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Investigation of Ageing Effects using Probabilistic Safety Assessment

Proceedings of the European Workshop on Probabilistic Safety Assessment

organised at Gösgen-Däniken Nuclear Power Plant, Switzerland, 9th & 10th November 2010

edited by

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1 Introduction

1.1 Goal of the Workshop

To present and discuss the developed methods and approaches and the results obtained for application of reliability and PSA techniques on evaluation and management of NPP ageing.

1.2 General Context

In countries with nuclear industry are carrying out various programs for managing material degradation of NPP structures, systems and components (SSC) and for related topics such as Long Term Operation (LTO) and Plant Life Management (PLiM). International exchange on these activities or international programs related to these topics have been initiated through the IAEA, the OECD-NEA and through projects partly sponsored by the European Commission (EC) in the EU.

In order to facilitate the analysis of power plant performance, many organizations have initiated efforts to collect data about nuclear power plants. IAEA has started the activity of data collection in 1970, and the Power Reactor Information System (PRIS) was implemented. PRIS constitutes now the most complete data bank on nuclear power reactors in the world and is being used as an essential source of information on nuclear power.

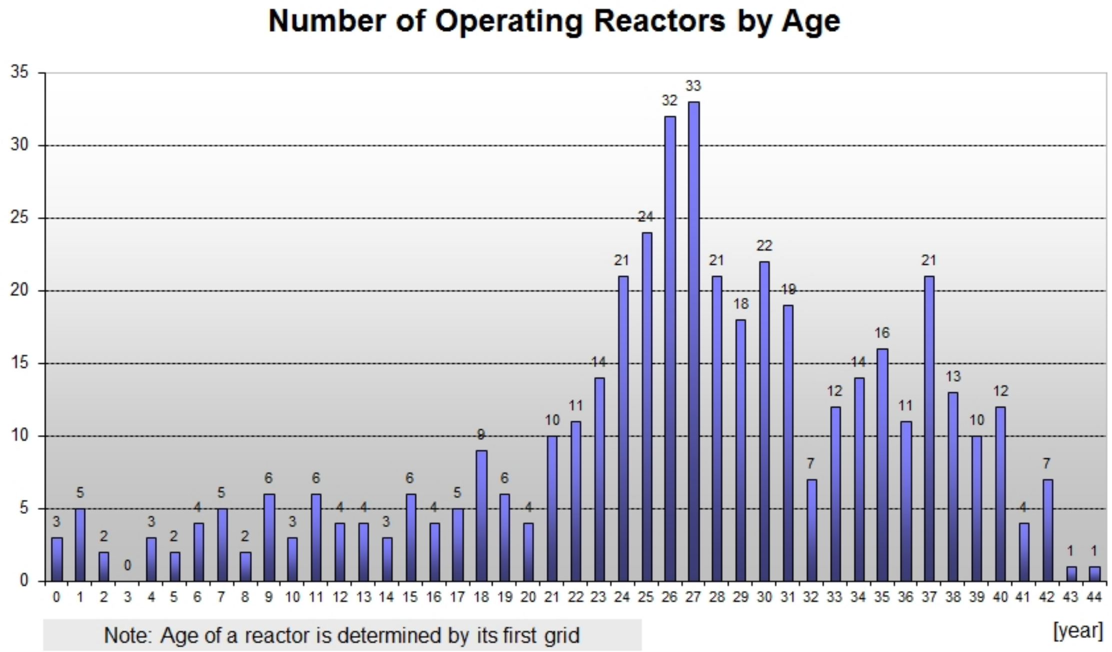

Figure 1: Ageing profile of nuclear operating reactors

According to PRIS statistics (PRIS, 2011), currently are 440 nuclear power reactors in operation in 30 countries, with a total net installed capacity of 374.093 GW(e). These plants, which have an average age greater than 20 years, were initially licensed to operate for 30 to 40 years. In order to meet the growing global demand for electricity, particularly to support economic development, it is projected that about 2300 GW(e) of new generating capacity would need to be built over the next 20-30 years. Since the costs for such projects will be significant and since the recent economic downturn might delay or even cancel many of these projects, the extension of the service life of existing NPP beyond their design lifetime becomes an extremely interesting issue. More and more utilities, nowadays, take into consideration the long-term operation policy.

IAEA PRIS data concerning aging profile of nuclear generation shows that more than 81% of operational reactors have more than 20 years of operation, which means that in the next decade ageing management and long term operation issues will became one of the key points of nuclear safety, and the evaluation of ageing effects on the overall plant safety will become a necessity. The fact that the ageing phenomena have certain effects on equipment is a non-arguable fact, and as the time is passing, is very likely that the component performances will be degraded.

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.

The activities related to ageing evaluation are usually performed in the frame of the following programs:

- Periodic Safety Review,
- Ageing Management,
- Maintenance Optimization,
- Long Term Operation.

In the last years, the PSA tools have reached a certain level of development and their results were increasingly being used as an integral part of the safety related decision making process. It was demonstrated that PSA results can bring important issues to successfully complement deterministic analysis.

PSA studies could be used as a safety evaluation tool, to help with identification and prioritization of ageing issues and optimization of ageing management activities. For applying PSA to characterize potential risks associated with ageing effects, PSA should be as realistic as practical, and appropriate support data should be available.
The requirement to accomplish the safety goals during the whole lifecycle of the nuclear installations (including the extended lifetime) sustained the idea of using the Probabilistic Safety Assessment (PSA) for ageing evaluation. Also, together with maintaining the established safety goals, it was recognized the necessity of prioritization of the Ageing Management or Long Term Operation actions, and the results of PSA could be used successfully for this task.

Use the PSA models for risk-informed decision making in case of ageing evaluation demand solving of many issues, e.g. how realistically are taken into account ageing issues in PSA models, what data are available for analysis, how representative they are with regards to the ageing assessment, and if any modifications or revisions of PSA assumptions are needed and how they could be managed.

1.3 Ageing PSA Projects

The motivation for initiating the JRC Institute of Energy project “Use of PSA for Evaluation of Ageing Effects” (APSA), was the fact that current standard PSA tools do not adequately address important ageing issues, which could have a significant impact on the conclusions made from PSA studies and applications, especially in case of plants operating in advanced aged conditions or in LTO conditions. By analyzing the standard PSA and the demands of performing age evaluations, the main characteristics for the APSA model were drawn as follows:

- APSA model should explicitly models ageing effects in component failure rates, which generally cause the failure rates to increase with age (the classic PSA use the assumption that component failure rates are constant).

- APSA model should explicitly calculate the ageing effects and age dependence on the core damage frequency and systems unavailability (the classic PSA calculates constant values for the core damage frequency and systems unavailability).
- APSA model should explicitly model the effects of test and maintenances in controlling the ageing of components.
- Standard PSA neglects the components that have small failure probabilities, not taken into account the fact that these probabilities could suffer dramatic changes in time (this reflects the situation of passive components); in APSA model these components are taken into account.

1.3.1 Objective of the APSA Project

The JRC IE project has established the bases for Ageing Probabilistic Safety Assessment (APSA) Network. Main objective of the APSA Network is to use common resources of Network participants for identification, development and demonstration of methods and approaches which could help PSA developers and users in the following activities:

- to investigate and to evaluate the effects that ageing phenomena could induce on the plant performance,
- to incorporate the effects of ageing on equipments into current PSA models,
- to provide the necessary support for identification and prioritization of reliability monitoring actions / approaches to assure that potential decreasing of reliability of SSC would be identified and corrected in time and
- to promote the use of PSA for ageing management and risk-informed applications in LTO.

