IAEA-EBP-LTO-13

18-03-05

EXTRABUDGETARY PROGRAMME ON SAFETY ASPECTS OF LONG TERM OPERATION OF WATER MODERATED REACTORS

MINUTES OF THE PROGRAMME'S WORKING GROUP 2 SECOND MEETING

16-18 November 2004

Vienna, Austria

INTERNATIONAL ATOMIC ENERGY AGENCY

1. INTRODUCTION

The number of Member States giving high priority to extending the operation of nuclear power plants beyond their initial license is increasing. Decisions on long-term operation (LTO) involve the consideration of a number of factors. While many of these decisions concern economic viability, all are grounded in the premise of maintaining plant safety. The IAEA recognized this new industry initiative; therefore, in the 1990's, it developed comprehensive generic guidance on how to manage the safety aspects of physical ageing. It was recognized, however, that internationally agreed-upon, comprehensive guidance was needed to assist regulators and operators in dealing with the unique challenges associated with the LTO issue.

In response, the IAEA initiated this Extrabudgetary Programme (Programme) on 'Safety aspects of long term operation of water moderated reactors' (original title was 'Safety aspects of long term operation of pressurized water reactors'). The Programme's objective is to establish recommendations on the scope and content of activities to ensure safe long-term operation of water moderated reactors. The Programme should assist regulators and operators of water moderated reactors, and, in particular WWERs, in ensuring that the required safety level of their plants is maintained during long term operation, should provide generic tools to support the identification of safety criteria and practices at the national level applicable to LTO, and should provide a forum in which MS can freely exchange information.

The Programme activities are guided by the Programme Steering Committee (SC), follow the overall SC Programme Workplan and SC Terms of Reference, [1], and are implemented in 4 Working Groups (WG). The WGs focus on:

- general LTO framework (WG 1);
- mechanical components and materials (WG 2);
- electrical components and I&C (WG 3);
- structures and structural components (WG 4).

Further detailed information on the Programme could be found at: http://www-ns.iaea.org/projects/salto .

The 2nd meeting of WG 2 was organized at the IAEA in Vienna, Austria, 16-18 November 2004. The purpose of the 2nd meeting of WG 2 was to:

- Review the input from Country Information Reports (CIRs)
- Review the IAEA draft report IAEA-EBP-LTO-03 Rev.1 [2]
- Assign review action items (develop/agree review plan)

The Agenda for the Meeting is provided in Appendix I. The list of participants is provided in Appendix II. and the presentations made during the meeting are provided in Appendix III.

2. MEETING SUMMARY

Mr. Radim Havel opened the meeting. Mr. Havel provided an overview of the Programme's status, outlined the expected outcome from the WG 2 meeting, and stressed the important role of WG-2 in the SALTO Programme. Mr. Havel then turned the meeting over to Mr. Tom Taylor, the Secretary for WG2 who then chaired the meeting for Mr. Vladimir Piminov who was unable to attend the meeting.

The meeting started with a series of presentations by each country that summarized the information contained in the Country Information Reports (CIRs).

2.1. National presentations

The information presented for each CIRs is briefly summarized below, complete CIR reports for each country are available at the Programme ftp server.

BULGARIA
In accordance with the Safe Use of the Nuclear Energy Act (SUNEA) and the newly released Ordinance for the order of issuing licenses and permits for safe use of the nuclear power it is obligatory for each power unit and any other nuclear facility to obtain a long-term Operation License. On the basis of Safety Analysis Report, the license is issued for long-term activities with a validity of up to 10 years.

Different screening criteria (safety relevance, availability relevance and availability of spare parts, replaceability, costs, accessibility, operational performance, calculations and analysis) are applied for the identification and selection of critical components that are within the long term operation.

The great number of Design modifications generates the need of establishing and maintaining a strict consistency between the plant documentation, installed equipment and design requirements. KNPP Configuration management system consists of several components - document control and records management, equipment configuration control, design change, design requirements.

Evaluation of Rest Life Time (RLT) of Kozloduy NPP units 3 & 4 was executed by a Consortium between Framatome ANP GmbH and Atomstroyexport, Russia. The results show that there are no general problems that might affect the plant operation during the expected 30 years of operation. Furthermore, for the biggest part of the most-important components, it was found that they could operate significantly longer without any major interventions. Some of the activities recommended in result of the performed RLT evaluations of units 3&4 have already been performed and some activities are in process of execution.

The non-destructive in-service inspection of the plant equipment has been performed in compliance with documents, updated in 2000 and approved by Bulgarian Nuclear Regulatory Agency (BNRA). The scope of inspection is expanded, so that to cover systems and components, installed during the modernization and reconstruction of the units. A programme for qualification of in-service inspection systems is developed and accepted by BNRA and now it is in process of implementation.

The maintenance activities are planned in accordance with the applicable requirements of the Bulgarian legislation. They are carried out during the annual outages of the units. The maintenance of the equipment is carried out by qualified personnel in accordance with the requirements of the maintenance instructions developed for each specific type of equipment.
Monitoring and surveillance specimen programmes:

The reactor body state is crucial for the life time of the NPP. Prevention and monitoring measures in connection with the embrittlement issue of the Reactor pressure vessel (RPV) are: Neutron Flux Reduction on RPV wall, Annealing, Prevention of cold overpressure of the RPV.

The leak before break (LBB) concept was investigated and applied for the main primary pipelines Dn 500 and 200 mm. The full set of systems includes three built on different principles, sensitive and diverse between themselves systems: Acoustic Leak Detection System (ALUS); Confinement Air Cooler Condensate Monitoring System (SKVB); Moisture Leak Detection System (FLUS).

The updated "Safety Analysis Report" (SAR) of the units is the basic document for issuing the operation license to NPP. For all units of Kozloduy Nuclear Power Plant there are plant-specific PSA developed and updated on a regular basis. Additional analyses concerning the assurance of the unit's safe and reliable future operation are under development.

All these activities are intended to substantiate the possibility of operating the units until the end of their design life at minimum and to keep open the possibility of lifetime extension as far as this would be technically feasible, acceptable and economically reasonable.

CZECH REPUBLIC

Legal framework in the CR is covered by Law No. 50/1976 Coll., Building Act (licence without validity limitation) and Law No. 18/1997 Coll., Atomic Act (operating licences with necessity of periodic renewal).

It follows from the definition of SSC set for LTO and current legislation that the set of SSC for LTO will consist of the following equipment subsets:

- **A –** SSC of 1 and 2 safety classes from the categories of:
  - **A.1 –** SSC which could be replaced with difficulties
  - **A.2 –** SSC whose availability has to be ensured for decommissioning
- **B –** SSC of 1, 2, and 3 safety classes required by regulatory body
- **C –** SSC which can cause considerable maintenance and investment.

For supporting of CM and LTO we are collecting and, if necessary, we are going to reconstitute Design Bases for both NPPs. Projects are in progress.

Measures mitigating ageing of NPP equipment are applied by the operators on the basis of evaluation of lifetime consumption. The inspections of the operated nuclear facility are ensured by its operator in accordance with the In-service Inspection Programme. The Programme serves for ensuring and implementation of periodic and operative in-service inspections of machine equipment, electrical and control system equipment and building equipment included into the list of Specific and Restricted Technical Equipment. Inspection activities are carried out mostly during unit outages for refuelling, but also out of them. The In-service Inspection Programme is approved by regulatory body and is a licence document for the power plants.

NDT methods are qualified within the meaning of ENIQ. Augmented Inspection Programs are held for issues as erosion/corrosion (CHECWORKS computer program), steam generator tubing (on the basis of N16 measurement in steam and NA24 in surface blowdowns, the rate of deposit formation on the heat-exchanging surfaces, the eddy-current inspection), intergranular stress corrosion cracking (for SG primary header).

Programmes of qualification of equipment important from the viewpoint of safety have been introduced in all the NPPs operated in the CR. An important aspect of these programmes in relation to LTO is the fact that the required qualified lifetime of the equipment subject to qualification is the same as the design lifetime of the NPP (30, 40 years).

For ensuring of LTO, it will be therefore necessary to reassess the state of qualification for equipment with qualified lifetime shorter that that required for LTO. General principles used in the qualification programmes are based on current international practice as described in the standards and guidelines of IAEA, IEC, and IEEE. Currently, the qualification programmes are in the phase of maintaining of the qualification state for the qualified equipment and remedial measures for the non-qualified equipment.

The system of programme of surveillance over safe operation covers implementation of functional tests, inspection activities, diagnostics, in-service inspections (ISI), engineering inspection, and special tests and experiments and other inspection activities applied on all systems and equipment important from the viewpoint of nuclear safety. When setting the time intervals of the implementation of the required inspection activities as well as when defining the criteria of acceptability, the data of SAR, Operational Limits and Conditions, and technical conditions of suppliers and manufacturers of the defined equipment groups are respected.

Except of the standard surveillance specimen programme designed by OKB Gidropress, a supplementary surveillance programme has been designed and applicated.
The following conclusions or analyses relate to the planned 40 years of lifetime in SAR for NPP Temelin: Evaluation of Strength and Lifetime of the Most Important Machine Components, Reactor Vessel Irradiation, Limit Values of Pressure and Temperature. In the current version of SAR NPP Dukovany, there is only a summary of PTS analyses in RPV chapter related to the 40 years of assumed reactor operation showing marked dependence on time out of the time-conditioned (by the assumed operation duration) analyses.

A number of other analyses (seismic calculations, evidences of resistance against extreme meteorological conditions, fall of an aircraft, pressure waves, strength calculations of component stress under emergency conditions and abnormal states count with certain material characteristics and properties of the stressed buildings and structures which are also time-dependent in general. Obviously, expert statement of material specialist would be sufficient for a relatively short period in the order of several decades.

The analyses of residual lifetime assessment of the main machine components towards fatigue stress and other degradation mechanisms are updated periodically according to the real operational data and therefore respect the real plant state regardless of the design assumptions of the way and duration of operation. Periodic updating of these analyses and also of all other analyses mentioned in SAR is assumed and performed periodically within the framework of ten-year reviews of SAR (and PSR) regardless of extension of the period of NPP operation.

FINLAND

The Nuclear Energy Act (990/87) and the Nuclear Energy Decree (161/88) regulate the use of nuclear energy in Finland. General safety requirements for nuclear power plants are presented in the Council of State Decision (395/91). By virtue of the Nuclear Energy Act and the Council of State Decision, the Radiation and Nuclear Safety Authority (STUK) issues detailed regulations, the YVL Guides, relating to the safety of nuclear power plants. It is prescribed that it shall be the licence-holder's obligation to assure the safe use of nuclear energy and the licensee is obliged to demonstrate that safety principles are met. For further safety enhancement, actions shall be taken which can be regarded as justified considering operating experience and the results of safety research as well as the advancement of science and technology.

In Finland the operating licence of a nuclear power plant is granted for a fixed term. Usually the period of validity of the licence is ten years. STUK controls the operation of a nuclear facility to ensure that the operation of the facility is safe and complies with the licence conditions and the approved plans. The systems, structures and components important to safety shall be designed, manufactured, installed and operated so that their quality level and the inspections and tests required to verify their quality level are adequate considering any item's safety significance.

Assessment of nuclear power plant safety continues systematically after the granting of the operating licence during operation. Regulatory control of operating nuclear power plants contains reviews and inspections such as periodic inspections which are specified and registered by STUK in a plant-specific program, inspections which the power company is obliged to request in connection with measures carried out at the plant or which STUK conducts at its discretion, and safety assessment based on operating experience and on safety research as well as other information obtained after the granting of the operating licence.

The periodic safety review (PSR) is required to be carried out to renew the operating licence. A Safety Guide NS-G-2.10 on the Periodic Safety Review of Nuclear Power Plants provides guidance on a systematic assessment of the effectiveness of the ageing management programs as well as on overall plant safety. An objective of the PSR is to determine the actual condition of SSCs important to safety and whether it is adequate for them to meet their design requirements.

Ageing management practices in the Loviisa NPP applicable to long-term operation are summarized in the CIR. In connection with the renewal of the operating licence, the plant construction and safety are assessed to comply with the current requirements. The original design lifetime specified for the Loviisa power plant was 30 years and some components were given component specific lifetimes. The issues affecting the engineering lifetime of the plant have to be identified and assessed for the continued and especially for the extended operation of the plant.

In connection with the operating licence renewal STUK supervises whether a systematic plant life management program has been established to ensure adequate safety functions of the SSCs during the applied operating period. Effectiveness of the plant life management procedures and functioning of the program as well as technical aspects of ageing management activities are evaluated. The plant shall have inspection and control programs that monitor the effects of ageing of the SSCs, instructions to ensure required integrity and functional capability of the SSCs and maintenance programs to prevent degradation and to maintain the SSCs' ability to perform their intended function.

HUNGARY
Summary not available

RUSSIAN FEDERATION
Summary not available

SLOVAK REPUBLIC
As regards legislation conditions of possible life extending of Slovak NPP’s, operation license is open referring to its duration in principle and no legal restriction have not existed for extending up to now. But there are several conditions in this area existing which fulfilment is necessary after all.

Basically, preparing and submitting PSR in each ten years interval is necessary to enable NPP operation including basic ageing processes evaluation according to regulations of Slovak Regulatory Authority (No 121/2003 „Coll on nuclear safety assessment“, No 167/2003 „Coll on requirements for nuclear safety of nuclear installation“). Regulatory body of SR has issued guidelines “Ageing management of NPP–requirements” recently defining basic requirements laid on AMP according to IAEA methodology. Present effort is concentrated on complex legislative document proposal elaboration dealing with possible extending operation life behind NPP design values (in the frame of new AMP R&D project).

Systematic approach for LTO and AMP including SSCs scope development is not available up to now in SR. SSCs selection for AMP elaboration was performed based on safety criteria (IAEA methodology and NPP SSCs safety classification) in new AMP R&D project. Output from project should provide proposal of legislative documents covering LTO area.

Slovak NPP SSCs safety classification is based on Decree 5/77, Decree 6/1980 of former Czechoslovak Nuclear Regulatory Body (CSKAE), Regulation 66/1989 of Slovak Authority for Labour Safety and Regulation 317/2002 of Slovak Nuclear Regulatory Body. Appendix 2 of Regulation 317/2002 defines criteria necessary for SSC categorisation into three safety classes. Specific NPP SSCs safety classification was developed based on this Decree in each Slovak NPP.

At present following organisational structure is being developed which should be valid both for EBO NPP and for EMO NPP from 1st January 2005. New structure consists of Technical Support Division, which contains Equipment Configuration Changes Management Department. This department shall take care of design administration, equipment configuration changes management and ageing management (including qulification problems). Therefore it shall contain three groups: Technical Engineering Group, Equipment Configuration Changes Administration Group and Design Administration Group. Technical Engineering Group should cover the problems of ageing management programmes mainly. Basic rules proposals have been prepared for this area on the level of NPPs recently.

Ageing management, besides other activities, is strongly required to enable life extending of Slovak NPP units. An effective ageing management of systems and components can be accomplished under a systematic programme that integrates existing subprograms. In Slovakia such complex programme hasn’t existed up to now, but several potential parts of complex ageing management programs have been available in each operating NPP for many years (in-service inspection program, diagnostic systems utilization, maintenance and repair system, special fatigue damage monitoring system, errosion-corrosion monitoring programme etc.).
New systematic AMP approach should be developed in the frame of R&D project “Ageing management and lifetime optimisation of NPPs with WWER 440 reactors”, which started in 2002 and will be finished in 2005.

Ageing mitigation measures have been applied by “case to case” procedure up to now. It means that if some ageing problem had been identified in NPP operation mitigating measures were developed for solution of this problem.

The inspections of the operated nuclear facility are ensured by its operator in accordance with the In-service Inspection Programme. The Programme serves for ensuring and implementation of periodic and operative in-service inspections of machine equipment, electrical and control system equipment and building equipment included into the list of Specific and Restricted Technical Equipment.

Inspection activities are carried out mostly during unit outages for refuelling, but also out of them. The In-service Inspection Programme is approved by regulatory body and is a licence document for the power plants.

The qualification of NDT systems is under way now. Augmented Inspection Programs are held for issues as erosion-corrosion (computer program application and thickness measurement) and steam generator tubing.

Programmes of qualification of equipment important from the viewpoint of safety have been introduced in all the NPPs operated in the SR. General principles used in the qualification programmes are based on current international practice as described in the standards and guidelines of IAEA, IEC, and IEEE. Currently, the qualification programmes are in the phase of maintaining of the qualification state for the qualified equipment and remedial measures for the non-qualified equipment.

The diagnostics systems are applied for primary piping, MCP, RPV and RPV internals, DG, TG, main steam piping etc. Standard surveillance specimen programme for NPP RPVs, designed by OKB Gidropress, was replaced by more sophisticated surveillance programme. Several corrosion loops are available (primary piping). Nondestructive material properties tests including hardness measurement application has been used several times to identify mechanical properties. Destructive tests have been utilized only in cases where part replacement carried out.

