Ageing aspects incorporation into PSA level 1 model for Armenian NPP Unit 2

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Abstract
At present the operating organizations pay a significant attention to safe operation of nuclear installations. Efforts are put world over to upgrade tools for minimization of probability of accidents with possible radiological consequences. One of the most effective tools is the probabilistic safety assessment (PSA). PSA is a standard tool for NPP safety analysis. The major advantage of PSA is the possibility of in-depth qualitative and quantitative analysis of NPP actual configuration with definition of factors introducing a significant contribution to the general risk of reactor core damage.

However main lack of the PSA current models is neglect of equipment ageing effects. Neglecting of ageing effects in PSA could lead to incorrectness of risk profile and influent on safety decision making. Based on this and appropriate experience in the field of PSA models development and review activities were started aimed to incorporate ageing aspects into PSA models for Armenian NPP Unit 2. This paper is addressing issues to be covered by forthcoming analysis in the framework of ISTC project on ageing aspects incorporation into PSA level 1 model for Armenian NPP Unit 2 and participation of Armenia in APSA Network activities.

Keywords: ageing of equipment, Probabilistic Safety Assessment (PSA), bath-curve, reliability models.

1. Introduction

1.1 Background

At present operating organizations pay a significant attention to safe operation of nuclear installations. Efforts are put worldwide to upgrade tools for minimization of probability of accidents with possible radiological consequences. One of the most effective tools is the Probabilistic Safety Assessment (PSA).

Requirement for PSA as a licensee submittal was established by Armnian Nuclear Regulatory Authority (ANRA) and reflected on the 2nd level of legislative pyramid (see Figure 1).

PSA for ANPP was initiated with development of PSA level I for the ANPP Unit 2 internal initiating events on the base of Risk Spectrum code. The work was performed by the Italian SOGIN company with involvement of ANPP personnel and local subcontractors.

General review of PSA model developed by SOGIN was done by Armenian specialists (ANPP, NRSC) in co-operation with international experts (CENS, RELCO, Kola NPP, Atomenegroproject, Moscow Obninsk Nuclear Technical University, etc). It reveals some lacks and incompleteness of performed study. Detailed review and upgrading of PSA level 1 model for ANPP was initiated in 2004 by NRSC and ANPP with technical expert support (including internal review) sponsored by DOE(Jacobsen Engineering, Risk Engineering).

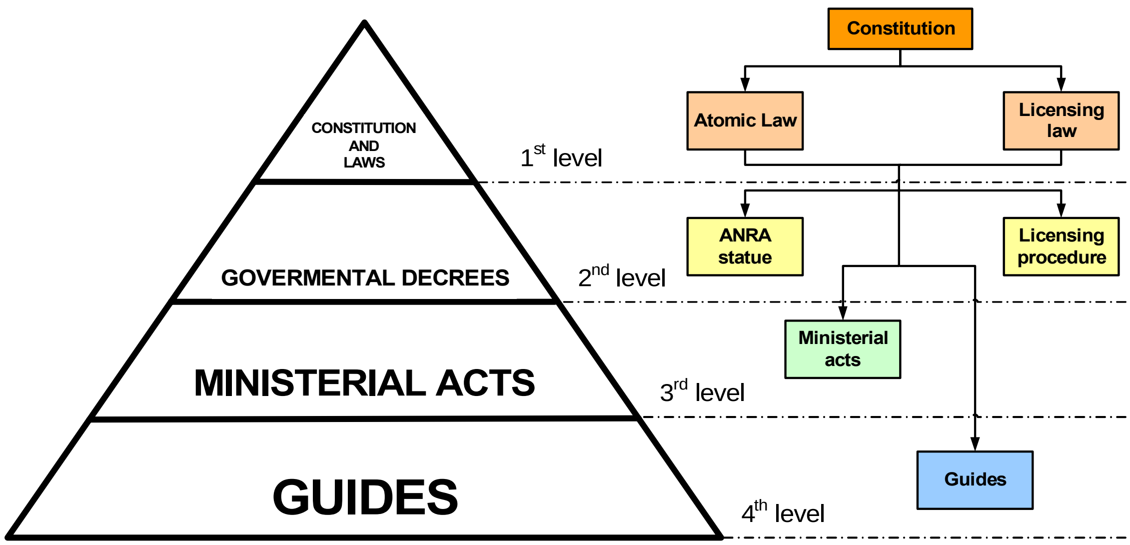

Figure Error! No text of specified style in document.**: Reflection of PSA submittal in legislative**