APSA model can be used as a demonstration of current safety level (monitoring) and/ or evaluation of risk profile for prioritization of ageing issues (using predictive extrapolation). The main issues of Ageing PSA development are related to the following:

- identification of appropriate data sources for performing ageing analysis,
- review of initiating events frequencies, considering the ageing effect,
- modelling the ageing impact on Common Case Failures (CCF) probabilities,
- reliability data analysis and parameters estimation for aged active components,
- application of physical reliability models for aged passive components,
- modification/ adaptation of computer codes for APSA applications,
- qualitative/ quantitative assessment of ageing impact,
- predictive extrapolations (including sensitivity and uncertainty analysis),
- application of the results in risk informed decision making processes.

1.3.2 Participants of the Workshop

The workshop was mainly addressed to the professionals from the EU new member states and candidate countries, as well as non EU Countries as Armenia, Ukraine and Russia. The workshop participants came from the following countries or international/ European Organizations: IAEA, EC JRC, Switzerland, Hungary, Slovenia, Romania, Bulgaria, Republic of Armenia, France, Russian Federation, USA, Lithuania, Republic of South Africa and Germany. There was a large variety in organization types, including utilities, safety authorities, support organizations and research institutes, providing a large variety in the presentations.

2 Workshop Summary

JRC – IET Petten organized with the support of the Gösgen-Däniken NPP, Switzerland, an EC Workshop on Investigation of Ageing Effects using the Probabilistic Safety Assessments. The workshop took place at Gösgen-Däniken NPP, Switzerland, and was dedicated to applications of reliability and PSA techniques on evaluation and management of NPP ageing.

At the beginning of the workshop, M. Nitoi welcomed the participants and presented the agenda of the meeting. She thanked Gösgen-Däniken NPP representatives for their support in organizing the event and wished for all participants a successful meeting. After a round table presentation of the participants, including specification of their organization and their field of expertise, as their activities related to PSA and AM/LTO, the sessions of the workshop were started.

The workshop contained a general session, dedicated to activities of different organizations in PSA field, and a technical session, focused on the results obtained in application of reliability and PSA techniques on evaluation and management of NPP ageing.

2.1 Workshop Presentations

"<u>Welcome at the plant site. Introduction of Gösgen-Däniken NPP activities</u>" paper was presented by *J. Klügel* on behalf of Gösgen-Däniken NPP General Plant Manager. J. Kluegel, as the host of the workshop, gave a welcoming presentation. The presentation started with information about Switzerland plants (type of reactor, power output, commercial operation date), specific information about the Gösgen-Däniken NPP and about developed PSA studies (scope of studies and PSA applications implemented at Gösgen-Däniken NPP). The date of commercial operation (1979) for Gösgen-Däniken NPP classifies the plant in the generation of mid-age plants. Important dates were also specified, as the date for first project study (1966), site license (1972), foundation of company (1973), and start of commercial operation date (despite the accident of TMI 2) – in 1979. The Gösgen-Däniken NPP is a nuclear unit with a PWR reactor (vendor: Siemens/KWU), with Gross Nominal Electrical Power Output of 1035 MWe. The plant has 472 employees. As a result of really good operation experience, the last Reactor Scram was in 11.12.1990.

PSA results are used in many applications, as the following: safety evaluation (as traditional application of PSA results), operational incident analysis, system of Probabilistic Safety Indicators, and financial risk analysis. The *safety evaluation of the plant* include support of plant upgrades, evaluation of plant modifications, evaluation of outage times AOT. *Operational incident analysis* include precursor analysis, scheduled and unscheduled maintenance, classification of events, and drawing annual risk profile that use offline risk monitor applications. PSA results are used to classify events (CCDP>10-8/a = INES0, >10-6/a =INES 1). The system of *probabilistic safety indicators* deals with risk peaks – max. CCDP, annual cumulative CCDP, unavailability on demand- combination of availability and reliability, management performance assessment system. *Financial risk analysis* includes use of PSA results for estimation of technical risks. In this analysis, PSA reliability data and data of initiating events are used as input for the evaluation of technical risks and associated financial consequences.

The steps of developing PSA were also specified, as follows:

- 1990-1994 – First PSA Project, Level1/ Level2 for power operation (internal initiating events, internal and external hazards); Level1+ for shutdown operational modes
- 1997 – Update of shutdown model (new third independent spent fuel pool cooling train)
- 2001 Update of seismic PSA
- 2003 Update/Upgrade of PSA model. Living PSA model
- 2004/2005 – International PSA Peer Review (Industry level) of level 1 PSA for power operation
- 2005 Update of shutdown PSA, update of seismic PSA
- 2008 Update/Upgrade of PSA – part of periodic safety review

After presentation, it was concluded that PSA results are intensively used at the plant, and some of the applications performed could be used as successful examples for other plants.

"<u>Overview of IAEA Activities and Recent Publications Relating to Safety Assessment</u>" paper was presented by M. Nitoi on behalf of *Irina Kuzmina*. The presentation had as goal to provide information on the activities being conducted by the Division of Nuclear Installation Safety (NSNI) in the area of nuclear safety and to provide information on significant recent publications on safety assessment including PSA and SAM. The mission of NSNI was specified as follows:

- to enhance the global nuclear safety regime and to ensure appropriate levels of safety throughout the total lifetime of all types of nuclear installations in EU MS by ensuring the availability of a consistent, needs-based and up to date set of safety standards, and assistance in their applications;
- to enable MS seeking to embark on nuclear power production programmes to develop appropriate safety infrastructures through the availability of IAEA guidance, assistance and networking;
- to enable MS to build improved competence frameworks for the safety of nuclear installations and to enhance their capabilities for capacity building as the foundation for strong safety infrastructure

For the safety review service, it was presented the core services provided, area for services and an overview of the activities in this area of NSNI. The generic reactor safety reviews performed by IAEA were specified also, as below:

- GRSR Projects - conducted in 2007- 2009
- UK HSE - Screening of Four New Reactor Safety Cases submitted for the consideration of the UK Health and Safety Executive/NII against GSR-4: ACR1000, AP1000, ESBWR, EPR
- ATMEA1 - Screening of Conceptual Design Safety File and its innovative features against GSR-4 and NS-R-1 of new AREVA-MHI Reactor ATMEA1
- AP1000 - Screening of AP1000 Safety and Environmental Report and its innovative features against GSR-4 and NS-R-1
- APR1400 - Screening of KHNP APR1400 Safety and Environmental Report against GSR-4 and NS-R-1

The goals for review of accident management programme (RAMP) service provided by IAEA were presented as follows:

- to provide advice and assistance at the utility/ nuclear power plant (NPP) level in effective plant specific Accident Management Programme (AMP) preparation, development and implementation.
- to conduct peer review by teams with selected independent international experts.

RAMP has the following specific objectives:

- to explain to licensee personnel principles and possible approaches in effective implementation of AMP;
- to perform an objective assessment of the status in various phases of AMP implementation;

- to provide licensee with suggestions and assistance for improvements of AMP.

It was specified that RAMP can be performed with the following options:

- *Seminar on AMP* (PRE-RAMP), with the aim to introduce and educate on the basic components of AMP development;
- *Review of accident analysis for accident management* (RAAAM), having the aim to review the completeness and quality of accident analysis covering BDBA and severe accidents;
- *Review of AMP* (RAMP), with the aim to review the quality, consistency and completeness of AMP

Area and scope of RAMP were detailed, as follows: selection and definition of AMP, accident analysis for AMP, assessment of plant vulnerabilities, development of severe accident management strategies, evaluation of plant equipment and instrumentation, development of procedures and guidelines, verification and validation of procedures and guidelines, integration of AMP and plant emergency arrangements, staffing and qualification, training needs and training performance, AMP revisions. It was highlighted that safety assessment competence is the key to making the right decisions in design, operation and licensing. Safety assessment education and training programme (SAET) is a three step process, containing formulation of knowledge requirements, development of training programmes and maintenance or improving of knowledge/ skills.

