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Ageing PSA as a support for effective ageing management

Keywords

ageing, ageing management, probabilistic safety assessment, prioritization of management activities

Abstract

Because of the growing operational age of nuclear power plants, the ageing management of structures, systems and components used in these plants is gaining an important role. Technical systems are subject to time-dependent and operationally caused ageing phenomena with modifications of originally given characteristics and, thus, of relevance in terms of safety. Especially physical ageing is of importance. Therefore, a comprehensive ageing management is required. In the context of an integrated safety management it has to be shown how to integrate the safety related issues of ageing into probabilistic safety assessment (PSA). In particular the question is to be answered whether the effort for the execution of an ageing PSA is justified, in particular if the safety significant effects of ageing can be identified and quantitatively estimated. Method for prioritization of the components in the nuclear power plant considering implication of their ageing on safety of the nuclear power plant is presented. On the basis of an actual report on ageing management in German nuclear power plants and a literature survey, this paper tries to estimate the necessity and value for the introduction of an ageing PSA in Germany.

1. Introduction

As of February 15, 2014, 435 nuclear power plant units are in operation in 30 countries. The data in *Figure 1* (as provided by the International Atomic Energy Agency - IAEA) show that the vast majority of nuclear power plants worldwide is more than 20 years old and more than 150 units are already more than 30 years in operation.

Moreover, many countries are currently extending or planning to extend the operating lifetimes of their nuclear power plants to 60 years or even more. Therefore, it seems to be of great significance to identify the possible impact of ageing on the safety in the long term.

Currently, 73 nuclear power plants in the US have already received 20-year operational life-time extensions, another 18 units are currently under review by U.S. NRC [17]. Several applications for a licence extension beyond 60 years are expected before 2020.

Several potential ageing effects on reactor pressure vessels, piping, cables and plant concrete structures must be considered by the regulators and addressed by industry in order to assure plant safety [17].

In anticipation of subsequent license renewal applications, the staff undertook a comprehensive review. Categories of items considered during this review included the use of probabilistic safety assessment (PSA) to risk-inform scoping and management of ageing effects [27].

For the units which have approached the end of initial design lifetime and especially for those which are planning to extend the lifetime, it has to be demonstrated that the plant safety level will remain adequate until the end of operation, and to do that, is necessary to evaluate the effects of ageing phenomena on the plant performance and safety.

Ageing, which could be understood in this context as a „general process in which characteristics of components, systems and structures ('equipment') gradually change with time or use, eventually leads to degradation of materials subjected to service conditions and could cause a reduction in component and systems safety margins" below limits provided in plant design or regulatory requirements." [1].

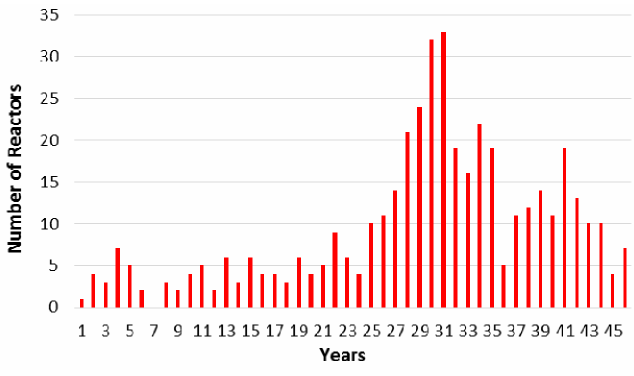

Figure 1. Number of operating reactors worldwide by age end of 2013; age of a reactor is determined by its first grid connection

