Post-Fukushima PSA modelling: Best-estimate, plant-specific considerations vs. conservative requirements

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Abstract: This paper presents a comparative study of variations of a nuclear power plant specific PSA model given various hazard inputs as well as modelling assumptions and considerations. The first part of the paper considers one of the measures of the Goesgen NPP investment program for increasing safety margins for the plant's long-term operation. Specifically, one of the foreseen measures - the implementation of an automatic seismic trip system, its consideration within the plant's PSA model and its effect on the plant risk - is being discussed in details. Further on, the turbine missile hazard and the specificity of this hazard to the Goesgen NPP is being addressed. Consequently, the PSA modeling assumptions are argued. As a third comparison case, the tornado hazard - as newly prescribed by the regulator - is compared against the old one and the one given by the U.S. NRC. Additionally, two seismic hazards - both suggested by the regulator - are considered and compared in the models. The results direct a risk reduction that can be achieved by the instalment of the automatic seismic trip system as well as indicate the wide variation of calculated risk given various inputs' (over-) conservatism.

Keywords: NPP, Post-Fukushima measures, plant-specific PSA.

1. INTRODUCTION

The nuclear power plant Gösgen (KKG) is a 3-loop KWU PWR 1060 MWe single-unit NPP that was put in commercial operation in 1979. Since 1979, regulatory safety requirements and hazard presumptions have increased significantly, especially after the Fukushima Daiichi event. In addition, technological obsolescence led to a lower safety level compared to new NPPs. For KKG, increased high earthquake hazard presumptions led to a significant reduction of safety margins. Therefore, the main issue to prepare the KKG for long-term operation was the issue to improve the control of severe earthquake events. Based on a thorough integrated risk-informed decision-making (IRIDM) process, a multi-measures investment program to enhance safety systems with focus on the special emergency system was initiated.

A comparative study of variations of the KKG PSA model given considerations of new safety equipment as well as modelling assumptions of various hazard inputs is presented within this paper.

Firstly, the paper addresses one of the measures of the KKG investment program for increasing safety margins for the long-term operation of the KKG i.e., the implementation of an automatic seismic trip system (ASTS) with associated triggering of a turbine trip as an additional limiting system. The current state of the plant is such that the High Confidence of Low Probability of Failure (HCLPF) value of the KKG regarding the triggering of reactor trip by a seismically induced damage is assessed to be 0.09g. In order to be able to perform a safe shutdown, the control rods should already be inserted at low peak ground accelerations (PGA) in the range of 0.02-0.03g. The study summarizes the modelling and implementation of the above described, new ASTS within the KKG plant PSA model as well as discusses the possible implications on plant risk assessment. Regarding the latter, the paper presents a sensitivity study of few possible scenarios of realistic, plant-specific modelling of the ASTS and its risk reduction impacts. One of the major foreseen impacts of the ASTS is its effect on the human reliability analysis (HRA) modelling.

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Secondly, the plant-specific analysis on turbine missile probability is briefly presented. The derived failure probability is much less than the generic one, being frequently used by the operators throughout the world. Consequently, the rationale of the turbine hazard being screening from the PSA model is discussed. A comparative analysis, given the possible plant-specific failure probability vis-à-vis the generic values, is presented.

Thirdly, the impact of the newly suggested tornado hazard by the regulator is considered. Namely, in the latest version of the PSA-related guideline issued by the regulator, a new tornado hazard is suggested. These new assumptions are reflected in the PSA model and compared against the old ones.

At the end, seen as a fourth point, the newly-suggested seismic hazard (ESREL-2015) [1] is compared against the previous one (PEGASOS2004). Again, a comparative analysis of the plant PSA model is performed given the two different hazard assumptions.

2. ANALYSIS AND MODELLING ASSUMPTIONS

The analysis is organized into several parts, where the various assumptions and modelling considerations are described.

The plant PSA model is prepared with the RISKMAN® analysis tool. It is a small fault tree (FT) – large event tree (ET) linking approach software. The plant's PSA model *REF0* [2] is used as a nominal reference model in this study. This model is based on the PEGASOS seismic hazard assumption, the older (ENSI A05-2009 [3]) tornado hazard assumption, the ASTS is still not modelled within, and the turbine hazard is screened out based on plant-specific analysis. The obtained PSA Level 1 and 2 risks for this model *REF0*, CDF and LERF, are 2.04E-5/yr and 8.91E-6/yr, respectively.

2.1. Automatic Seismic Trip System - Preliminary Analysis and PSA-Model

Currently, at KKG, there is option for manual actuation of the reactor trip from the “secured area”, i.e. the emergency bunkered (ZX) building. The idea, foreseen with this measure, is to extend this manual actuation with an additional option for automatic initialization of the reactor trip from the ZX building. Consequently, along the already existing high level of protection given the available automatic reactor trip from the “unsecured area” (the electrical (ZE) building), there will be a provision for automatic reactor trip of the plant directly from the ZX-building also in case of relatively rare-events that are to be coped with by the special emergency safety systems. Given the automatic reactor trip from the ZX-building, new criteria/limit values will be installed such that small LOCAs and secondary side leakages would be possible to cope with. These criteria / limit values are already available within the reactor protection system (RPS) logic. Currently, they are processed either in the ZE- or the ZX-building. All the contacts from the “unsecured area” are being currently interconnected for the initialization of reactor trip.