Detailed review and upgrading of Risk Spectrum model was performed mainly in compliance with general review results and was completed in October 2006.

Main improvements in model are as follows (by tasks):

1. Initiating event analysis
   - New initiating event definition was established
   - Engineering investigation was performed
   - Master Logic Diagram analysis was performed
   - As a result 14 new IE groups were established in comparison with SOGIN model
2. Accident sequence analysis
   - Logic of transient and LOCA ETs was corrected
   - ETs were constructed for newly established IEs
   - ET/FT interface was corrected for some initiating events (e.g. loss of offsite power, break of MSH etc.)
3. Success criteria analysis
   - More detailed and plant-specific RELAP and MELCOR models were used
   - Best estimate analysis were performed
4. System analysis
   - New systems were modeled according newly established IEs and ETs
   - Logic of modeling of some systems was corrected
   - Detalisation level was increased
   - Components boundaries were corrected

5. Human reliability analysis
   - Detailed analysis of human errors was provided
6. Data Analysis
   - Plant specific data were used
   - Data were collected according new boundaries of components

1.2 Scope and objectives of existing PSA study

The main objective of performed work is to develop a safety-related decision-making tool for Armenian nuclear community members (operators and regulator) and to use it for following safety-related activities:

- Modernization list prioritization based on importance, sensitivity and MCS analysis
- Modernization licensing using incorporation of licensing safety issue to the PSA model and analysis of results (positive & negative impacts)
- Identification of necessity of particular safety-related modifications
- Improvement of plant procedures (normal & emergency)
- Demonstration of compliance with the international criteria (CDF<1E-04 1/year) [1], etc.

Scope of performed work was limited by the following assumptions:

- Undesired event – Damage of fuel located in reactor core
- Plant condition – Power operation (both turbines are in operation)
- Initiating events – Internal initiating events (excluding explicit modeling of "B" type human errors)

Also analysis are available for external events and so called internal hazards (fires, floods). However the results of external and internal hazards analysis are not incorporated in the PSA level 1 model developed in Risk Spectrum for internal events.

2. Ageing aspects incorporation into PSA

2.1. Usage of PSA methods

The PSA methodologies are standard tools for NPP safety analysis. The major advantage of PSA is the possibility of in-depth qualitative and quantitative analysis of NPP actual configuration with definition of factors introducing a significant contribution to the general risk of reactor core damage.

Current practice in PSA field shows that probabilistic methods/models are quite matured and could be/are a base for quantitative risk-informed decision making both for nuclear utilities and regulatory bodies.

However the applicability of PSA model for either purpose is limited by detail degree of the given model. Level of detalization must be appropriate not only for systems and plant response modeling, but also for equipment and human reliability data estimation. Current PSA techniques have number of fields of uncertainties regarding reliability data estimation process, such as

- Common cause failure analysis
- Human Reliability analysis
- Ageing influence on equipment reliability (Failure rate increase trend)

There are quite a lot of done in the fields of common cause failures and human reliability analysis, and models describing those aspects are always exist in current PSA models.

However aspects of ageing are not addressing in PSA models worldwide. Meanwhile neglecting of ageing aspects during PSA model development could affect the correctness of risk profile (see fig.2 with dependency of component failure rate from time) obtained in qualification results and reduce practical value of the model as a risk-informed decision-making tool.