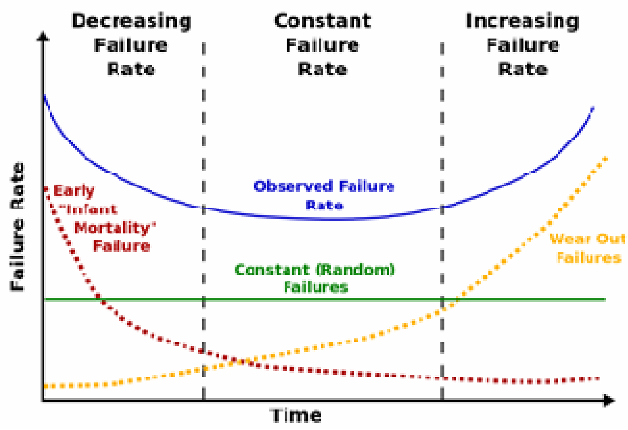

Figure 2. Failure rate

Ageing phenomena can have one or both of the following effects [1]: The failure rate of component or a system can increase with time or the components degrade and no longer fulfil design requirements.

Especially physical ageing is of importance. They are caused by, e.g., embrittlement, fatigue, corrosion, wear and tear or by a combination of these factors.

Consequently, the effective failure rate is described by the so called "bathtub" curve presented in *Figure 2*, which comprises of three parts: "infant mortality" phase, period of normal operation and wear-out phase. It is generally assumed that ageing is taking part in the third phase [2].

Ageing management in nuclear power plants comprises the entirety of measures taken to control the above mentioned ageing phenomena that could be detrimental to the safety of a nuclear power plant.

In Germany the recently issued nuclear safety standard KTA 1403 [16] specifies the requirements for ageing management that encompass the technical and organizational measures with respect to an early detection of ageing phenomena relevant to the safety of nuclear power plants and to maintaining the actually required quality condition. It is underlined that the development of the state of the art of science and technology regarding ageing shall be pursued and assessed.

Additionally, in Germany it is required by law that the licensees have to conduct an overall periodic safety review of the operating nuclear power plant every ten years. Part of this periodic safety review is to perform a level 1 probabilistic safety assessment (PSA) for all operating states and a level 2 PSA for full power.

The probabilistic safety assessment (PSA) is seen as a complement to the classical deterministic safety assessment and, in the meantime, PSA is used as a tool to assess the safety level of a plant in the frame of international and national licensee and regulatory activities.

In the context of an integrated safety management the question arises if it is possible also to integrate the safety related issues of ageing into PSA. So far, the ageing effects are not explicitly included in the PSA provided today in the frame of periodic safety reviews.

2. Incorporating ageing into PSA

In 2004 the so-called "Ageing PSA (APSA)" project of Joint Research Centre (JRC) of the European Commission started. The APSA Network was created after it was recognized that current standard PSA tools do not adequately address important ageing issues, and this could have a significant impact on the conclusions drawn from PSA studies and applications, especially in cases of operational aged plants. The network brings together operators, research institutions, industry and consultants, who have their own research program in the area or are interested by the subject [8].

In 2006 a preliminary report was published by the APSA project, summarizing the general procedure of incorporating ageing effects into PSA [1]:

- Selection of structures, systems and components sensitive to ageing,
- Modelling of ageing mechanisms,
- Modification of PSA parameters/models and calculation of PSA results.

The first step consists of selecting relevant structures, systems and components (SSCs) which shall be considered. Detailed analysis of every single component in a plant would be too difficult and extremely expensive. In addition to the analytical complexity, data gathering also involves a major effort, particularly for projecting the effects of ageing in the future.

Therefore, a “simplified method that screen out less risk-significant components and prioritize the most risk-significant ones would be very useful in this context” [1]. The risk-significance can be estimated by looking at, for example, the influence of the reliability of a particular component on the overall PSA results.

A qualitative approach for the selection of SSCs is described in [13] including a list of ageing mechanisms.

Moreover, a guideline has recently been issued [14] providing a practical approach and recommending methods to be used in the selection and prioritization of SSCs sensitive to ageing. The guideline proposes a simplified approach for SSC selection by using an integrated decision-making process which incorporates both risk and traditional engineering judgment as given by an expert panel [14]-[15].

The results of the application of this simplified approach carried out for two particular systems of the TRIGA research reactor are given in [14] and demonstrate the feasibility of the approach and it allows to identify components that are important from the risk point of view but are not much sensitive to ageing (pneumatic valves from primary circuit) and components that are not important for safety but are very sensitive to ageing [15].