A seismic trip via an ASTS will give a lead-time before other reactor trip initiators. With a tripping time < 0.3s of this new ASTS it is achieved that the rod insertion is triggered before a greater load of the reactor pressure vessel (RPV) internals, the control rod drives or the fuel assembly structures occurs. Several load cycles are required (load hysteresis) before damage to a system, structure or component (SSC) can be caused while maintaining the same force. Currently, the ageing of the plant (conventional part) as well as the seismic enhancement of external power supply performed some time ago have a detrimental effect on the plant safety in the event of an earthquake, as an automatic reactor trip is expected only above the Operating Basis Earthquake (OBE) of the original design, which is 0.07g. The chosen trigger criterion of 0.02g for the new-to-be-built ASTS ensures a sufficient safety margin until reaching the OBE excitation level of 0.07g (narrow band spectrum) at the reactor building foundation level. The installation of an ASTS and the triggering of the rod insertion with turbine trip at a trigger value well below the OBE in KKG complies with the recommendations of the International Atomic Energy Agency (IAEA) for a safety-related approach for older nuclear power plants given the seismic risk assumptions increase significantly based on new findings.

One of the major foreseen impacts of the ASTS is its effect on the human reliability analysis (HRA) modelling. Namely, at its current stage, the KKG PSA model presumes a general guaranteed failure of all post-accident operator actions (OAs) by seismic accelerations > 0.6g. The successful operation of the ASTS renders this rather conservative HRA treatment as irrelevant. Realistically seen, the psychological stress of the staff after an earthquake differs no longer from the burden of any other initiating event for a major accident. This leads to a significant increase in human reliability in personnel actions. Meanwhile, the Swiss regulator ENSI has adopted the new seismic hazard assumptions ENSI-2015 as the basis for deterministic safety assessment and risk assessment. Thus, a sensitivity calculation comparing the old (PEGASOS) versus the new (ENSI-2015) seismic hazard is also performed. Additionally, a consideration is made of completely removing and thus eliminating the risk impacts of the seismic hazard > 0.6g from the PSA model. Namely, the basis for the KKG plant special safety systems enhancement program is the possibility to control design extension conditions 4a in case of an earthquake of PGA=0.6g, i.e. plant target HCLPF of 0.6g (broad-band spectral shape) which corresponds to the ENSI-2015 exceedance frequency 10-5/a. Given the Black Swan theory (the next Black Swan Earthquake for KKG [4] is calculated to be 0.54g) as well as the Theory of Records, earthquakes that could endanger the plant safety during the remaining operation life of the KKG (presumably 2039) can be excluded.

In the present case, when assessing the risk benefits of the ASTS, the baseline PSA comparison model must be able to map the adverse effects of reactor trip in the event of earthquakes under uncontrolled constraints, otherwise a realistic, cost-benefit assessment of the change is not possible. Under the present KKG conditions without a targeted reactor trip of the RPS in the event of earthquakes such that malfunction of structures, systems and components (SSCs) and in particular of control systems can occur up to reactor shutdown by the normal reactor protection criterion. If the reactor trip does not take place with clearly defined boundary conditions in regards to the plant condition, there is no guarantee that the event sequences postulated in the PSA with the associated success criteria will actually reflect with sufficient accuracy the response of the plant to an earthquake. This aspect is given (too) little attention in the established PSA methodology, as there are generally limited potential adverse interactions between earthquake-related safety systems and normally operating systems. While it is possible to model damages on the secondary side, that can occur as a result of earthquake-induced component failures, the system state changes (performance parameters, initial state of the systems), which may occur before damage occurs, are not considered. However, these can lead to a change in the event sequences and to other conditions of effectiveness (success criteria) of the systems. It is not completely excluded, for example, that a power transient is induced by control rod malfunctions, which leads to an increase in the thermal power before the reactor shutdown. This would also result in a higher level of residual heat after shutdown, resulting in different requirements for residual heat removal. However, this can also lead to other processes, e.g. in the KKG for (guaranteed) opening of a pressurizer safety valve (PSV) with an increased probability of remaining open. For this reason, an extended reference model has been developed (*REF2*). As a conservative scenario it is postulated that, as a result of an earthquake and until the actuation of the trip various malfunctions can occur in the power control of the reactor, which can then lead to control rods ejection. This power increase leads to an increase in pressure in the primary circuit and possibly to the response of the first PSV. Since the boundary conditions of this "pre-shutdown transient" cannot be determined exactly, a conservative assumption is that the PSV remains open (induced LOCA, "TMI" scenario). This scenario can only occur in the case of small to medium earthquakes (up to ca. 0.6g), since very strong earthquakes can be considered prerequisite for a loss of offsite power (LOOP), given the fact that in such cases the earthquake loads reach the capacity limits of the external power grid but also the design operating basis earthquake (OBE) capacity limits of non-nuclear SSCs are significantly exceeded and also a failure in a short time after the occurrence of the earthquake is to be expected. The assumption that there will be delayed reactor trip in case of earthquakes with PGA below 0.6g with opening of the first PSV (with probability 1.0) can therefore be regarded as a sufficiently justified conservative scenario for the extension of the reference model (*REF0* → *REF2*). On this basis, it is possible to determine the usefulness of the ASTS.

Once the preliminary analysis are being conducted, the PSA-modelling, i.e. the consideration of the ASTS within the PSA-model follows. The functional logic of the ASTS is presented on Figure 1. The measuring unit, i.e. the measuring transducers (2 pro redundancy train) of the ASTS are presented above the red dotted line. Everything below the red-dotted line belongs to the TXS unit. The TELEPERM®-XS (TXS) is the I&C system platform of AREVA for the safety instrumentation and control in NPPs. In the case of KKG, one of its applications is the reactor power limitation function, via which the new-to-be-installed TXS is to be implemented. In other words, the seismically initiated reactor trip will be realized via the reactor power limitation function (STEW-RESA).

