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An Approach to Classify the Risk of Operating Nuclear Power Plants – Case Study: Neckarwestheim Unit 1 and Unit 2

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Abstract: A level-2 Probabilistic Safety Assessment (PSA) is an integrated approach to investigate the progression of severe accidents up to containment failure and release of radionuclides into the environment. The results of a standard level-2 PSA include the frequencies associated with various containment failure modes (release categories) along with the environmental release quantities for various radioisotopes (source terms). The extended level-2 PSA approach discussed in this paper merges the standard level-2 PSA results into an integral metric for risk assessment by estimating the integral risk of activity of radiological release to the immediate vicinity of the plant. Risk is defined as a product of the released activity and the release-category frequency, integrated over all possible release categories.

This approach was recently used to assess the risk of severe accidents for the Neckarwestheim Unit 1 (3-loops, 840 MW_e) and the Neckarwestheim Unit 2 (4-loops, 1400 MW_e) nuclear power plants, which entered commercial operation in 1976 and 1989, respectively. The results have demonstrated that neither the core damage frequency nor the core damage profile necessarily is an adequate indicator of plant risk. Furthermore, neither the absolute frequencies of release categories nor the relative proportions of the release category frequencies necessarily provide a balanced picture of severe accident risk as represented by the integral activity of release.

Keywords: Core Damage, Severe Accidents, Level-2 PSA, Risk

1. INTRODUCTION

As generally known, engineers and scientists use the term *risk* to describe events with negative effects and to quantify damages, respectively. Usually, this term is perceived as the product of two factors – *occurrence frequency* and *extent of damage* of an event. Thereby, the frequency of occurrence is the occurrence probability of the event over a specific time period. The extent of damage is the quantitative degree of possible consequences or damage caused by the event. However, evaluation of the influences on risk requires a closer and separate examination of these two factors. Failing uniform definitions for the extent of damage, damage quantification and risk evaluation, respectively, are often geared to the special demands of technical or scientific uses. In addition, the public perception and acceptance of risk based on occurrence frequency and extent of damage is influenced by complex human input variables, shown impressively by Krämer [1], but whereupon this paper does not go into further details. Here, it is presented how the damage potential of Nuclear Power Plant (NPP) accidents in Germany is quantified generally and especially for the NPP Neckarwestheim Unit 1 (GKN I) and Unit 2 (GKN II). In addition, this paper will discuss the importance that GKN ascribes to the risk identified by the quantification of damage for the safety review and safety upgrade of the two units.

The paper introduces shortly the NPP Neckarwestheim with the two units at first. It then presents which types of damage are quantified, the scope covered by the analyses and the methods used. Afterwards the essential results are described and the conclusions are discussed.

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2. THE NPP NECKARWESTHEIM

The NPP Neckarwestheim is located in an abandoned quarry on the river Neckar near the city of Neckarwestheim in the State of Baden-Württemberg in southern Germany. At this plant site, two Siemens-Kraftwerk Union AG (KWU) pressurized water reactors are operated by Energie Baden-Württemberg Kernkraft GmbH (EnKK).

The nuclear power plant Neckarwestheim Unit 1 (GKN I) is a three-loop plant with two turbo generators, which entered commercial operation in May 1976. The reactor thermal output is 2,495 MW_{th}, corresponding to an electric power output of 840 MW_e. GKN I is the only plant in Germany that produces the usual three-phase current of 50 Hertz as well as current of 16.7 Hertz used by the grid of the German railway (Deutsche Bahn AG). The current for the railway is produced by a separate turbo generator. The nuclear power plant Neckarwestheim Unit 2 (GKN II), a four-loop plant with one turbo generator of the so-called *Konvoi* generation, is the newest NPP operating in the German fleet and started commercial operation in 1989. The reactor thermal output of 3,850 MW_{th} corresponds to an electric power output of 1,400 MW_e. This plant does not produce current for the railway but the three-phase current can partly be converted on site by a converter plant into railway current.

Wet cell-type cooling towers are used by GKN I, and a wet–dry hybrid cooling tower with forced ventilation is used by GKN II, in contrast to most other NPPs in Germany. These cooling towers of the two units have a compacter construction and a lesser height; the hybrid cooling tower produces substantially less fogging than the usual natural-draft cooling towers.

The nuclear steam supply system of both units is enclosed by a large spherical steel shell that forms the containment. The containment shell and the components outside the shell are enclosed by a reinforced concrete reactor building. The annular gap between the containment shell and the reactor building is referred to as the Ringraum. Both units are equipped with Passive Autocatalytic Recombiners (PARs) in order to minimize the potential for build-up of combustible gases inside the containment during severe accidents. In addition, in both units, a Filtered Containment Venting System (FCVS) has been implemented that enables manual relief of containment pressure to prevent containment overpressure failure and consequential uncontrolled and unfiltered release of radioactive material into the environment. Some plant parameters of both units important to severe accident progression and containment response are listed in Table 1.