Safety guides on PSA have as objective to provide recommendations for performing or managing a PSA project for an NPP and using it to support the safe plant design and operation. The recommendations aim to provide technical consistency of PSA studies to reliably support PSA applications and risk-informed decisions. An additional aim is to promote a standard framework that can facilitate a regulatory or external peer review of a PSA and its various applications. The scope of SG covers all plant operational conditions (full power, low power and shutdown), and all potential initiating events and hazards, i.e.:

a) internal initiating events caused by random component failures and human errors;
b) internal hazards (e.g. internal fires and floods, turbine missiles, etc.);
c) external hazards, both natural (e.g. earthquake, high winds, external floods, etc.) and man-made (e.g. airplane crash, accidents at nearby industrial facilities, etc.).

As radioactivity source is considered the reactor core. The SG is intended primarily for use by operating organizations of NPP, utilities and their support organizations, but may also be used by regulatory bodies to facilitate preparation of the relevant national regulatory requirements.

Applications of PSA level 1 results has a dedicated chapter in safety guide – chapter 10 – that provides the key recommendations for a number of Level-1 PSA applications. The following applications are covered:

- Use of the PSA for design evaluation,
- Risk informed technical specifications,
- Risk monitors,
- Risk informed in-service inspection,
- Risk informed in-service testing,
- Graded quality assurance,
- PSA-based safety performance indicators,
- PSA-based event analysis,
- Risk informed regulation.

SG on deterministic safety analysis has as objectives to provide recommendations and guidance on deterministic safety analysis for designers, operators, regulators and technical support organizations. The deterministic safety analysis could be used in:

a) Demonstrating or assessing compliance with regulatory requirements;
b) Identifying possible enhancements of safety and reliability;
c) Obtaining increased operational flexibility within safety limits for nuclear power plants.

SG on SAM programmes has as objectives to provide recommendations on meeting the requirements for accident management, including managing severe accidents and to provide recommendations for the development and implementation of an accident management program. TECDOC1511 has as objective to provide information regarding the technical features (termed "attributes") of major PSA elements (IE, AS, HR, etc.) which are appropriate for carrying out various applications, including a 'Base Case PSA'. The document covers Level 1 PSA applications, at nominal power, considering only internal initiating events. External hazards like earthquakes, tornadoes, and other natural and man-induced hazards are not included. The shutdown and low power operation modes are out of the scope of the report. General (a minimum set of the attributes needed to perform a state-of-the-art PSA with the aim to assess the overall plant safety, or a typical "Base Case PSA") and special attributes (they provide enhanced capabilities supporting certain applications of a PSA) were specified for each application. It was specified that the general attributes are applicable to all PSA applications. The document contains a MATRIX mapping the special attributes to PSA applications. The categories of PSA applications were presented, each of them with their purpose and the procedure to be followed to achieve the adequate quality.

The documents issued by IAEA are expected to support the use of PSA in various applications and planning of a PSA project, making sure that appropriate PSA quality is achieved, or assessing the applicability of an existing PSA for use in an application. It is expected that the TECDOC 1511 will serve as a complementary technical reference for PSA-related services and review missions being conducted by the IAEA on request of the Member States.

Other IAEA publications were also presented, all aimed at improving the quality of the PSA, so that they could support decision making efficiently and reliably. The list included: IAEA-TECDOC-1101 on “Framework for a quality assurance programme for probabilistic safety assessment”, IAEA-TECDOC-1106 on “Living probabilistic safety assessment (LPSA)”, IAEA-TECDOC-1135 on “Regulatory review of PSA Level 1” and IAEA-TECDOC-1229 on “Regulatory review of PSA Level 2” developed jointly with OECD/ NEA. It was specified the fact that all documents are available free of charge in PDF-form at the IAEA publications web-page.

It was concluded that IAEA Safety Standards provide an internationally agreed platform for NPP safety considerations and that Safety Review Services is an effective tool for promoting the Safety Standards application. TECDOC-1511 provides in a well structured way useful information on PSA features for applications. Also it was highlighted the fact that GRSR – review of safety cases for new NPP designs on the basis of Safety Fundamentals and Safety Requirements is promoting the harmonization of safety approaches and possibly licensing activities of Member States.

“Plant Safety Operation project activities at JRC IE” was presented by M. Nitoi on behalf of *C. Bruynooghe*. The presentation started with specification of the JRC mission to be a trusted provider of science-based policy options to EU policy makers to address key challenges facing our society, underpinned by internationally-recognised research, together with the IE core competences.

An overview on the tasks and structure of the European Commission was given, with special emphasis on the Joint Research Centre (JRC) and in particular the Institute for Energy (IE), which is one of the 7 institutes of the JRC. In the first part of the presentation it were presented very briefly the obligations coming from the Euratom treaty and subsequent legislation and how JRC supports other more political general Directorates of the EC in their nuclear activities, i.e. nuclear related legislation and funding to non-EU member countries to improve the safety of their NPP.

Then the activities of the Safety of Present Nuclear Reactors (SPNR) Unit were presented in more detail, as below:

- The European Clearinghouse, which is related to operational experience feedback;

- The Plant Operation Safety Action (POS), which is related to activities on plant operation safety including network activities on AM/LTO and the JRC participation to NULIFE, and activities related to the TACIS & PHARE programs.

The participation to DG-RTD programmes and to SNE-TP Industrial Initiative, ENER was included in the presentation. It was emphasised the fact that NULIFE document issues a list of the top priority targets for Gen II/III – LTO as the following:

- European harmonised plant design and safety justification methodology,
- Integrity assessment,
- Ageing mechanisms of Structures-Systems-Components,
- Ageing monitoring,
- Prevention and mitigation of ageing,
- Pre-normative research, codes and standards and
- Safety issues in instrumentation & control and electrical systems.

The framework of activities and actions related to safety area for unit of "Safety of Present Nuclear Reactors" were presented. More specific, the on-going networks which operate under POS umbrella (ENIQ, APSA, SENUF) were presented, with their main activities.

ENIQ is a Network of 40 organizations, collaborating on Non-Destructive Examination, Qualification and Risk Informed In-Service Inspections. It was emphasised that the working meetings attract increasingly JRC visibility, and the activity is completely in line with the prioritisation of SNE-TP and IAEA. The network operates through 2 working groups, RI-ISI working group and Qualification WG.

SENUF Network promotes harmonisation on maintenance issues or other engineering issues. 15 organizations are member of the Network. It is intended to introduce in TWG a Fire safety specific sub-network, accordingly to Gen II/III of the SNE-TP Industrial Initiative. In 2010 the activities were focused on spare part issues, LTO organisation and regulation frame. A workshop was organized, with involvement of utilities, and having as subject practical application of the NS-G2-10 IAEA safety guide on preparation for Periodic Safety Review.

The last part of the presentation was dedicated to on-going research programs and activities related to LTO with JRC involvement, mainly the LONGLIFE Project on treatment of long term irradiation embrittlement effects in RPV Safety Assessment.

EURATOM FP7 was also part of the presentation, with specification of its objectives:

- Improved knowledge on LTO specific phenomena relevant for European reactors;
- Assessment and proposed improvements of prediction tools, codes and standards and

- Elaboration of best practice guidelines on irradiation embrittlement surveillance.

As concluding remarks, were specified the future actions planned to be performed. They are dedicated to reinforce the modelling, to continue activities of APSA, ENIQ, SENUF, to participate to calls coming from SENTP TWG Gen II/III.

"Ageing PSA Network activities" was presented by *M. Nitoi*, the coordinator of the Network. EC JRC Network on Ageing PSA aimed to use the common resources of Network participants to identify, develop and demonstrate methods and approaches which could help PSA developers and users:

- to promote the use of PSA for ageing management and risk-informed applications in LTO;
- to incorporate the effects of equipment ageing into current PSA models to perform engineering analysis;
- in case where age-dependent PSA could not be applied (absence or non-adequacy of ageing probabilistic model, lack of data, etc.), to specify and prioritize reliability monitoring actions/ approach to assure that potential decreasing of reliability of SSC would be identified and corrected in time.