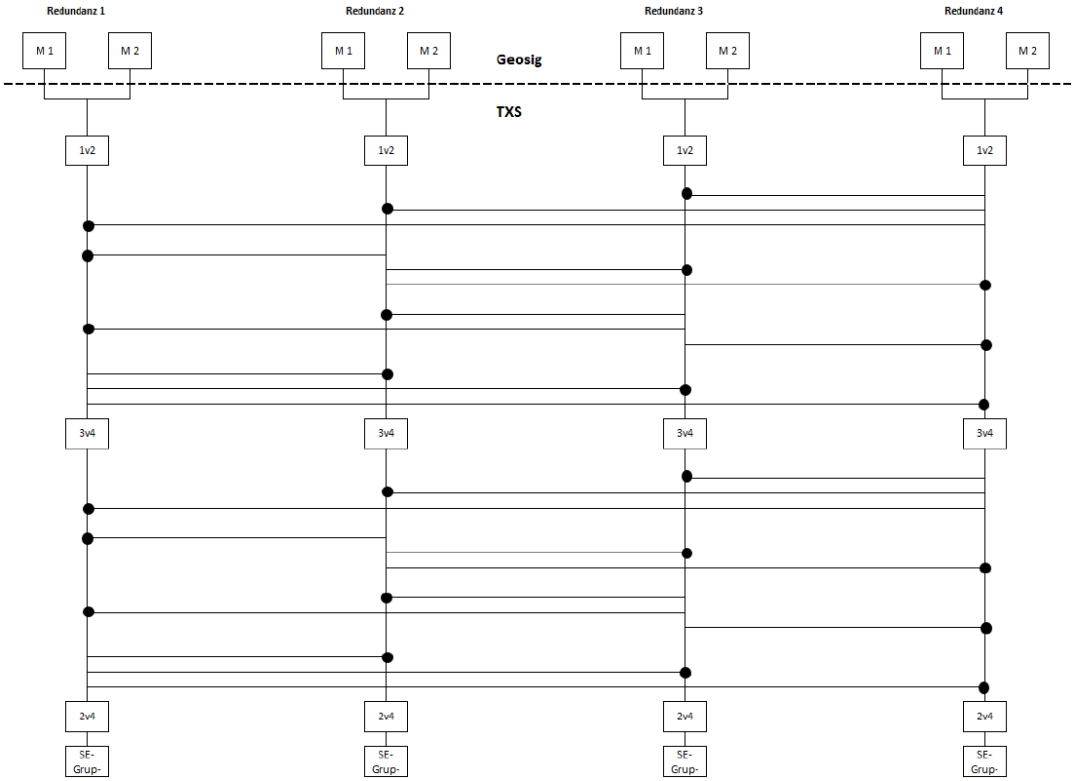

Figure 1. Logic of the ASTS [5]

If one of two measurements pro redundancy train indicates PGA > a_{limit}, the redundancy will generate a signal via the 1/2 logic to the following 3/4 logic for further processing. If 3/4 redundancies have indicates PGA > a_{limit}, then the signal "a_{limit} exceedance" is triggered by the 3/4 logics in the respective redundancies. This signal is passed on to the next logic level characteristic for the reactor power limitation function, where it is checked by 2/4 logic whether at least 2/4 redundancies have exceeded a_{limit}. If this is the case, the control rod group, which is assigned to the redundancy, is inserted. The fulfilment of the 2/4 logic per redundancy train thus results in the insertion of one control rod group (not all control rods). For the complete insertion of all control rods, triggering of the seismic reactor trip in all four redundancies is required.

For the purpose of probabilistic assessment of the failure probability of the ASTS, a fault tree (FT) model was constructed [6]. The evaluation shows that the reliability of the system both in terms of failure per demand as well as with respect to probability of spurious actuation is determined by the TXS-extent of the system, i.e. the unreliability of the GeoSIG data acquisition and measurement transducers is negligible. The overall unreliability of the system thus results from the unreliability of the TXS scope. Thus, the ASTS failure probability per demand is assessed to be 1.4E-5/d [6]. By additional consideration of CCF potential within the TXS software, the final failure probability of the ASTS is calculated to be 1.14E-4/d [5]. The annual probability for spurious actuation of the ASTS is calculated to be 2.1E-4/yr. However, given the implicated conditional core damage frequency (CCDF) of ca. 6E-16/yr, the risk of spurious actuation can be completely neglected.

The integral improvement of the reliability of the seismic reactor trip is implemented as a factor (1.14E-4 instead of 1.0 previously) directly in fragility module in seismic Top Event (TE) SCRA. This top event models the seismic failure probability of the control rods insertion and comprises three different components: the fuel spacer grid, the RPV-internals as well as control rods drive mechanism.

One additional aspect is also being considered as a consequence of the new ASTS, i.e. the human error probability (HEP) psychoshock model due to earthquake (TE: SHEP). Namely, the Swiss regulator prescribes a HEP model in case of earthquake such that: for accelerations PGA<0.2g the HEP values are the same as for those for internal events; for 0.2g < PGA < 0.6g the HEP value increases linearly as a function of the PGA, such that for PGA ≥ 0.6g => HEP = 1.0. This is seen as an especially conservative requirement from the aspect of the operators. The impact of the ASTS on human (operator) behavior is modelled such that in a case of successful rod insertion, the top event SHEP, which models the ENSI-A05 operator action model for NPPs without ASTS, is set to be guaranteed success. In other words, the adaptation of the HEP values, as prescribed by the regulator's guideline A05, is not needed anymore. This corresponds to the situation that in the case of an automatic reactor trip following an earthquake, the operator actions in accordance with the plant's emergency operating procedures (EOP) are executed same as in the case of other plant internal IEs. Mental psychological pressure, complexity and stress do not differ significantly from the situation in other accident scenarios. The available time for OAs are also the same as for other plant internal accidents. This takes into account that the hardware dependencies of the OAs within the KKG-PSA model are explicitly modelled for all OAs, so that the impact of seismically induced technical errors on the success of the OAs are considered.