The second step consists of modelling the ageing mechanisms, e. g. embrittlement, fatigue and so forth.

The third step consists of modifying the PSA parameters or the structure, since one now can have components which were not considered previously in a "classical" PSA, e. g. passive components.

Two main options exist when the basic methods for consideration of ageing in probabilistic safety assessment are considered.

The first option, which is known as stepwise constant failure rates, includes modification of probabilistic safety assessment models in sense that the ageing contribution is added to the initial models, which consequently causes also the modified results, when evaluation is performed [26]. The failure rates are determined as constant in determined time intervals, but as the time intervals go on, the failure rates increase, if the ageing contribution to the failure rates increases [24]. The second option includes modification of the resulted minimal cut sets in sense that the ageing contribution is added to the resulted minimal cut sets [25]-[26].

A case study on incorporation of ageing effects into the PSA model and a discussion on the use of PSA to evaluate the SSC ageing effect on overall plant safety are provided in [22] as part of task 7 of the APSA project. The possible impact of age-related degradation on the component reliability and on the plant risk profile is demonstrated using the PWR Large LOCA PSA model as an example.

The incorporation of age-dependent reliability parameters and data of SSC into the PSA model is also addressed in [18] presenting the application of implicit and explicit reliability models for incorporating ageing effects. These are dynamic system reliability methodologies such as the GO-FLOW chart and the Petri-nets-based method-Analysis of Topological Reliability of Digraphs (ATRD) as a cell-to-cell analysis procedure.

These alternative methodologies have been applied to a repairable pump and a non-repairable check valve in a segment of the VVER 1000 safety system fault tree in order to evaluate the ageing impact on the unavailability of the residual heat removal system and the unavailability sensitivity [19]. The ageing degradation and restoration have been modelled using linear functions.

Furthermore, a new analytical model has recently been developed [9]. The main advantage of this model is that it simultaneously integrates the contributions of component ageing, effects of test and maintenance activities as well as the test strategy (sequential, staggered) in deriving the system mean unavailability.

The obtained results generally indicate the fact that risk-informed surveillance requirements differ from existing ones in technical specifications as well as show the importance of considering ageing data uncertainties in component ageing modelling.

Internationally, there is no standardized single approach to incorporate ageing into PSA. The greatest obstacle seems to be the sparse empirical data and resulting from that, large uncertainties. Especially the chosen initial assumption (e. g. “same-as-old” vs. “same-as-new”) can have drastic impact on the results [10].

Further difficulties arise in the modelling of ageing mechanisms. “[It] is difficult to distinguish equipment failures and equipment failures, [whose] causes are connected [to] degradation due to ageing” [6]. On the other hand “it is difficult to define the basic elements of the evaluation, which are the components themselves, as they are mostly made of several parts or subcomponents, which may degrade through time and age differently…” [6].

3. Data availability and analysis

In order to conduct a PSA, one needs some sort of reliability data of studied components, for example in form of failed components in a given operation time interval (failure rate). One way of obtaining this data is operational experience. Ageing effects, however, can result in an increasing failure rate. The ageing PSA tries to model this time dependence of the failure rate, in contrast to the "classical" PSA where the failure rate is assumed to be constant. Several models exist, the simplest being the so called linear

ageing, where the constant ageing is assumed to be described by a linear function instead.
In general, there are different types of ageing which can occur with different types of age dependent failure rates $\lambda(w)$. The linear method, the exponential method and the Weibull method are the most common methods used for modelling of components ageing.
Data obtained from operational experience can be studied using these models and the resulting time-dependent failure rates can be used to compute the associated unavailability of the considered structure, system or component.
Practical methods to analyze component and system reliability data with focus on frequentist and Bayesian approaches are discussed in guideline elaborated in the APSA framework [23].