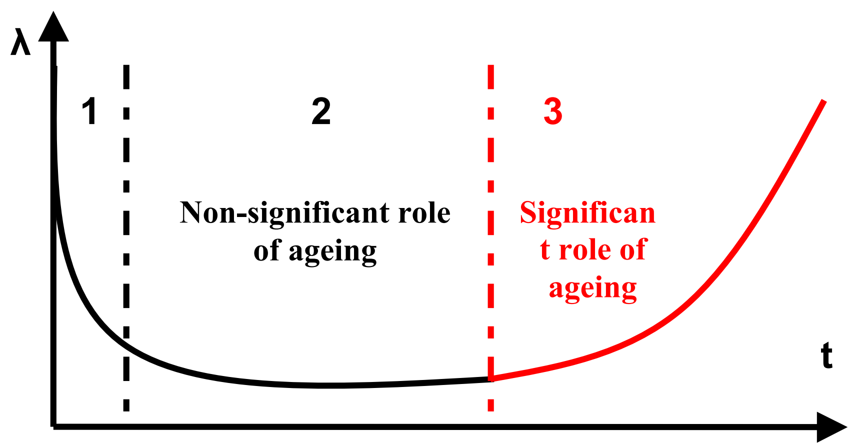

Figure 2: Bath tube curve

Based on this it was considered to apply for the ISTC project (2007-2010) aimed to address issues regarding Ageing aspects incorporation into PSA models.

2.2. ISTC project on ageing aspects incorporation into PSA

The main objective of the ISTC project is to use NRSC and international experience to address equipment ageing influence on total risk profile for Armenian NPP. The project implementation is foreseen by Nuclear & Radiation Safety Center in Armenia and Armenian Nuclear Power Plant in co-operation with foreign collaborators.

Following institutions are defined as a foreign collaborators in the project:

- European Commission, Joint Research Center, Institute of Energy (IE)
- Brookhaven National Laboratory (USA)
- Argonne National Laboratory (USA)

Analysis of PSA tasks in regard with ageing shows that ageing ascpets could affect key elements/attributes [3] of PSA models such as initiating events analysis and system analysis.

2.3. Main tasks

Incorporation of ageing effects in PSA model envisages the following tasks to be addressed:

- Analysis and adaptation of a methodology for incorporating ageing effects in PSA model for WWER-440 type units
  Under this task the analysis of potential ageing mechanisms and their effect on WWER equipment is foreseen. One of the deliverables of this task analysis should be adaptation of existing theoretical base to the WWER equipment and materials, which will allow to develop a guideline/methodlogy for λ-failure intensity increase trend analysis for WWER units.

- Analysis of ageing effect on initiating events.
  The analysis of IE occurrence frequency variation specified with system reliability models (being the so called specific IE related to system failures) should be reconsidered as well as it is possible to have an addition of IE list with IE screened earlier due to low occurrence frequency.

- Reliability analysis of the ANPP safety important systems in regard with ageing effects.
  During activities under this task it is foreseen to perform a) addition of system component list with passive components based on the analysis of system reliability characteristics in regard with ageing effects, b) analysis of possible success criteria change of system operation due to ageing effects, c) analysis of possible change of equipment normal operation limits due to ageing effects, d) Analysis of possible increase of system component common cause failure probability due to ageing effects, e) analysis of possible increase of system failure probability due to the same ageing monitoring method. This task solution will result to significant change of system analysis results in comparison with current situation.

- Development of probabilistic safety analysis mathematical model for the Armenian NPP and performance of calculations taking into account the ageing effects.
  This task is assuming incorporation of performed analysis into existing PSA model and performance of calculations aimed to estimate a) core damage frequency of the Armenian NPP depending on the operational life and b) risk profile for the Armenian NPP depending on the operational life. Main deliverable from this task will be a report with conclusions and recommendations package on the WWER NPPs and Armenian NPP safety upgrading in regard with ageing effects.