2.2. Turbine Missile Hazard and Implicated Plant Risk

The potential for main turbine overspeed, and thus a turbine missile event, gets a constant attention in the process industries, and especially heightened awareness in the nuclear industry after the Salem Unit 2 event in 1991. Instead of applying the generic turbine missile probability values available in the public databases (e.g. in [7]), KKG together win an external consultancy company developed its own plant-specific analysis on turbine missile probability as well as the failure probability analysis of the turbine-generator overspeed protection system. The benefits of conducting a plant-specific reliability analysis of specific hazards vis-à-vis the option of using the generic databases are emphasized. Specifically, the results of the turbine missile plant-specific analysis in NPP Goesgen indicate that the turbine missile risk would have been overestimated by at least three orders of magnitude if generic data were to be used.

The two general categories of turbine missile failures are usually referred to as “design overspeed” (up to approximately 120% of the rated speed) failures and “destructive overspeed” (any speed above the design overspeed) failures.

The turbine is designed as single stage high-pressure (HP) and 3-stages low-pressure (LP) turbine. As part of the planned replacement strategy, all 3-LP turbines were replaced by Siemens AG in 2013. In addition to that, a plant-specific probabilistic assessment was performed in order to provide information on rotor burst probability, resulting from hypothetical load case, for use in safety analysis of nuclear power plants. The most significant source of turbine missile is a burst-type failure of bladed LP-rotor. At KKG, turbine blades bursts are not considered as a "turbine missile" event since it is proofed that these blades would be contained within the casings (housing). Hence, only rotor (shaft) bursts are accounted as potential for generating turbine missiles. Failures of the HP and generator rotors would be contained by relatively massive and strong casings, even if failure occurred at maximum conceivable overspeed of the unit. Moreover, these missiles would be much less hazardous than the LP rotor, due to low mass and energy and therefore, was not considered. The most critical load case considered for crack growth failure of LP-rotor is that turbine reaches 120% overspeed during each start-up. This case covers the operating speed and all maximum overspeed excursions, which may occur in normal operation of the unit. Within the discussed study, the rotor rupture probability is defined as the probability of the crack growth to critical flaw size at design overspeed of

120% after 1000 start-up cycles. The Monte Carlo method was used to evaluate this failure probability. Although numerous conservative assumptions were made, the probability of the crack growth to critical flaw size at design overspeed of 120% is estimated to be well below the generic value estimates given by the U.S. NRC and the Swiss regulator. The methodology for evaluation of probability is described in the following sections. The overall probability of turbine missile damage *P* can be calculated as follows:

$$P = P_1 \cdot P_2 \cdot P_3 \tag{1}$$

where P_1 is the probability of external turbine occurrence; P_2 is the probability of missile striking a critical area; P_3 is the probability of damage due to the strike. The focus in this paper is to estimate P_1. The probability P_1 can be calculated as follows:

$$P_1 = P_{1r} \cdot P_{2r} \cdot P_{3r} \tag{2}$$

where P_{1r} is the probability of rotor burst up to 120% of rated speed due to crack growth to critical size; P_{2r} is the probability of casing penetration given a burst of the rotor up to 120% of rated speed; P_{3r} is the probability of turbine running up to 120% of rated speed. For the purpose of this analysis, it is conservatively assumed that P_{2r} and P_{3r} are 1.0.

The rotor rupture probability, P_{1r}, is defined as the probability of the crack growth to critical flaw size at design overspeed of 120% after 1000 start-up cycles. In order to evaluate the failure probability P_{1r}, a Monte Carlo simulation technique involving successive deterministic fracture mechanics calculations using randomly selected value of fracture toughness was used. The results after 1E+7 simulations performed direct a 1E-7 probability for a rotor burst given 1000 start-ups.

This turbine missile probability given design overspeed conditions was subsequently used by KKG to derive the conditional plant-specific failure frequency [8]. In KKG, the conservatively assessed failure frequency is calculated to be:

$$F_{design} = 1E-7/1000d \cdot 3d/y = 3E-10/y \qquad (d = \text{demands}) \tag{3}$$

This conservative estimate is based on the assumptions that not more than 3 relevant transients per year took place on average; and the crack growth rate over the number of load cycles has a linear growth behavior. According to the principles of linear fracture mechanics, crack growth is caused by the stresses that a component undergoes during operation. The analysis [9] describes as the most effective influence the thermal stresses that the turbine is exposed to during the speed and power increase. Opening the turbine control valves increases the flow of steam to the turbine rotor. The outer layer of the rotor is thus raised to the steam temperature, while the middle layer is delayed by heat conduction in the direction of the outer layer temperature. The faster the steam temperature is raised, the greater the resulting stresses. These stresses are kept small taking into account the operation model according to the plant's operating procedures (POP). The temperature display in the main control room limits the permissible speed and power gradients by allowing the possible temperature or power amounts. It is conservatively assumed that these allowances are not credited. This leads to counting of the first effective load cycle as part of the annual start-up of the turbine after the revision, i.e. 1/y. The operational shutdown procedure used for the annual overhaul & refueling outage is rated as follows: Starting with a small power gradient (approximately 5 MW/min), the turbine run down is released after turbine trip (TUSA), i.e. without further steaming. The turbine rotor is therefore not exposed to significant tension. For this reason, the shutdown is not counted as a load cycle. Fault-related transients that lead to load shedding (load shedding to own demand, load shedding by pump failures or faults) are designated as the second effective load cycle with 1/y. This value is calculated as follows from the statistics of the last 20 years and is conservatively derived from the statistics of the last 20 years, in which the plant had 17 such occurrences. Reactor scram (RESA) transients with automatic TUSA or TUSA transients are not counted because the steam flow to the turbine is shut off completely and the shaft cools convectively. The operational starting procedure following all power transients is included as the third load cycle. This leads, according to the transients operating manual, starting the system 0-100%, to the above-mentioned 17 operations, i.e. conservatively 1/y. The complete discussion above, i.e. the values discussed, justify an approach of 3 load cycles per year.