4. Method of prioritization of ageing from results of probabilistic safety assessment

Method for prioritization of the nuclear power plant components due to ageing based on the results of probabilistic safety assessment is developed and presented. Ageing is a process, where the properties of systems and processes may degrade through the time and age.
The change of the core damage frequency for different replacement and surveillance intervals is calculated from the new importance coefficient. Components are sorted depending on their contribution to the change of the core damage frequency resulting from their ageing.
The method of assessment of ageing from results of probabilistic safety assessment is presented in reference [26], while the data are analysed also in [24]. The mathematical formulation of the method is based on the TIRGALEX database [24] about components ageing rates. The change of the failure rate $\Delta\lambda_i$ of component i due to the ageing is given with expression:

$$\Delta\lambda_i = \lambda_i - \lambda_{i0} \quad (1)$$

where
λ_{i0} - failure rate of equipment i (no ageing considered)
λ_i - failure rate of equipment i with ageing considered
$\Delta\lambda_i$ - the increase of failure rate of equipment i due to ageing.

The change of the component unavailability Δq_i with consideration of the ageing is given as:

$$\Delta q_i = q_i - q_{i0} \quad (2)$$

where
q_{i0} - unavailability of equipment i (no ageing considered)
q_i - unavailability of equipment i with ageing considered
Δq_i - the increase of unavailability of equipment i due to ageing.

For a linear ageing failure rate, the average unavailability increase [25] due to the ageing for tested equipment is:

$$\Delta q_i = \frac{1}{4} a_i (L_i - T_i) T_i + \frac{1}{6} a_i T_i^2 \quad (3)$$

where
a_i - ageing rate of equipment i
T_i - test interval of equipment i
L_i - replacement (overhaul) interval of equipment i.

The overhaul or replacement interval L is the interval at which the component is replaced with a new one and the age of the component is restored effectively to a value of zero.
The surveillance interval T is interval at which the component surveillance is performed, in order to assure operational status with minimal repair being performed. The component is basically in the same condition after the test as before the test. The replacement interval of equipment i (L_i) is obtained [25] as:

$$L_i = \frac{1}{\lambda_{i0}} \quad (4)$$

If there is no surveillance tests expected on the component between replacements, then T is set equal to L. In the case when the mean time to failure of the component is larger than the facility lifetime and there is no surveillance test expected the formula for the unavailability increase [25] will be:

$$\Delta q_i = \frac{1}{2} a_i t_0^2 \quad (5)$$

where t_o - facility lifetime.

To calculate the core damage frequency change ΔCDF as a function of the component ageing changes Δq_i, a Taylor expansion approach was utilized to express ΔCDF as a function of the Δq_i:

$$\Delta CDF = \sum_i S_i \Delta q_i + \sum_{i>j} S_{ij} \Delta q_i \Delta q_j + \sum_{i>j>k} S_{ijk} \Delta q_i \Delta q_j \Delta q_k + ... + S_{12..n} \Delta q_1 \Delta q_2 ... \Delta q_n \quad (6)$$

where

ΔCDF - change in core damage frequency

S_i - standard Taylor expansion coefficients, importance of equipment i

Δq_i - change of the component/system unavailability.

The Taylor expansion coefficients S_i in Equation (6) are obtained as multi order derivates of the *CDF* and are termed as a core damage frequency sensitivity coefficients or core damage frequency importance coefficients.

With the consideration of the first order Taylor coefficients the Equation (6) is simplified as:

$$\Delta CDF = \sum_i S_i \Delta q_i \quad (7)$$

Change of the core damage frequency ΔCDF_i resulting from the change of the component unavailability Δq_i will be:

$$\Delta CDF_i = S_i * \Delta q_i \quad (8)$$

The Fussel-Vesely (*FV*) importance measure gives fractional contribution to the system unavailability.

$$FV_i = \frac{CDF - CDF(q_i = 0)}{CDF} \quad (9)$$

The importance coefficients S_i of equipment i can be obtained from the Fussel-Vesely importance measure as:

$$S_i = \frac{FV_i * CDF}{q_i} = \frac{CDF - CDF(q_i = 0)}{q_i} \quad (10)$$

where

CDF - core damage frequency

$CDF(q_i=0)$ - core damage frequency when unavailability of equipment i is set to zero

FV_i - Fussel-Vesely importance measures for equipment i

q_i - unavailability of equipment i.