Presentation contained the background motivation to initiate the EC JRC IE project, the on-going activities of the Network, the results obtained and also the planned activities for the coming year. The possible impact to PSA results were presented, with their implications for IE frequencies (contribution from active and passive components ageing), for safety systems unavailability (due to ageing failures, failures of non-redundant parts, unplanned maintenance), and for common cause failures CCF (occurrence of intersystem CCF). As practical implications of ageing effects, CDF, Risk Profile and RI factors used as decision criteria in many applications could be changed in time, and these changes have to be considered in decision making process. The expected results for the network are the following:

- Developed set of feasible approaches/ models, methodological guidelines;
- Proved feasibility of the proposed approaches/ models;
- Provided support & training in correct application of the project methodological guidelines;
- Establishment of European best-practice for assessment of ageing effects;
- World-wide dissemination of project results, leading to a better understanding of important issues in modelling of ageing phenomena using PSA models.

The main activities of the Network were presented, with their results and further planned activities. Summary of the Network activities was presented, with specification of published peer review papers, performed case studies that addressed all the tasks from action plan, with their developed reports (Task 3 - INR + JRC -EUR

23446 EN, NRSC, Task 4- JRC IE & VVER-440 partners, Task 5 - NRI Rez, Task 6 - KKG/SW, Task 7- IRSN/FR & Statwood Cons./US -EUR22483EN, JSI, JRC IE & INPE/RF -EUR23079EN) documents issued for end-user support (Guideline for selection of SSC to be considered in Ageing PSA, Guideline for Analysis of Data Related to Ageing of NPP Components and Systems). The Training on Advanced Time-Dependent Reliability Data Analysis was specified also, with his main sessions. The activities dedicated to dissemination of the results were presented. As major challenges were specified the following:

- Modelling the ageing as common-cause failure;
- Investigation of ageing effects on CCF parameters;
- Finding suitable data for ageing analysis;
- Assessment of ageing for passive components (data, approach, age-dependent reliability models) and
- APSA results application (AM, LTO).

The concluding remarks are presented below:

- On system availability level, ageing could induce the modification of system success criteria, could increase the Common Cause Failure (CCF) probability for highly redundant systems, and could change the list of contributors to overall system unavailability. On overall plant level, ageing could induce the modification of initiating event frequencies, probability of mitigation of undesired events (probability of unavailability for safety systems), the occurrence of inter-systems CCF, the dominant sequences, and could change the list of contributors to the accident sequences.
- Practical applications that use the addressing of ageing effects in PSA could be related to:
  - prioritization of ageing management issues and LTO activities using APSA findings;
  - predictive evaluation of plant safety level, aiding in focusing the resources for ageing mitigation where and when is necessary;
  - prioritization of maintenance activities using APSA results.
- The resulting knowledge from the project running should help PSA developers and users to efficiently incorporate the effects of equipment ageing into current PSA tools and models, to identify and/ or develop most effective corresponding methods, to focus on dominant ageing contributors and components and to promote the use of PSA for ageing management and for risk-informed decisions.

"Application of Bayesian methods for age-dependent failure analysis" was prepared by *R. Alzbutas*, from Lithuanian Energy Institute. The paper presented the general framework for assessment of age-dependent failure analysis in case of ageing systems, structures and components. The framework is based on Bayesian approach and its ability to incorporate prior information and on idea that ageing can be considered as inducing time-dependent change of systems parameters. Bayesian update combines prior knowledge or expectations regarding behaviour of a statistical parameter with actual observations of the behaviour. Proposed approach is able to deal with sparse and rare failure events as is the case of electrical components, piping systems and other systems considered as having high reliability. In case study of electrical, instrumentation and control components, the proposed framework was applied to analyse age dependencies in failure rate together with treatment of uncertainties of age-dependent model selection. Unfortunately, the author was not able to sustain the presentation, so M. Nitoi presented the summary of the paper, and no discussions were made. The paper was included on the CD and distributed after the workshop along with all presentations.

B. Lydell, from Scandpower Inc. presented the paper "A-PSA & Analytical Challenges: Results and Insights from the Integration of Passive Component Reliability in PSA". The paper was based on practical insights from Piping Reliability Analysis (1994 – 2010), and had the aim to provide a technical perspective on the investigation of aging effects using PSA. The work had initially aimed to validate assumptions/ estimates loss-of-coolant-accident (LOCA) frequency in WASH-1400 (NUREG-75/014). The steps of the analysis were:

- Create database & evaluate service experience with safety-related piping – 1970 to date
- Determine feasibility of augmenting PFM with statistical models of piping reliability – foster a deep understanding of the unique reliability attributes & influence factors & develop a solid analysis framework.

Data requirements & data quality related to integration of passive component reliability in PSA constituted a first part of the presentation. Methods used for Piping Reliability Analysis, as practical insights were presented. Early results were specified, as follows:

- Series of Technical Reports (SKI Report 95:58, SKI Report 97:26, SKI Report 97:32, SKI Report 98:30),
- Microsoft® ACCESS Database – 'SKI-PIPE',
- Workshops & Seminars organized (1996-98),
- Planning for continuation as an international cooperative effort to collect and evaluate pipe failure data
    - OECD/NEA: OPDE (2002-2011), CODAP (2011-2014),
    - SCAP-SCC (2006-2010).

It was commented that even if there exists a strong statistical basis for estimating reliability parameters for degraded states, the following should be taken into account: correlation of service experience with pipe population data, correlation of service experience with mitigation practice, detailed considerations of uncertainties (data completeness, plant-to-plant piping design differences, modelling), Bayes analysis framework, and conditional failure probabilities.

It was mentioned that the major advances made in piping reliability analysis were supported by the following issues: a robust pipe failure database, Bayes method for uncertainty, Markov model to evaluate integrity management strategies. A robust database could be used as a learning tool, for knowledge preservation, or to improve the understanding of inspection and mitigation programs and their beneficial impact on structural integrity, and as resource for methods development (calibration of assumptions, validating results, advancing the state-of-the-art). Some insights about Markov model were given during the presentation, as follows:

- Markov Model was originally developed for EPRI RI-ISI Program;
- Many applications of the model (applied to 26 plant specific RI-ISI programs in U.S. and South Africa, applied to PBMR to support new ASME Code development for in-service inspections, applied in NUREG-1829 LOCA frequency update, and recently applied to LWR to guide efforts to reduce internal flood and HELB contributions to CDF);
- Degradation related parameters estimated from service experience and Bayes models are the same as those used for the base failure rates;
- Test and inspection parameters are estimated using simple and easy approach to quantify models.

Related to evaluation of ageing effects using PSA, 2 approaches were presented. The basic approach has as requirements: CCIII PSA Model (Integrated all modes, internal flooding, fire, external events, seismic, L2) and existence of a Strong Degradation Mechanism database. The expected outputs are the results decomposition by degradation mechanism/ ageing effect (aging effect assessment on CCDP/CLERP, aging factor assessment and projection of aging effect on risk metrics). The advanced approach has the same requirements as for the basic approach, plus an enhanced PSA Software Platform with ‘AM Module’. The outputs are the same as for ‘Basic’ approach, but streamlined & integrated. The steps in assessing the ageing factors were specified, as follows:

- development of qualitative data (completeness is essential);
- performing data screening and assessing existence of trends (cumulative failure plots, binned data-by age, test different hypotheses could be used);
- performing parameter estimation (characterize uncertainties, validate results-statistically & qualitatively, determine time dependency -if any);
- performing data specializations as warranted;
- assessing impact of past, current & future inspection strategies (e.g., leak detection, ISI, RI-ISI) by applying Markov model.

Feasibility of using PSA for ageing was highlighted, given the following issues:

- Unique requirements for data quality & completeness
- Essential that a Degradation Mechanism knowledgebase 'co-exists' with database
- Analyst must be sufficiently trained in Degradation Mechanism assessment
- Maturity of PSA model is essential (is necessary to have at least CCII with plant-specific internal flooding model that includes HELB consideration, and CCF parameter considerations)
- Consensus guidelines for risk metrics to be used
- Implementation of 'Basic' vs. 'Advanced' approach
- After discussions, it was recommended to be performed an international 'benchmark' exercise for Basic Approach.