In a subsequent, second-phase and in order to cover the region of postulated “destructive overspeed” failures, analysis of the failure probability of the turbine overspeed protection system was performed [10]. The following two initiator groups can be defined as relevant for the accident scenarios that are related to turbine missile events in the area of destructive overspeed:

IE_1 - all transients and system states where the turbine is not synchronized with the grid or disconnected from the grid. These transients include:

a. Faulty (spurious) opening of the generator circuit breaker (GCB);
b. Faulty (spurious) opening of the block circuit breaker;
c. All the transients related to opening of the block and / or generator circuit breaker by the protective functions;
d. Speed control during the starting process of the turbine.

IE_2 - all transients that lead to TUSA.

The transients of the initiator groups are adopted from the available data (statistics) and conservatively estimated. In the IE_1 transient group of transients no transients of the type a.) or b.) were counted at the KKG. The transients of type c) dominate. The number of these events is 44. The trigger rate is after 38 years of operation at 44/38y = 1.15/y. The operating time of the turbine (disconnected from the power grid) is acc. [11] conservatively estimated at 6 h per year. With 4 independently malfunctioning turbine control valves, the from current KKG PSA model [12] adopted distribution function *TSE1VT* (spurious opening failure rate of a StV=1.63 E-6/h) results in an additional contribution of 4E-5/y. This latter contribution is negligible compared to the other former (1.15/y). Hence, the initiating frequency of group IE_1 is set conservatively with f_1 = 1.25/y. The initiator frequency of group IE_2 is set conservatively to f_2 = 1/y.

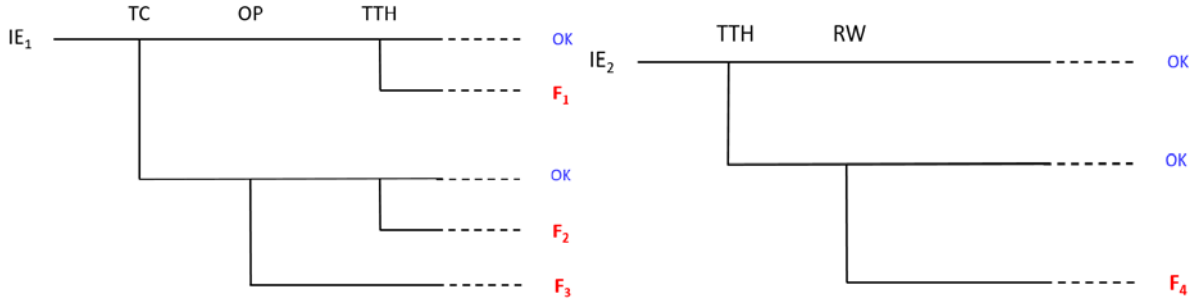

Figure 2. Event trees for the possible scenarios given initiator group IE_1 and IE_2

The TE: "TC" represents the failure probability of the turbine governor SE10 C010. This failure probability is assessed to be 1.5E-4/d. The TE: "OP" represents the failure probability of the turbine overspeed protection system. This failure probability is derived as 1.02E-5/d [10]. The TE "TTH" represents the failure probability of the non-closure of a StV and the assigned turbine stop valve (SSV) of one of the 4 trains. In this top event, the CCF potential has been also studied, and consequently two common cause component groups (CCCGs) have been implemented. The TE "TTH" failure probability is calculated to be 4.40E-8/d. With TE: "RW", the so-called power reversal protection of the generator is modelled here. For a grid disconnection to occur, the generator breaker should spuriously open. In this context, the distribution OG1=2.66E-04/d is adopted for this TE: "RW" [12].

The risk contribution of turbine missile as a result of a overspeed scenario is quantified through the conditional core damage frequency (CCDF). Taking into consideration the ET for the first initiator group, depicted on Figure 2, the following value can be derived for the failure frequency of the turbine due to destructive overspeed conditions:

$$F_{TZK1} = F_1 + F_2 + F_3 \approx IE_1 \cdot \left[Q(TTH) + Q(TC) \cdot Q(TTH) + Q(TC) \cdot Q(OP) \right] \approx 5.7E-08/y \quad (4)$$

Regarding the ET for the second initiator group, depicted on Figure 2, the following value can be derived for the failure frequency of the turbine due to destructive overspeed conditions:

$$F_{TZK2} = F_4 \approx IE_2 \cdot Q(TTH) \cdot Q(RW) \approx 1.2E-11/y \quad (5)$$

Both the F_{design}, related to the turbine missile frequency due to design overspeed scenario, as well as the F_{TZK2} are negligible in comparison to the F_{TZK1}. Hence, the CCDF is calculated for F_{TZK1}. In this sense, it is conservatively assumed that at a destructive overspeed, the rotor debris will likely penetrate the casings and exit. The affected buildings in which safety relevant SSK are housed, are the following: the electrical building - ZE, the emergency diesel generator 1&2 building - ZK01, as well as the emergency feedwater injection building ZV. Furthermore, it is conservatively assumed that all three buildings are hit simultaneously and all PSA-relevant SSK are destroyed with a conditional probability of 1.0. The resulting CCDF given the reference model *REF1* becomes:

$$CCDF\left(REF0 : IE -> F_{TZK1}\right) \approx 2.3E-11/y \quad (6)$$

With this quantitative estimate, it has been shown that the CDF contribution is below 1E-9/y. Consequently, the risk of turbine missile due to a destructive overspeed can be screened out according to the Swiss Regulator [1]. In contrast to this turbine missile frequency F_{TZK1}, calculated based on plant-specific analysis, the U.S. NRC [6] prescribes a frequency of ca. 1E-4/yr. This frequency is used later in this paper, for the purpose of comparative analysis within section 3.