The relative error of expression given with Equation (10) can be large in case when the values of *CDF* and $CDF(q_i = 0)$ are close to each other indicated by the small value of *FV* importance measure [5].

Results of prioritization of ageing from results of probabilistic safety assessment are presented in *Table 1* and *Table 2* [6]. Results include change of core damage frequency due to ageing for different test intervals (*T*) and replacement intervals (*L*), given in months, for a selected probabilistic safety assessment model. The reference nuclear power plant used in the study has a core damage frequency of 1.00E-05/reactor year.

The first column in these tables contains identification of the event which includes: number of the basic event and type (Common Cause Failure (*CCF*), Diesel Generator (*DG*), Emergency Core Cooling System (*ECCS*), Motor Operated Valve fails to remain open (*EMSMV*) or fails to operate (*ESIMV*)).

In the second column the basic event unavailability with the Fussel-Vesely importance measure (*FV*) is provided and ageing rates are given in the third and fourth column.

The fifth column contains the value of the sensitivity coefficient S_i of the corresponding basic event. The value of the sensitivity coefficient S_i is obtained with application of Equation (10) using basic event unavailability q_i and FV_i given in first and second column of *Table 1*.

In the sixth column the average unavailability increase Δq_i due to the ageing is given which is obtained with the application of Equation (3) for a given surveillance interval *T* and replacement interval *L*.

The last column in *Table 1* contains ΔCDF_i representing the increase of the *CDF* resulting from the ageing of the particular component. The ΔCDF_i is obtained with the application of Equation (8) and is product of sensitivity coefficient S_i and unavailability increase Δq_i of the corresponding basic event given in previous two columns.

Basic events in *Table 1* and *Table 2* are sorted based on the value of ΔCDF_i [6], [28].

Table 1 and *Table 2* contain only 10 basic events, which have largest ΔCDF_i.

The results in *Table 2* show that basic event identified with the largest ΔCDF_i is the event corresponding to the *CCF* of the diesel generators. Other basic events identified in *Table 2* correspond to the failure of the valves of the *ECCS*, *DC* bus of class 1E power system and valves of the components cooling system [6].

Table 1. Results of consideration of ageing

Basic event	q_i	FV_i	α_i	S_i	T = 1m, L = 18m	
					Δq_i	ΔCDF_i
BE01 (*CCF DG*)	2.81E-05	3.05E-03	4.11E-10/h^2	1.09E-03/ry	9.41E-04	1.02E-06/ry
BE02 (*ECCS* valve)	1.20E-05	7.81E-04	4.11E-10/h^2	6.51E-04/ry	9.41E-04	6.12E-07/ry
BE03 (*ECCS* valve)	2.40E-06	1.56E-04	4.11E-10/h^2	6.50E-04/ry	9.41E-04	6.12E-07/ry
BE04 (*DC* bus)	2.40E-05	8.83E-03	3.43E-11/h^2	3.68E-03/ry	7.84E-05	2.88E-07/ry
BE05 (*ECCS* valve)	2.40E-06	3.73E-05	4.11E-10/h^2	1.55E-04/ry	9.41E-04	1.46E-07/ry
BE06 (*ECCS* valve)	1.20E-05	1.86E-04	4.11E-10/h^2	1.55E-04/ry	9.41E-04	1.46E-07/ry
BE07 (*ECCS* valve)	2.40E-06	3.13E-05	4.11E-10/h^2	1.30E-04/ry	9.41E-04	1.23E-07/ry
BE08 (*ECCS* valve)	1.45E-03	1.89E-02	4.11E-10/h^2	1.30E-04/ry	9.41E-04	1.23E-07/ry
BE09 (*ECCS* valve)	2.96E-05	2.83E-04	4.11E-10/h^2	9.56E-05/ry	9.41E-04	9.00E-08/ry
BE10 (*PCS* valve)	1.45E-03	1.29E-02	4.11E-10/h^2	8.90E-05/ry	9.41E-04	8.37E-08/ry