Table 1: Selected Plant Parameters Important to Severe Accident Progression and Containment Response in the PSAs of GKN I and GKN II

Plant Parameter		Dimension	GKN I	GKN II
Thermal power		MW_{th}	2,497	3,850
Containment steel shell	inner diameter	m	50	56
	free volume	m^3	48,000	70,659
Containment failure pressure for static $^{(+)}$ and transient $^{(*)}$ loads	lowest value	MPa-abs.	0.67 $^{(+,*)}$	0.774 $^{(+,*)}$
	50% percentile	MPa-abs.	1.05 $^{(+,*)}$	1.53 $^{(+)}$ 1.70 $^{(*)}$
	highest value	MPa-abs.	1.3 $^{(+,*)}$	2.8 $^{(+)}$ 3.12 $^{(*)}$
Total capacity of passive autocatalytic recombiners		kg/hr	142	192
Ratio of reactor cooling system water volume to power		m^3/MW_{th}	0.095	0.11
Ratio of containment free volume to power		m^3/MW_{th}	19.2	18.35
Ratio of zirconium mass to containment free volume		kg/m^3	0.39	0.45
Ratio of fuel mass to containment free volume		kg/m^3	1.50	1.69

3. DAMAGE QUANTIFICATION

Probabilistic Safety Analysis (PSA) methodology is used for damage quantification of NPPs. It determines on various levels the occurrence frequency and the potential extent of damage to the plant and in the environment. The PSA methodology has matured and is used worldwide as an established and proven approach to quantification of damage, which can occur in the NPPs themselves or can be caused in the environment as a consequence of damages in NPPs. The PSA integrates the impacts of various design, construction and operational features into a systematically integrated and logically consistent process that can be used to search for potential plant design and operational vulnerabilities including the impact of various uncertainties.

Three levels in the quantification of potential damage associated with the operation of NPPs are distinguished: *Level-1 PSA* is concerned with components and systems as well as with analyses of how initiating events may lead to core damage through combinations of various random and common cause failures, including operator errors. The level-1 PSA also investigates the availability of the active functions of containment isolation. *Level-2 PSA* is concerned with the phenomenological aspects of severe accidents and the assessment of possible containment failures after core damage, and the extent of the direct radiological releases at the containment failure locations. *Level-3 PSA* consists of an analysis of the transport and dispersion of radionuclides through the environment to assess the biological, ecological, and economical consequences of various accidents. Within the German framework, level-1 PSA results are sometimes referred to as core-damage-risk, and that of level-2 PSA as the release-risk. But the damage components of the analyses, in all cases, are limited only to the plant itself. The classical environment-risk is ultimately reached as part of the level-3 PSA only.

The information and data needed as input for conducting a PSA are related to the uncertainties associated with various aspects of the data and PSA models. The magnitudes of these uncertainties increase as the consequences propagate from level-1 to level-3 PSA.

In Germany, plant specific level-1 and level-2 PSAs are performed; however, level-3 PSAs are not required. The scope of level-2 PSAs is limited to internal initiating events and full-power operation only. The level-1 and an extended level-2 PSA of GKN I were finished in July 2007 [2] and of GKN II in November 2009 [3]. For both plants, the level-1 PSA was conducted using identical methods by AREVA NP, and the extended Level-2 PSA also using identical methods by Energy Research, Inc. (ERI). A goal of these studies has been to achieve consistency with international practices that form the current state-of-the-art and with the specifications as outlined by the German PSA guidelines [4].

The extension of the Level-2 consists of a *risk approach* to estimate the global consequences outside the plant, allowing for the assessment of the effectiveness and efficiency of the safety-relevant equipment and measures for mitigating the consequences of severe accidents, which was the main objective of the extended level-2 PSA studies. This risk approach uses the integral activity of radiological release to the immediate vicinity of the plant, evaluated over all containment failure modes following core damage. It accounts for the integral risk within the uncertainty margins of the level-2 PSA but excludes the large uncertainties associated with the transport and dispersion of radionuclides and their biological and economical effects typically inherent in the level-3 PSA. It is shown in [5] that insights derived from the results of this risk approach and the quantified biological risk are comparable as a first approximation.

This paper pursues the question of whether the classical results of the level-1 and level-2 PSA are necessarily adequate indicators of the integral plant risk.

4. APPROACH: INTEGRAL RISK OF ACTIVITY OF RADIOLOGICAL RELEASE TO THE IMMEDIATE VICINITY OF THE PLANT

The typical results of the level-1 PSA as well as of the level-2 PSA are related to the plant only. In case of the level-1 PSA, these are the conditional frequencies of plant states with imminent core damage as a consequence of failure of the residual heat removal required for the control of accident initiating events. The results of level-2 PSAs typically include the conditional probability of the various containment failure modes and the frequencies associated with these modes (i.e., release categories) along with the magnitude and times of release of radioisotopes (i.e., source terms). The quantities of radionuclides, e.g., mass fractions of the initial core inventory, released at particular locations of the plant, are not a measure of the global damage impact on the environment surrounding the plant. However, the activity associated with these radionuclides is expected to better reflect the potential consequences of NPP severe accidents. Therefore, the total activity released from all locations with containment failure modes is a more adequate metric for the global damage in the environment.