2.3. Tornado Hazard and Implicated Plant Risk

The Swiss regulator requires a probabilistic evaluation of tornado hazards and tornado-induced failures as part of the plant's PSA.

Table 1. Frequency of tornadoes impacting the KKG site - ENSI A05 edition 2009

Tornado scale	A05 Data 2009				KKG site calculations		
	Frequency† [/yr]	Annual frequency (events per /yr km^2)	Width of tornado [km]	Max travel of tornado [km]	Freq. of tornadoes within site [/yr]	Freq. of tornadoes outside the site impacting the site [/yr]	Total frequency of tornadoes impacting the site [/yr]
F0 and F1	2.30E+0	1.84E-4	0.07	3.8	2.55E-5	2.97E-4	3.225E-4
F2	2.20E-1	1.76E-5	0.15	5.1	3.46E-6	4.53E-5	4.876E-5
F3	6.30E-2	5.04E-6	0.32	19	1.78E-6	6.44E-5	6.618E-5

Table 2. Frequency of tornadoes impacting the KKG site - ENSI A05 edition 2018

Tornado scale	A05 Data 2018				KKG site calculations		
	Frequency† [/yr]	Annual frequency (events per /yr km^2)	Width of tornado [km]	Max travel of tornado [km]	Freq. of tornadoes within site [/yr]	Freq. of tornadoes outside the site impacting the site [/yr]	Total frequency of tornadoes impacting the site [/yr]
E0	1.54E+0	1.23E-04	0.035	2.6	1.43E-05	1.250E-04	1.393E-04
E1	6.91E-1	5.53E-05	0.082	6.9	8.11E-06	1.605E-04	1.686E-04
E2	1.99E-1	1.59E-05	0.124	10.2	2.81E-06	7.725E-05	8.006E-05
E3	5.81E-2	4.65E-06	0.343	17.5	1.75E-06	5.663E-05	5.838E-05
E4	1.30E-2	1.04E-06	0.383	23.1	4.39E-07	1.767E-05	1.811E-05
E5	1.25E-3	1.00E-07	0.45	53.4	5.03E-08	4.279E-06	4.329E-06

† Defined over an area of 12,500 km^2

The occurrence of tornadoes should be assumed to be uniformly distributed with a rectangular area of 12,500 km^2 around the Swiss NPPs. Within its 2009 edition of the A05 guideline, the Fujita scale was prescribed by the regulator. Table 1 summarizes the specifications of the tornado hazards for the KKG. Then, in 2018 there was a new edition of the same guideline. In the newer one, the extended Fujita scale is required to be considered (Table 2). It is however interesting to compare both of these tornado hazards, prescribed by the Swiss regulator for the Swiss NPPs, against the ones assessed for the US NPPs. The following figure depicts the comparison between the latest tornado hazard, i.e. ENSI 2018, and the U.S. NRC tornado hazard assessment for the U.S. NPPs [13].

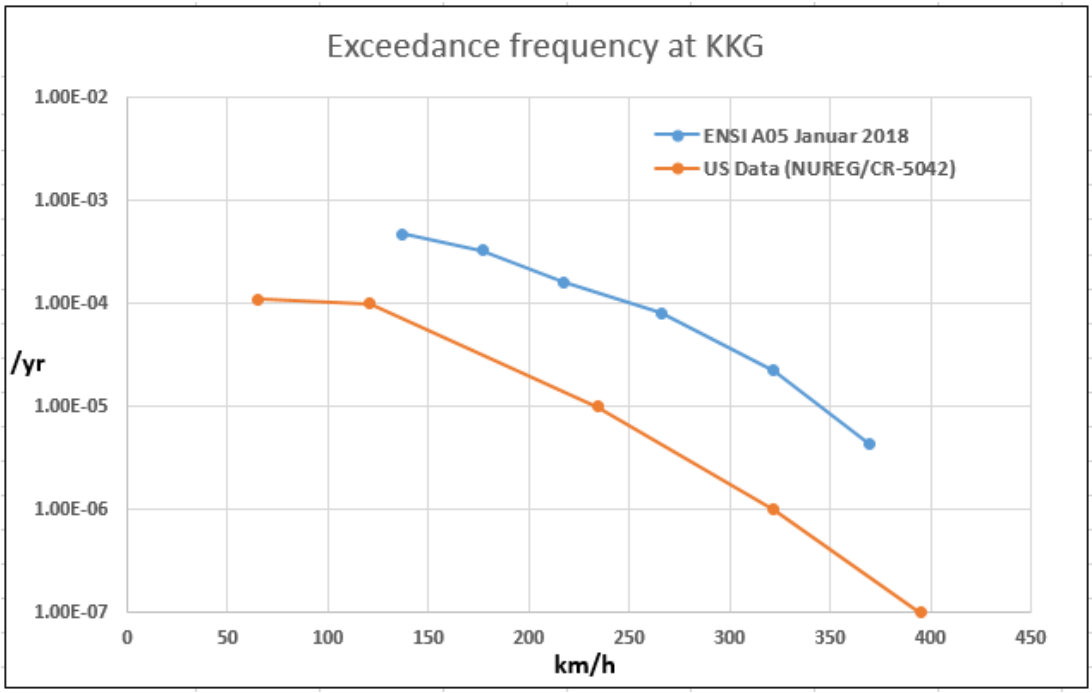

Figure 3. Comparison between the latest ENSI A05 and the U.S. NRC tornado hazard assumption

2.4. Seismic Hazard and Implicated Plant Risk

As part of the continuing development of the safety analyses of Swiss nuclear power plants, in 1999 the Swiss regulator called on the nuclear operators to redefine the seismic hazard according to the most advanced methodological fundamentals, and in particular comprehensively quantify the fuzziness of the calculation results. Hence, the PEGASOS project was initiated by the NPPs operators. The ambitious project was founded on the strictest requirements of a method newly developed in the USA at the time. To date, no other European country has carried out a study of this kind.