It was highlighted the fact that there are major improvements in the treatment of piping failure in PRA in the last years. It was concluded that there is essential to have a mature PSA model, to have co-existence between Degradation Mechanism knowledgebase and database to obtain good reliable results.
It was specified that collecting & maintaining a sustained and consistent data collection effort is resource intensive & time-consuming, but there are significant benefits of it, if is desirable to have a robust basis for evaluation of aging effects. It was concluded also that a 'strong' database fosters a good understanding of aging effects, and there exists solid technical basis for assessing aging factors using statistical reliability methods.

"Overview of recommendations for data and models requirements for passive components ageing assessment in PSA" was presented by *I. Dinu*, from Cernavoda NPP. The document intended to make a review regarding typologies, failure mechanisms, failure modes, and failure rate calculation methods used when passive components are included in current Probabilistic Safety Assessment studies. The issue of passive component failure is increasingly important as many power plants are confronted with the problem of aging on one side and the necessity of safe and reliable long term operation on the other side. This requires the systematic and proactive identification of degradation and aging mechanisms of critical passive components that are not addressed by normal Preventive Maintenance strategies and mitigation of these inherent major problems which will otherwise impact on the plant assets.

The failure rate calculation methods for passive components were specified, as below:

Statistical analysis at different time interval – the method implies statistical estimates of component failure rates, and is based on data collected from SSC service experience, at a given operational moment. The major problem to apply the method would be the scarcity of data, because the passive components have only visual and

ND inspection, and not PM. A variation of this approach is to augment statistical estimates of component failure parameters with models that express the problem in terms of a failure rate and a conditional probability for each failure mode of interest, at a given time moment.

Physical Modelling - the process of detailed physical modelling requires the governing equations for each specific mechanism to be defined.

Bayesian Updating - combines prior knowledge or expectations regarding behaviour of a statistical problem with actual physical observations of the behaviour, for a better estimate of expected behaviour. By updating the theoretical failure rate estimate (known as the “prior”) in this fashion, it is obtained a more accurate failure rate prediction (the “posterior”) than either the theoretical prediction or a purely empirical prediction based on failure observations.

Markov model - explicitly model the interactions between failure mechanisms that produce failures and the inspection, detection and repair strategies that can reduce the probability that failures occur, or that cracks or leaks will progress to ruptures before being detected and repaired. The model starts with the representation of the “piping system” as a set of discrete and mutually exclusive states. At any moment in time, the system could change the state in accordance with whatever competing processes are appropriate for the plant state. The states refer to various degrees of piping system degradation, starting from indications of degradation and progressing to leaks and ruptures. The change of state is given by the various failure mechanisms and could imply the inspection and repair activity performed before progression into rupture. This method was found as meeting the requirements of an up-to-date analysis of piping reliability, some of these requirements being to:

- account for statistical evidence and engineering insights of plant experience;
- evaluate the impact of changes in the In Service Inspection strategy, like adding or removing of the locations to the existing ISI program or even changing from fixed to randomly selected locations from one inspection interval to another;
- address uncertainties in the reliability assessment and account for it in estimating pipe ruptures and in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF).

The method was applied for piping system, but the basic approach can be applied to any passive component for which enough inspection and findings are provided during its operational life. It was specified that CNE Cernavoda is participating to a number of international initiatives related to ageing management of NPP, as well as cooperative projects. One of these projects is the COG joint project “RI-ISI Pilot Study” – where the objective is to adapt and apply the EPRI risk informed in-service inspection methodology to several systems at a COG identified site. The methodology that is adapted and will be specific to COG fleet of nuclear facilities refer to a series of specific systems: PHT, SDCS, ECCS, Main Steam. Results obtained will be documented in a final report:

- Fuel Channel R&D program managed by COG focusing on addressing the current operational need to improve confidence in the fitness for service of CANDU pressure tubes and developing industry standards for pressure tube integrity;
- Ageing PSA – current project, containing several investigation areas of effects of ageing into Probabilistic Safety Assessment studies;
- Some passive components considered in CNE Cernavoda PSA, like Heat Exchangers, were summarized, in the perspective of methods, failure modes, failure mechanisms, and ageing considerations.
- The modern PSA analysis and Risk Informed In-Service Inspection strategies implemented intend to answer to questions related to:
  - time evolution of failure rate, considering the current maintenance practices;
  - mitigating actions to be taken in order to account for reliability deterioration in time;
  - effects of these actions over the failure rate and over the CDF/ LERF.

G. Petkov, from Technical University of Sofia presented the paper <u>"A Case Study on Incorporation of Ageing Effects into the PSA Model of NPP with VVER1000"</u>.

The paper presented the results of a case study on incorporation of ageing effects into the PSA model of the Russian PWR - WWER-1000, study that was carried out within the framework of the EC-JRC Ageing PSA Network Task 7. The ageing impact was presented, first on the plant level (effect visible by CDF changing, risk profile - contribution of initiating event groups to the CDF, list of the dominant minimal cut sets MCS, results of SSC prioritization due to ageing), and after, on the system level (effects related to reliability/ availability, list of the dominant MCS, results of SSC prioritization due to ageing) and on the component level – appropriate mathematical models within existing computer codes for reliability and risk assessment.

Basic model for ageing model include mathematical parameters, for which it is possible to obtain appropriate data, however the PSA software do not use such parameters. This makes these models more difficult for ageing incorporation into the PSA, modelled by current software as Risk Spectrum (RS), and their use should not be widely applicable. The difference between age and time was presented, considering that ageing of the component does not proceed at the same speed as time. Also the age generally incorporates the effect of the surveillance, maintenance, and replacement of the subcomponent or whole component, while the time incorporates non-ageing (accelerated, sudden and unexpected) degradation as well.

The approaches that can be used to incorporate the ageing effects into PSA models are summarized below:

Minimal Cut Sets Modification Approach includes simple modification of the previous MCS by addition of the ageing contribution to the new MCS. The procedure consists of two steps:

- reliability database update,
- MCS recalculation.

However, this procedure missed all changes of the PSA/ system reliability models due to creation of new BE because of passive equipment ageing or replacement during reconstruction and modernization. Therefore, some ageing contributions are missed, and this first procedure was considered not appropriate.

Step-wise approach consists of extension of the existing PSA and reliability models with addition of ageing as an independent contribution. The deficiency of such approach is the fact that contribution of ageing may not be independent from the already considered contribution to failure.

- Age-dependent degradation - passive SSC, not included in the initial PSA model;
- Age-independent degradation - active SSC, included in the initial PSA model.

If the dependence is important, such approach is not appropriate to be used. As output of the procedure it is possible to obtain:

- CDF/ risk/ reliability/ availability profile,
- list of the dominant MCS,
- results of SSC prioritization due to ageing.

The SSC prioritization for consideration of ageing was done in four iterative steps: extension of the SSC (components) included into the existing models (if L<M<N), extension of the SSC failure modes included into existing models, updating of the SSC reliability data and performing SSC ageing sensitivity study.

Interface between *PSA Software and Ageing Reliability Parameters Specification* was evaluated, taking into account the following aspects:

- The PSA models are updated periodically and taking into account the ageing that affect the reliability of one or more SSC.
- The regulators urge plant operators to update and calculate more frequently risk models in order to check, catch and incorporate as well degradation as ageing processes.

- The methods for age-dependent reliability parameters specification include mathematical parameters, for which is possible to obtain appropriate data.

It was concluded that the parameters of ageing cannot be included directly to the probabilistic models within the existing computer codes for PSA, because for this purpose the codes have to possess the integrated capabilities for calculations with consideration of ageing into the PSA models. Alternatively, the ageing contribution can be calculated outside of the PSA computer code and the evaluated contributions of ageing could be inserted after into the PSA models. At each selected time interval, the average failure rates or failure probabilities are updated for the equipment under investigation and the evaluation of the PSA is performed with insertion of calculated average probabilities into the PSA model.