Here, risk is defined mathematically as follows:

$$R_c = \sum_i \sum_d \sum_s [f_i \cdot P(\mathrm{i}|\mathrm{d})] \cdot P(\mathrm{d}|\mathrm{s}) \cdot C(\mathrm{s}|\mathrm{c}) \,, \qquad (1)$$

where R_c is the risk of consequence measure c [consequence/year], f_i is the frequency of initiating event i [per year], P(i|d) is the conditional probability that initiating event i will lead to plant damage state d, P(d|s) is the conditional probability that plant damage state “d” will lead to source term (release) “s”, and C(s|c) is the expected value of the conditional consequence measure “c”, given the occurrence of source term (release) “s”.

In the present studies, the conditional consequence measure c of severe accidents is the released activity Q_r associated with various radiological releases. This activity is defined as the number of decays per second, i.e. Becquerels (Bq), of a particular radioisotope r, that is:

$$Q_r = \lambda_r \cdot \chi_r \, I_r = \frac{0.6931 \, N \, \chi_r \, I_r}{A_r \, \tau_{1/2,r}}$$

where Q_r = activity of radioisotope r [Bq],
λ_r = radioactive decay constant for radioisotope [s^{-1}],
χ_r = fraction released to the environment for the fission product to which the radioisotope r belongs to,
I_r = total initial inventory of radioisotope r in the fuel [kg],
$\tau_{1/2,r}$ = half-life of isotope r ($= \ln 2 / \lambda_r = 0.6931 / \lambda_r$) [s],
N = Avogadro number ($= 0.6022 \cdot 10^{24}$ mol^{-1}), and
A_r = atomic weight of isotope r [kg/mol].

The model that is used in this calculation of activities accounts for the radioactive decay and daughter build-up of 60 representative, risk-dominant radioisotopes. Each isotope can be mapped to one of the ten radiological groups that are defined for purposes of the source term calculations.

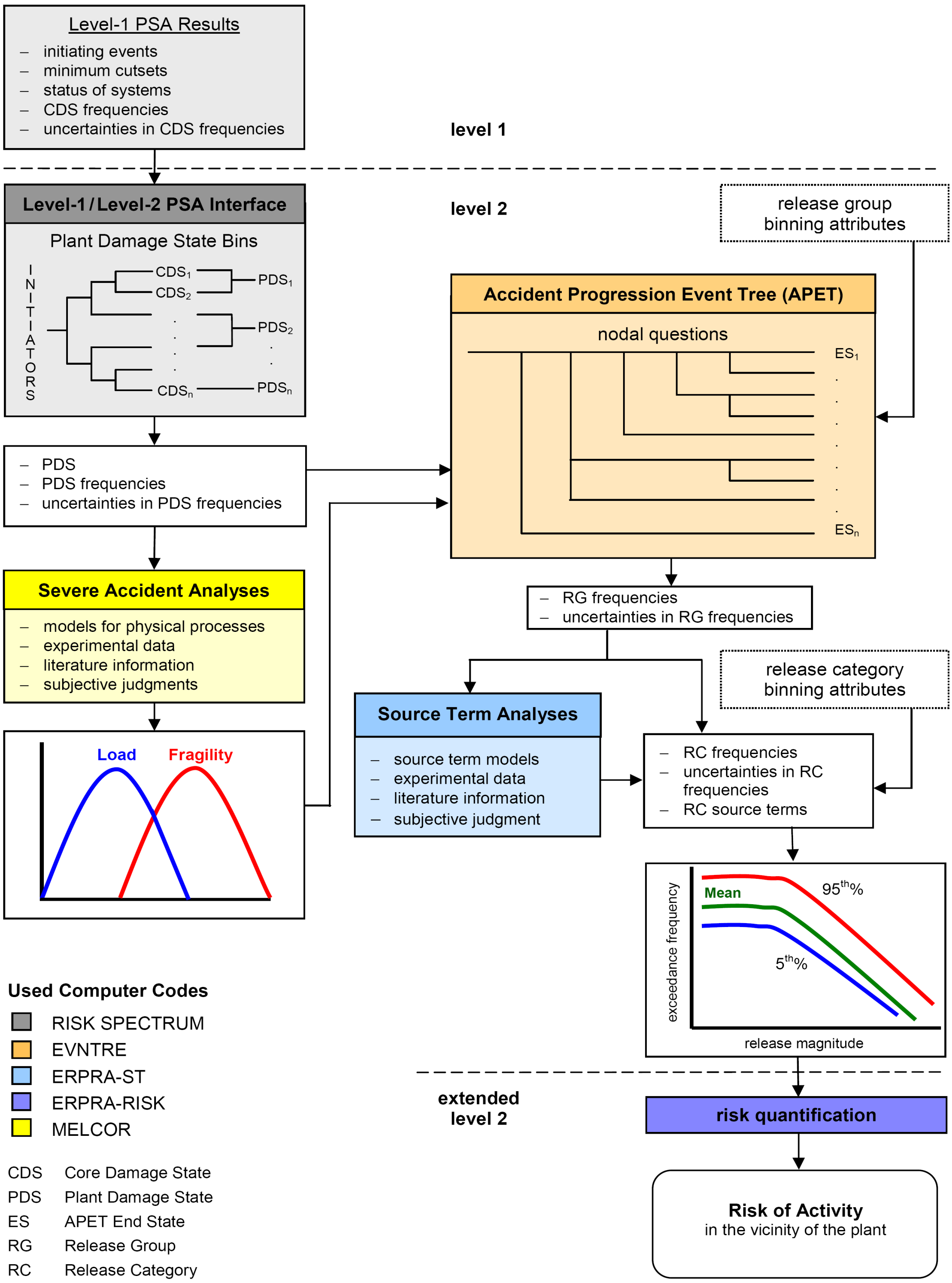
Therefore, the risk metric used in the extended level-2 PSAs for GKN is the integral risk of activity, defined as a product of the source term frequency [per year] and the release activity [Bq], integrated over all unique source terms without specifying the point of release; therefore, the results may be interpreted as the risk of activity in the immediate vicinity of the plant. This time-independent risk-metric is a characteristic plant property, which merges the large number of individual and intermediate outcomes of the level-1 and level-2 PSA into one integral meaningful number.