In its concluding statement on the PEGASOS project, the regulator came to the conclusion in 2004 that the PEGASOS project met the methodological requirements and even achieved a new state of the art with regard to various aspects (quality assurance, extension of the method to characterization of site effects). However, the regulator also noted that the range of uncertainties reported in the PEGASOS results is quite large and could be reduced by further investigations. With the aim of reducing the fuzziness of the PEGASOS results, in 2008 the nuclear power plant operators launched the PEGASOS Refinement Project (PRP). The PRP took into account newly available findings from earthquake research and the results of new measurements of seismological soil properties at the nuclear power plant sites. The review by ENSI showed that the PRP as a whole had made significant progress and that the project was a significant step forward, particularly with regard to the main project priorities. However, in the course of the review, it has become increasingly apparent that although earthquake characterization has progressed, key issues have not been adequately addressed.

As a result, the reported earthquake hazard results could not be accepted. Consequently, a new, modified earthquake hazard assumption (ENSI-2015) was prescribed by the regulator in 2015. Figure 4 depicts the comparative analysis among the three different hazard assumptions for the KKG site.

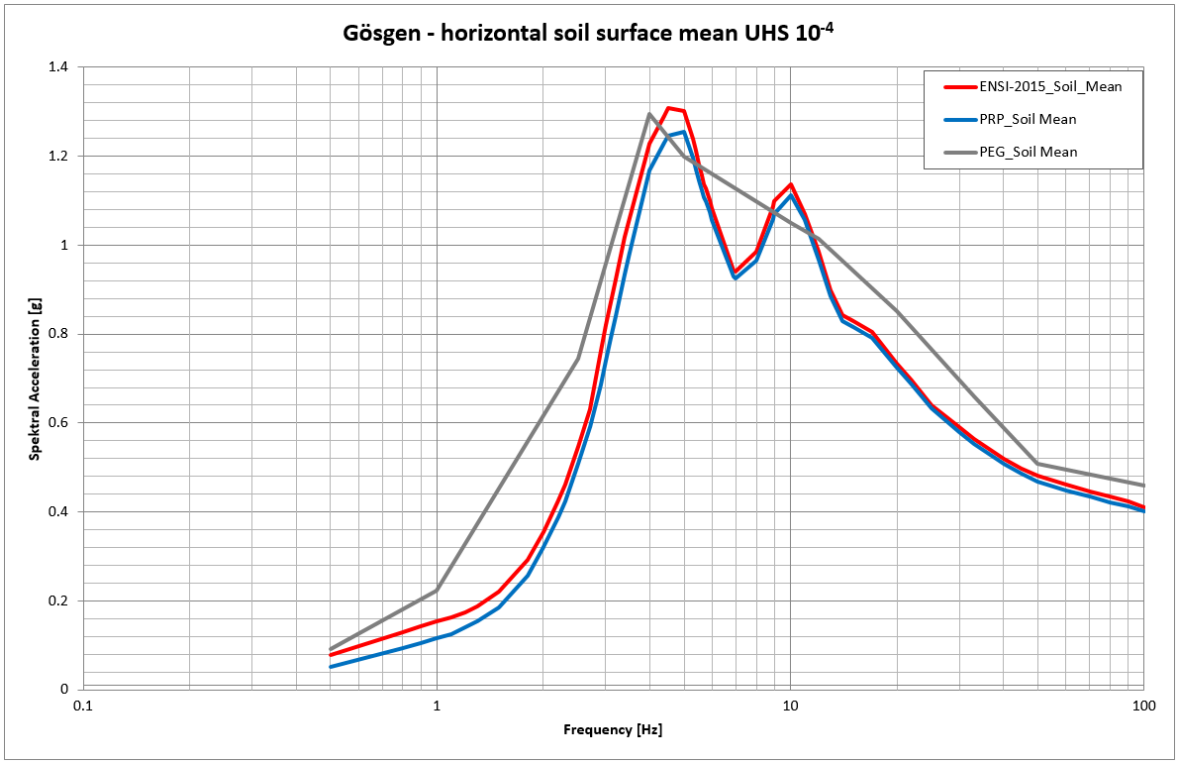

Figure 4. Spectral acceleration comparison among the PEGASOS, PRP and ENSI-2015 for KKG site

3. MODELS DESCRIPTION AND QUANTIFICATION

Based on the analysis assumptions made in the previous chapter, a comparative analysis is conducted among the following models.

i. ***REF0*** - a basis reference model for the comparative purposes within this paper; It is based on the PEGASOS seismic hazard assumption, no ASTS considered and as a consequence the psychoschock SHEP model regarding the OA HEPs is present; The turbine missile hazard is being screened out based on the plant-specific assessment of the risk implications; Regarding the tornado hazard - the older (ENSI 2009) is considered;

ii. ***REF1*** - a somehow adapted reference model based on *REF0*, such that conservatively the turbine missile hazard prescribed by the U.S. NRC is considered, i.e. the turbine missile risk is not screened from the model; In addition, the latest tornado hazard (ENSI 2018) is considered;

iii. ***REF2*** - based on REF1, with an additional conservative extension, given the discussion in section 2.1 (stuck-open first PSV in case of earthquakes < 0.6g);

iv. ***MODEL1*** - first of the two models where the benefits of realistic, best-estimate and plant-specific analysis are included within the PSA modelling assumptions; Namely, based on *REF0* with the difference that the ASTS is implemented and consequently, the SHEP impact is removed;

v. ***MODEL2*** - second of the two models where the benefits of realistic, best-estimate and plant-specific analysis are included within the PSA modelling assumptions; Namely, based on *MODEL1* with the difference that the ENSI-2015 seismic hazard is implemented instead of the PEGASOS.