Special loading measurement systems exist for fatigue damage monitoring system of primary components consisting of a)off-line fatigue calculation of components critical points and b)special fatigue monitoring systems (stratification). Standard chemical regimes monitoring are available for primary and secondary circuits.
Plant specific safety analysis connected with planned (extended) NPP life-time should contain:

a) RPV integrity analysis
b) RPV internals ageing analysis
c) fatigue damage analysis
d) thermal ageing analysis, seismic analysis of pipings
e) LBB analysis and postulated piping break consequences

SWEDEN
Nuclear power in Sweden has been a very prominent issue in the political debate since the 1970’s. In 1997, an Act on the phase out of nuclear power was taken in the Parliament. This Act authorizes the Government to shut down a nuclear reactor as a consequence of conversion of the energy system. The location, age, design and importance for energy system of a particular reactor shall be considered when taking such a decision. Based on the new Act Barsebäck 1 was shut down in November 1999. Barsebäck 2 will be shut down in 2005. The next plant in line is to be identified in a few years time.

At the same time an intensive activity is going on dealing with application to increase the power outputs from 8 of remaining 10 reactors. An additional special nuclear power tax has increased the economical burden together with new nuclear regulations. The new regulations issued that will be in force from 2005-01-01 will furthermore cost some hundreds of millions of euros in modernization and upgrading of the remaining power plants. To get better economics of the plant when these new regulations will be in force there will application for the up rating the power.

Besides new regulations from the nuclear regulatory authorities there are also new regulations from other regulatory authorities that must be met. This means for example modernization and replacement of fire detection and fire extinguish systems.

In –service inspection and in-service testing are required to based on the relative risk for fuel damage and based on qualified technology. Equipment, procedures and personnel must be qualified together. This has led to focus on ISI of components in the RCPB (reactor coolant pressure boundary) and certain internal parts in the RPV (reactor pressure vessel). ISI has led to replacement or repair of welds and components in the RCPB and internals in the old reactors. Almost all of the defects found up to now have been faults left from the installation when no such ISI techniques were available.

Long-term operation in Sweden is based on periodic safety review (PSR) of the plant. The PSR is required to be performed at least every ten years. Oskarshamn 1 for example have done three such PSR:s up to now. The PSR must cover the ten years since the last PSR took place and predict what has to be performed to guarantee safe operation for the next ten years of operation. The regulatory authority will when determine how long time this operation could take place based on the PSR.

Long time operation is an on going activity in the same way as nuclear safety. Modernization of a nuclear power plant is the basis for LTO in Sweden. Modernization is mainly based on new regulatory requirements, operational experience, utility policy or goals, spare parts not to be found, obsolete equipment, aging of components and materials.

UKRAINE

Nowadays the most power units of Ukrainian NPPs have reached a half or more of determined by the project 30-year operation term.

The operating organization (Operator or Utility) with the support provided by the Government of Ukraine has developed the «Comprehensive Program of Activities on the Running NPP Power Unit Service-life Extension». The program includes four groups of efforts: development of missing normative and technical documentation, implementation of measures to assess the technical condition of the major equipment and building structures, funding sources, implementation the periodic safety assessment activities.

Approved in Ukraine approach means, that aging and life management is an economically and technically expedient way of prolongation of NPP operating terms. Extra life time must be defined according the evaluation results of technical conditions of equipment and on the safety grounds.

For realization of this approach following measures are being fulfilled:

- Analysis and accounting of operation regimes of equipment;
- Monitoring of equipment metal conditions and building constructions;
- Planning and realization of measures of long-term operation of basic equipment and building constructions;
- Optimization of technical service and repair to enlarge safety and reduce impact of damaging surfaces;
- Data gathering, analysis, development and saving of technical and radiation conditions of elements;
- Licensing.

With this purpose the type program of aging management was developed and is presently on approval in regulatory body.

The works object of aging management of power units elements of Ukrainian NPPs is to provide a required level of safety during all term of operating (incl. beyond the project) as well as to reach a maximum affectivity in its operation by carrying out measures for timely detection and softness of degradation caused by elements aging to be sure in their integrity and capacity of work.

As a result of such actions is development and application of technically and economically expedient measures to prevent failures of NPP power unit elements caused by aging of their elements.

USA

The legal framework for the safe regulation of nuclear activities is created by Atomic Act (1954) as amended and by requirements stipulated by 10 CFR 50, Regulatory guides, industrial standards (ASME), ACI, AISC and other official documents.

For legal and economic considerations, the design lifetime in USA is stipulated by 40 years. With the aim to enable license renewal (LTO) the licence renewal rule 10 CFR 54 was issued by the US regulatory body – NRC in 1991. This rule defines all conditions of licence renewal process (scope, documentation, regulatory review etc.). Set of guidance documents was issued specifically the Generic Aging Lessons Learned (GALL) report, the Standard review plan for license renewal (SRP-LR), Regulatory guide 1.188, and Interview staff guidance.

On the basis of the following significant NRC determinations: 1) existing regulatory process is adequate for ensuring safety of operating plants; 2) current licensing date base is adequate for further operation and 3) issues relevant to the current operation will be addressed by the regulatory process, the license renewal scope is focussed to:

- safety related systems a components (pressure boundary, shut down of reactor and prevent or mitigation of off site consequences
- non-safety related SAC which can adversely influence safety functions
- SSC relied for compliance with five regulations (10 CFR 50, 48,49,61 and 63)

The basis for Compliance with 10CFR50.24 is the Standard Review Plan for License Renewal - (SRP-LR) which includes the following guidance for review:

- Acceptance Criteria
- Review Procedures
- Evaluation Findings
- Implementation

The SRP-LR also references GALL Report for generic aging evaluation as Technical Basis document. The GALL report contains specific requirements for aging management programs. The US CIR for WG2 contains a detailed summary of the objectives and requirements for each aging management program.

During discussion of the CIR, it was agreed that the CIR reports would be revised to address the following topics:

- Sections 1.0 and 2.0 of the CIR reports need only contain a brief description of the laws and regulations that directly affect WG2 activities. It is assumed that WG1 will address the global regulatory requirements for LTO.
- Each CIR should provide a brief description of the pertinent research projects relative WG2 LTO activities. The listing should include the project titled, objective, a point of contact and the scheduled start and completion dates for the project.
- Each CIR should include a list of any unresolved technical or regulatory issues that are pertinent to WG2 activities.

2.2 Discussion of the review process for WG2 CIRs

WG2 agreed to a similar review approach as that adopted by WG1. Four Review Groups were created to review the specific CIR sections identified below. Each review group identified a leader to facilitate the in-depth discussion and ensure that the review process was conducted in a timely fashion so that the review schedule for WG2 (see below) would be completed on time.

Review Group	Assigned CIR Sections
Group 1 – Bulgaria, Hungary and EC Group Leader – Sandor Ratkai	Sections 1.0 & 2.0 and Section 3.2
Group 2- Czech Republic and Slovak Republic Group Leader – Robert Krivanek	Section 3.1.1 to 3.1.3
Group 3 – Russia and Ukraine Group Leader – Sergey Malkov	Sections 3.1.4 to 3.1.7
Group 4 – Finland and Sweden Group Leader - Fredrik Barnekow	Sections 3.1.8 to Sections 3.1.10

During the discussion of the review process for WG2 CIR reports, WG2 members agreed to the following review schedule.

Action Item	Scheduled Date for Completion
Revise CIRs Based on 2nd WG2 Meeting	Dec. 15, 2004
Review Groups Complete Review of CIR Sections	Feb. 15, 2005
WG Leader & Secretary Complete Review Report – Combine all Review Groups	April 15, 2005
Receive Feedback from WG2 members on Review Report	May 1, 2005
Finalize Review Report	May 15, 2005
3rd WG Meeting	May 30 or June 6, 2005

The review schedule was developed taking into account the WG2 schedule and the overall program schedule including the next WG2 and WG3 meeting which is planned to be held May 30 to June 3rd at the Oskarshamn nuclear power plant in Sweden.

2.3 Review the draft Standard Review Process Rev.1

WG2 members reviewed Section 5.0 of the draft IAEA-EBP-LTO-03, Standard Review Process Rev.1 [2]. This report has been revised to include guidance for conducting the reviews of the CIRs and describes a process that is to be used to develop draft WG reports based on the review of CIRs. The objectives of the review process outlined in the revised report are as follows:

- Identify and define the common elements of national practices for maintaining safe operation during LTO.
- Identify and define differences of national practices for maintaining safe operation during LTO.
- Identify generic challenges (LTO related) to be resolved.
- Develop guidance on developing and improving programmes and practices to support safe LTO.

The WG2 members agreed with the objectives of the review guidance, however, there was considerable discussion on the use of Appendix V in of [2]. Many WG2 members did not agree with the use of the Safety Factors and Elements from the IAEA Periodic Safety Review document. Many WG2 members also felt that there was not good agreement between the CIR report table of contents and the Safety Factors/Elements provided in Appendix V. During the discussion of the evaluation of CIRs with reference to the Safety Factors/Elements provided in Appendix V of the revised of [2], the following viewpoints emerged.

All WG2 members agreed that the topics described in the Safety Factors/Elements presented in Appendix V are necessary for safe operation of a nuclear power plant.

However, several members of WG2 felt that many of the Safety Factors/Elements presented in Appendix V were topics that are best used as prerequisites for a good LTO program and should not be addressed in by the SALTO programme. These members expressed the technical opinion that only those safety factors specific to LTO should be addressed in the SALTO project.

Several members of WG2 also felt that there was no clear distinction between normal operation, prerequisites to LTO and issues specific only to LTO. Therefore any safety factor that needed to be addressed as part of LTO should be evaluated in Appendix V.

WG2 members resolved the issues discussed during the review of [2] as follows. WG2 members reviewed Appendix V and agreed upon which Safety Factors should be evaluated in Appendix V. The results of the WG2 review are included in Appendix IV to this report. WG2 members also agreed to the following change in review process. The review groups would review the CIR reports and only completed the review Table (Appendix IV to of [2]). The WG2 Leader and Secretary would complete the evaluation of Safety Factors/Elements presented in Appendix V as part of the process in developing a consolidated Review Report which will integrate the reports from all of the review groups.

3. ACTION ITEMS

The following actions items resulted from the meeting.

1. WG2 members agreed to provide one page summaries of presentation material by November 25, 2005.
2. WG2 members agreed to revise CIRs by December 15,2004
3. Mr. Taylor agreed to complete a draft of the meeting minutes by the first week in December.

4. WG2 members agreed to provide any comments on the draft within one week from the date of receipt of the draft.

4. REFERENCES

[1] Minutes of the Programme's 1st Steering Committee Meeting, IAEA-EBP-LTO-01, Vienna, 2003 (internal EBP report).

[2] Standard review process Rev.1, IAEA-EBP-LTO-03 Vienna, 2004 (internal EBP report).

[3] Programme's Working Groups Workplans, IAEA-EBP-LTO-08, Vienna, 2004 (internal EBP report).

APPENDIX I. AGENDA

16-Nov-2004

9:00	Introduction and Agreement of Agenda Items	T. Taylor
9:20	Presentation & Review Sections CIRs	T. Taylor (Lead)
9:30	Sections 1&2 will be reviewed first	Bulgaria
9:45		Czech Republic
10:00		Finland
10:15		etc.
	Discussion on Section 1&2 (30 minutes)	
	Section 3.0 will be reviewed next	Bulgaria
		Czech Republic
		Finland
		etc.
	Section 4.0 will be reviewed last	Bulgaria
		Czech Republic
		Finland
		etc.

In order to complete the reviews in a timely manner each member is requested to read all CIRs prior to the meeting. The presentation will be a brief overview of information, followed by questions. The objective of the review is to help achieve consistency of information. It is likely that all members will need to revise their respective CIR to address questions by other members.

10:30 – 11:00	*Coffee Break*
11:00 – 12:30	Continue with Review
12:30 – 14:00	*Lunch*
14:00 – 15:30	Continue with Review
15:30 – 16:00	*Coffee Break*
16:00 – 17:30	Continue with Review
18:30	*Reception*

17- Nov-2004

9:00 – 10:30	Continue with CIR Review	T. Taylor
10:30 – 11:00	*Coffee Break*	
11:00 – 12:30	Continue with Review	
12:30 – 14:00	*Lunch*	
14:00 – 14:45	Agree upon Action Items for CIRs	T. Taylor
14:45 – 15:30	Review – IAEA Report	T. Taylor
15:30 – 16:00	*Coffee Break*	
16:00 – 17:30	Continue Review of IAEA Report	

18-Nov-2004

9:00 – 12:30	Wrap Up Discussion, Agree on Assignments and on next meeting	T. Taylor

APPENDIX II.
LIST OF PARTICIPANTS

IAEA	**Mr. Radim Havel**	**NSNI-ESS**

BULGARIA

Ms. Teodora Ribarska
Kozloduy NPP
3321 Kozloduy
Bulgaria
Tel.: +359 973 7 2067
Fax: +359 973 8 0126
E-mail: ribarska@npp.cit.bg

CZECH REPUBLIC

Mr. Jiri Zdarek
UJV a.s.
25068 Husinec
Rez No.130
Czech Republic
Tel.: + 420 266173544
Cell: + 420 602761553
Fax: + 420 220940519
E-mail: zda@ujv.cz

Mr. Robert Krivanek
Dukovany NPP
Dukovany 269
675 50 Dukovany
Czech Republic
Phone 00 420 568 815 200
Fax 00 420 568 815670
Email Krivar1.edu@mail.cez.cz

FINLAND

Ms. Ritva Korhonen
Fortum Nuclear Services
Ltd. Rajatorpantie 8
01600 Myyrmaeki, Vantaa
Finland
Tel.: + 358 10 4532483
Fax: + 358 10 4533355
E-mail: Ritva.Korhonen@fortum.com

HUNGARY

Mr. Sandor Ratkai
Paks NPP
P.O. Box 71
H-7031 Paks
Hungary
Tel.: +36 75 508576
Cell: +36 20 9522241
Fax: +36 75 507036
E-mail: ratkai@npp.hu

LITHUANIA	**Ms. Jelizaveta Korikova** Parko 19/1 - 69 31132 Visaginas Lithuania Tel: +370 386 24427	***Interpreter***
RUSSIAN FEDERATION	**Mr. Sergey Malkov** Rosenergoatom Bolshaya Ordynka 24/26 109017 Moscow Russian Federation Phone: 00 7 095 2206368 Fax: 00 7 095 2206364 Email: malkov@rosenergoatom.ru	
SLOVAK REPUBLIC	**Mr. Miloslav Hrazsky** VÚJE, Trnava, a.s. Okruzna 5, Trnava 91864 Slovak Republic Tel.: + 421 805 605 226 Fax: + 421 805 42750 E-mail: hrazsky @vuje.sk	
	Mr. Frantisek Kalmancai Mochovce NPP 935 39 Mochvce Slovakia Phone: +421 36 636 3506 Fax: +421 36 639 1120 Email: kalmancai.frantisek@emo.seas.sk	
SWEDEN	**Mr. Fredrik Barnekow** OKG AB 572 83 Oskarshamn Sweden Tel.: + 46 491 786450 Cell: + 46 070 3996142 Fax: + 46 491 787558 E-mail: Fredrik.Barnekow@okg.sydkraft.se	
UKRAINE	**Mr. Sergiy Kostenko** SNRCU State Nuclear Regulatory Committee of Ukraine Arsenalnaya St. 9/11 Kiev 01011 Ukraine Tel.: + 380 44 5583624 Fax: +380 44 5587839 Email: kostenko@inspect.snrc.gov.ua	

Ms. Zoya Gubenko
NAEK "Energoatom"
Vetrova, 3
Kiev 01032
Ukraine
Tel.: +38 044 2010920
Fax: +38 044 2944292
E-mail: z.gubenko@direkcy.atom.gov.ua

USA

Mr Tom T. Taylor ***Leader***
Pacific Northwest National Laboratory (PNNL)
2400 Stevens, Mail Stop K5-26,
Richland, WA 99352,
USA
Tel.: +1 509 375 4331
Fax: +1 509 375 6736
E-mail: tt.taylor@pnl.gov

EUROPEAN COMMISSION

Mr. Claude Rieg
EC, Directorate General JRC
Westerduinweg 3
P.O. Box 2
1755 ZG Petten
The Netherlands
Tel.: +31 224 565153
Fax: +31 224 565 637
E-mail: rieg@jrc.nl

APPENDIX III.
PRESENTATIONS' HANDOUTS

Appendix IV
Results of WG2 Review of Safety Factors/Elements

PLANT DESIGN	Evaluation			
	4	3	2	1
A detailed description of the plant design, supported by layout, system and equipment drawings (This Element should be used for Scoping)				
Are plant design drawings referenced or reviewed as part of application for LTO?				
List of Systems, Structures and Components (SSCs) important to safety and their classification (the list of SSCs should include only those within the scope of LTO – Shoul be a procedure for Scoping and Screening)				
Is there a peer reviewed technical basis document used as the basis for identifying SCCs for LTO?				
Is the list of SCCs important to safety reviewed by the regulatory authority?				
Documented design basis (original and updated). (Used for LTO Scope)				
Is the design basis available for peer review as part of the process for LTO licensing?				
Is the plant design basis documentation available for regulatory audit?				
Has the design basis been peer reviewed and approved by the regulator?				
Significant differences (strengths and shortcomings) between the present plant design and the current standards (used for comparison). (This needs to be a decision at the begin of LTO)				
Does the application for LTO, reference a formal process for review and commitment to update plant SCCs important to safety during LTO?				
ACTUAL CONDITION OF SYSTEMS, STRUCTURES AND COMPONENTS	Evaluation			
	4	3	2	1
List of SSCs important to safety and their classification.				
Is the list of SSCs based upon a well defined criteria that uses PRA or design bases?				
Was an independent peer review conducted by subject matter experts of the screening criteria basis?				
Are the screening criteria and the results of the screening criteria approved by the regulatory authority?				
Information about the integrity and functional capability of SSCs important to safety, including material case histories.				