5. METHODOLOGY AND SCOPE OF THE PSA STUDIES AT GKN

Figure 1 shows schematically the methodology used for the PSA studies at GKN and is herein described in brief; a more detailed overview of the extended level-2 PSA methodology including the quantification of uncertainties is given in [6].

Level-1 PSA. As can be seen, the level-2 PSAs use the level-1 PSAs, which were produced by Siemens-KWU and updated by AREVA using standard PSA tools, as the starting point. The numerous event sequences result in distinct end-states of the event trees and the associated relative frequencies and uncertainty distributions. The end-states are characterized by the unavailability of all required safety functions within the specific event progression, including active containment functions and preventive accident management measures for controlling potential accidents.

Figure 1: Methodology and Scope of the PSA Studies Performed at GKN (Overview)



Interface of level-1 and level-2 PSA. Since the number of end-states that are identified is too large to analyze the physical processes consequent to each of them individually, in the first step in performing a level-2 PSA, the end-states of the event trees of the level-1 PSA are collapsed, by the use of binning attributes and the corresponding states, into Plant Damage States (PDS), which determine the core damage characteristics with potential implications for containment response and radiological releases. The PDS frequencies and associated uncertainties are determined by a level-1 PSA event tree expanded by the binning attributes defined to classify the plant damage states. Yet, the level-2 PSA of GKN is based on a structured interface between the level-1 and level-2 PSAs which practically decouples the two levels and their associated uncertainties.

Deterministic accident analyses provide the technical basis for the detailed assessment of the phenomenological response of various accidents that was subsequently used to quantify the event progression uncertainties and their impact on the challenges to the reactor containment. These analyses were performed based on plant-specific MELCOR 1.8.5 and 1.8.6 computer code calculations for sequences from representative PDS that were selected on the basis of their dominant frequencies and potential consequences, in particular including sequences representative of all potential modes of containment bypass.

Accident progression analysis, consisting of the decomposition of various severe accident processes and their integration within an accident progression event tree (APET), forms the central building block of the level-2 PSA model. The APET represents the probabilistic model that considers all physical and chemical processes influencing the progression of accidents and containment failure and release modes and their associated frequencies. The APET branch fractions, i.e., conditional probabilities, are related to highly uncertain deterministic phenomenological processes with considerable uncertainties, e.g. Zircaloy oxidation, hydrogen production and combustion, vessel failure mechanisms, high-pressure melt ejection, temperature-induced failure of hot leg piping and steam generator tubes, direct containment heating, vessel lift-off, ex-vessel fuel–coolant interactions, thermal loading of the containment-sump intake pipe, molten-core–concrete interactions (MCCI), combustible gas transport in the containment, combustion inside the containment and the filtered containment venting system and over-pressurization of containment due to steam and non-condensable gases in various time frames. The APET also includes branching points for accident management measures for the recovery of system functions and operator actions beyond the level-1 PSA.

