defined as a state during power operation in which a constant set of components is unavailable.

d. The conditional core damage frequency of the *i*-th component unavailability configuration, during which one or more components are unavailable, is denoted in the following as $CCDF_i$ and shall be determined as follows:

   - with an approximation,
   - with a more precise calculation, if the approximation shows that the $CCDF_i$ of a component unavailability configuration for the year in question represents a relevant risk peak, or if the same component unavailability configuration occurs several times in a single year. In the

[4] A latent failure is a failure that remains undiscovered until the affected (standby) component is actually demanded or functionally tested. In cases where no exact time for the beginning of the unavailability can be determined, half of the time interval between the last two (functional) tests shall be assumed.

latter case, a more accurate calculation shall be performed by re-quantifying the zero maintenance model setting the corresponding components in the model as unavailable.

e. The incremental conditional core damage probability $ICCDP_i$ of the i-th component unavailability configuration shall be estimated as follows:

$$ICCDP_i = (CCDF_i - CDF_{Baseline}) \cdot \frac{\Delta t_i}{8760\,[\text{hours / year}]} \qquad (1)$$

whereby Δt_i is the duration of component unavailability configuration in hours and $CCDF_i$ is the conditional core damage frequency per calendar year.

f. The $ICCDP_j$ of the j-th reactor trip shall be estimated as follows: In the zero maintenance model, the frequency of the corresponding initiating event shall be set to 1 and the frequency of other initiating events shall be set to zero. In case of simultaneous component unavailabilities, the corresponding components shall be set to unavailable in the zero maintenance model.[5]

g. The incremental cumulative (annual) core damage probability $IC_{um}CDP$ is defined as follows:

$$IC_{um}CDP = \sum_{i=1}^{m} ICCDP_i \qquad (2)$$

whereby m is the sum of all component unavailability configurations and all reactor trips that occurred during the calendar year.

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5 For the sake of simplicity (and as a conservative assumption), the risk of a reactor trip estimated in this way is designated as an $ICCDP_j$.

A3.2 Probabilistic Assessment of Reportable Events

This section describes the probabilistic assessment of reportable events.

a. The event to be evaluated shall be assessed by an incremental conditional core damage probability $ICCDP_{Event}$ as follows:

- If the event represents an unplanned component unavailability configuration, then the $ICCDP_{Event}$ is the sum of all $ICCDP_i$ of the k unavailability configurations occurring during the unplanned unavailability configuration:

$$ICCDP_{Event} = \sum_{i=1}^{k} ICCDP_i \qquad (3)$$

  If there is a time overlap of two unplanned component unavailability configurations, then only one $ICCDP_{Event}$ shall be calculated, which takes into account the unavailability of all related components during the overlap of the unplanned unavailability configurations.

- If the event represents a reactor trip, then the $ICCDP_{Event}$ shall be calculated in accordance with Chapter A3.1.

- If the event involves a component unavailability, then the potential impact on the frequency of initiating events and on the probability of Common Cause Failures (CCF) shall be considered.

A3.3 Report on the Probabilistic Evaluation of Operational Experience

The report on the probabilistic evaluation of operational experience (as part of the systematic safety evaluation as per Article 33 Paragraph 1 KEV), which also comprises information on component unavailabilities (Article 37, Appendix 5 KEV), shall cover the following:

a. Documentation of the version of the PSA model applied;

b. Brief description and justification of any special modelling assumptions concerning human reliability analysis and/or CCF;

c. Characteristics of the year under review (date and duration of outages, $CDF_{Baseline}$ applied);

d. Representation (as per Appendix 1) and evaluation of PSA-relevant plant modifications implemented during the year under review;

e. Discussion of the annual evaluation of operational experience according to Chapter 6.6. In order to do so

- the value of the two probabilistic safety indicators ($IC_{um}CDP$ and $CCDF_{i,\,max}$) for at least the last 5 years,

- the contributions to $IC_{um}CDP$, and
- the approximate time graph for *CCDF*

shall be depicted graphically;

f. List of unavailable components including the name of the unavailable component, a brief description of the cause of the component unavailability, its start time and duration;

g. The following data in tabular form for each identified component unavailability configuration (this shall also be sent electronically to ENSI):

- Reference number for each component unavailability configuration,
- Name of the unavailable component(s),
- Brief description of component unavailability configuration,
- Start of component unavailability configuration (date and time),
- End of component unavailability (date and time),
- Conditional core damage frequency of component unavailability configuration i ($CCDF_i$),
- Incremental conditional core damage probability of component unavailability configuration and/or of reactor trip i ($ICCDP_i$),
- Cause (select one of the 4 categories; repair, maintenance, test, reactor trip) for every $ICCDP_i$.