An advantage of this method is that it can be used in standard PSA, and the PSA database could be updated for any ageing equipment and any year of interest. However, the PSA software developers do not recommend change the data locally because it is dangerous without a seamless interface between PSA software modules, and it involves unreasonable long time spent for updating boundary condition sets in PSA models.

Risk spectrum was used to recalculate the failure rate/ unavailability of all components with different ages (for each failure modes), in different plant operating states with different boundary conditions and for many time/ age points of the unit, to monitor the NPP ageing. This made the application of the methods quite cumbersome and tedious for ageing incorporation into the PSA models, and therefore, their use are practical just for case studies with limited applicability.

Another part of the presentation was dedicated to R-DAT, which is a Reliability Data Analysis update tool (it uses generic and installation specific reliability data). The background for developing R-DAT was specified, as it provides Bayesian analysis capabilities needed in support of risk and reliability assessment of complex systems, and is intended to give common interface between RS products. The idea to develop a seamless interface of RS with the simple and basic software for database support was taking into account by RS developers, and they added a new feature to the new version of the RS PSA – module R-DAT and MS Excel Import and Export feature. Two new developing features of Risk Spectrum that could be used for taking into account ageing dependency are: correlation between parameters and success block diagrams.

The last part of the presentation was dedicated to presentation of the case study. The discussion on the sensitive use of PSA to evaluate the SSC ageing effects on the overall plant safety was provided using the WWER-1000 large LOCA PSA model as an example. The ageing data included time-dependent reliability models for certain mechanical, electrical and instrumentation & control components. Event tree model developed took into account the following factors:

- operational state: power operation and hot shutdown,
- brake location (100<Dy≤850mm): hot legs or cold legs.

The procedure of modifying the existed PSA reference model was presented, as including the following seven steps:

1. *Step 1:* Identification of BE which correspond to the components sensitive to ageing – by importance measure calculation (FV, FC, RDF and RIF) for which age-dependent reliability data are available.
2. *Step 2:* Creation of House Events for the same model with three different databases to trigger the analysis cases and activate the exchange events for each particular age point where the CDF calculation has to be done (for case study purposes were considered age points of 8, 14 and 20 years of operation).
3. *Step 3:* Specification of 4 exchange events for each BE identified on the Step 1, each exchange event corresponds to the component unavailability at the time point 8, 14 and 20 years in operation and it's linked to the corresponded house events; Optionally it is possible to specify large groups of exchange events by tagged BE.
4. *Step 4:* Determination of attributes for created exchange events taking into account failure mode, operating state, unit age considered for calculation, test and maintenance strategy, type and parameters of reliability model;
5. *Step 5:* Specification of the parameters (failure rate and probability), linking them to the exchange events and input the initial values for the point estimations and distribution functions.
6. *Step 6:* Creation of a correspondent CCF group for each exchange event in case when initial BE is considered for the CCF group; the CCF model parameters (e.g. β-factors or MGL) remain the same in the CCF group modelling. CCF failure probability, then, changed with the unit age proportionally to the changes in a probability of independent failure.
7. *Step 7:* Quantification of the CDF for a particular age point as soon as all modifications for all identified components and BE are made; for each analysis case in a Boundary Condition Set specification the corresponding House Event was set up to "true".

Depending of the purpose of the calculation the following risk measures could be quantified:

- CDF changing as a function of unit age,
- modification of risk profile (contribution of IE groups to the CDF) as a function of unit age,
- modification of the list of the dominant MCS,

- changing in risk importance measures (Risk Increasing Factor, Risk Decreasing Factor, etc.).

The total change of CDF for Large LOCA in POS0 (full power operation) for the last six years of operation (considered value for 20 and 14 years) was calculated as ∆CDFLLOCA=CDF20 years-CDF14 years, and the obtained value confirms that large scale modernization on the both of Kozloduy NPP units succeeds to prevent the SSC ageing processes and to reduce the overall plant risk.

Concluding remarks:
The results obtained gave evidence that the ageing contributors should be treated distinctively on the basis of importance measures, system unavailability, dominant accident sequences and IE CDF portions in the overall plant risk (total CDF).
Considering sensitive incorporation of ageing effects into PSA models can help in the selection, prioritization of SSC susceptible to ageing, improving maintenance measures, replacing important components and performing consistent ageing management as a part of a risk-informed decision-making process.
The methods for age-dependent reliability parameters specification included mathematical parameters, for which is possible to obtain appropriate data, however, the estimated ageing parameters cannot be included directly in the probabilistic models within the existing computer codes for PSA.
The codes have to possess the integrated capabilities for calculations of ageing considerations. New modules and features of the most popular PSA software RiskSpectrum as R-DAT, Excel Import and Export, correlation between parameters and success block diagrams could be very useful for preparing extended ageing reliability database and successful incorporation of ageing effects into the PSA models.

J. Kluegel presented the paper "Activities of NPP Gösgen in the field of Ageing PSA".
Applications and applied research activities at NPP Gösgen are related to: probabilistic safety indicators, time-trend analysis (active components), pipe rupture frequency (Markov models), probabilistic lifetime assessment (fatigue evaluation). The fact that NPP Gösgen has an unlimited operational license was specified, together with the fact that ageing effects are permanently monitored, using a large investment program for replacement in-time of passive components in order to avoid undue risks and to convince plant owners that safety goals are maintained. Regulatory requirement for the plant is: *„No risk increase with ageing".* PSA results are used to assess the technical risks, with link to financial risks (production loss).

The ageing management (AM) program is based on deterministic criteria; it covers all safety classified equipment, components and structures and its efficiency needs to be assessed. The NPP Gösgen established a system of probabilistic safety indicators before the introduction of regulatory requirements: global indicators (annual risk profile, evaluation of risk peaks), detailed indicators (unavailability on demand of safety important systems), probabilistic cost-benefit analysis.

The activities where PSA results are successfully used were presented:

- Scheduled on-line maintenance at Gösgen is permitted due to the deterministic design (6*100%), and improved configuration control is established to avoid inadvertent maintenance, with the following criteria:

  - no risk peaks with a CCDF above 1.E-5/y;
  - CCDF for scheduled maintenance plus assumed forced repair (over 24h) should remain below 5.E-5/y and the overall contribution (under this assumption) shall not exceed 25% of the zero-maintenance cumulative core damage frequency;
  - the contribution to the annual cumulative core damage frequency due to maintenance should remain below 5.E-7/y (regulatory requirement A06).

- Technical departments get „acceptable time windows“ for online maintenance:

  - 25% of AOT (repair time) can be used for preventive online maintenance activities.
  - Additionally, train revisions (a single redundancy) can be performed once a year.
  - Work is only allowed in a single redundancy.

The process is controlled by plant operations (on-line, just in time) and by the annual event analysis and the procedure assures low risk profiles and acceptable maintenance times.

The developed procedure for performing trend analysis of the APSA network was implemented at NPP Gösgen and introduced as part of the Periodic Safety Review (2008) as follows:

- Prediction of CDF for the end of lifetime was made, observing a small reduction due to ongoing learning effects (DFR), and with inclusion of a check for “passive” components (LOCA frequency).
- Assessment of LOCA frequency was performed (comparison with LOCA-frequencies obtained by a Bayesian method and with data used in Gösgen PSA), together with the development of a Plant Specific Piping Failure Database as part of participation in the OPDE project (via regulatory body). As a result a more detailed database following mainly the structure of OPDE was created.
- Development of a Markov model using plant specific information from in-service inspections.
- Quantification and investigation of alternate inspection practices.

All reportable incidents are evaluated by the help of PSA, and the results support the identification of the "true safety significance of events", and the classification of the events (Swiss "INES" classification depends on the results of risk evaluation).