Principally, the APET addresses only systemic and phenomenological questions that are fully independent of events modelled in the level-1 PSAs. Beyond a few PDS-dependent questions, the APET facilitates a complete separation between the level-1 PSA analysis of frequencies and their associated aleatory uncertainties, which result from the finite number of observations of a changing system, and epistemic uncertainties, which result from knowledge uncertainties, as described by subjective probability distributions. The APET is solved using the EVNTRE computer code.

The containment damage states as end-states of the APET analysis with the conditional probabilities of the various containment failure modes are collapsed, according to certain characteristics and their attributes, into bins, i.e., release groups, and these, in turn and within the source term analysis, into release categories. The criterion used for distinguishing the bins and release categories is that the release characteristics, i.e., magnitudes, activities, and time of release to the environment, should be similar for all outcomes collapsed into a specific bin or release category, leading to an APET bin definition consisting of primary system pressure at the time of vessel breach, containment failure mode, containment failure time, status of MCCI, conditions in the reactor cavity and core damage time. The dominant characteristics for the subsequent classification into the release categories listed in Table 2 were time and mode of containment failure.

The **source term analysis** determines the quantity of radioisotopes released to the environment as fractions of the initial core inventory, and the uncertainties associated with these release fractions, for each release group individually in a first step and then on an average basis for each release category in a second step.

Risk integration. The release frequencies and properties, including the uncertainties and release quantities, i.e. source terms, of the radionuclides represent the end product of a typical level-2 PSA. Using the risk approach given by Equation 1, the integral risk of activity of radiological release to the immediate vicinity of the plant as an extension to the level-2 PSA is calculated.

Sensitivity analyses are performed to examine the sensitivity of the PSA results to differing assumptions or boundary conditions.

Importance Analyses are aimed at examining which results of the PSA are most sensitive to the uncertainties associated with relevant model parameters. For selected pairs of input and output variables, the correlations between the uncertainties of output variables and the uncertainties of input variables are determined.

Table 3 provides an overview of the numeric scopes of the PSAs of GKN I and GKN II.

Table 2: Definition of the Release Categories Used for the GKN level-2 PSAs

Release Category	Containment Failure Mode	Description of Release Path
RC-A	LOCA outside containment	Large containment bypass → Ringraum → Unfiltered release
RC-B	Uncovered SGTR	Release via uncovered steam generator tubes
RC-C	Early containment rupture	Containment failure at or before vessel breach → Ringraum → Unfiltered release
RC-D	Containment isolation failure	Containment failure before core damage → Ringraum → Unfiltered release
RC-E	Covered SGTR	Release via covered steam generator tubes
RC-F	Sump line failure	Containment failure after vessel breach → Ringraum → Unfiltered release
RC-G	Late containment rupture	Containment failure long after vessel breach → Ringraum → Unfiltered release
RC-H	Basemat melt-through	Release via penetration of concrete basemat
RC-I	Unfiltered containment venting	Containment venting with loss of filtration capability
RC-J	Filtered containment venting	Containment venting to stack with filtration
RC-K	No containment failure	Small containment leakage → Ringraum → Filtered or unfiltered release

LOCA: Loss of Coolant Accident
SGTR: Steam Generator Tube Rupture

Table 3: Numbers to Characterize the Details of the Analyses for the GKN I and GKN II Studies

Parameters and Results of the PSA		GKN I [quantity]	GKN II [quantity]
Bins used to characterize PDS		9	8
PDS		60	56
MELCOR accident analysis runs performed for at least 48 hours		12	18 [a)]
Severe accident phenomena and containment challenges quantified		32	37
Questions in Accident Progression Event Tree (APET)	independent questions to import PDS from level 1	9	9
	dependent questions on accident progression	35	32
	summary questions on the states of branches defined by previous answers	9	8
Bins to combine APET end states into release groups		6	7
Release groups with individual source term analysis		73	84
Release categories as combined release groups		11	11
Radionuclide groups used in source term calculation		10	10
Risk-dominant radionuclides to calculate the activity-based plant risk		60	60
Importance analyses	runs of the importance code	5	5
	individual correlations checked	518	537
Sensitivity analyses on the influence on model parameters		14	14

PDS: Plant Damage State

[a)] MELCOR runs performed for at least 60 hours after initiating event

6. RESULTS AND INSIGHTS

The results of damage quantification in the various levels of PSA are determined by the product of two different kinds of factors, namely the frequency of occurrence of the damage and the extent of damage. Therefore, ideally PSA results on each level should be examined and discussed not as a whole but individually with respect to these different contributors. The relevant properties are compiled in Table 4.