Appendix 4 Procedure for Determination of the FV and RAW Importance Measures of Components

a. To determine the *FV* value of a component, all basic events assigned to the component in question in the current plant-specific PSA model shall be taken into account.

b. To determine the *RAW* value of a component, all basic events assigned to the component in question in the current plant-specific PSA model shall be taken into account.

c. When determining the risk measures *FV* and *RAW*, it shall be taken into consideration that the unavailability of components may have an influence on the initiating event frequencies and on the probability of CCF[6].

d. It shall be shown that the number of components just failing to meet the selection criterion is small. In particular, for components just failing to meet the selection criterion, *FV* and *RAW* shall be determined based on re-quantification of the entire PSA model.

e. If *FV* and *RAW* are not determined based on re-quantification of the entire PSA model, then the uncertainty in the computational approximation shall be discussed.

f. The *FV* and *RAW* values of a component for *FDF* and *LERF* shall be determined in a similar way to those for *CDF*.

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[6] For the assessment of the impact of the CCF probability, for example the following approaches are acceptable:

- The FV/RAW value of the relevant CCF group is included as an additional basic event when calculating the FV/RAW value of components.
- Balancing Method [K. Kim, D. I. Kang, and J.-E. Yang, *On the use of the balancing method for calculating component RAW involving CCFs in SSC categorization*, Reliability Engineering and System Safety, 2005, Vol. 87, p. 233 - 242]

Appendix 5 Definition of Terms

Terms used in this guideline are defined below:

Baseline Core Damage Frequency ($CDF_{Baseline}$)

CDF when no component is assumed as unavailable. The determination of the value is described in Appendix 3.

Component Unavailability Configuration

State during power operation when a constant set of components is unavailable.

Conditional Core Damage Frequency ($CCDF_i$)

Conditional core damage frequency of the *i*-th component unavailability configuration. The determination of the value is described in Appendix 3.

Incremental Conditional Core Damage Probability ($ICCDP_i$)

Incremental conditional core damage probability of the *i*-th component unavailability configuration or reactor trip. The determination of the value is described in Appendix 3.

Incremental Cumulative Core Damage Probability ($IC_{um}CDP$)

Incremental cumulative conditional core damage probability. The determination of the value is described in Appendix 3.

Plant Modification

Any change to a system, structure, or component, or procedure that affects nuclear safety.

PSA-Relevant

Structures, systems, components, operator actions are PSA-relevant if they are considered in the PSA model prepared in accordance with the guideline ENSI-A05.

Zero Maintenance Model

Modified PSA model in which the basic events representing component unavailability due to test, maintenance or repair are set to zero (always available) in the model. The zero maintenance model provides the baseline *CDF* ($CDF_{Baseline}$).

Appendix 6 List of Acronyms

CCDF	Conditional Core Damage Frequency
CCF	Common Cause Failure
CDF	Core Damage Frequency
ENSI	Eidgenössisches Nuklearsicherheitsinspektorat (Swiss Federal Nuclear Safety Inspectorate)
FDF	Fuel Damage Frequency
FV	Fussell-Vesely
ICCDP	Incremental Conditional Core Damage Probability
$IC_{um}CDP$	Incremental Cumulative Core Damage Probability
KEG	Kernenergiegesetz (Nuclear Energy Act)
KEV	Kernenergieverordnung (Nuclear Energy Ordinance)
LERF	Large Early Release Frequency
PSA	Probabilistic Safety Analysis
PSR	Periodic Safety Review
RAW	Risk Achievement Worth

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