Generally the Project implementation shall provide:

- Identification of actual risk profile in regard with the ANPP equipment ageing.
- Prioritization of modifications related to the ANPP safety.
- Development of recommendations for correcting of the ANPP equipment testing and maintenance procedures and schedules.

- Development of recommendation for correcting of ageing monitoring methods.

The developed methodology shall allow incorporation of ageing effects in PSA models for other operating WWER-440 units. The obtained analysis results will allow specifying the degree of ageing effect on WWER-440 NPPs safety.

3. Interlinks with APSA Network

NRSC is also participating in the APSA European Network activities. Role of NRSC is foreseen in 4 out of 8 tasks of APSA European Network (mandate 2006-2008). Following tasks are held with NRSC involvement:

- Task 2. Analysis of main PSA tasks with regards to Aging PSA
- Task 3. Selection of the SSC to be considered in Aging PSA
- Task 7. Incorporation of Age-depended reliability parameters and data into PSA model.
- Task 8. Aging PSA development and applications

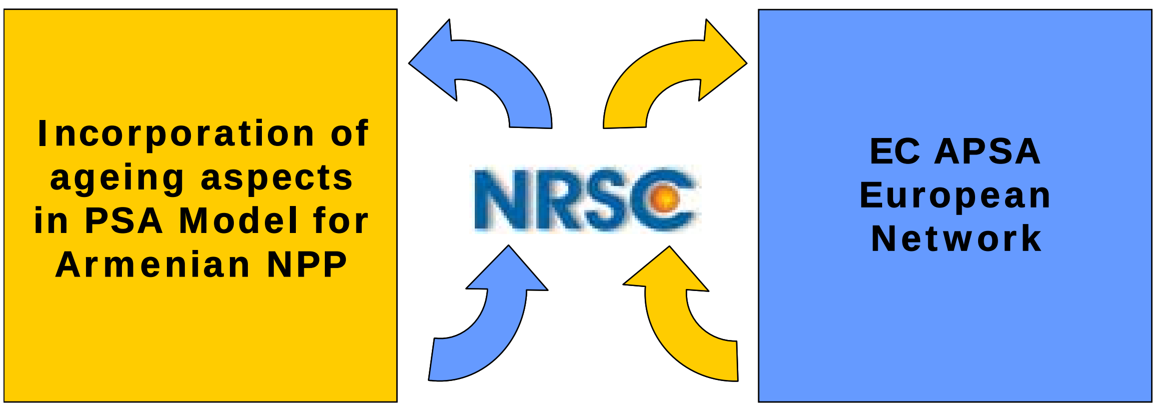

Figure 3: Interlinks between ISTC project and APSA Network activity

Actually those tasks will be developed in parallel with ISTC project development and they are quite consistent from information point of view. The decision was made to harmonize activities in both projects for more efficient information exchange (see fig.3)

4. Information from current PSA study in support to APSA development

Analysis of all systems and components presented in the current PSA model is extremely time consuming process and is not assure effect needed. For efficient usage of resources it was decided to select most important fields (initiating events, systems, components, etc.) for further analysis in regard with ageing.

Upon recent PSA model completion some information is available to support APSA development. The final PSA model quantification demonstrated that core damage frequency is 7.58E-05 1/year. Risk profile for the Armenian NPP as to the results of PSA level 1 model analysis is shown in figure 7.1. [4]