Table 4: Terms Important for Damage Quantification and PSA on Various Levels

PSA Level	Analyses Concerns	Quantified Damage Factors		PSA Results	Characteristics of Final States of PSA Results
		Damage Occurrence Frequency	Damage Extent of Plant or Environment		
1	*Plant* components, systems	Accident initiators and accident initiating events	Unavailability of equipment and measures of the plant to control accident initiators and initiating events	Frequency of various core damage states and their total frequency	Imminent core damage
2	*Plant* severe accident phenomena	Core damage states binned by similar behaviour of severe accident progression, containment challenges, and radionuclide release	Failure probability of containment and fractions of radionuclide core inventory released at locations of containment failures	Frequency of various release categories and their source terms	Containment failures and associated radionuclide releases
2 ext	*Plant* severe accident phenomena of activity release	Release categories	Release probability of activity to vicinity of the plant by release categories	Frequency of total activity released to vicinity of the plant by all release categories (activity risk)	Total activity in the immediate vicinity of the plant

The binning process of core damage states at the level-1/level-2 interface does not use cut-off criteria that could eliminate significant contributions to the core damage frequency of the level-1 PSA or to the results of the level-2 PSA. Therefore, the illustration of the essential results of the level-1 PSA can be confined to the PDS used in the level-2 PSA.

The analysis of PDS for GKN I and GKN II show that the total mean PDS frequency is very low at a level of about 10^{-6} per year, whereupon the GKN I and GKN II results lie about a factor of 2 over and under this level, respectively, with comparable uncertainty ranges of about one order of magnitude between the 5% and 95% percentiles. The uncertainty ranges increase with the further development of the PDS within the level-2 and the extended level-2 PSA.

6.1 Impact of relative PDS frequencies on the results of the extended level-2 PSA of GKN I

Table 5 shows the relative contributions of the initiating events to the total PDS frequency of GKN I. Based on this table, the focus within the context of plant safety optimization is directed upon the dominance of the PDS associated with station black-out (SBO). For this initiating event one has to balance to which extent either the initiating event frequency or the unavailability of plant safety systems or operator actions could be reduced with an appropriate effort–benefit ratio. Such efforts may result in a reduced core damage frequency; however the potential impact of such an effort on the environment-risk cannot be assessed on basis of a level-1 PSA.

In the level-2 PSA, a transition takes place, as displayed in Table 6, from the PDS and their associated frequencies including uncertainties to the release categories and their frequencies including uncertainties. It is noteworthy here, that the total PDS frequency and the sum over all release-category frequencies are identical. Column 3 of Table 6 further shows the time associated with the onset of radiological release in each release category. Four time frames are distinguished: (1) very early time frame from the onset of core damage to reactor pressure vessel (RPV) failure, (2) early time frame near the time of RPV failure, (3) intermediate time frame lasting approximately 12 hours from RPV failure, and (4) late time frame from the end of the intermediate time frame until the end of the level-2 mission duration, which is typically 48 hours in the GKN analyses from start of accident initiation. A typical duration of the very early time frame, which varies depending on the particulars of the accident sequence, is about 10 h. Not presented here are the additional time periods from the occurrence of the initiating event to the onset of core damage. These time periods, e.g., for STGR events can be considerable, i.e., 20 h or more. However, these time periods are associated with large uncertainties and are therefore not appropriate as reference values for safety-relevant decision criteria.

Table 5: Relative Contributions of Initiating Events to the Total PDS Frequency of GKN I

Initiating Event Leading to Core Damage	Contribution to Total PDS Frequency [%]
Station black-out (SBO)	51.6
Other transients including ATWS	16.3
Very small and small LOCA in the reactor coolant system	12.8
Pressurizer LOCA	11.2
Steam generator tube rupture	6.7
Transients initiated by internal flooding	1.1
LOCA outside containment	0.3
Sum	**~ 100**

ATWS: Anticipated Transient without Scram
LOCA: Loss of Coolant Accident

The released fraction of radioisotopes is not discussed here. All release categories are associated with different release fractions of all relevant groups of radionuclides. However, released fractions of radionuclides are not a direct measure of the extent of biological, ecological, and economical damage associated with the radiological release. Therefore, pragmatic safety goals based on the relative contribution of each release category to the total PDS frequency or the absolute frequency, such as the Large Early Release Frequency (LERF) or the Large Release Frequency (LRF), are not necessarily the appropriate way to evaluate their contribution to the overall plant risk as also indicated in Table 6. At this point, a first finding from comparing Tables 5 and 6 is that the accident progression of most PDS that dominate the total PDS frequency, e.g., transients and primary-side LOCAs, lead to late filtered containment venting. The containment is assessed to remain intact and unvented for about 10% of core damage events. Further insights to the GKN I results are presented in [6].

The extended level-2 PSA calculates the conditional release activity associated with the radiological release of each release category and the integral risk of activity of radiological release in the immediate vicinity of the plant. Table 6 provides the relative contribution of each release category to the integral risk of activity. Since noble gases decay quickly and are not as consequential as aerosols such as caesium or barium, their otherwise dominating contribution to the integral risk of activity is not discussed in this paper. The risk dominance of the noble gases would mask any findings from sensitivity analyses and will not be useful in interpreting the results. Furthermore, noble gas releases following severe accidents in existing plants cannot be mitigated with a proper risk–benefit ratio.