Table 2. Results of consideration of ageing (continued)

Basic event	T = 18m, L = 18m		T = 1m, L = 72m		T = 6m, L = 72m		T = 72m, L = 72m	
	Δq_i	ΔCDF_i	Δq_i	ΔCDF_i	Δq_i	ΔCDF_i	Δq_i	ΔCDF_i
BE01	1.15E-02	1.25E-05/ry	3.82E-03	4.14E-06/ry	1.28E-01	1.39E-04/ry	1.84E-01	2.00E-04/ry
BE02	1.15E-02	7.49E-06/ry	3.82E-03	2.48E-06/ry	1.28E-01	8.32E-05/ry	1.84E-01	1.20E-04/ry
BE03	1.15E-02	7.48E-06/ry	3.82E-03	2.48E-06/ry	1.28E-01	8.31E-05/ry	1.84E-01	1.20E-04/ry
BE04	9.59E-04	3.53E-06/ry	3.18E-04	1.17E-06/ry	1.07E-02	3.92E-05/ry	1.53E-02	5.64E-05/ry
BE05	1.15E-02	1.79E-06/ry	3.82E-03	5.93E-07/ry	1.28E-01	1.99E-05/ry	1.84E-01	2.86E-05/ry
BE06	1.15E-02	1.78E-06/ry	3.82E-03	5.92E-07/ry	1.28E-01	1.98E-05/ry	1.84E-01	2.85E-05/ry
BE07	1.15E-02	1.50E-06/ry	3.82E-03	4.98E-07/ry	1.28E-01	1.67E-05/ry	1.84E-01	2.40E-05/ry
BE08	1.15E-02	1.50E-06/ry	3.82E-03	4.98E-07/ry	1.28E-01	1.67E-05/ry	1.84E-01	2.40E-05/ry
BE09	1.15E-02	1.10E-06/ry	3.82E-03	3.65E-07/ry	1.28E-01	1.22E-05/ry	1.84E-01	1.76E-05/ry
BE10	1.15E-02	1.02E-06/ry	3.82E-03	3.40E-07/ry	1.28E-01	1.14E-05/ry	1.84E-01	1.64E-05/ry

Table 3. Summary of results due to consideration of ageing

T	L	ΔCDF
T = 1m	L = 18m	4.58E-06/ry
T = 18m	L = 18m	5.60E-05/ry
T = 1m	L = 72m	1.86E-05/ry
T = 6m	L = 72m	6.22E-04/ry
T = 72m	L = 72m	8.96E-04/ry

In *Table 3* the overall increase of the core damage frequency ΔCDF, obtained with Equation (7), for given surveillance interval T and replacement interval L is presented. Obtained results in *Table 3* show that extension of the test interval T and replacement interval L results in an increase of the ΔCDF.

The increase of the core damage frequency ΔCDF is relatively large compared to the baseline CDF of CDF = 1.00E-05/reactor year.

The sensitivity and uncertainty of the ΔCDF due to the changes and uncertainties of other parameter is not investigated within this study [28]-[29].

5. Evaluation of the ageing management of the structures, systems and components in German nuclear power plants

Ageing management in nuclear power plants comprises the entirety of measures taken to control any ageing phenomena that could be detrimental to the safety of a nuclear power plant. In Germany the nuclear safety standard KTA 1403 [16] specifies the requirements for ageing management that encompass the technical and organizational measures with respect to an early detection of ageing phenomena relevant to the safety of nuclear power plants and to maintaining the actually required quality condition until the end of life time.

Already in 1996 [3] ageing management in nuclear power plants including deterministic versus probabilistic based ageing management has been discussed. Against the background of the growing operational age of nuclear power plants, the ageing management of the structures, systems and components used in these plants is gaining an even more important role.