A new project, Probabilistic Assessment of Plant Lifetime was presented, with the motivation for initiating it and with expected results. Lifetime of a nuclear power plant is driven by the lifetime of the most vulnerable components/ structures that cannot be the subject of replacement (Reactor vessel and internals, critical piping, large components -pressurizer, steam generators). Besides irradiation, different modes of fatigue mechanism (including environmental factors) are driving the lifetime. Knowledge of lifetime allows to prepare replacement programs in cases there such replacement is feasible and needed. Longevity curves are defined as the conditional probability of failure of a component in dependence of age. A model similar to the traditional fragility function model in seismic PSA would be applied (double-lognormal) The idea was to characterize the plant lifetime as the minimal HCLPF of the lifetime distributions of critical components. Gösgen NPP offers a three-month internship for this project.

"Verification of ageing trends by screening of safety significant operational events database" was presented by *A. Rodionov*, from IRSN. The presented study investigated reportable events caused by ageing of SSC in France and Germany, using the IRSN and GRS event databases. The used methodology was composed by three steps:

1. selection of events (screening of the database with keywords and selection of relevant events),
2. classification of selected events by groups and statistical analysis,
3. summary of lessons learnt (conclusions and recommendations).

303 events were selected as ageing related, from 11972 events reported, during the period from 01 January 1990 to 30 June 2010. As examples for ageing-related events the following were specified:

- Primary coolant leak incidents caused by thermal fatigue,
- Pneumatic control valves failures on the relief valves to atmosphere,
- Ageing-related malfunction of circuit breakers,
- Formation of whiskers on circuit boards used in I&C

The results were presented in categories of components, components from primary circuit, mechanical components, electrical and instrumentation & control components (number of events, contribution of degradation mechanisms were specified for each group). The report will be issued as an EUR report next year. There were questions related to the ranking of AM that leads to failure, difficult to be made due to lack of clear information. Discussions were related to the possibility of issuing recommendations for improving the event reporting system.

K. Dusko, from Jozef Stefan Institute, presented the paper "Age-dependent unavailability modelling of safety systems integrating effects of test and maintenance". Activities connected with the age-dependent PSA of NPP included the following:

- Modelling of ageing of passive systems, structures and components (*NUREG/CR-5632*);
- quantification of the ageing induced risk using PSA (*NUREG/CR-5510, NUREG/CR-5378*);
- ageing-related failure analysis of NPP operational data (*NUREG/CR-4747, NUREG/CR-5248*).

The objectives of APSA activities at JSI were specified as being:

- to examine the options for direct and separate inclusion of ageing in PSA in order to compare the effectiveness of the models and the applicability of the results;
- to develop an analytical age-dependent unavailability model, that would overcome the limitations of FTA;
- to conduct trade-off analysis between system unavailability and T&M costs;
- to assess the implications of ageing data uncertainties on the system unavailability calculations using the developed analytical unavailability model.

Two main options for consideration of ageing in PSA were considered: method of stepwise constant failure rates and method of prioritization of components due to ageing using the PSA results. Due to limited parameter options in risk spectrum, the method of stepwise constant failure rates was utilized and calculations were made for three different time points.

The effects of ageing were presented separately at component, system and plant level. The details are given below:

- Ageing consideration at component level:
  - Containment Spray System (CSS) selected as an example;
  - 11 parameters representing the selected components considered;
  - results indicate that some equipment can be non-important, if no ageing is considered, and very important if ageing is considered and vice versa;
  - risk factors may considerably vary depending on ageing consideration.
- Ageing consideration at system level (CSS):
  - FT model comprised of 24 BE and 22 gates;
  - results show the increase of system unavailability with one order of magnitude for the considered period of 15 years;

- FV changed between 11% and 2 orders of magnitude;
  - RDF changed between 0% - 97%;
  - RIF changed between 2% - 90%;
- Ageing consideration in all PSA model (at plant level):
  - Model comprised of thousandths of gates, thousandths of BE, hundredths of FT and 16 ET with 16 IE;
  - IE frequencies were changed (due to ageing) for two groups of IE
  - The linear method for consideration of ageing was implemented;
- Results of prioritization of components due to ageing indicate the following:
  - the increases of CDF are relatively large;
  - it is questionable how those large increases of CDF are comparable to changes due to other parameter changes ;
  - more experience and analyses are needed to confirm or reject the applicability of the method for prioritization of components due to ageing based on the results of PSA in the real application of the results.

It was concluded that the contribution of ageing seems to be an interesting issue for modelling and quantification. Sensitivity calculations were performed for specific BE (MDP and MOV failure) of the FT model (QTOP – HPSIS failure probability).

Related to ageing impact on components, it was observed that if sequential is replaced by staggered testing, relatively high reduction in Qsys is observed. The impact of the component ageing data uncertainty on system unavailability calculations was performed via:

- age-dependent unavailability model considering T&M;
- A Monte Carlo simulation-based computer code;
- RiskSpectrum software.

It was observed that the uncertainty of system unavailability is rising with the extension of the test interval (*Ti_opt* -> *Ti_TS*). After discussions, it was concluded that the "more complex" the system is, the higher the ageing impact becomes; and if large uncertainties are associated with the ageing parameters, they have a substantial impact on the quantitative results. Discussions highlighted the necessity of developing an analytical unavailability model that simultaneously integrates:

- the impact of T&M activities,
- testing strategies (sequential vs. staggered),
- component ageing.

S. Poghosyan, from Nuclear & Radiation Safety Center, presented the paper "Time-dependent reliability assessment for VVER equipment". The aim of the investigation performed was to address T-D aspects in reliability calculations for critical VVER equipment in order to estimate the degree of ageing influence on component reliability. Medzamor NPP Unit 2 was used as an example, and the Guidelines for Analysis of Data Related to Ageing of Nuclear Power Plant Components and Systems (EUR 23954 EN) were used as a base for data processing. Selection of components for further consideration was done using risk importance parameters:

- Fussel-Vesely importance (F-V)
  - Provides current contribution of SSC to overall risk
  - Selection criteria: F-V > 0.005
- Risk increase factor (RIF)
  - Shows potential importance of SSC, given that SSC failed
  - Selection criteria: RIF > 1E+2.

Newly installed SSC were screened out from the model. This includes essential service water system, fast steam isolation valves, Diesel-driven feedwater system and reverse motor generators. WinBUGS was used for data processing, and the following models were verified:

- For Poisson model (failure rate)
  - constant model
  - loglinear model
  - power model
- For Binominal model (probability of failures on demand)
  - constant model
  - log-log model - $\ln(-\ln(1-p(t)) = \theta1 + \theta2t$
  - logit model - $\ln (p(t)/(1 - p(t))) = \theta1 + \theta2t$
  - probit model - $\Phi\text{-}1(p(t)) = \theta1 + \theta2t$, where $\Phi\text{-}1(p(t))$ - the inverse function of cumulative normal distribution

Best-fit model was found based on p-value (=0.5).

T-D reliability data analysis showed the existence of an increasing trend for some components. A decreasing trend was observed for other components and the constant model was applied. Some interesting issues were observed during the analysis. DG failures recorded at VVER shows that sometimes it is difficult to clarify applicable failure modes (fail to start, fail to run) and it was concluded that the uncertainties in root cause analysis could lead to inadequate interpretation of ageing-trend analysis for DG reliability parameters. Sensitivity study was performed in order to verify used assumptions (sensitivity study was done by combining statistics of both failure modes). Reprocessing of combined data revealed that the best fitted model is a loglinear model with p-value equals 0.13. Incorporation of ageing/ T-D aspects in reliability models will allow prediction of risk profile, hence to enhance efficiency of safety-related decision making on equipment resource assessment, in ageing management programs and safety-related modernizations strategy.