As a consequence, the integral risk of activity for GKN I is extremely low. The mean percentage of risk of release of the entire core inventory, i.e., the ratio of the integral risk of activity and risk of activity of release[1] assuming the entire initial core inventory of aerosol-type and gaseous radionuclides (excluding noble gases) is released, is about 0.06 %. This underlines the efficiency of safety-relevant equipment and measures for mitigating the consequences of severe accidents at GKN I.

From comparing the results in Tables 5 and 6, further conclusions can be drawn. It is apparent that some of the PDS with the lowest fractions at the total PDS frequency show the highest contributions the integral risk of activity. Sequences for which the containment integrity is compromised prior to core damage, i.e., SGTR events, containment isolation failure due to internal flooding, and LOCA

[1] The risk of activity of release assuming the entire core inventory is released is defined as the product of the entire initial core inventory in [Bq] of the 60 risk-relevant radioisotopes analyzed and the total PDS frequency.

outside containment, dominate the integral risk of activity, i.e., the quantification of the extent of damage to the environment, with more than 80 %. In contrast, those PDS that dominate the total PDS frequency do not play a significant role in the integral risk of activity. As a consequence, a safety optimization solely based on the results of level-1 and level-2 PSAs does not necessarily improve plant safety with respect to the potential extent of damage in the environment, i.e., level-1 and level-2 PSA results might not be a sufficient basis for environment-risk reduction.

Table 6: Release Categories, Associated Times of Release, Fractions of the Total PDS Frequency, and Fractions of the Integral Risk of Activity in the Immediate Vicinity of the Plant GKN I

Release Category (RC)			Fraction [%]	
RC	**Description**	**Period of Start of Release**	**Total Frequency**	**Risk of Activity [a]**
RC-B	Uncovered SGTR	very early	< 0.1	3.0
RC-E	Covered SGTR	very early	6.7	51.4
RC-D	Containment isolation failure [b]	very early	1.4	12.5
RC-A	LOCA outside containment	very early	0.3	21.5
RC-C	Early containment rupture	very early to early	< 0.1	1.6
RC-F	Sump suction liner failure	intermediate	< 0.1	< 0.1
RC-G	Late containment rupture	intermediate to late	0.2	0.3
RC-H	Basemat melt-through	late	0.5	0.6
RC-I	Unfiltered containment venting	late	4.0	7.4
RC-J	Filtered containment venting	late	77.5	1.6
RC-K	No containment failure	late	9.3	<< 0.1
Sum			**~100**	**~100**

SGTR: Steam Generator Tube Rupture
LOCA: Loss of Coolant Accident

[a] without noble gases; [b] due to internal flooding

6.2 Impact of relative PDS frequencies on the results of the extended level-2 PSA of GKN II

Table 7 shows the relative contributions of the initiating events to the total PDS frequency of GKN II. The release categories, their time frames for the beginning of radiological release, and the fraction contributed by each release category to the total PDS frequency as well as to the integral risk of activity are compiled in Table 8. The relative contributions of the release categories to the total PDS frequency for GKN I and GKN II are compared in Figure 2.

Table 7: Relative Contributions of Initiating Events to the Total PDS Frequency of GKN II

Initiating Event Leading to Core Damage	**Contribution to Total PDS Frequency [%]**
Very small and small LOCA in the reactor coolant system	36.6
Pressurizer LOCA	32.4
Transients including Station Blackout (SBO) and ATWS	20.8
Steam generator tube rupture	9.3
Transients initiated by internal flooding	0.8
Sum	**~100**

Table 8: Release Categories, Associated Times of Release, Fractions of the Total PDS Frequency, and Fractions of the Integral Risk of Activity in the Immediate Vicinity of the Plant GKN II