After defining the models of interest, each with its specifics, they are quantified. Table 3 summarizes the results of the quantification.

Table 3. PSA models quantification

REF0		***REF1***		***REF2***		***MODEL1***		***MODEL2***	
CDF [/yr]	LERF [/yr]	CDF [/yr]	LERF [/yr]	CDF [/yr]	LERF [/yr]	CDF [/yr]	LERF [/yr]	CDF [/yr]	LERF [/yr]
2.038E-5	8.914E-6	2.061E-5	8.924E-6	1.428E-4	1.468E-5	1.299E-5	2.900E-6	9.411E-6	2.035E-6

3. RESULTS: PSA MODELS RELATIVE COMPARISON

The L1 and L2 PSA risk measure, i.e. CDF and LERF respectively, are applied as comparative measure among the different models and the comparison is depicted on Figure 5.

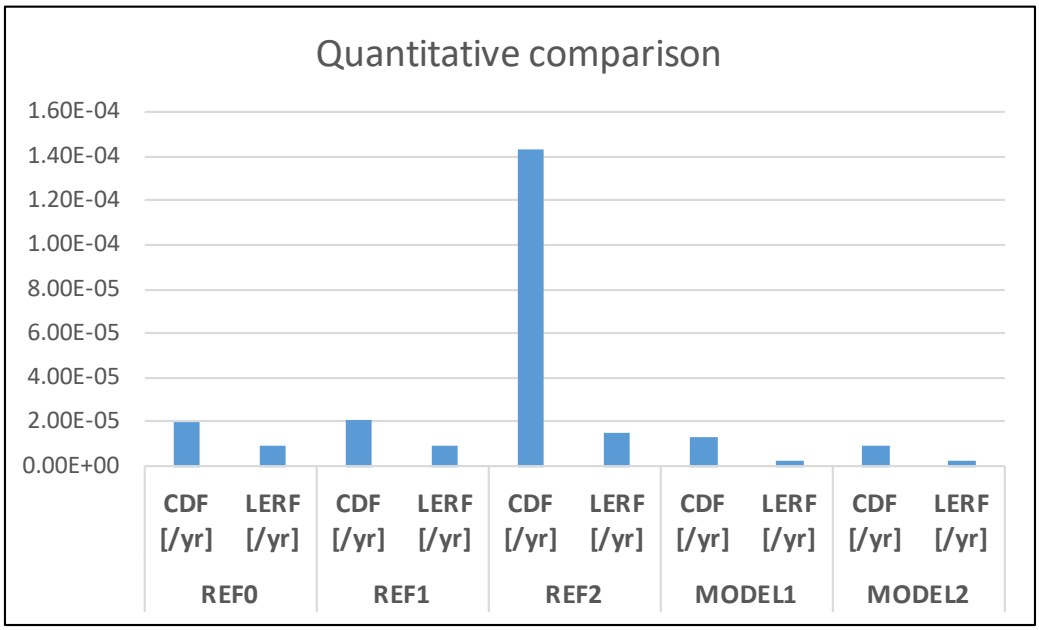

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Figure 5. L1 & L2 PSA quantitative comparison

From the comparison, it is obvious that the highest difference in risk is observable between the *REF2* and any of the other models. This is due to the fact that the extended reference model, *REF2*, is quite conservative from the aspect of seismic accident progression assumptions (stuck-open PSV in case of smaller earthquake). Also, when the *REF0* and *REF1* are compared with each other, it is obvious that for the KKG PSA model, the effect of the conservative assumptions regarding the turbine missile frequency and the new tornado hazard is negligible (ca. 1.5 % rel. difference). However, the idea behind this was to show the potential benefits of installing the ASTS (*MODEL1*), and, consequently removing the impact of the psychoshock model SHEP on the OA HEPs. In this direction, the relative difference in risk reduction (in aspect of CDF) is ca. one order of magnitude. Further on, if the *MODEL1* is compared against *MODEL2*, then a relative risk difference, i.e. CDF reduction of ca. 28% is observable. This difference is mainly due to the change of seismic hazard assumption, i.e. changing form PEGASOS to the ENSI-2015 seismic hazard.

4. CONCLUSIONS

This paper presents a comparative analysis case-study considering best-estimate and plant-specific PSA models vis-à-vis generic, out-of-date and, consequently, conservative hazard assumptions and modelling requirements. The analysis is organized into several parts, where the various assumptions and modelling considerations are described. Firstly, the implementation of the ASTS with associated triggering of a turbine trip as an additional limiting system is described as well as the modelling considerations within the KKG model and the beneficial consequences in terms of plant risk reduction are discussed and quantified. Later on, the turbine missile hazard and the specificity of this hazard to the Goesgen NPP via conduction of plant specific analysis is being addressed. A comparison against the out-of-date, generic databases is being conducted. Thirdly, the impact of the newly suggested tornado hazard by the regulator is considered. At the end, seen as a fourth point, the newly suggested seismic hazard is compared against the previous one. Again, a comparative analysis of the plant PSA model is performed given the two different hazard assumptions.

The results direct, in the first line, a considerable risk reduction that can be achieved by the instalment of the ASTS. In addition to that, the effect of the new tornado hazard as well as the conservative, generic turbine missile failure frequency are being considered for comparative purposes. Although the differences (in comparison with the old hazards) are much higher, the plant seems so exhibit a high resilience to these changes. Namely, the change in plant risk in in the order of ca. 1.5%. Their effect on the KKG risk given the plant's PSA model seems to be negligible. Further on, the change of plant risk given the implementation of a new seismic hazard (which is somehow more favourable for the KKG site than the previous one) is investigated. A considerable reduction in risk of ca. 28% (in terms of CDF) is observable by implementing this new seismic hazard instead of the old one.

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