It was concluded that VVER-specific database development is necessary, and in this respect, activities were initiated at IAEA (database interface is ready), and some data were collected and used during case study for Task 4. It was concluded that the guideline (EUR 23954 EN) is detailed and can be used for data processing, but it requires some additional information. Following issues of EUR 23954 EN guideline should be detailed:

- criteria for acceptable p-values,
- information/ decision making procedure for poor fitting cases.

For T-D / ageing trend analysis it is required to have detailed information on observed failures, and it is also important to take into account the so called “burn-in” failures.

An interesting plant tour was organized in the afternoon of November, 10, for the workshop participants. The information center of Gösgen NPP is a very modern one, providing all the necessary information in an interactive and attractive way.

2.2 General Conclusions

The workshop demonstrates the level of interest and progress with Ageing PSA development in each organization. It could be stated that there is a variety of topics under development as well as a level of details in presented results. Based on the presentations and participants experience, there were discussions about topics considered interesting for further development. The most interesting conclusions are presented below:

Many participants would like to see more practical oriented results and examples, in particular, the links with maintenance practice and reliability data analysis and applications of the results. Also it would be good to develop similar evaluations with active components, regarding the ageing of passive components. In order to perform age dependent reliability assessments, current PSA reliability data are not enough and need to be completed by additional data categories. Collection and processing additional data is time and resource consuming process, and to reduce the cost of data collection, this process has to be improved on multipurpose bases.

In most of the countries the assumption about constant failure rate is taken without validation, probably due also to the fact that performing such validation, additional amount of data are required. Possible approaches and methods for validation are proposed in the guideline on reliability parameters estimation, together with identification of the potential difficulties related to the data interpretation (model checking, assumption on renewal process, etc.). Gösgen NPP PSA team uses the trend analysis from 2008. The recommendation of Dr. Klugel was to perform a trend assessment using specific data with 10 years periodicity.

Assessment of passive components ageing was debated intensively among the participants, and discussions included the types of criteria to be considered when deciding an inclusion of a passive component in PSA, as follows:

- passive components considered in Periodic Inspection Program or In Service Inspection Program should be considered;
- evidence of passive components failures from the specific utility or from similar plants (this involve the searching of events in the internal plant database, from processes like reports of abnormal conditions, plant work requests, operator logs and other internal sources)
- expert judgments comprised in technical documents issued by NRC, EPRI, IAEA and their applicability.

It was agreed that passive components have the effect to create an IE much more likely than adding a failure to unavailability of components. It was concluded that AFMEA could be used to decide if the AM is really important, and how large the effect can be (hybrid model, empiric model), and SAFIRE could be used for calculation of rupture failure for piping. Physical modelling could be used in the assessment of passive components, assuming the uncertainties with Monte Carlo analysis, but their modelling in PSA could be difficult (in case of I&C system).

For further development it was considered interesting to discuss different approach that could be used to assess and model the ageing of passive components, and the difference between them (after comparing). Related to activities dedicated to incorporation of Age-Depended Reliability Parameters and Data into PSA Models, it was remarked that many case studies were dedicated to this activity, wishing to demonstrate possible ways to incorporate ageing effects (time-dependent reliability models) into PSA, and using mainly the stepwise failure rate approach for implementation of these effects in the model. It was suggested as appropriate to initiate actions to improve the software, to make it suitable for ageing evaluations – issuing recommendations based on the case studies results and insights could be the next step.

It was concluded also that the incorporation of ageing effects into PSA models can help in the selection, prioritization of SSC susceptible to ageing, improving maintenance measures, replacing important components and performing consistent ageing management as a part of a risk-informed decision-making process. The ageing contributors should be treated distinctively on the basis of importance measures, system unavailability, dominant accident sequences and risk profile.

At the end of the workshop, M. Nitoi expressed her gratitude for the host of the event, for organizing the workshop in such a good way. She also thanked the participants for taking the time to attend the meeting, to prepare the presentations and to engage in the discussions. The evaluation round table, plus the comments made by the participants indicate that the workshop has been a success, not only regarding the very interesting technical programme, but also regarding the premises and hospitality.

APPENDIX A: Workshop Agenda

9th November 2010:

9:00 – 9:30	Opening of the Workshop. Introductions and welcome greetings
9:30 – 9:50	Jens Kluegel - Welcome at the plant site. Introduction of Gösgen NPP activities
9:50 – 10:30	Christiane Bruynooghe – Plant Safety Operation project activities at JRC IE
10:30 – 11:00	Coffee break
11:00 – 11:40	Mirela Nitoi – Ageing PSA Network activities
11:40 – 12:20	Irina Kuzmina – Overview of IAEA Activities and Recent Publications Relating to Safety Assessment
12:20 – 13:00	Robertas Alzbutas - Application of Bayesian methods for age-dependent failure analysis
13:00 – 14:00	Lunch
14:00 – 14:40	Bengt Lydell - A-PSA & Analytical Challenges: Results and Insights from the Integration of Passive Component Reliability in PSA
14:40 – 15:20	Irina Dinu - Overview of recommendations for data and models requirements for passive components ageing assessment in PSA
15:20 – 15:50	Coffee break
15:50 – 16:30	Alexander Getman – Reliability of the NPP mechanical components and optimization of their lifetime management - probabilistic approach
16:30 – 17:10	Guergui Petkov - A Case Study on Incorporation of Ageing Effects into the PSA Model of NPP with VVER1000

10th November 2010:

9:00 – 9:40	J. Kluegel - Activities of NPP Gösgen in the field of Ageing PSA
9:40 – 10:20	A. Rodionov – Verification of ageing trends by screening of safety significant operational events database
10:20 – 10:50	Coffee break
10:50 – 11:30	Kancev Dusko - Age-dependent unavailability modelling of safety systems integrating effects of test and maintenance

11:30 – 12:10	Shahen Poghosyan - Time-dependent reliability assessment for VVER equipment
12:10 – 12:30	Discussion & closure of workshop
12:30 – 13:30	Lunch
13:30	Technical visit to Gösgen-Däniken NPP (2 groups)

APPENDIX B: List of Participants

Number of participants: 21

Participants' geographical spread: 14 countries (EU, Switzerland, Armenia, Russian Federation, USA, Republic of South Africa):

Armenia	1	Romania	2
Austria (IAEA)	1	Russian Federation	1
Bulgaria	3	Germany	1
Slovenia	1	France	1
Hungary	1	South Africa	1
Lithuania	1	Switzerland	4
Netherlands (EC JRC)	1	USA	1

Following is a list of workshop participants including their affiliation.

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APPENDIX C: List of Acronyms

AMP	Accident Management Program
CCDF	Conditional Core Damage Frequency
LTO	Long Term Operation
PLiM	Plant Life Management
EC	European Commission
JRC	Joint Research Centre
IET	Institute for Energy and Transport
PSA	Probabilistic Safety Assessment
CCF	Common Case Failures
PRIS	Power Reactor Information System
NPP	Nuclear Power Plant
APSA	Ageing Probabilistic Safety Assessment Network
MCS	Minimal Cut Sets
NSNI	Division of Nuclear Installation Safety
RAMP	Review of accident management program
SAET	Safety assessment education and training program
SG	Safety guides
SAM	Severe Accidents Management
DM	Degradation Mechanism
CDF	Core Damage Frequency
LERF	Large Early Release Frequency
RI-ISI	Risk Informed In Service Inspection
SSC	Structures, systems and components
R-DAT	Reliability Data Analysis

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Abstract
JRC – IET Petten organized with the support of the nuclear power lant Gösgen-Däniken, Switzerland, an EC Workshop on Investigation of Ageing Effects using the Probabilistic Safety Assessments. The goal of the workshop was to present and discuss the developed methods and approaches and the results obtained for application of reliability and PSA techniques on evaluation and management of NPP ageing. 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. The workshop contained a general session, dedicated to activities of different organizations in PSA field, and a technical session, focused on the results obtained in application of reliability and PSA techniques on evaluation and management of NPP ageing. Based on the presentations and participants experience, discussions about topics considered interested to be developed further were organized. The arising conclusions are presented.

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