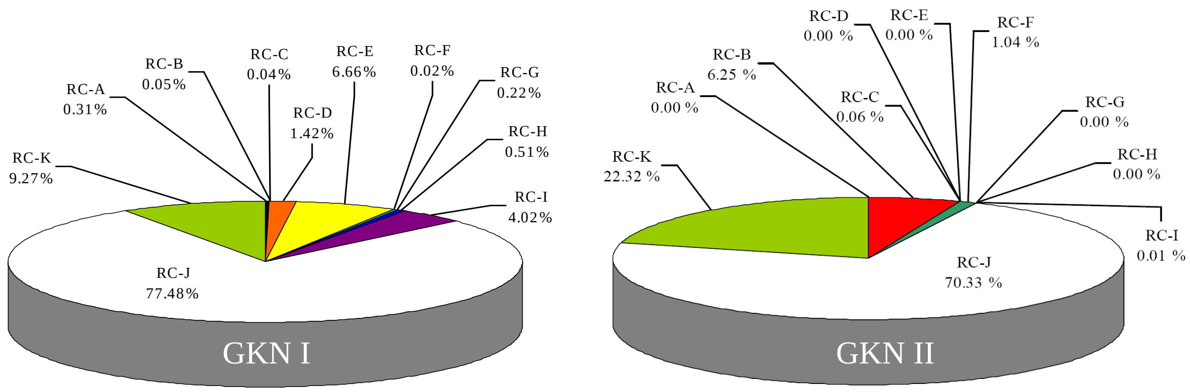
Release Category			Fraction [%]	
RC	**Description**	**Period of Start of Release**	**Total Frequency**	**Risk of Activity [a)]**
RC-B	Uncovered SGTR	very early	6.3	99.1
RC-E	Covered SGTR	very early	0	0
RC-D	Containment isolation failure	very early	<< 0.1	< 0.1
RC-A	LOCA outside containment	very early	<< 0.1	<< 0.1
RC-C	Early containment rupture	very early to early	< 0.1	0.2
RC-F	Sump suction liner failure	intermediate	1.0	0.4
RC-G	Late containment rupture	intermediate to late	< 0.1	< 0.1
RC-H	Basemat melt-through	late	0	0
RC-I	Unfiltered containment venting	late	< 0.1	< 0.1
RC-J	Filtered containment venting	late	70.3	0.2
RC-K	No containment failure	late	22.3	<< 0.1
Sum			**~100**	**~100**

SGTR: Steam Generator Tube Rupture
LOCA: Loss of Coolant Accident

[a)] without noble gases

A first comparison of Tables 7 and 8 shows, that, as is the case for GKN I, the accident progression characteristics associated with most PDS that dominate the total PDS frequency, e.g., primary-side LOCAs and transients, lead to late filtered containment venting or even no containment failure. Again, it is obvious that the contribution of the frequency-dominant PDS to the integral risk of activity is extremely low. In a comparable way to GKN I, especially certain PDS with minor contribution to the total PDS frequency, i.e., those with SGTR, dominate the integral risk of activity in the immediate vicinity of the plant. Again, similarly to GKN I, this risk was found to be extremely low. The mean risk of severe accidents at GKN II is about 0.6 % of the mean risk of the release of the total core inventory (excluding noble gases). This demonstrates the effectiveness of safety-relevant equipment and measures in mitigating the consequences of severe accidents at GKN II.

Figure 2: Relative Contributions of Release Categories to the Total PDS Frequency



Altogether, the insights found for GKN I are confirmed by the findings for GKN II. Therefore, the following general conclusions can be drawn.

7. CONCLUSIONS

The integral approach of the PSA is not only a useful but also a necessary supplement to the individual deterministic approaches to guarantee a save and economic operation of NPPs. The PSA enables to determine the extent of damage of the plant and the environment comprehensively and to quantify the uncertainties of the results. The use of three PSA levels is a pragmatic approach connected with the knowledge base and uncertainties of the phenomena to be quantified for the various levels. These phenomena are related for level-1 to the well-known behaviour of components, systems, and operators, for level-2 to partly known severe accident progression and containment challenges, and for level-3 to only marginally known transport behaviour and effectiveness of radionuclides in the environment. Conclusions from a PSA should only be drawn if the evaluation of damages is finished completely or is well-founded clear at least. Otherwise decisions can later turn out to be integrally less efficient.

The classical environment-risk of NPPs as a complete evaluation of damages in the plant and in the environment is ultimately arrived at by the level-3 PSA only, which, however, suffers from large uncertainties. Therefore, in this study, the level-2 PSA was extended by a risk approach that estimates the integral risk of activity of radiological release to the immediate vicinity of the plant as a metric for the global consequences to the environment outside the plant. This risk metric is a characteristic plant property, which merges the large number of individual and intermediate outcomes of the level-1 and level-2 PSA into one meaningful integral number.

The integral risk of activity of release was found to be extremely low and comparable for GKN I and GKN II. The PSAs for both GKN plants demonstrate that those PDS, which dominate from the perspective of relative frequency, have only a marginal influence on the integral risk of activity. In contrast, this risk is dominated just by certain core damage states with minor relevance in the PDS profile of the level-1 PSA. To bring safety improvements only into line with frequency-dominant contributions of the level-1 PSA might be less efficient for the integral plant safety because these improvements may have only marginal influence on the risk of activity. Simplifying characteristics or parameters, e.g., total core damage frequency or large early release frequency, cannot compensate interrelations of the damage developments analyzed plant-specifically.

In conclusion, the results of the PSAs of GKN I and GKN II have demonstrated that neither the core damage frequencies nor the core damage profiles are adequate indicators for the integral risk of activity. Furthermore, neither the absolute frequency of each release category or of groups of release frequencies, e.g., LERF or LRF, nor the relative proportions of the release category frequencies necessarily allow a conclusion to be made about the integral risk of activity.

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