Are fabrication QC records and in-service inspection records of SCCs reviewed by the regulatory authority on a regular basis?				
Does the regulator use lessons learned from material failures to produce generic issues?				
Have design changes ever resulted from records of service failures kept at the plant?				
Information on existing or anticipated obsolescence of any SSCs important to safety.				
Is a well defined system used to document and control design changes in the plant?				
Is the technical basis for design changes compared against the design basis, justified and approved by the regulatory authority?				
Findings of tests which demonstrate the functional capability.				
Are the results of tests reviewed/approved by the regulatory authority?				
Is there a well structured system for recording and maintaining functional records?				
Are the acceptance criteria for functional tests based upon design criteria or other well defined basis?				
Results of inspections.				
Are the results of previous examinations reviewed prior to subsequent examinations?				
Are the test results trended, if appropriate?				
Are the results of examinations reviewed by the regulatory authority?				
Is there a well defined process for regulatory review and approval of examinations that are not conducted in accordance with established standards?				
Is there a well defined process for review and approval of examinations that are do not meet acceptance standards?				
Maintenance records.				
Does work order documentation include design review and root cause analysis?				
Are "as found" and "as left" information included in the maintenance records?				
Does the regulator review plant maintenance records for potential future problems?				
Description of the present condition of SSCs important to safety.				
Do existing preventative maintenance frequency's depend on plant experience or manufactory's recommendations?				
Are the preventative maintenance trends maintained for critical SCCs				
Are current preventative maintenance inspections addressing all know aging degradation mechanisms?				
Description of the support facilities available to the plant both on and off the site, including maintenance and repair shops.				
Does the plant maintenance infrastructure include facilities necessary for extended operation?				

EQUIPMENT QUALIFICATION **Is Important to Long Term Operation – This is Important to normal Operation)**	Evaluation			
	4	3	2	1
(Scoping Procedure) List of equipment covered by the equipment qualification programme and a list control procedure.				
Is there a well defined basis for developing a the list of equipment that is covered under an equipment qualification program, such as design basis?				
Qualification report and other supporting documents (e.g. equipment qualification specifications, qualification plan).				
Is the basis used for acceptance criteria for the equipment qualification reviewed an accepted by the regulatory authority?				
Is the qualification report used to support meeting the design basis criteria?				
Verification that the installed equipment matches the qualified equipment.				
Is there a well defined configuration control process that ensures that installed equipment matches qualified equipment?				
Procedures to maintain qualification during the installed life of the equipment.				
Is there a well defined qualification surveillance that ensures equipment qualification is maintained during the life of the plant?				
Is the frequency of surveillance supported by analyses?				
Mechanisms for assuring compliance with these procedures.				
Does the regulatory authority periodically audit compliance to procedures that are part of the equipment qualification program?				
Surveillance programme and a feedback procedure to ensure that ageing degradation of qualified equipment remains insignificant.				
Are the surveillance procedures based upon internationally accepted practices?				
Is there a well developed process for incorporating service experience into surveillance inspections?				
Does the surveillance program acceptance criteria including a trending analyses?				
Monitoring actual environmental conditions; identification of “hot spots”.				
Do the surveillance procedures include analysis of data relative to the surrounding environment?				
Does the surveillance procedures contain criteria to account for environment?				
Analysis of the effect of equipment failures on equipment qualification and appropriate corrective actions to maintain equipment qualification.				
Does the equipment qualification procedure include a "lessons learned" analysis?				
Protection of qualified equipment from adverse environmental conditions.				

If adverse environmental conditions are noted (e.g., water chemistry out of specification) are supplemental inspections required to determine effect on qualified equipment?				
Do the plant technical specifications contain criteria to alert when environmental conditions are adverse?				
Physical condition and functionality of qualified equipment (confirmed by walk downs).				
Does the regulatory authority review or participate in walk downs for equipment condition?				
Records of all qualification measures taken during the installed life of the equipment.				
Is there a formal process or requirement for maintaining qualification records?				
Are the qualification records reviewed as part of the process for LTO?				
AGEING	Evaluation			
	4	3	2	1
Programme policy, organization and resources.				
Is there a specific organization (Regulatory or other) authorized to develop policy, review technical basis documents and develop resources to address issues related to LTO?				
Documented method and criteria for identifying SSCs covered by the ageing management programme.				
Are there a set of technical basis documents available for review that describe methods and criteria for identifying SSCs?				
Does the regulatory authority review the implementation of methodology for identifying SCCs?				
List of SSCs covered by the ageing management programme and records which provide information to support management of ageing.				
Are there a specific set of technical basis documents that are used as the basis for ageing management of SSCs?				
Are the records of surveillance programs used for aging management available for review/audit?				
Evaluation and documentation of potential ageing degradation that may affect the safety functions of SSCs.				
Does the aging management process review and include operational history in assessing degradation mechanisms?				
Does the ageing management process require a degradation assessment?				
The extent of understanding of dominant ageing mechanisms of SSCs.				
Does the process of determining aging management programs include a review of root cause analysis to understand the cause for degradation?				
Availability of data for assessing ageing degradation including baseline, operating and maintenance history.				

Are plant specific and industry experience data used reviewed against the planned aging management programs to ensure the technical basis for aging management programs is adequate?				
Effectiveness of operational and maintenance programmes in managing ageing of replaceable components.				
Is there a formal regulatory review of operational and maintenance programs as part of the LTO license?				
Are plant design criteria used a basis for acceptance criteria in operational and maintenance programmes?				
Do operation and maintenance procedures contain criteria for retirement for cause?				
The programme for timely detection and mitigation of ageing mechanisms and/or ageing effects.				
Are the selection criteria for SCCs based upon an assessment of degradation mechanisms for the specific SCCs?				
Is the frequency of inspections based upon an assessment of degradation mechanisms for the specific SCCs?				
Acceptance criteria and required safety margins for SSCs.				
Are the acceptance criteria based upon international accepted standards?				
Are the acceptance criteria required to demonstrate that the required safety margins are maintained?				
Awareness of operational condition of SSCs, including actual safety margins, and any life limiting features.				
Do the technical specifications, used during LTO, for operation take into account the actual physical condition of SSCs?				
Do the technical specifications, used during LTO, for operation require a plant shut down if the physical condition of SCCs challenge safety margins?				

Additional DETERMINISTIC SAFETY ANALYSIS Used for LTO	Evaluation			
	4	3	2	1
A compilation of the existing deterministic safety analyses and its assumptions.				
Is there a set of accident analyses performed in the existing safety analysis report (or equivalent document)?				
Is the scope of accident analyses comparable with the scope required for a modern nuclear power plant?				
Is a conservative approach applied for DSA (i.e. single failure criteria, selection of conservative parameters, list of postulated initiating events)?				
Have deterministic safety analysis and assumptions reviewed / approved by regulator?				
Postulated initiating events (for the existing safety analyses and a comparable list for a modern nuclear power plant)				
Are postulated initiating events for the existing deterministic safety analyses comparable with the list for a modern nuclear power plant?				
Are initiating events reviewed / approved by regulator?				
Anticipated operational occurrences.				

Are anticipated operational occurrences included in the scope of the safety analysis ?				
Is the scope of anticipated operational occurrences comparable to the list for a modern nuclear power plants?				
Beyond Design Basis accidents (BDBA)				
Are the BDBA included in the scope of the deterministic safety analyses?				
Is the list of BDBA in existing deterministic safety analyses comparable with the list for a modern nuclear power plants				
Are BDBA reviewed with respect to plant ageing?				
Limits and permitted operational states.				
Are there safety limits and permitted operational states determined for the safety and related equipment comparable with those for a modern nuclear power plant?				
Are safety limits and permitted operational states reviewed / approved by the regulator?				
Analytical methods and computer codes used in the existing deterministic safety analyses and comparable methods for a modern nuclear power plant, including validation.				
Are the analytical methods and computer codes used in the existing deterministic safety analyses comparable with those methods used for modern NPPs, including code validation?				
Radiation dose and release limits for accident conditions (SSCs important to Offsite release).				
Is radiation dose and radioactivity release limit determined for those analyzed accident conditions in the scope of deterministic safety analyses?				
Guidelines for deterministic safety analyses, including for single failure criterion, redundancy, diversity and separation.				
Are there formal guidelines available to conduct deterministic (licensing) safety analysis?				
Are deterministic safety analyses a formal regulatory requirement?				
Are deterministic safety analyses reviewed by regulatory body?				
Does the regulator have an appropriate level of expertise to review PSA?				

PROBABILISTIC SAFETY ANALYSIS **(As Used for Risk Informed ISI or possible change of CLB)**	Evaluation			
	4	3	2	1
Completeness of the PSA Level 1 model				
Does the PSA model involve low power and shut down operational states?				
Does the PSA model involve internal hazards?				
Does the PSA model involve external hazards?				
Does the regulator have appropriate level of expertise to review PSA?				
PSA assumptions				

Are postulated initiating events for the existing PSA comparable with the list for a modern nuclear power plant?				
Risk informed ISI				
Are the risk insights used to assess maintenance activities at NPPs?				
SAFETY PERFORMANCE	Evaluation			
	4	3	2	1
System for identifying and classifying safety related incidents (incident reporting).				
Is there a formal regulatory process for reporting, classifying and resolving safety related incidents?				
Does the process for reviewing safety related incidents include an incident critique that identifies corrective actions and lessons learned?				
Are there legal requirements to report safety related incidents that involve time constraints?				
Is there a formal regulatory review and acceptance of the methods used for recording safety related operational data?				
Trend analyses of safety related operational data.				
Is there a formal requirement for trending safety related operational data?				

Does the regulatory authority review or audit the trends of safety related operational data?				
Feedback of safety related operational data into the operating regime.				
Is there a formal process for including feedback of safety related operational data?				
Analyses of safety performance indicators such as:				
the frequency of unplanned trips while a reactor is critical				
the frequency of selected safety system actuation/demands				
the frequency of safety system failures				
safety system unavailability				
)				
failure cause trends (operator errors, plant problems, administration, control problems)				
the backlog of outstanding maintenance				
the extent of repeat maintenance				
the extent of corrective (breakdown) maintenance				
Is there a formal process for analyzing safety related performance?				
Records of the integrity of physical barriers that prevent release of radioactivity.				
Is there a formal surveillance program that verifies the integrity of physical barriers that prevent release of radioactivity?				

USE OF OPERATIONAL EXPERIENCE FROM OTHER PLANTS AND OF RESEARCH FINDINGS	Evaluation			
	4	3	2	1
Arrangements for feedback of experience relevant to safety from other nuclear power plants and other relevant non-nuclear plants.				
Is there a formal mechanism for evaluating the operating experience of other nuclear power plants as part of the process for LTO?				
Assessments of and actions on the above experience.				
Is there a requirement to demonstrate that the plant specific and industry operational history experience will be adequately managed during the period of LTO?				
Arrangements for the receipt of information on the findings of relevant research programmes.				
Is there a formal mechanism for distributing research results relevant to LTO to power plants?				
Assessments of and actions on the research information.				
Are plants required to participate in industry research activities related to LTO?				

PROCEDURES (that are within the scope of the SALTO program)	Ealuation			
	4	3	2	1
Formal approval and documentation of plant procedures .				
Does the regulatory authority review and approve plant procedures?				
Adequacy of these procedures in comparison with good practice.				
Are power plants required to justify procedures against "good practices"?				
Formal system for modification of a procedure.				
Is there a formal system or process for review and approved of procedure modifications, especially those that identify expectations to national codes?				
Evidence that these procedures are followed.				
Does the regulatory authority audit procedures and audit work to ensure that procedures are followed?				
Arrangements for regular review and maintenance of plant procedures.				
Is there a formal system or process for regular review and maintenance of the plant procedures?				

Kozloduy NPP Plc

АЕЦ КОЗЛОДУЙ ЕАД

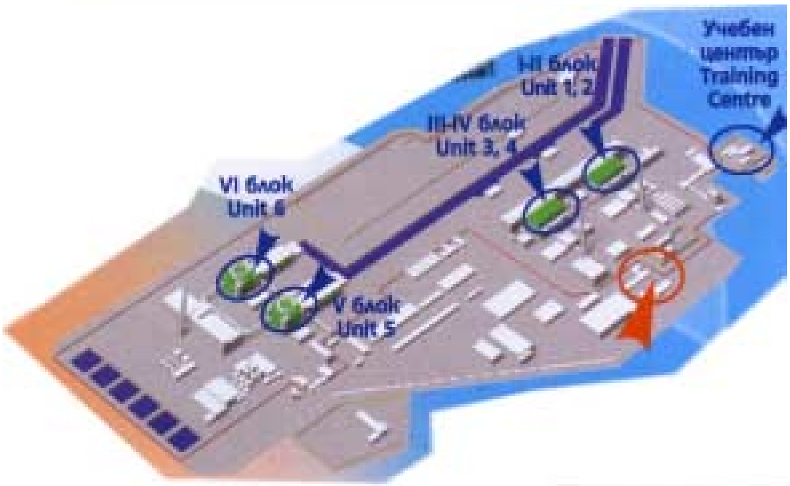
IAEA EBP on Safety Aspects of Long Term Operation of Pressurized Water Reactors

Country Information Report

T. Ribarska
KNPP

2nd WG 2 Meeting, Vienna 16 – 18 November 2004

Kozloduy NPP plc



1

Section 1: APPLICABLE LAWS SPECIFIC TO MATERIALS AND COMPONENTS FOR LONG TERM OPERATION

The term Long Term Operation is not yet defined in Bulgarian laws and regulations.

Laws:

- "Safe Use of the Nuclear Energy Act "(SUNEA);
- "Act on Environmental Protection";
- "Act on Energy and Energy Efficiency".

www.kznpp.org

2

Section 1: APPLICABLE LAWS SPECIFIC TO MATERIALS AND COMPONENTS FOR LONG TERM OPERATION

"Safe Use of the Nuclear Energy Act "(SUNEA)

In 2002 the licensing requirements underwent significant changes based on the new Safe Use of the Nuclear Energy Act (SUNEA) adopted in June 2002.

Under the SUNEA it is obligatory for each power unit and any other nuclear facility to obtain a long-term Operation License. The license is issued for long-term activities with validity up to 10 years on the basis of Safety Analysis Report. The License renewal is preceded by a Periodic Safety Review.

www.kznpp.org

3

In the ***Safe Use of the Nuclear Energy Act*** new requirements to the licensing process are introduced. The Act envisages two types of authorizations to be issued by BNRA: license for operation of nuclear facility and permissions for site selection, design, construction and commissioning of nuclear facility and for any changes of the facility's design connected to modification of the systems, structures and components, important to nuclear safety and radiation protection. Prerequisites for issuance of a license for operation of nuclear facility are given in Article 35.

Bulgarian REGULATIONS :

REGULATION No. 3 on Providing Nuclear Power Plants Safety During Design, Construction and Operation;

REGULATION for the order of issuing licenses and permits for safe using of the nuclear power - newly released / May 2004/ ;

REGULATION on Safety by Spent Nuclear Fuel Management - newly released / August 2004/ .

Russian REGULATIONS :

OPB-88/97 - General Considerations for Ensuring Safety of NPPs; (item 5.1.14. Assumes that an operational utility may require lifetime extention of a NPP unit)

PNAE G-7-008-89 - Guidelines for Construction and Safe Operation of Equipment and Pipelines at NPPs; The last revision of the code was issued in 2000. According to article 2.1.11 of the code SSC's operational lifetime can be extended for a period that exceeds the lifetime defined by the manufacturer of the SSC, on base of the so called "technical solution" developed by NPP management with participation of the designer and manufacturer.

MPK – CXP – 2000 Russian Guide for 'Determination of the Rest Life Time of WWER RPVs in operation'

STANDARDS:

Bulgarian National Standards, Russian Standards and ISO standards are accepted as national standards according to the existing practice.