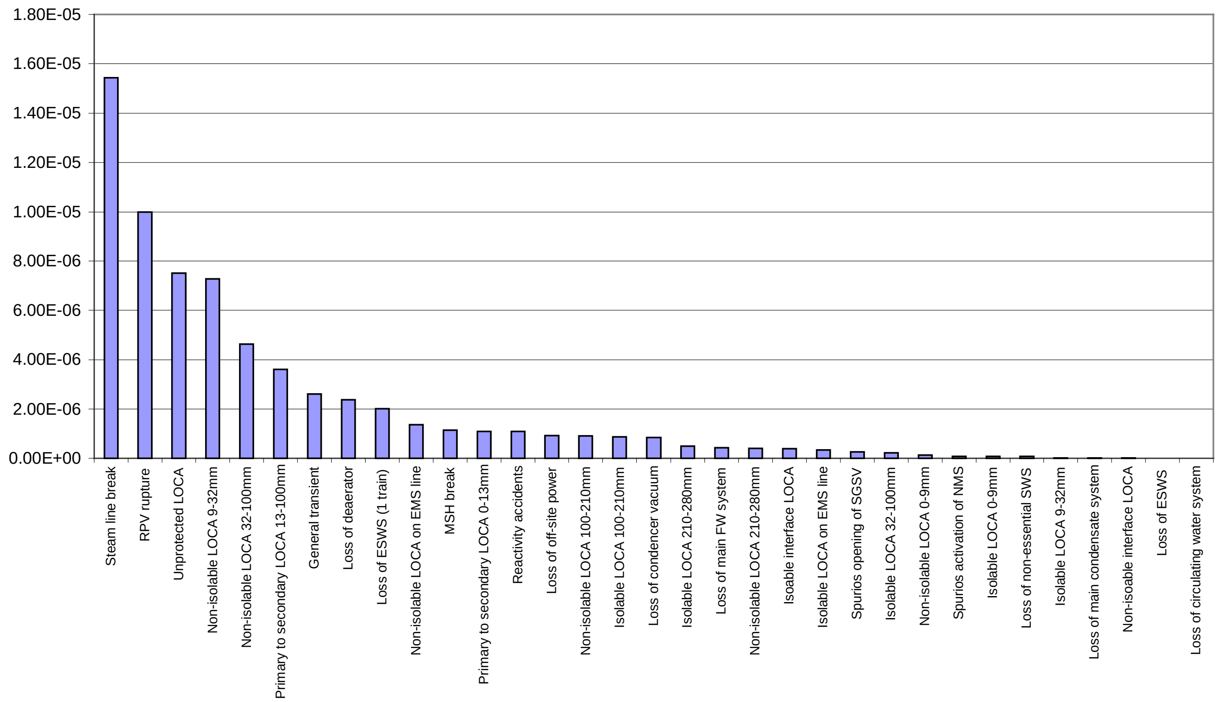

Figure 4: Current risk profile for the Armenian NPP

Results presented on Figure 4 shown that first 10 initiators have about 90% of overall CDF. Hence addressing of safety problems related to mentioned contributors will lead to significant safety increase (risk decrease). MCS detailed analysis shown in table 1 below:

Table 1: MCS groups analysis for Armenian NPP Level 1 PSA

j	IE	CDF_j	Scenario description	ΣMCS	$\%CDF_j$
1.	Steam line break upstream FSIV	1.54E-05	Steam release on elevation +14.7 of turbine Hall (place of break) leads to failure of all equipment located at this elevation (conservative assumption)/ Failure of equipment leads to unavailability of secondary feed&bleed function. CD occurs in case of primary feed&bleed failure.	1.41E-05	91
			CD due to other failures (mainly COP problem due to FSIV failure)	1.37E-06	9
2.	RPV rupture	1.00E-05	RPV rupture directly leads to CD due to insufficient core cooling capabilities.	1.00E-05	100
3.	Unprotected LOCA	7.50E-06	Very large LOCAs directly lead to CD due to insufficient core cooling capabilities.	7.50E-06	100
4.	Non-isolable LOCA 9-32mm	7.28E-06	CD due to failure of spray system to cool down boron tank	6.79E-06	93