The German Commission on Reactor Safety (RSK) has elaborated a recommendation regarding the management of ageing processes at nuclear power plants [6] describing the principles on the procedure regarding the management of ageing processes at nuclear power plants and detailed requirements to manage ageing processes in nuclear power plants. RSK recommended that an annual report on ageing management should be submitted to the competent supervisory authority. In order to reach a standardised proceeding with regard to ageing management on a broad knowledge base, the RSK recommends to evaluate the plant-specific reports of the plant operators generically. The results obtained from the evaluation have to be considered in the ageing management of the different plants. For this purpose, corresponding procedures have to be specified [7].

The main objective of a German project [11] was to further develop the technical decision basis for a standardised national assessment of the effectiveness of ageing management in German nuclear power plants from a methodical point of view and to carry out an up-to-date generic assessment of the effectiveness of the ageing management systems implemented in German plants for safety-relevant structures, systems and components.

For this purpose, recent operating experience with regard to the ageing management of structures, systems and components in German nuclear power plants was evaluated and the technical basis expanded by evaluating the operating experience of foreign nuclear power plants and analysing the state of the art in science and technology with respect to selected degradation mechanisms. The results are shown in *Figures 3-4* for nuclear power plants with pressurized and boiling water reactors, respectively.

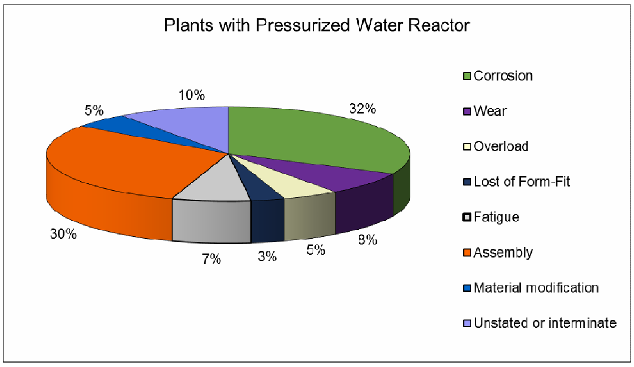

Figure 3. Ageing categories for plants with PWR in Germany

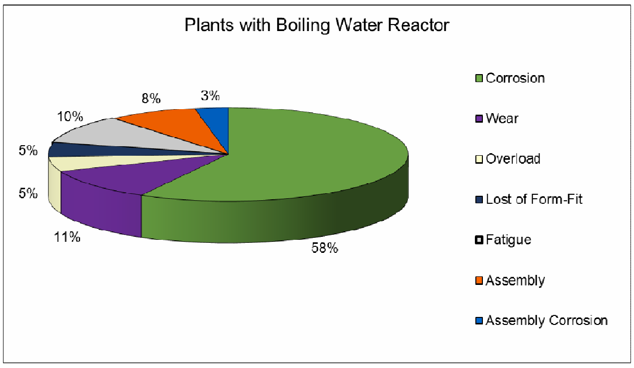

Figure 4. Ageing categories for plants with BWR

As a further task of the German project, licensee reports on ageing management were assessed from a generic point of view and proposals have been elaborated for an improved standardised national assessment in the future [11].

The results of the evaluation of operating experience show that the measures that have been initiated to detect, monitor and control safety-relevant ageing-induced changes in SSCs in German nuclear power plants have so far proved to be effective. The comparative evaluation of a representative cross-section of recent licensee reports on ageing management showed in particular differences in the kind and detail of representation. There were no indications of any deficits in the ageing management of the SSCs [11].

For the future standardised national assessment of the ageing management of structures, systems and components in Germany, the assessment according to KTA 1403 [16] regarding the reports to be prepared on ageing management is seen as suitable. Supplementary approaches are seen to be the generic evaluation of operating experience with the ageing behaviour of SSCs, detailed examinations of the implementation of ageing management systems in the plants, and the reflection of the approaches on the results of recent international projects.