A great number of Bulgarian National Standards are being revised and modified according to EU Standards.(In most cases Russian Standards are stronger than EU).

Section 1: APPLICABLE LAWS SPECIFIC TO MATERIALS AND COMPONENTS FOR LONG TERM OPERATION

IAEA PRINCIPLES, STANDARDS AND CODES:

IAEA, INSAG – 8 Safety Assessment Principals
IAEA, INSAG – 3 Safety Goals
IAEA, 50-SG-012 Assessment Procedure
IAEA, TECDOC 640 Database
IAEA, Safety Requirements and Guidelines

8

Section 2: Regulatory Requirements Applicable to LTO

2.1. The process used in developing the scope of systems, structures and components (SSCs) that are within the long term operation (LTO) review

Different screening criteria are applied for the identification and selection of critical components, such as:

- Safety relevance
- Availability relevance
- Replaceability
- Costs (price for the component, cost for repair or replacement as well as for prolonged or unscheduled outage)

9

www.kznpp.org

Section 2: Regulatory Requirements Applicable to LTO

- Accessibility (personnel dose, repair during operation possible or during outage)
- Availability of spare parts
- Operational performance.
- Calculations and other methods, as specified in tasks of the Specific Work Program
- Analysis of information on spare parts (availability, prices).

The selection process for the critical equipment is done by engineering judgment based on the above mentioned criteria and information sources.

10

www.kznpp.org

Section 2: Regulatory Requirements Applicable to LTO

2.2. Configuration Control Practices used to Control Design Basis

The great number of Design modifications, resulting from implementation of new technologies and systems and the installation of equipment of different suppliers generate the need of establishing and maintaining a strict consistency between the plant documentation, installed equipment and design requirements.

KNPP Configuration management system consists of several components:

- Document control and records management
- Equipment configuration control
- Design change
- Design requirements

11

Section 2: Regulatory Requirements Applicable to LTO

Document control and records management

- Control of documentation, including review, receipt, introduction, issuance of new revision, distribution, restoration of original copies, keeping and tracking the status of documents;
- Management of records, which defines the methods, responsibilities and requirements for development, archiving, keeping and accountability of the records and archive documents;
- The system is based on the Standard ANSI/NIRMA CM 1.0 – 2000, Configuration Management for Nuclear Installations.

www.kznpp.org

12

Section 2: Regulatory Requirements Applicable to LTO

Equipment configuration control

In order to enhance the process for equipment configuration control, a new database for Operational management (DB OM) is under implementation. The design of the new database includes four modules:

- "Equipment" module
- "Operation" module
- "Design changes" module
- "Design requirements" module

A relation will be established between the registered documents and the registered equipment tag numbers by integrating the "SmartDoc" database and "Operational management" database.

www.kznpp.org

13

Design change control

There is an approved procedure that covers the activities related to control of all design changes. Each design change (called Technical Decision –TD) is assessed by qualified plant personnel with regard to:

- conforming to legislation requirements
- impact on the original design
- impact on plant safety
- impact on operation activities
- impact on maintenance activities
- conditions for implementation
- design change tests

Design requirements management

The original Russian design documentation is recorded in SmartDoc database. Each of the new design documents undergoes assessment and approval by competent plant personnel before the implementation of the project.

The original plant design is used as input for the newly developed design projects. A review is made for conforming the requirements of regulatory documents in the fields of nuclear safety, industrial safety, radiation protection, fire protection, boilers and pressure vessels, quality assurance and the requirements for equipment operation and maintenance.

Section 2: Regulatory Requirements Applicable to LTO

www.kznpp.org

2.3. Aging Management Programs

Evaluation of Rest Life Time(RLT) was executed in 2002 by a Consortium between Framatome ANP GmbH and Atomstroyexport, Russia. It comprises an evaluation of the residual service life of components/systems subject to acceptance by international experts, identifying the need for further investigations/calculations in certain cases, and finding solutions for improvements that achieve a consensus of safety and economy.

16

Section 2: Regulatory Requirements Applicable to LTO

www.kznpp.org

The project consists of :

Evaluation of the residual service life of representative (safety- and availability-relevant) components and systems of the two units using state-of-the-art techniques;

Elaboration of a computerized system structured to handle all relevant component or system data, from plant erection, involving relevant data gathered during plant operation and in-service inspection and replacement activities as well;

Generating an aging management program that permits detection, evaluation and mitigation of the relevant aging degradation mechanisms and identification of plant locations where they are likely to occur.

17

Section 2: Regulatory Requirements Applicable to LTO

The components addressed in the main scope of the project are:

- All components of the primary circuit;
- Components of the secondary circuit (piping, valves and supports);
- Spent fuel storage pool;
- Hoisting equipment;
- Electrical equipment (generator, transformers (main and house), batteries, switchgears, cables and diesel generators);
- Civil structures (reactor building, turbine building, auxiliary building.

18

Section 2: Regulatory Requirements Applicable to LTO

The general evaluation of the plant is accomplished. The results show that there are no general problems that might affect the plant operation during the expected 30 years of operation. Furthermore, for the biggest part of the most-important components, it was found that they could operate significantly longer without major interventions.

Final report includes assessment of the equipment life-time, measures to assure 30, 35 and 40 years for operation and presents RLT management program with the corresponding corrective actions to be implemented into the plant preventive maintenance program.

19

Section 2: Regulatory Requirements Applicable to LTO

Powerful tools were implemented for continuous follow-up process of RLT management:

FAMOS – system for monitoring and calculation of the fatigue of the key components and pipelines in the confinement;

COMSY - condition oriented computerized system for monitoring and prediction of the wall thickness due to erosion-corrosion phenomena of secondary pipelines, based on real working conditions

AMDB - Computerized Database for Ageing Man-agement and Evaluation of the Residual Lifetime

20

Section 2: Regulatory Requirements Applicable to LTO

As a further result of the Rest Life Time evaluation of SSC executed in 2002, activities were categorized in analyses, inspections, monitoring, repairs and replacements had been recommended and were included in the ageing management programs:

- “Program for assuring the life time of units 3&4 of Kozloduy NPP”;
- “Program for surveillance of units 3&4 Reactor Pressure Vessels”
- “Program for future activities for seismic qualification assuring of civil structures, technological equipment and distribution systems of units 3&4 of Kozloduy NPP”.

21

Section 3: Operational Approaches Applicable to LTO

www.kznpp.org

In-service Inspection Practice/Programs

The non-destructive metal control of the Units equipment has been performed in compliance with documents, updated in 2000 and approved by BNRA . The accumulated operation experience, the new international approaches and modern requirements are reflected in these documents:

Procedures for operation control of the basic metal, cladding and welded joints of equipment and piping in KNPP ;

Individual programs for each reactor vessel, based on the data, obtained during the operational control of the metal and its initial state;

Programme for neutron metal embrittlement monitoring of the reactor vessels ;

Programme for Improving the Eddy-Current Control at Kozloduy NPP for the Period 2000 – 2005 г.

22

Section 3: Operational Approaches Applicable to LTO

www.kznpp.org

Qualification of in-service inspection systems

For that purpose a programme was developed and accepted by BNRA, called "Programme for qualification of the non-destructive control of components important for the safety of Kozloduy NPP" having identification No. ЦДИ.П.КБК.212/01, which covers, at present, the following facilities and systems:

- Eddy current control (ECC) of the heat exchanging tubes of the Steam Generator (SG) of WWER-440/1000;
- Ultrasonic control (USC) of end lap welded joints of the Main Circulation Pipeline (MCP) Du 500;
- Reactor internal ultra sound and eddy current inspection of WWER-440;

23

Section 3: Operational Approaches Applicable to LTO

- USC of end lap welded joints of the MCP Du 850;
- ECC of the collectors of SGs W-1000M;
- USC of the welded joints of nozzles Du 200/100 to MCP Du 500;
- USC of the of end lap welded joints of the MCPs Du 400/300 of the pressure compensation systems and Emergency Core Cooling System of Units 5 and 6;
- Reactor internal and external ultrasonic inspection of reactors WWER-1000.

The first three procedures are completed, as the Qualification Authority has already issued the relevant certificates for the methods of control and for the personnel.

The work on the rest of the procedures continues and the completion deadlines are during 2004 - 2005.

24

Section 3: Operational Approaches Applicable to LTO

Maintenance Practices

Long term maintenance plan is developed for 5 years and includes proposals for improvement of both investment and maintenance programmes;

Short term or annual outage plan for plant equipment is developed and completed six mounts before outage date;

Short term plans in terms of work order request to restore detected defects during normal operation of the units.

25

www.kznpp.org

Section 3: Operational Approaches Applicable to LTO

Maintenance activities are carried out during an annual outage. Type of equipment maintenance is defined on the base of the requirements of Bulgarian legislation, taking into account the following factors:

- Design basis;
- Requirements of the equipment manufacturer;
- Technical specification and operating instructions;
- Data from diagnostic and monitoring systems;
- Data received from factory acceptance test and on-site acceptance test;
- Own and international experience;
- ISI data;
- Vibration monitoring data, chemistry control data, ets.

26

www.kznpp.org

Section 3: Operational Approaches Applicable to LTO

- Preventive maintenance improved through: thermovision; expanded vibrocontrol; control on the status of the breakers and the electric processes; control on the status of the insulation of the electric cables and penetrations;
- A computer system for defect monitoring introduced;
- Maintenance instructions developed for a specific type of equipment;
- Records of the performed maintenance activities are filled in forms and stored in accordance with the requirements of Quality assurance programme.

27

www.kznpp.org

Section 3: Operational Approaches Applicable to LTO

Monitoring and surveillance specimen programmes

- Reactor pressure vessel integrity (RLT and PRIMAVERA)

For monitoring and prediction of the RPV material behavior in irradiation conditions a special long term international project was established in 2002 named PRIMAVERA with the participation of FORTUM and VTT - Finland, Kozloduy NPP, Kurchatov Institute – Moscow, ROSENERGOATOM - Russia, JRC Institute in Petten - Netherlands. The data obtained will be used for forehead prediction of the material behavior in irradiation conditions, re-irradiation after annealing as well for neutron flux measurement and validation of calculation models.

- Prevention and monitoring measures in connection with the embrittlement issue of the RPV:
  - » Neutron Flux Reduction on RPV wall
  - » Annealing
  - » Prevention of cold overpressure of the RPV

28

www.kznpp.org

Section 3: Operational Approaches Applicable to LTO

- Primary Circuit Piping integrity

  The full set of systems include three built on different principles, sensitive and diverse between themselves systems:
  - » Acoustic Leak Detection System (ALUS), installed between 1994-1997, modernized in 2003;
  - » Confinement Air Cooler Condensate Monitoring System (SKVB), installed between 1999 and 2003;
  - » Moisture Leak Detection System (FLUS) installed between 2001 and 2003 with sensors located on the pipe lines with Dn100 and Dn200, on the largest pipe lines of primary circuit (Dn500), selected equipment of the primary circuit and in addition in the premises of hermetic compartment, where the leak from primary circuit piping is likely to occur.
- Additional operator support monitoring systems

29

Plant-specific safety analyses

The updated “Safety Analysis Report” (SAR) of units 3&4 has been the basic document for issuing the operation license.

The development of Units 5&6 SAR has been progressing along with the implementation of Modernization Program.It is expected to be completed in the middle of 2006. The new SAR includes an updated list of postulated events.

There are plant-specific PSA developed and updated on a regular basis.

The progress achieved in PSA Level 1 development is a good basis for developing of new probabilistic analyses and PSA applications. Efforts now are focused on PSA level 2 developments that already started for units 3 and 4.

PSA level 1 for units 5&6 was completed in 1995. At present, an update of PSA-Level 1 is underway and will be completed within this year. The study includes:

- Internal initiator analysis;
- Flooding risk analysis;
- Fire risk analysis.

As part of the modernization program for units 5&6 terms of reference for PSA, Level 2 development have been approved as a next step of plant activities in this field.

Section 3: Operational Approaches Applicable to LTO

Additionally to the above-mentioned analisys the following documents are under development:

- Justification of the strength of the reactors internals according to the current norms;
- Operation reliability analysis for the primary circuit piping with diameter DN 200 and DN 500;
- Determination of the parameters inside the containment in case of a primary circuit large diameter pipe break;
- Static and dynamic strength analysis of the primary circuit and implementation of the necessary restrain;
- Determination of the resource of the reactor vessels.

32

www.kznpp.org

Thank you for your attention!

33

IAEA - EBP

SAFETY ASPECTS OF LONG TERM OPERATION OF WATER MODERATED REACTORS

WORKING GROUP 2
Materials and Mechanical Components

CZECH REPUBLIC
COUNTRY INFORMATION REPORT

Robert Křivánek, ČEZ a. s.

TABLE OF CONTENTS

Working Group 2 – Materials and Mechanical Components

1.0 Applicable Laws Specific to Materials and Components for LTO

A. Generally Binding Regulations on the Basis of Which Fundamental Licenses for Nuclear Power Plant Operation Are Issued:

1. Law No. 50/1976 Coll., Building Act [2].
State Administration Body: The relevant building office.
Licence Issued: Occupancy permit decision.
Period of Licence Validity: No validity limitation.
Necessity of Licence Alteration With Operation Extension: None (occupancy permit decision has no time limits).

2. Law No. 18/1997 Coll., Atomic Act [1].
State Administration Body: State Office for Nuclear Safety (SONS)
Licences Issued:
- Operating licence for a nuclear facility in accordance with Article 9(1)d) of the Atomic Act,
- Operating licence for IV category workplace in accordance with Article 9(1)d) of the Atomic Act,
- Permit to reach nuclear reactor criticality again after refuelling in accordance with Article 9(1)e) of the Atomic Act.
Period of Licence Validity: All licences are time-constrained (the permit according to Article 9(1)e) is a once-only permit to reach criticality after refuelling, the licences according to Article 9(1)d) are issued for a determined period – currently for about five years).
Necessity of Licence Alteration with Operation Extension: The permit according to Article 9(1)e) has to be obtained again for commencement of every new campaign. The licences according to Article 9(1)d) have to be obtained for the period during which the nuclear power plant will be operated.
The enumeration of OZPP concerning the area of utilization of nuclear energy and ionizing radiation is given in Annex No. 5 in [1].

It follows from the above mentioned that [1] and its executive regulations are the decisive legal rules for long-term operation of a nuclear power plant. The reason for it is the necessity of continuous renewal of licences for nuclear power plants according to the Atomic Act, as these licences are time-constrained. From the point of view of a long-term operation of a nuclear power plant, the single fact of continuous renewal of operating licences for the nuclear power plant is less important than the fact that no exact concrete safety criteria of acceptability (from the point of view of the Atomic Act) for nuclear power plant operation are defined. As it is, the Atomic Act stipulates from the holder of safety will adopt when assessing the acceptability of a nuclear power plant operation from the safety point of view are formulated.

As regards [2], only the conditions of occupancy permit decision validity have to be observed as these decisions are not time constrained.

2.0 Regulatory Requirements Applicable to LTO

2.1 The Process Used in Developing the Scope of Systems, Structures and Components (SSCs) that Are within the LTO Review

To define the selected set of SSC for LTO is an important initial step in the development of the Programme of LTO Assurance. The scope of equipment monitored within the framework of lifetime management system is further specified under the MPO III project.

It follows from the definition of SSC set for LTO and current legislation that this set will consist of the following equipment subsets:

A – Within the framework of **PP 053 Lifetime Management** the set of monitored SSC of 1 and 2 safety classes is defined and is divided into the categories of:
A.1 – components which could be replaced with difficulties;
A.2 – equipment and systems whose availability also after the end of electricity generation in the NPP has to be ensured (which means that A.2 proceeds from the requirements of the Study of NPP Decommissioning).

B – Set including selected SSC of 1, 2, and 3 safety classes, not included into A as yet, which will be required by SONS within the framework of assessment of equipment readiness for LTO when renewing the licence beyond the limits of its design life.

C – Set including large wholes (SSC), not included into **A)** and **B)**, which can cause considerable maintenance and investment costs in case of implementation of the expected versions of operation duration (i.e. operation of 10, 15, 20, 25, and 30 years beyond the design life limits).

Preparation of SSC set for LTO is planned to be performed during this year (2004). It will be solved by UJV ŘEŽ (Nuclear Research Institute in Řež) under MPOIII project – stage 2.

2.2 Configuration Control Practices used to Control Design Basis

Change Management

The “Nuclear Power Plant Division (UJE) Design Administration” process is a part of the “Configuration Management”. “Complex Assessment of Equipment Configuration Changes – ZKZ” is a sub-process of the “UJE Design Administration” process. This sub-

Within the framework of this sub-process, assessment of technical solution of TZi, TZp, TŘN and evaluation of ZKZ impacts are ensured.