			CD due to other failures (EMS failure, failure of support systems ESWS, AC power, etc.)	4.86E-07	7
5.	Non-isolable LOCA 32-100mm	4.63E-06	CD due to failure of spray system to cool down boron tank	3.52E-06	76
			CD due to failure of EMS to supply water to primary circuit.	1.01E-06	22
			CD due to other failures (failure of support systems ESWS, AC power, etc.)	1.01E-07	2
6.	Primary to secondary LOCA 13-100mm	3.61E-06	CD due to failure of EMS to supply water to primary circuit.	2.92E-06	81
			CD due to other failures (primary depressurizing function, failure of secondary heat removal systems)	6.95E-07	19
7.	General transient	2.61E-06	CD due to failure to close TG stop valves and additional failuyre of FSIVs (COP effect).	1.24E-06	48
			CD due to consequential grid failure (ANPP producing about 40% of overall energy in Armenia) with failure of DGs and diesel-make-up system.	7.55E-07	29
			CD due to other failures (failure of secondary heat removal systems)	6.10E-07	23
8.	Loss of deaerator	2.38E-06	Steam release leads to failure of Turbine Hall equipment including FSIVs (conservative assumption). Failure of TG stop valves to close after scram leads to COP effect due to unavailability of FSIVs.	2.26E-06	95
			CD due to other failures (failure of secondary heat removal systems)	1.20E-07	5
9.	Loss of ESWS (1 train)	2.03E-06	Loss of 1 train of ESWS leads to unavailability of DG train (insufficient cooling of DGs). CD due to consequential grid failure (ANPP producing about 40% of overall energy in Armenia) with failure of second train of DG and diesel-make-up system.	1.83E-06	90
			CD due to other failures (failure of TG stop valves and FSIVs, failure of secondary heat removal systems, etc.)	2.01E-07	10
10.	Non-isolable LOCA on EMS line	1.37E-06	This IE leads to unavailability of EMS train. CD due to failure of second train of EMS.	9.98E-07	73
			CD due to failure of spray system to cool down boron tank	3.34E-07	24
			CD due to other failures (failure of support systems ESWS, AC power, etc.)	4.28E-08	3

Risk increase factor analysis (RIF) was performed for systems' attributes to assess importance of systems modeled in current PSA study.

$$RIF_i = \frac{Q(P_i = 1)}{Q} \tag{1}$$

Where $Q(P_i)$ – conditional core damage frequency in case of "i"system failure, Q- is base case core damage frequency value. Based on importance analysis all systems were divided by 5 groups see table 2 below.

Table 2: Importance of the systems based on RIF analysis

Groups	RIF	Systems
Group #1	10^5-10^4	DC, Fast steam isolation valves (FSIV), AC
Group #2	10^4-10^3	Primary overpessure protection (PORVs), Essential service water (ESWS)
Group #3	10^3-10^2	TG stop valves (SK TG), Primary Isolation Valves (GZZ), EMS, Reactor control and protection (SUZ), Spray system, Steam dump valves (BRUA and BRUK), Emergency seismic system (ASN)
Group #4	10^2-10	Emergency feedwater (EFWS), Main feedwater (MFWS), Demineralizated water (DWS), Normal primary make-up (NMS)
Group #5	10-1	Intermediate MCP cooling (PKGCN), Diesel feedwater, Circulating water (CWS), Main condensate system (KEN), Residual heat removal (secondary system), DG sequencer, Boron preparation, Non-essential service water system.

5. Conclusions

Ageing aspects could lead significant changes in risk profile obtained for Armenian NPP Unit 2. Launched ISTC project and participation in APSA Network will lead to efficient and fruitful investigation in the field of incorporation of ageing aspects into PSA model and particularly for ANPP Unit 2 PSA study. Existing PSA experience and receiverd results about system and IEs importance is a good base for forthcoming analysis, which will lead to the following deliverables:

- Developed methodology of incorporation of ageing effects into PSA model adapted to WWER-440
- Definition of level of influence of ageing on nuclear safety for typical WWER-440 reactors
- Identification of ANPP risk profile by taking into account of ageing aspects
- Prioritization of modifications related to ANPP safety
- Correction of procedures and schedules of test and maintenance activities carrying out at ANPP. Especially risk informed in service inspection development based on new reliability information gained for passive components (e.g. piping, tanks, etc.)
- Correction of ageing management methods using on ANPP

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