6. Concluding remarks

The activities related to ageing evaluation are usually performed in the frame of periodic safety reviews, ageing management, maintenance optimization and long term operation. Basic methods for modelling of ageing are the linear method, the exponential method and the Weibull method are presented [26].

Probabilistic safety assessment is a standardised tool for assessment of safety of nuclear power plants. It is more and more included in the risk-informed decision-making process. Due to the potential impact of ageing on the performances of structures, systems and components, the identification of ageing effects and implementation of appropriate methods for their mitigation has become of increasing interest in the last years. Recently, a regulatory document has been issued underlining that probabilistic safety assessments for a nuclear power plant are to account for the cumulative effects of ageing degradation of SSCs on overall systems and plant safety performance [4].

As a result, there is a lot of research activities worldwide concerning ageing and its integration in probabilistic safety assessment. One of the most advanced projects is the European APSA project.

Realizing the fact that neglecting the impact of ageing effects in PSA models could lead to incorrectness of risk profile and to wrong safety decisions, there is currently a sustainable effort to incorporate the ageing effects in PSA studies.

Therefore, it becomes extremely important to develop a robust and efficient approach for a systematic and gradual SSC screening and prioritization, with the aim to identify and prioritize all structures, systems and components which require time-dependent reliability models in the PSA.

Experience shows that the contribution of ageing into the probabilistic safety assessment is a difficult issue at the current stage of developed models and availability of empirical data which lead to large uncertainties of the results.

The evaluation of ageing within the probabilistic safety assessment is difficult [6] mostly because it is difficult to

- distinguish equipment random failures and equipment failures, which causes are connected with degradation due to ageing,
- define the basic elements of the evaluation, which are the components themselves, as they are mostly made of several parts or subcomponents, which may degrade through time and age differently one from another and which can be partly exchanged or renewed or inspected.

Even PSA models without systematically incorporated ageing effects may be adequate and useful for certain limited ageing related applications like setting priorities in ageing management.

In a case study described in [21] the result of risk extrapolation is an increase of core damage frequency up to almost an order of magnitude. The authors of this study acknowledge the main problems discussed above (methodology, data availability, etc.).

Another study described in [12], discusses ageing at the level of electrical systems of NPP Cernavoda. Using generic data and assuming linear ageing model, unavailability have been computed and compared to unavailability without incorporating ageing. The resulting system unavailability including ageing can be up to more than the order of magnitude bigger [12].

Furthermore, a comprehensive ageing PSA plant-specific model has been developed for the Armenian NPP Unit 2 based on the results of time-dependent reliability analysis [20] The attempt was to use ageing PSA results for plant system classification and to compare the obtained results with the safety classification using the base case PSA model (i.e. neglecting ageing factor).

The comparative analysis has shown that the steam dump to the condenser, the SG seismic make-up system and the emergency feed water system have a higher rank if ageing factors are taken into account. Thus, it was recommended to adjust test and maintenance strategies in order to pay more attention on systems which appear to be vulnerable to ageing effects.

The Fukushima accidents have led to fundamental changes in the use of nuclear power in Germany. On August 1, 2011, an amendment of the German Atomic Energy Act came into force. It consists basically of an accelerated step-by-step phase-out until 2022. Eight older nuclear power plants were shut down immediately. The remaining nuclear power plants will be stepwise shut down; the last three will stop their commercial operation at the end of 2022 having reached an age of 34 years.. Against this background, long-term operation is not an issue anymore in Germany. However, for the nuclear power plants in operation it is still necessary to ensure the required quality of safety-related SSCs by an ageing management process, though the specified operational lifetime is not enlarged.

The necessity and value for the introduction of an ageing PSA in Germany is under these boundary conditions not seen anymore.

On the other hand, generic studies as explained above show the potential of an ageing PSA to describe ageing related effects on the overall risk of the plant. With a further development of mathematical models, increasing knowledge and a broader empirical data basis, more precise statements could be possible and support risk-informed decisions on changing test intervals or replacement intervals.

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