"PP038 Design Basis" work procedure describes two sub-processes, which have the following targets:

- Evaluation of impact and establishing of DB requirements on ZKZ
- Management of the process of DB preparation and updating

The sub-process of "Evaluation of ZKZ Impact on Design and Establishing of Requirements on ZKZ" is an integral part of "ZKZ Complex Assessment" process. A design basis specialist will evaluate the impact of **every** ZKZ on DB and, if need be, formulates the DB safety requirements on ZKZ.

At present, DBs are not sorted in a unified database in UJE. With regard to this situation, ME042 "Evaluation of ZKZ Impact on Design" methodology has been prepared, which provides retrieval of available DB information about modified equipment fulfilling a safety function inclusive of check of functional requirement fulfilment. The safety requirements following from evaluation of ZKZ impact on DB are an integral part of PKP. Control mechanisms are applied during the process of "Modification Preparation and Implementation". The observance of the DB safety requirements is verified during the check of agreement of the design documentation with the technical specification and when evaluating deviations from the technical specification.

ETE (NPP Temelin) and EDU (NPP Dukovany) Design Bases Reconstitution Project

For the sake of better observation of the way DBs are handled, their better monitoring and management, preparation of SDB covering all professions for the current state of ETE and EDU design according to RFMEA methodology in DART environment is in progress.

All-profession Safety Design Bases (SDB) for the current state of ETE design have been prepared under a Contract for Work (SoD) for **ETE** since 2001. SDB project has been created according to RFMEA methodology in DART environment adapted to the conditions of the Temelin Nuclear Power Plant. The project state is as follows:

- 2002 - Contract for Work for ETE was signed
- Subject-matter of performance was broken down into 8 phases (a sub-contract to SoD is concluded for each phase)
- December 2002 – 1st phase completed
- December 2003 – 2nd phase completed
- June 2006 – expected completion of the 8th phase according to the present SoD

ESFAS; XL - bubbler/condenser tower system, suppression pool trays and HDA; EP1-4- RTS, RLS and their supporting systems: ESW, ZN I kat - power supply of 1st category of reliability, ZN II kat – emergency busses, US – high pressure air system)

On termination of SoD within the framework of MPO3 project, the remaining DBDs will be added until completion of SDB on the basis of a new SoD.

The projects are managed in such a way that the product structure will make possible subsequent extension from SDB to EDB (Engineering Design Bases).

2.3 Ageing Management Programs

EDU Ageing Management Programs:

- generator
- turbine
- intermediate rods of drives of safety and control rod assemblies
- main circulation piping
- pressurizer
- steam generator
- main circulation pump
- loop isolating valve
- reactor pressure vessel
- reactor internals
- piping systems of high-pressure and low-pressure cooling system
- piping of active water treatment 1
- piping systems of the secondary circuit (E-C)
- ion-exchanger and mechanical filters of TR, (TM), TD, TE systems

ETE Ageing Management Programs:

- turbine
- main circulation piping and connected high-energy piping
- pressurizer
- steam generator
- main circulation pump - casing
- main circulation pump – inner parts
- reactor pressure vessel
- reactor internals
- piping systems of high-pressure and low-pressure cooling system
- piping systems of the secondary circuit – steam
- TQ 10.. cooling system exchanger

2.3.1 Research/ Process Providing Basis for Applicable Ageing Effects on Structure and Component Intended Function(s)

- VERLIFE
- MPO III – Methodology and tools of lifetime management (ending in 2006)
- Integrity and lifetime of heterogeneous welded joints, acquisition of unconventional material characteristics for calculation of lifetime and fracture toughness of these welds.
- Preparation of stand for simulation of corrosion conditions in confined spaces of SG.
- Taking of micro-specimens from RPV weld cladding, for fluence determination at present

2.3.2 Ageing Mitigation Measures Applied in NPP (Change of Operation Parameters, Component Design Change, Component Material Change, Material Properties Recovery Methods Applied)

Measures mitigating ageing of NPP equipment (sub-sets) are applied by the operator on the basis of evaluation of lifetime consumption, and can be divided into three basic groups:

- Operational and technical measures during equipment operation – relation to the operating procedures. Solutions taken for RPV on the basis of SP+DSP+PTS analyses are characteristic for this area.

- Equipment design modifications

NPP operator considers equipment (sets, components) design modifications to be an effective way of dealing with the unfavourable trend of lifetime consumption. In this area, sealing assembly of SG primary headers and sealing assembly of the main circulation pump have been modified in EDU: metal packing has been replaced by comblike packing with graphite film – this solution can be deemed very successful and used for other sealing assemblies of NPP equipment.

- Selection of more objective methods, or obtaining of more objective information, used for evaluation of lifetime consumption

Research tasks are being solved under the auspices of:

a) SONS ("Checking of Defects of the Type of Non-adherence of RPV Weld Cladding"; "Sampling of RPV Weld Cladding to Ascertain Fast Neutron Fluence")
b) PHARE ("Verlife"; "Cladding")
c) MPO - grant programme ("MPO3 – Ensuring of Long-term Lifetime - Method and

- Criteria from the viewpoint of accumulation of low-cycle fatigue damage in VERLIFE
- When detecting thinning of E-C wall by pipeline measurement process – cutting out and replacement of a piping section – original material
- Suggestions contained in Operational Safety Report 1, 2 (replacement of flanged joint sealing, replacement of fitting at pressurizer injection, suggestions to specify NDT).

3.0 Operational Approaches Applicable to LTO

3.1 Normal Operational Practice/Programs Applicable to Ageing Management

3.1.1 In-service Inspection Practices for Passive Components

Approaches, Principles of Operational NDT Inspections Used

The inspections of the operated nuclear facility are ensured by its operator in accordance with the approved In-service Inspection Programme.

The Programme serves for ensuring and implementation of periodic and operative in-service inspections of machine equipment, electrical and control system equipment and building equipment included into the list of Specific and Restricted Technical Equipment. Inspection activities are carried out mostly during unit outages for refuelling, but also out of them.

In-service inspections mean: inspections of materials and welded joints (NDT inspections) and non-material inspections set by quality assurance documentation, equipment checks, diagnostic inspections and measurements etc.

The nuclear power plant operator is responsible for preparation, implementation and evaluation of the in-service inspections. The In-service Inspection Programme is binding for all organizations taking part in preparation and implementation of the in-service inspections. The programmes of inspections of the individual Units during outages are made more precise in annual plans. The In-service Inspection Programme is prepared within the intention of [3], [4], and [5]. It is approved by SONS, being a licence document for the power plants.

The individual inspections can be carried out gradually during a standard or augmented overhaul as well as out of the overhauls thus that the programme as a whole is fulfilled in the set intervals. Based on this, the plans of inspection activities performing for the individual outages are prepared.

The following regulations are used as a basis for determining the inspection periodicity:

1. Regulation for Construction and Safe Operation of Equipment of Nuclear Power Plants, Experimental and Research Nuclear Reactors and Sets (Moscow 1973)
2. Principles and Conditions of Use of Soviet Standards and CSN and Regulations for

- the results of evaluation using risk-oriented approaches for selection of components for in-service inspections.

With regard to the requirements on establishing, with high degree of reliability, the data concerning the structural imperfections detected during non-destructive inspections that are the input data for establishing of the residual lifetime, the mechanized inspections using the ultrasonic and eddy current methods for specified components important for safety are qualified using the methodologies of ENIQ and IAEA.

NDT method qualification within the meaning of ENIQ is applied on the following components:

- Reactor pressure vessel bodies (welded joints, basic materials, areas under weld cladding)
- Steam generators (threaded connections, heat-exchanging tubes, header material)
- Pipelines with heterogeneous welded joints
- Steam and feedwater piping on the elevation of 28 m – ETE site

Personnel Qualification

The qualification of workers performing non-destructive crack detection corresponds to Std 101/E/95 qualification and certification system, which is fully consistent with EN 473, or with Std 301/E/95 and Std 201/E/95 national standards. The requirements on the qualification of workers performing other inspection activities are determined by the qualification catalogue of the NPP operator.

Testing, Inspection Methods

The prescribed NDT inspections are carried out according to the written procedures developed for each testing method and written instructions developed in the step-by-step form for every monitoring point.

NDT methods used are VT (visual testing), MT (magnetic testing), PT (penetration testing), UT (ultrasonic testing), RT (radiographic testing), ET (eddy-current testing), LT (leak testing), performed either manually or automatically.

Standard Testing Equipment

The apparatuses and measuring instruments for in-service inspections must comply with the requirements given by the regulations in force [6] and [7].

Non-standard Testing Equipment

Apparatuses and equipment enabling mechanized or automated inspection in connection with remote manipulators or positioning mechanical systems can be used for performing of inspection activities.

determining them, we proceeded from the quality assurance individual programmes, general evaluation regulations, and producers' recommendations.

Should any inadmissible defects be found, further solution is determined by the "Technical Solution of Defect" developed in accordance with the Quality Assurance Programme for Operation.

Schedule of Future Work

The requirement of long-term operation of NPP components makes higher demands on in-service inspections in the following areas:

1) Knowledge and understanding of degradation mechanisms inclusive of development of models simulating incubation time, initiation and propagation of imperfection. At the same time, specific attention is paid to degradation mechanisms like erosion/corrosion and stress corrosion monitored within the framework of augmented in-service inspections (see also 3.1.1.1 to 3.1.1.3);
2) Improvement of detection and more accurate establishing of dimensions of imperfection indications, especially for highly stressed austenitic and heterogeneous welded joints which can be inspected with difficulties;
3) Reaching of qualitatively higher level of monitoring point selection for passive components by means of the methods of risk-oriented inspections (see NURBIM project [9]) with the possibility of proposing a change in inspection period, sensitivity analyses for establishing of trends for the individual parameters, and evaluation of the economic effect of the change in inspection schedule with the risk level maintained or even lowered;
4) Use of the methods of mathematical modelling of especially UT/ET signal (see SPIQNAR project [8]) for development of new or improved inspection procedures, e.g. for checking of inspection areas with difficult geometry, lowered accessibility of the monitoring point, or material through which the signal goes with difficulties (e.g. heterogeneous or austenitic welded joints);
5) Application of frequency analysis methods, signal processing and mathematical modelling of especially UT signal by means of modern program resources (see e.g. SPIQNAR project [8]) to reach improvement of signal-to-noise ratio in heterogeneous or austenitic welded joints which can be inspected with difficulties;
6) Use of the methods of mathematical modelling of especially UT/ET signal for the sake of technical substantiation within the framework of NDT qualifications.

Legislation Applied, Standards and References Used: [3], [4], [10], [11].

lifetime, and general strategy of proceeding in relation to the operational conditions are determined on the basis of this prediction.
From the point of view of the safety of nuclear power plant operation, the programme goal is to ensure with acceptable probability that the integrity of the pressure piping of the secondary circuit will not be lost during the power plant operation. To fulfil this goal, the following basic tasks have to be implemented:

- Identification of the systems prone to erosion/corrosion
- Determination (prediction) of erosion/corrosion rate for the given component
- Identification of piping components for inspections and use of suitable inspection methods
- Analysis of the data measured inclusive of their archiving
- Ensuring of basic pre-operational data; should they be missing, it is necessary to proceed in an alternate way consisting of establishing of the real state and accessible operational information as an input for the first – replacement evaluation
- To perform, on the basis of the evaluation criteria, evaluation of the piping integrity and its residual lifetime until next measurement inclusive of recommendations for repair or replacement of the degraded piping section

Currently, CHECWORKS verified computer program identifying sensitive components on the basis of a model into which operational and material quantities enter is used for prediction of erosion/corrosion damage in the Dukovany and Temelin Nuclear Power Plants. Then the model is being made more exact continuously by data from inspection measurements of component wall thickness or measurement of the chemical composition of the material.

The selection of inspection points is performed on the basis of:

- Modelling by CHECWORKS program and its predictive analysis of erosion/corrosion damage
- Operator's experience of the system, especially with thinning already indicated, leakages of medium, or component fractures
- Engineering opinion
- Other operators' experience, especially from analogous nuclear power plants

Moreover, attention is also paid, when selecting the components, to the representation of components from all secondary circuit systems which have been found to be sensitive to erosion/corrosion, if possible, and to the representation of all kinds (geometries) of components that occur in the evaluated systems.

The Chemistry Department of **ETE** has prepared a system monitoring the corrosion state of power plant equipment, evaluation of corrosion development, and management of anticorrosive measures. Within the framework of this general programme, specific sub-programmes for monitoring of the following systems or components have been prepared:

- Managed monitoring of the state of corrosion of steam generators, TQ (spray system) exchangers under quality assurance individual programme, and monitoring of other components if requested by their administrators.
- Chemical analyses monitoring corrosion products in the system of the secondary circuit, cooling circuits etc. (Fe) and DG cooling system (Cu) are carried out and evaluated.

Different methods of corrosion state evaluation are used for the above-mentioned activities and effort is made to evaluate the state comprehensively, taking into account history and chemical regime of the operational medium. Activities like monitoring of Ryznar index, media chemical composition; monitoring of corrosion product behaviour; use of corrosion coupons; measurement by means of Corrater probes; acquiring of photographic documentation of the equipment condition connected with its archiving enabling evaluation of development in time; evaluation of biological regeneration etc. are carried out. We are trying to minimize corrosion losses by optimization of the chemical regime, its management, and use of conditioning agents.

This programme has its "reporting" system for departments concerned and persons responsible.

Regular monitoring in **EDU** E-C has been performed from 1995 (Checmate – Checworks) on the following pipelines, inclusive of the measurement of the chemical composition of the given component:

- Steam piping from SG
- Condensate from HPH1 TGx1 into FWT
- Condensate from HPH1 TGx2 into FWT
- Suction of residual heat removal pumps
- Delivery of residual heat removal pumps
- FWT draining into condenser
- Suction of feedwater pumps
- Feedwater pump delivery to HPH
- Feedwater from HPH to the main feedwater header (HNK)
- Feedwater from HNK into SG
- Condensate from LPH 5 into feedwater tank
- Condensate from condensate pumps into LPH 1
- Condensate from LPH 1 into LPH 2
- 6th TG bleeding point
- 7th TG bleeding point

- 8th TG bleeding point
- Heating steam into separator

3.1.1.2 Augmented Inspection of Steam Generator Tubing

Inspections of the SG heat-exchanging tubes are set by the method as well as by the scope in the Guidelines for SG Operation (eddy-current inspection, leakage test – described below).

The NPP operator has applied augmented inspections of steam generator tubing in cases when the outputs of the inspections performed in a standard way have not brought about the expected outputs which could be used for SG operation, especially the relation between the inter-circuit leakage detected in SG during operation and identification of a leaking heat-exchanging tube during Unit outage.

An **inter-circuit leakage** in an operated SG is evaluated on the basis of N16 measurement in steam and NA24 in surface blowdowns. If an inter-circuit leakage is found during operation, then during Unit outage the leaking tube is searched by means of a leakage test with nitrogen overpressure from the secondary side and leakage into the primary header filled with boric acid solution. This "bubble method" has usually good results, but there are certain exceptions being solved currently by the operator by augmented inspections – helium leakage test of the heat-exchanging tubes with helium overpressure at the SG secondary side and helium detection at the outlets of the heat-exchanging tubes from the side of the primary header inner surfaces. Helium test is preceded by a special preparation of SG consisting of drying of the SG secondary side to the pressure values in mbar units.

The standard leakage test has an expected sensitivity of 1.10^{-4} mbar.l.s^{-1}; the augmented helium leakage test supposes the sensitivity of 1.10^{-6} mbar.l.s^{-1} and, on the basis of the current results, this augmented inspection of the SG tubing can be taken as offering good prospects.

Further we evaluate the rate of deposit formation on the heat-exchanging surfaces by means of sampling the SG tube surfaces. We also take stock of impurities around SG and thus gain an idea of the potential of sludge and deposit formation, which then can be inductors of corrosion processes. Measurement and evaluation of "hide-out" and "hide-out return" effects is performed with calculation of pH in crevices. SG protection and maintaining of its lifetime is the target of the management of the chemical regime in the secondary circuit.

Method for establishing of real chemistry in the SG crevices has to be developed. Its understanding and subsequent possibility of its management are a way to the elimination of the problem of corrosion in crevices and thus also to cracking of headers and tubes under clamps.

The eddy-current inspection with inner cylindrical probe with circumferential coil winding (Bobbin coil - BC) is supposed to be the standard method of the inspection of the integrity of SG heat-exchanging tubes. Phase-amplitude analysis of multi-frequency

The inspection of the SG header bridges is performed in two steps:

- Inspection by a BC probe with higher sensitivity
- In case of BC probe indication a follow-up inspection by means of a rotary probe is performed.

3.1.1.3 Augmented Inspection for Specific Degradation Mechanisms Such as Intergranular Stress Corrosion Cracking

Augmented inspections must be understood as inspections that were not specified in the original accompanying technical documentation of the equipment but followed from scientific and technical progress or from operational experience and are introduced with the aim of ensuring sufficient evidence of the technical condition of equipment inclusive of inputs for the evaluation of lifetime consumption. The following solution can be mentioned as an example of the NPP operator's approach:

In connection with the damaging found in the area of the socket part of the SG primary header and caused by intergranular stress corrosion cracking, the NPP operator has adopted, among other, measures concerning inspections and, in addition to the originally required scope, has introduced eddy-current inspections of the area of threaded connections, "cups" under the threaded connections, inner surface of the header welds, and ultrasonic inspection of the welds and basic material of the header – with the aim of obtaining sufficient information about possible damaging of the primary headers by intergranular stress corrosion cracking during in-service inspection performance. These NDT inspections are complemented by chemical inspections in the area of "cups" under the threaded connections and technological measures for bolt tightening with the aim of preventing the conditions of intergranular stress corrosion cracking initiation. This approach – introduction of augmented inspections – enables timely detection of possible damaging of the SG primary header by intergranular stress corrosion cracking and the supplementing measures limit the conditions of intergranular stress corrosion cracking initiation.

3.1.2 Maintenance Codes or Practices for Active Components

SM 026 Regulation – DLHM (long-term tangible property) Maintenance

Purpose and target – The purpose is the management of the DLHM maintenance process with the target of economical ensuring of the required equipment functions and ensuring of activities associated with exercising of rights and discharge of duties concerning the entrusted property in accordance with the operational and design documentation, with the national and international legislation, and with the recommendations of the international nuclear power engineering bodies.

SM 027 Regulation – Technological Maintenance Preparation

Purpose and target - The purpose of the technological preparation is:

- To ensure the necessary documentation of repairs for safe and economical repair execution (PP 055 Preparation Document)
- To ensure the necessary technical, technological, and material support of repairs (timely issuance of the necessary requirements for contract that are technically clearly and correctly specified; order forms for ensuring of material, spare parts or aids; checking of the existing stock of spare parts and materials, checking of the existing aids inclusive of their usability and their reservation)
- To ensure feedback for optimization of costs of maintenance and further strategic goals of the company (projection of knowledge acquired from own implementation and international "good practice")

The target is to ensure proper, safe and economical property maintenance.

PP 047 Work Procedure – Preparation and Management of Maintenance Programme

Purpose and target – This document sets procedure and responsibilities during the preparation and management of maintenance programme in ÚJE (Nuclear Power Plant Division). The target of the document is to set uniform rules of preparation of maintenance programmes for the maintenance of DLHM of UJE.

ME 148 Methodology – Equipment Maintenance Categories and Methods

Purpose and target – This methodology sets the sequence of activities when assigning categories and methods of equipment maintenance. The target of the methodology is to reach

- uniform and effective approach to equipment maintenance,
- minimum maintenance cost while maintaining high operational safety,
- minimum equipment failure rate.

PP 048 Work Procedure – Execution of Maintenance and Repairs

Purpose and target – This procedure sets the main principles and procedures which will lead, during ensuring of maintenance work, to a high maintenance quality, technical and describes the way of keeping and filing of maintenance documents into the memory document files. Lists of documents are the documentary outputs. Their form and content are defined by other regulations, especially by PP Preparation Document.

ME 070 Methodology – Methodology of PPO Preparation in EDU

Purpose and target – The methodology sets uniform approach to preparation of work procedures in EDU.

Repair Work Procedures (PPO) are prepared for maintenance of any primary circuit equipment, where the work procedure of the maintenance as such is described in detail. These procedures are approved by ITI for equipment subject to [4].

At the same time, PPO are being entered into the Passport database so that in case of issuing of any work order (PP) appropriate Repair Work Procedure could be generated from the database for this order. We expect the linking of PP and PPO database to be completed at the end of 2006.

Until completion of the Passport database, the current, written form of PPO will be used; several hundreds of them have been prepared until now. Overwhelming majority of the procedures has been developed by the appropriate preparation workers. Nevertheless, for some components PP are used that have been developed by the repair supplier (Energoservis company for the reactor), and the company considers the information provided in PPO to be its know-how.

PPO proceeds from manufacturer's recommendations, or from technical conditions of the given equipment. As during ca 20 years of operation pronounced modification steps were taken, PPO must reflect the current state of equipment configuration (see the use of comblike sealing etc.). PPO updating is always ensured by the author of PPO.

3.1.3 Equipment Qualification Practices

Programmes of qualification of equipment important from the viewpoint of safety have been introduced in all the nuclear power plants operated in the Czech Republic. An important aspect of these programmes in relation to LTO is the fact that the required qualified lifetime of the equipment subject to qualification is the same as the design lifetime of the NPP (30, 40 years). For ensuring of LTO, it will be therefore necessary to reassess the state of qualification for equipment with qualified lifetime shorter that that required for LTO.

METHODS, APPROACHES, PRINCIPLES USED

General principles used in the qualification programmes are based on current

Design Inputs
Qualification, as a managed multidisciplinary process, has begun by preparation of the QA programme and defining of the design inputs like:

- ✓ Ambient medium parameters for normal operation and for postulated initiation events of LOCA or HELB type
- ✓ List of equipment subject to qualification
- ✓ Location in NPP, activity required for fulfilment of the system safety function, and time for which the function will be required under conditions arising during design basis accidents and after their ending for each equipment from the list
- ✓ NTD, methods and procedures used for qualification of the individual equipment groups

EDU
The process of EDU equipment qualification began in 1995. ÚJV ŘEŽ, VÚJE Trnava, and EGPI Uherský Brod together with EDU specialists took part in preparation of the list of equipment to be qualified. The main criterion of machine equipment selection for qualification is the fulfilment of safety functions. For the determined set of safety functions, a minimum scope of systems and equipment ensuring these functions was selected inclusive of supporting systems which are absolutely necessary for the functioning of the main system like lubrication, cooling etc.

Ambient medium parameters were established for all rooms and compartments in all spaces in which the qualified equipment is located. The values of the thermo hydraulic and radiation parameters of ambient medium after design basis accidents of LOCA and HELB type were calculated, operational parameters of the individual rooms were developed by ÚJV ŘEŽ on the basis of monitoring of these spaces. The seismic response spectrum was established in accordance with IAEA recommendation by Stevenson & Associates, 3E Praha Engineering, and Energoprůzkum.

ETE
The process of ETE equipment qualification began in 1996 on the basis of recommendation of the mission of HNUS US company in 1991. NTD, methods and procedures, ambient medium parameters, original scope of equipment for qualification to the ambient medium and the required safety functions were determined in the 1st phase of the equipment qualification programme by the main suppliers, Energoprojekt, in 1996. These design inputs were updated in 1998, 2001 and the last updated state dates from 03/2003. The scope of equipment for seismic qualification and the required sub-category of equipment seismic resistance were determined by the main suppliers, Energoprojekt Praha, in supplement No. 406 to the initial design in 1995. This scope of equipment for

- Assessment of the state of qualification of the individual equipment components from the list on the basis of an analysis of the available accompanying, design or qualification documentation.
- Preparation and execution of remedial measures (replacement, modification, or decision on additional qualification) for non-qualified equipment.
- Preparation and execution of qualification tests, analyses and other activities for checking of the qualification state of equipment designed for additional qualification.

EDU
Methodologies of qualification eligibility assessment were developed and all equipment components from the qualification list were evaluated in the 2nd phase of EDU qualification process. The results of this process are reports of the course of inspections and qualification tests inclusive of qualification certificates issued that either confirm the functionality and resistance of the equipment or describe its shortcomings and the remedial measures suggested.

ETE
The equipment qualification programme was developed by the supplying company of Stevenson and Associates in 2002. The qualification of NPP Temelin equipment relates to the safety systems and their components. Seismic qualification relates to the NPP Temelin equipment in general; it belongs into so called 1st category of seismic resistance. The equipment in the Equipment Qualification Programme is specified into qualification groups according to the equipment type and qualification requirements on equipment. A database of equipment to be qualified is prepared within the framework of the Qualification Programme. ETE equipment is qualified for long-term operation. A small group of conditionally qualified equipment (technical and formal reservations) has been specified in the Qualification Programme.

Description of Current State of Problem Solution
Currently, the qualification programmes are in the phase of maintaining of the qualification state for the qualified equipment and remedial measures for the non-qualified equipment.

EDU
At the beginning of 2002, the programme of solution of the remedial measures following from the qualification started. The following tasks concerning the machine equipment area have been established and are solved by remedial measures:

- Ensuring of qualification of the electric drives of the individual systems

To carry out "final qualification" of the conditionally qualified equipment in 2004-2006. In 2004 and 2005, equipment qualified with a reservation of technical character will be qualified; between 2004 and 2006, qualification of equipment qualified with a reservation concerning "thermal ageing" only will be completed.

Schedule of Future Work

Completion of remedial measures, maintaining of equipment qualification in accordance with the methodology until the end of the NPP lifetime.

EXTERNAL LEGISLATION APPLIED:

[1], [3], [25], [26], [27], [28], [29], [30], [31]

Internal Legislation Applied

- SM 021 Regulation "Design Administration"
- 041 Procedure "Equipment Qualification"
- ME 060 Methodology "Specification of Qualification Requirements for New Equipment Orders"
- ME 061 Methodology "Maintaining of Equipment Qualification"
- ME 083 Methodology "Selection of Equipment for Qualification"
- ME 084 Methodology "Qualification Documentation Preparation"

3.1.4 Component Function Tests

The system of programme of surveillance over safe operation covers implementation of functional tests, inspection activities, diagnostics, in-service inspections (ISI), engineering inspection (OTK), and special tests and experiments and other inspection activities applied on all systems and equipment important from the viewpoint of nuclear safety. When setting the time intervals of the implementation of the required inspection activities as well as when defining the criteria of acceptability, the data of Operation Safety Report (PpBZ), Operational Limits and Conditions (LaP), and technical conditions of suppliers and manufacturers of the defined equipment groups are respected.

The defined equipment groups include:

- restricted technical equipment,
- restricted technical equipment in nuclear power engineering,
- specific equipment,

while their boundaries comply with the boundaries of systems and equipment which are modelled and analysed in PSA studies.

5) Equipment in-service inspections (non-destructive testing, metrology, special techniques, QA individual programmes)
6) **Special tests and experiments**

We will further deal with items 1) and 6) only.

For 1) Functional Tests of Systems and Equipment

Functional tests of the defined systems and equipment proceed from the requirements of PpBZ, LaP, and from the equipment suppliers' and manufacturers' requirements specified in the operating procedures and accompanying technical documentation. The Operation Section is responsible for preparation and implementation of the functional tests.

Functional tests include:
1A) Tests of controls and protections
1B) Operability tests
1C) Prevention (operation monitoring)

1A) Tests of Controls and Protections

Tests of setpoint values and time delay and tests of operability of the instrument channels are included in the scope of the tests of controls and protections.

Setpoint verification tests are required in the safety systems or sub-systems to verify that the values of setpoint and delay are within limits prescribed by LaP. All the tests, either for equipment according to LaP or other, are described in the appropriate methodologies and operating procedures and also an accurate schedule of the execution of the relevant tests is determined.

1B) Operability Tests

Operability tests ensure that the tested system or equipment are able to perform the function set by the design within the meaning of inspection requirements defined in LaP and within the meaning of criteria set in PpBZ.

These tests contain the following activities:

- Check of initial conditions at test beginning
- Start of active equipment and monitoring of the main parameters within the framework of acceptability criteria (the test duration must be sufficiently long to reach stabilization of parameters and to verify the functionality of the system, equipment and its individual components)
- Check of the control and functionality of motor-operated and air-operated fittings; check of their full or partial lift within the meaning of inspection requirements
- Check of proper operation of automatic
- Monitoring of selected parameters during the test.

the selected parameter values into printed forms of daily operation records. These inspections and their periodicity are set by the appropriate regulations (the regulations are broken down according to expertise).

For 6) Special Tests and Experiments

They are not included in the equipment test programme as a standard. In concrete cases of their execution, their unavoidability and justification must be assessed. A special procedure must be developed for every test or experiment (operative programme – OP in EDU).

The Heads of the EDU special departments in co-operation with the specialists of the Production or Maintenance Section are responsible for preparation of a special procedure. The execution date must be adapted to the functional test schedule as much as possible. This procedure must be reviewed independently by a qualified person different from the author of the programme draft to ensure that neither the operational limits and conditions nor design requirements are violated and that they will not reach dangerous condition.

All the tests performed are documented by appropriate reports and entries in the operation personnel logs and are archived according to the rules of the relevant managing documentation.

3.1.5 Applied Diagnostic Systems

The following on-line diagnostic systems are used within ÚJE:

Measurement of Vibrations of Selected Rotating and Electrical Machines

Diagnostics is continuously performed for bearing and rotor vibration of the turbogenerator (VIBROCAM Schenck TG system in ETE), main circulation pumps (RECOP-MCP system in ETE, ADASH 3100 system in EDU), feeding turbo-pumps (VIBROCAM Schenck TBN system in ETE), cooling water pumps (EGV-DSBQUW system in ETE), and dieselgenerators (EGV-DSDSA system in ETE, also torsional vibration is evaluated).

Measurement of Temperatures, Stress, and Displacement of Selected Components

Temperatures, stress, and displacement of selected components are measured continuously for the sake of monitoring of low-cycle fatigue consumption (MAFES-TSF, MAFES-DMS systems for primary circuit equipment, 21, 22, and 23 tests for steam piping in ETE; system of thermal technical measurements for determining of lifetime of the primary circuit in EDU).

Monitoring of Ultrasonic Signal in Primary Circuit

Continuous monitoring of the RMS value of ultrasonic signal for the sake of monitoring of leakages of the high-pressure and high-temperature piping of the primary circuit (LEMOP system in ETE), monitoring of acoustic emission for the sake of

Monitoring of vibrations, pressure pulsations, and coolant temperatures at the core outlet by means of noise analysis for diagnostics of RPV (reactor pressure vessel), reactor internals, and steam generators (RVMS system in ETE, SÜS system in EDU).

Monitoring of Pressure Pulsations in Separators-Reheaters (SPP)

Monitoring of origin and registration of duration of pressure pulsations in selected tubes on both sides of the separators-reheaters (SPP) (test No. 6 in ETE).

Monitoring of Deaeration Condition of Feedwater Tank Condensate

The target of the deaeration diagnostics is to determine the degree of wear of the main deaerator structures, that is the spray jets of the main condensate and bubbler steam distribution (test No. 20 in ETE).

Monitoring of Thermal Balances and Diagnostics of Secondary Circuit Components

Monitoring of thermal balances of the selected secondary circuit components (turbine, main turbine condensers, feeding turbo-pumps condensers, steam separators-reheaters, low pressure heaters, high pressure heaters, and heating water heaters) for the sake of diagnostics of their long-term degradation (test No. 7 in ETE).

Loose Part Monitoring System

Monitoring of occurrence of loose parts in the primary circuit (DMIMS system in ETE and EDU).

3.1.6 Surveillance Specimen Programmes (Irradiation Damage, Corrosion Loops)

Reactor Pressure Vessel (Programme until 2022)

The pressure vessel material becomes embrittled in the core area due to irradiation. This is why it is necessary to monitor changes of RPV material during operation. Every reactor pressure vessel (RPV) of VVER-440/V-213Č type has surveillance specimen programme for pressure vessel materials ensured by the design already. This programme was designed by the main design organization, i.e. by OKB Gidropress, Podolsk, at the turn of the sixties and seventies already on the basis of knowledge and possibilities of those times in the then Soviet Union and with regard to the reactor design, i.e. configuration of the pressure vessel and its internals.

The original, so called standard surveillance specimen programme (SSP), implemented in EDU in accordance with the accompanying technical documentation, does not quite meet the current requirements on content, purpose, and required results, especially from the viewpoint of their application on assessment of the residual lifetime of RPV, especially in the following aspects:

- High acceleration coefficient between flux falling on the surveillance specimens and inner vessel wall, ca 10
- Inaccuracies in establishing of the real flux falling on the individual surveillance

Therefore a new surveillance programme has been designed (ÚJV Řež and Škoda JS), whose conception takes into account the above-mentioned aspects; see report of ÚJV Řež Reg. No. 10 986 T (revision 1) "Execution Design of Supplementary Surveillance Specimen Programme for Reactor Pressure Vessel of NPP Dukovany".

Supplementary Surveillance Specimen Programme
Purpose of Supplementary Surveillance Programme (DSP)

The purpose of DSP is to ensure the possibility of assessment of the residual lifetime of the reactor pressure vessels during the whole time of their operation in accordance with the "Instructions and Recommendations for Assessment of the Lifetime of the Pressure Vessel and Internals of VVER NPP reactors during NPP Operation" issued by SONS in 12/1998, and with the possibility to ensure the necessary documents for potential plant life extension.

DSP Target

The outputs of tests and measurements and evaluation of their results must ensure the following data necessary for assessment of the residual lifetime of RPV:

- Continuous time dependence of changes in yield point, breaking strength, shift of critical brittleness temperature from the notch toughness tests and transition temperature from the fracture toughness tests, with knowledge of the real irradiation temperature
- Continuous time dependence of neutron fluence falling on surveillance specimens (summarily always after several campaigns).

Pressure Vessel of VVER 1000 Reactor

Standard surveillance programme (SP) has been started according to the design (Ae 10106/R1 and Ae 10036/R2) for the basic material, welds, and weld cladding. By reconstruction, surveillance specimens have been placed into boxes near to the inner reactor wall, so the acceleration coefficient of around 2 is expected, which complies with international requirements inclusive of establishing of the irradiation temperature. SP complies with NTD A.S.I. standard, section IV (VERLIFE).

After every campaign – in accordance with VERLIFE – neutron fluence behind the reactor pressure vessel is measured.

The materials of reactor internals have been subjected to accelerated irradiation tests within the framework of technical specification of Czech Power Enterprises. The results of the basic mechanical properties after fluence corresponding to 45 years of operation have been assessed as satisfactory.

important for safety whose operational safety and LTO depend, among other, also on the concrete values of these characteristics.

Changes or degradation of material properties as a result of long-term influence of the process medium, changes of operating parameters, irradiation by the neutron flux etc. can happen and these changes can cause limitation or lowering of the safety of the critical components and thus limitation of LTO.

The fundamental approach to establishing of data concerning the influence of operation over the exposed material is to assess its properties continuously by regular inspections or measurements in working conditions by such NDE methods that can evaluate the influence of the operation on mechanical properties change concretely. A set of information about the initial condition of the examined material (at the operation beginning) which should be available in the original (supplier's) documentation of the component important for safety at the equipment operator belongs among factors important for such evaluations. It means documents of material identification, containing also data about heats and charges used inclusive of thermal treatment, and values of Rp0,2, Rm, chemical composition and other properties. These data serve as a basic comparative criterion for assessment of changes of properties which can be ascertained by NDE methods.

Also the possibility to check the given material in the conditions of artificial ageing which usually enables execution of check of changes of the mechanical properties by classic destructive testing according to relevant standards on test specimens (tensile, bending, impact tests, hardness etc.) and comparison of the results with NDE method, possibly more accurate validation of the method of result evaluation of this NDE method, is important for realistic assessment of operation influence and for heightening of accuracy of operating changes assessment by NDE methods.

"ABIT" Method

NDE method used most often for assessment of the basic mechanical properties of the materials of NPP components in the Czech Republic is the so called "**ABIT**" (*Automated Ball Indentation Testing)* method based on evaluation of so called "indentation diagram" (stress-deformation curve) acquired by a special device when pressing a ball indenter into the component material under clearly defined conditions. The measurement equipment itself, method, as well as evaluation methodology are still being improved at present. "ABIT" method is mostly used in USA and Russia, but also in other countries for different commercial and research programmes.

The method enables to determine Rp0,2 and Rm of the material and has its advantages as well as disadvantages. The possibility of checking of the above-mentioned values in a non-destructive way under operational conditions belongs among its undisputable advantages. After using this method, several indents with diameter of about 1.2 mm and components with rather simple surface which enable to design and manufacture an aid or manipulator for fixed anchoring of the measurement system.

It is not suitable for welds in spots with complicated geometry (elbows, branches, reducers); however, it can be used for small material cut-outs (down to 10x10x5 mm) where a tensile test cannot be applied.

The current state of adoption of this method for NPP in the Czech Republic (the method is applied by ÚJV Řež, a.s.) is as follows:

Four measurement systems using **ABIT** method have been developed and are used:

1. Equipment for remote semi-automatic measurement of the properties of weld cladding metal of RPV VVER 440 under water and under working conditions during reactor outage
2. Equipment for remote semi-automatic measurement of the properties of metal of RPV VVER 440 from outside (in the gap between the RPV wall and biological shielding)
3. Equipment for manual measurement at DN 250 – DN 1000 piping systems
4. Equipment for remote semi-automatic measurement of the properties of metal of core supporting cylinder (pit) of the inner structure of VVER 440 reactor.

A manually operated ABIT system has been and can also be used in a number of other applications for laboratory specimens of materials aged or non-aged in hot cells etc.

The current state of development of this method in the Czech Republic is described in greatest detail in [32] document.

The non-destructive testing of material properties is usually not executed by the personnel and laboratories of NPP operators but is ordered from outside expert organizations and is performed as a supplementary method to requirements on destructive and laboratory testing.

Applied Inspection Methods in NPP:

- Measurement of material hardness (welded joints, basic materials, thermally influenced areas)
- Structural analyses, replicas
- Chemical spectral analyses by mobile devices
- Measurement of chemical composition of carbon steels used

Inspected NPP Components:

- transition piping of the high-pressure stage of the steam turbine
- spent fuel pool
- welded joints of the main circulation piping

inspection hole of the shielding container fitted above the inspection pit was used for measurement of indentation diagrams.

The Diagnostics Department monitors the **reactor internals vibrations** by regular measurement and evaluation of signals of SÜS system and selected outer ionization chambers performed during Unit operation at 100% power in one month's interval. At the same time, their changes and deviations from characteristics recorded for a long time are checked.

The monitored characteristics of the signals pertaining to displays of vibrations of Unit 1 reactor internals have been stable during operation up to now. Occasional deviations from the reference values have been unimportant only.

In case of problem (more often destructive testing):

- bolts of SG, PZR, MCP, SG threaded connections
- SG sealing (primary, secondary header, side manhole), MCP
- ESW piping inclusive of replacement of material (fibreglass)

Measurement of HV (HV5 or HV10) Hardness over Circumferential Welds

The target is the check of surface hardness of metals and comparison with the valid standards. From the point of view of further operation, minimum measured hardness values are important, mainly in welded joints that are subjected to heightened stress, especially in the areas of adapting pipes, bends, and branches.

The most suitable method is to measure the HV hardness profile over circumferential welds in a line perpendicular to the weld axis from the basic material (ZM) of the first section of the piping or component over the heat affected zone (TOZ), weld metal (SK), TOZ and ZM of the second section thus that the hardness of the individual belts of the heat affected zone is captured (also microhardness measurements can be performed in the vicinity of TOZ with spacing less than 1 mm). The measurement can be executed for every circumferential weld according to requirement following from calculation evaluation.

The measurement should include all areas determined by the calculation as the most stressed ones and also other areas to get sufficient quantity of data.

World references: [34], [35], [36], [37], [38], [39], [40].

3.1.8 Destructive Tests and Material Research Carried out during NPP Operation

Destructive tests are not performed by the personnel and laboratories of the NPP operators, they are ordered from outside expert organizations. We are giving a general

- metallographic analysis of the basic material
- evaluation of inside and outside surface defects
- chemical analysis of the condenser tube plate

2. SPP 1000 tubes – laboratory evaluation of the manufacturing defects of the tubes

Tests performed:
- mechanical and technological tests
- metallographic analysis
- establishing of corrosion attack
- chemical analysis

3. Two unit block (HVB) piping – verification of the material properties of the steels used, issuance of replacement metallurgical attests, surveillance specimens, slide valve damage

Tests performed:
- mechanical and technological tests
- metallographic analysis
- chemical analysis
- surface defect evaluation

4. Inspection pit plates

Tests performed:
- chemical analysis

5. Electrical penetrations – condition of electrical penetrations

Tests performed:
- colloidal prints
- metallographic analysis
- hardness measurements

6. Impulse piping – evaluation of the quality of the site sleeve welds

Tests performed:
- metallographic analysis
- mechanical and technological tests
- chemical analysis

7. SG – laboratory testing of M 60 bolts

Tests performed:
- metallographic analysis
- hardness measurement
- surface defect evaluation

8. Transport cask of the fuel system

Tests performed:
- chemical analysis

9. Pond of HVB I.

11. Laboratory inspections of supplies of materials for separator (Romania)

Tests performed:
- evaluation of inside and outside surfaces
- check chemical analysis
- mechanical and technological tests
- metallographic analysis of material

12. Working tests of welders – check of welding technology also for construction suppliers

Tests performed:
- mechanical and technological tests
- macroanalysis of welded joints
- MKK
- evaluation of SS faults detected

13. Within the framework of installation peraparation – test of intergranular corrosion of welded joints and liner, material condition after fire, galvanized steel profiles, steel net tensile test

Tests performed:
- MKK of the liner
- macroanalysis and microanalysis
- mechanical and technological tests
- mechanical and technological tests

Notice:

Ti tubes - HKT - main turbine condenser
- TN – feeding turbo-pump (here it concerns the tubes in TN condenser)

SPP 1000 tubes – steam separator
HVB piping – the tests mentioned have been carried out on the steam piping, feedwater piping, main steam supplying piping into the high-pressure stage, MSH 800 mm (from the intermediate machine hall)
Material for separator (Romania) – here it concerned the material for spare parts for separator supplied by SES Tlmače
MKK – intergranular corrosion
SS defects – welded joints defects.

3.1.9 Special Loading Measurement Systems (Temperature, Deformation etc.) Combined with Damage Calculation (e.g. On-line and Off-line Fatigue Monitoring)

- In addition to TMDS/MAFES diagnostic system, also TMDS/RECOP system (RCP rotating parts) can be used for lifetime assessment as well as its addition – HCC_VIB expert system. Vibrations of reactor internals are evaluated by means of TMDS/RVMS, which has also an expert system. Moreover, there are two systems evaluating "defect propagation" on the basis of the acoustic emission of TMDS/ACMS on-line and off-line.
- TMDS/LEMOP system for identification and localization of leakages evaluating acoustic signal can be used in a limited way.
- Measurement of temperatures on the outside surface of piping and components
- Measurement of stress on the outside surface of piping and components
- Measurement of global displacements of piping and components

The systems are used for selected equipment of the primary and secondary circuits, piping and components of BT1 and BT2 (according to [3])

The measurement outputs are processed by the systems for assessment of fatigue life of the NPP piping and components.

For selected components, the assessment is processed on-line, for others, always as batch after certain period.

- Every individual system uses specific methods of processing end evaluation
- Most special measurement systems are at the current world science and technology level
- The selection of equipment has been mostly performed in accordance with international experience with problematic areas where problematic stress of components begins. No special methodology of equipment selection for determination of equipment scope in the given area of special stress has been used
- The selected equipment set has been determined on agreement of research and development workplaces with the NPP operator
- The schedule of future work will depend on the requirements of the equipment administrators, on legislative requirements of the regulatory bodies, and economic aspects in relation to LTO
- No special outside legislation with the exception of the above-mentioned general regulations has been applied on the special load measurement and internal regulations and rules have been used.

3.1.10 Chemical Regimes Monitoring

application of calculating prediction of their formation. The smaller their quantity, the lower cost of maintenance, lower necessity for decontamination and subsequent equipment damaging, less radwaste and lower cost of power plant liquidation.

3.2 Plant-specific Safety Analyses Which May Have Been Based on an Explicitly Assumed Plant Life or Operating Period

Temelin Nuclear Power Plant

The following conclusions or analyses relate to the assumed plant life in the operational safety report (PpBZ) for ETE:

1) In paragraph 3.9.3 – ***Evaluation of Strength and Lifetime of the Most Important Machine Components***, where the values of residual lifetime and prognoses of accumulation of fatigue damaging consumption for systems and components during the period of active trial are set. These data relate to 40 years of operation for RPV a 30 years for other equipment.

2) In paragraph 4.3.2 – ***Reactor Vessel Irradiation***, where design value of maximum fluence of fast neutrons with an energy of ≥ 0.5 MeV on the pressure vessel for 40 years of vessel lifetime is mentioned and **in paragraph 5.3.1** – ***Pressure Vessel Materials***, where there are calculations of neutron fluxes and radiation fluences in the vessel wall and in test objects after the expected 40 years of planned operation.

3) In paragraph 5.3.2 – ***Limit Values of Pressure and Temperature,*** where increase of critical brittleness temperature and corresponding shift of limit temperature and pressure values are calculated for the planned 40 years of lifetime.

For 1)

The methods used for residual lifetime assessment in the prepared paragraph 3.9.3 comply with the approved "VERLIFE" project supported by EU *(Unified Procedure for Lifetime Assessment of Components and Piping in VVER NPPs)*, which has been transformed to A.S.I. standard, section IV, Calculation of Residual Lifetime of Equipment and Piping of Nuclear Power Plants of VVER Type. To evaluate three equipment sets of RPV, a more accurate method of classified operational modes has been used for calculation. More conservative calculations according to the design modes using DIALIFE SW have been used for the remaining RPV components and the main components of the primary circuit.

For 2)

For 3)
The analyses have been performed according to a Russian standard from 1989 [41] with using of mechanical properties of the exposed parts of the pressure vessel. Five selected points (sets) have been checked. The resistance test under the conditions of concrete modes has been made for nominal mode, hydraulic strength test of the primary circuit and several modes of the planned heating and cooling. Operational limiting curve of pressure vessel integrity has been calculated for emergency modes in accordance with a procedure used by Westinghouse Company. The test of resistance against brittle failure under the conditions of several selected design modes in the time after 40 years of operation has been carried out for five selected sets of the pressure vessel. This test has verified the resistance of all five sets of both Units in the selected modes. It is possible to say that, if the design fluences are not exceeded, even after 40 years of operation the pressure vessels of Unit 1 and 2 of NPP Temelin will be resistant against brittle fracture in the nominal mode, during hydraulic strength test of the primary circuit, and during planned heating and aftercooling.

Dukovany Nuclear Power Plant

In the current version of PpBZ EDU, there is only a summary of PTS analyses in RPV chapter related to the 40 years of assumed reactor operation showing marked dependence on time out of the time-conditioned (by the assumed operation duration) analyses. A number of other analyses (seismic calculations, evidences of resistance against extreme meteorological conditions, fall of an aircraft, pressure waves, strength calculations of component stress under emergency conditions and abnormal states) counts with certain material characteristics and properties of the stressed buildings and structures which are also time-dependent in general. Obviously, expert statement of material specialist would be sufficient for a relatively short period in the order of several decades.

In PpBZ, references to analytical evidences relating to component qualification that was done for a certain assumption of operation duration (so called qualified lifetime) are mentioned. These evidences have to be reviewed; it is sufficient to review the references to these evidences in the safety report.

The analyses of residual lifetime assessment of the main machine components towards fatigue stress and other degradation mechanisms are updated periodically according to the real operational data and therefore respect the real plant state regardless of the design assumptions of the way and duration of operation. Periodic updating of these analyses and also of all other analyses mentioned in PpBZ is assumed and performed periodically within the framework of ten year reviews of PpBZ (and PSR) regardless of extension of the period of NPP operation.

Also other activities centred on utilization of simulation methods of accelerated ageing for the fittings of the safety systems with electric drives and implemented in co-operation

The methodology of probabilistic safety assessment includes potential for solution of the specific problems of the reliability of the components important for safety in the late period of a nuclear power plant lifetime.

As regards the reliability of mechanical components, it especially means transition from the methodology of "standard" probabilistic assessment (PSA) to a more general assessment methodology including dynamic effects of ageing (APSA – ageing PSA). This requires replacing the rather simple component reliability model based on assumption of exponential division of time to failure by a more general model of three-parameter Weibull division.

Adaptation of the probabilistic model of the power plant to LTO conditions is the first step in a sequence leading to effective utilization of these methods when solving the given problems. After this step, application as such follows in all areas of utilization of probabilistic and reliability models like assessment of average operation risk, monitoring of immediate operation risk, reliability- and risk-oriented equipment maintenance, creation of source documents for risk-oriented decision making process, retrospective evaluation of operational events etc.

References: [42], [43], [44], [45].

4.0 Acronyms:

BT	Safety class
DB	Design bases
DBDs	Design bases documents
DLHM	Long-term tangible property
EDU	Dukovany NPP
ETE	Temelin NPP
HKT	Main turbine condenser
LaP	Operational Limits and Conditions
MPO	Project with financial support of Department of Commerce
OaB	Control and protection system
PKP	Complex assessment report
PP	Work Procedure
PpBZ	Operation Safety Report
PPO	Repair Work Procedure
SoD	Contract
SONS	State Office for Nuclear Safety
TZi	Investment technical specification
TZp	Operational technical specification
TŘN	Non-conformance technical solution

5.0 Compilation of a list of reference documents from which the above information was collected

[1] Law No. 18/1997 Coll., on Peaceful Utilization of Nuclear Energy and Ionizing Radiation

[2] Law No. 50/1976 Coll. – Building Act

[3] SONS Regulation No. 214/97 Coll. on Quality Assurance during Activities Connected with Utilization of Nuclear Energy and Activities Leading to Irradiation and on Establishing of Criteria for Inclusion and Dividing of Selected Equipment into Safety Classes

[4] ČÚBP (Czech Occupational Safety Office) Regulation No. 76/89 Coll. Ensuring of Safety of Technical Equipment in Nuclear Power Engineering as amended by Regulation No. 263/1991 Coll.

[5] SONS Regulation No. 106/98 Coll.

[6] Law No. 505/90 – Metrology Act

[7] Regulation No. 69/91 – Regulation to apply Law No. 505/90

[8] FIKS-CT-2000-000065: Signal Processing and Improved Qualification for Non-destructive Testing of Ageing Reactors (SPIQNAR)

[9] FIKS-CT-2001-00172: Nuclear Risk-Based Inspection Methodology for Passive Components (NURBIM)

[10] ČSKAE (Czechoslovak Atomic Energy Commission), Regulations for Inspections of Welded Joints and Weld Claddings for NPP, PK 1514-72, June 1976

[11] Regulation No. 174/68 Coll., ITI

[12] Recommendations for an Effective Flow-Accelerated Corrosion Program, NSAC-202L, EPRI

[13] Requirements for Analytical Evaluation of Pipe Wall Thinning, Section XI, Division 1, ASME Code Case N-597

[14] Assessment of Strength of Equipment and Piping of VVER Type Nuclear Power Plants, Section III, normative technical documentation, A.S.I., 1996

[15] ČÚBP Regulation No. 18/1979 Coll. for Restricted Technical Equipment – Pressure Vessels

[16] ČÚBP Regulation No. 19/1979 Coll. for Restricted Technical Equipment – Lifting Equipment

[17] ČÚBP Regulation No. 20/1979 Coll. for Restricted Electrical Equipment

[18] ČÚBP Regulation No. 21/1979 Coll. for Restricted Gas Equipment

[19] ČÚBP Regulation No. 48/1982 Basic Requirements for Ensuring of Safe Work and Technical Equipment

[22] NS-G-2.6 Maintenance, Monitoring, and In-service Inspections in Nuclear Power Plants

[23] 50-C/SG-Q5 Assessment

[24] 50-C/SG-Q13 Quality Assurance in Operation

[25] SONS Regulation No. 195/1999 Coll., on Requirements on Nuclear Facilities for Ensuring of Nuclear Safety, Radiation Protection, and Emergency Preparedness

[26] Nuclear Facilities Safety – Guidelines and Recommendations for Qualification of Equipment Important for Safety of VVER 440/213 Type Nuclear Power Plants, SONS, Prague, 12/1998

[27] Equipment Qualification in Operational Nuclear Power Plants: Upgrading, Preserving and Reviewing, IAEA Safety Report Series No.3, 1998

[28] ČSN IEC 60780: 2001 (35 6609) Nuclear Power Plants – Safety System Electrical Equipment – Qualification Verification

[29] ČSN IEC 980:1993 (IEC 60980:1989) Recommended Procedures for Seismic Qualification of Safety System Electrical Equipment for Nuclear Power Plants

[30] IEEE Std -323-1983 Standard for Qualifying Class 1 E Equipment for Nuclear Power Generating Stations

[31] IEEE Std 344-1987 IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations ASME QME-1-1994 Qualification of Active Mechanical Equipment Used in Nuclear Power Plants

[32] ÚJV Řež, a.s.: "Ball Indentation Tests and Method of Evaluation of Indentation Diagrams - Demonstration of Both Calibration and Testing", Rep. NRI Rez plc No: Z 435T, 4/ 1999

[33] ÚJV Řež Report: DITI 304/125 "Evaluation of Breaking Strength of Steel of Core Supporting Cylinder of EDU Unit 1 (08Ch18N10T) after ca 130 000 Hours of Operation by ABIT Method".

[34] F. M. Haggag: Field Indentation Microprobe for Structural Integrity Evaluation. U.S. Patent 4,852,397, August 1989.

[35] TEST Ltd.: Universal Automatic Hardness Tester TEST-10U, Instructions for Use (in Russian).

[36] D. Tabor: Hardness of Metals. Oxford: Clarendon Press, 1951.

[37] Y. Tirupataiah, G. Sundararajan: Metall. Trans. A 22A (1991), 2375.

[38] T. Ishibashi, S. Shimoda: JSME International Journal 31 (1988), 117.

[39] R. Hill, F.R.S., B. Storakers, A. B. Zdunek: Proc. R. Soc. Lond. A 423 (1989), 301.

[40] B. Storakers, P.-L. Larson: J. Mech. Phys. Solids 42 (1994), 307.

[44] Holý J., Statistical Support of Experiment with Fitting Accelerated Ageing: Methods, Commentaries, Suggestions, Appendix 7 to the Feasibility Study for Test Programme of Accelerated Ageing of Fittings with Electric Drives, NRI/FSR1/07.03, 2003

[45] Rodionov A., Holý J., *Selection of Data Processing Methods for Reliability Test Planning (Accelerated Ageing Tests),* "Lifetime Length, Ageing, and Degradation Models in Reliability, Health Service, Medicine, and Biology" conference, Petrograd, 7-9 June 2004

Country information report (Finland)
Loviisa NPP approaches to long-term operation

Ritva Korhonen, Fortum Nuclear Services Ltd

IAEA-EBP-LTO

2nd Meeting of WG2 on Mechanical Components and Materials

Vienna, November 16 - 18, 2004

1.0 Applicable laws specific to materials and components for long-term operation

- **The Nuclear Energy Act (990/87) and the Nuclear Energy Decree (161/88)**
  - general safety principles of the use of nuclear energy, the licensing, control of the use of nuclear energy, competent authorities
- **Council of State Decision (395/91)**
  - general safety requirements for nuclear power plants and plant structures
- **The YVL Guides**
  - detailed regulations relating to the safety of nuclear power plants
  - issued by the Radiation and Nuclear Safety Authority (STUK)
  - regulations that the holder of a licence shall comply with

1.0 Applicable laws specific to materials and components for long-term operation, cont.

- The general objective is to ensure nuclear plant safety so that nuclear power plant operation does not cause radiation hazards which could endanger safety of workers or population in the vicinity or could otherwise harm the environment or property.
- A nuclear power plant's safe design and the high quality of its components and operation contribute to the safety of the plant.

1.0 Applicable laws specific to materials and components for long-term operation, cont.

- It is the licence-holder's obligation to assure the safe use of nuclear energy.
  - The licensee is obliged to demonstrate that safety principles are met.
  - The maintenance and improvement of an achieved safety level require a highly advanced safety culture.
- STUK controls the design, construction and operation of a nuclear power plant
  - to ensure the operation of the NPP is safe and complies with the licence conditions and the approved plans
  - to ensure the operation also in other respects adheres to the Nuclear Energy Act and other regulations
  - the control of operation also involves the maintenance, repairs, inspections and tests of the NPP SSCs.

1.0 Applicable laws specific to materials and components for long-term operation, cont.

- The Government issues the licence for the operation of the nuclear power plant.
- STUK issues a statement with its safety assessment on the application for an operating licence to which a safety assessment is attached.
- The operating licence of a NPP is granted for a fixed term (usually ten years).
- When the length of the term is considered, particular attention shall be paid to ensuring safety and to estimated duration of operations.

1.0 Applicable laws specific to materials and components for long-term operation, cont.

- The application for an operating licence shall include, among other things,
  - the planned service life of the NPP,
  - the construction licence or the previous operating licence of the NPP;
  - outline of the technical operating principles and other arrangements whereby safety has been ensured,
  - a description of the safety principles that have been observed, and an evaluation of the fulfilment of the principle;
  - if the NPP has previously been granted an operating licence description of how the provisions in the previous operating licence have been complied with;
  - any other information considered necessary by the authority.

1.0 Applicable laws specific to materials and components for long-term operation, cont.

•The application for an operating licence for a NPP already in operation shall include the documents below only to the extent they have not been submitted before.

- The final safety analysis report
- A probabilistic safety analysis
- A quality assurance programme for the operation of the NPP
- The technical specifications (Technical and administrative requirements and restrictions for ensuring the safe operation of a NPP)
- A summary programme for periodic inspections

1.0 Applicable laws specific to materials and components for long-term operation, cont.

•The licence-holder is responsible for revising the documents submitted to STUK with the application of construction or operating licence when modifications in the NPP systems or structures or in the way the NPP is operated are carried out.

•When modifications will have an effect on safety and will involve changes in the plans or documents approved by STUK the licence-holder shall obtain an approval from STUK for these modifications.

1.0 Applicable laws specific to materials and components for long-term operation, cont.

•If licence conditions are modified to ensure the safe use of a NPP same procedure shall be followed as when the licence was granted or if an amendment to the operating licence is applied the same regulations shall be observed as in the case of a new licence, to the extent applicable.

1.0 Applicable laws specific to materials and components for long-term operation, cont.

•Provisions for the design requirements for ensuring nuclear safety

- For ensuring primary circuit integrity: 'The primary circuit of a nuclear reactor shall be designed so that the stresses imposed upon it remain, with sufficient confidence, below the values defined for structural materials for preventing a fast growth crack during normal operational conditions, anticipated operational transients and postulated accidents. The possibility of a primary circuit break due to other reasons shall be low, too.'
- The functions important to the safety of the SSCs shall be defined and the SSCs classified according to their safety significance.
- The SSCs important to safety shall be designed, manufactured, installed and operated so that their quality level as well as the inspections and tests required to verify their quality level are adequate considering any item's safety significance.

1.0 Applicable laws specific to materials and components for long-term operation, cont.

•Requirements for reliable operation of a nuclear power plant and its systems and components

- **Appropriate procedures shall exist for the operation, maintenance, in-service inspections and periodic tests as well as transient and accident conditions of a NPP.**
- **In all activities affecting the operation of a NPP and the availability of components, a systematic approach shall be applied for ensuring plant operator's continuous awareness of the state of the plant and its components.**
- **Operating experience from NPPs as well as results of safety research shall be systematically followed and assessed.**

Fortum

1.0 Applicable laws specific to materials and components for long-term operation, cont.

•Requirements for the licences for the devices and equipment; licence validity and licence conditions shall be specified

- **reactor pressure vessel**
- **control rods and control rod guide tubes**
- **electronic controls for controlling the power levels in nuclear reactors**
- **pressure tubes to contain the primary coolant in a nuclear reactor (operating pressure > 5.1 MPa)**
- **coolant pumps specially for circulating the primary coolant of nuclear reactors**
- **internal components specially for the operation of a nuclear reactor**
- **heat exchangers**

Fortum

1.0 Applicable laws specific to materials and components for long-term operation, cont.

•YVL Guides for requirements and control of pressure equipment (pressure vessels, piping and their support structures, valves and pumps)

- STUK controls the safety of pressure vessels (YVL 3.0)
- STUK approves manufacturers of nuclear pressure equipment for their duties and inspection or testing organizations for duties pertaining to the control of pressure equipment at nuclear facilities

•Requirements and provisions laid down for the operation in other legislation shall also be observed

- Pressure Equipment Act (869/1999)
  - technical requirements and demonstration of safety for conventional pressure equipment at a NPP

Fortum

1.0 Applicable laws specific to materials and components for long-term operation, cont.

•Other regulations (YVL Guides) ensuring the reliable operation of the SSCs

- Manufacturing requirements of mechanical components and structures to ensure high quality operations and that manufacturing is carried out according to manufacturing documents, the applicable regulatory requirements and decisions. The manufacturers of mechanical equipment and structures shall have a quality system whose functionality is monitored.
- Regulations on repairs, modifications, preventive maintenance and related inspections of SSCs at a NPP during operation and outages.
- The requirements for quality assurance programmes and valid instructions and procedures (for all activities which affect safety and relate to the design, construction and operation of a NPP).

Fortum

1.0 Applicable laws specific to materials and components for long-term operation, cont.

Long-term operation of SSCs is not specifically regulated by the laws or regulations.

Assessment of NPP safety continues after the granting of the operating licence constantly during operation.

Fortum

2.0 Regulatory requirements applicable to long-term operation (and ageing management practices in the NPP)

•Regulatory control of operating NPPs contains reviews and inspections (YVL 1.1)

- **periodic inspections specified and registered by STUK in a plant-specific program (contents reviewed annually, the inspection period - usually one year - defined for each item) and focused at the power company's activities important to safety**
- **special inspections which the power company is obliged to request when measures are carried out at the plant or which STUK conducts at its discretion; requirements and obligations presented in the YVL Guides**
- **safety assessment based on operating experience, safety research and other information such as development of NPP technology and changed safety requirements, obtained after the granting of the operating licence**

Fortum

2.0 Regulatory requirements applicable to long-term operation, cont.

•Fixed-term and event-specific reports are submitted to STUK for evaluation of safety enhancing measures and overall monitoring of the safety level.

•The need for and possibilities of improving safety are considered on the basis of a safety assessment made during operation.

•STUK annually draws up a plan for reassessment of the issues reviewed for an operating licence. The annual special safety assessment is made known to the licensee for possible action, if necessary.

2.0 Regulatory requirements applicable to long-term operation, cont.

•An overall safety assessment is made for the plant, e.g. in connection with the renewal of an operating licence.

- licence-holder's and STUK's safety assessments

•The documents approved by STUK in connection with the review of an operating licence application shall be updated, e.g.

- approved changes of the SSCs
- the results of PSA applied to the assessment of needs for technical specifications changes in connection with extensive plant changes
- case specific risk analyses performed by the licensee for decision making, regarding e.g. applications for exemption from technical specifications, urgent reparation and modification works and analyses of operational events.

2.0 Regulatory requirements applicable to long-term operation, cont.

•The periodic safety review (PSR) is required to be carried out to renew the operating licence.

- An overall safety assessment; the licensee shall provide
  - assessment whether the licence-holder's activities and the NPP comply with current regulations
  - report of operating experiences and ageing of the plant
  - report of development of organizational practices and competence of personnel; assessment of the safety culture of the plant
  - other reports required by STUK
- An assessment of ageing

2.0 Regulatory requirements applicable to long-term operation, cont.

•A Safety Guide NS-G-2.10 on the Periodic Safety Review of Nuclear Power Plants

- overall plant safety,
- effectiveness of ageing management programs

•The scope of a PSR includes all nuclear safety aspects of a NPP, i.e. all facilities and SSCs on the site covered by the operating licence and their operation.

2.0 Regulatory requirements applicable to long-term operation, cont.

- **An objective of the PSR is to determine the actual condition of SSCs important to safety and their adequasy to meet their design requirements**
  - Focus on reactor pressure vessel and reactor primary circuit, reactor coolant system, containment and protection automation, cables, electrical systems, etc.
- **The review shall confirm that the condition of SSCs is properly documented.**
- **When the safety of a plant is assessed, both probabilistic and deterministic safety analyses are used.**

2.0 Regulatory requirements applicable to long-term operation, cont.

- **Safety principles used in the design of NPPs are presented for systems important to safety in Guide YVL 1.0.**
- **SSCs important to safety**
  - their malfunction or breakage can significantly increase the radiation exposure of the plant's workers or the environment
  - they prevent the occurrence and propagation of transients and accidents
  - they shall mitigate the consequences of accidents

2.0 Regulatory requirements applicable to long-term operation, cont.

- YVL 1.0 requires regarding ageing of components and materials that in NPP design, the service life and the effect of their ageing on the safety of all safety significant structures, components and materials shall be assessed using sufficient safety margins.
- To ensure that the SSCs perform reliably, it must be possible to service, inspect and test them over the entire design service life of the plant.
- Provision shall be made for the surveillance of their ageing and their possible replacement or repair.

2.0 Regulatory requirements applicable to long-term operation, cont.

- All SSCs shall be so designed that they perform reliably under design-basis environmental conditions.
- The operability of a SSC under the environmental conditions in question shall be demonstrated by the necessary tests and analyses.
- Requirements for the environmental qualification of structures and components are prescribed in YVL Guides which apply to these structures and components.

2.0 Regulatory requirements applicable to long-term operation, cont.

•STUK controls the manufacture of pressure vessels and other mechanical components of NPPs (YVL 1.1).

•During design, manufacture, installation, inspection and testing of the reactor primary circuit, state-of-the-art know-how and technology shall be effectively used to prevent fast growth crack and any other serious failure.

•Frequently occurring loads and stress peaks transferred to the circuits' components, and the number of welded seams in the circuit should be limited.

2.0 Regulatory requirements applicable to long-term operation, cont.

•Radiation induced and chemical effects as well as thermal, hydraulic and mechanical loads shall be taken into account so that fracturing of the pressure vessel and other components of the primary circuit is highly unlikely.

•The safety margin for operational conditions and postulated accidents must be sufficient regarding brittle behaviour of structural materials.

•Uncertainty factors shall be taken into account when properties of structural materials, irradiation of the materials and the effects of irradiation on material properties, stresses and flaw sizes are determined.

2.0 Regulatory requirements applicable to long-term operation, cont.

- The primary circuit and its components shall be so designed that
  - important areas and details can be periodically inspected and their structural integrity and leaktightness evaluated
  - the periodic reactor pressure vessel materials testing program to determine the effects of radiation and the ageing of structural materials can be implemented.

2.0 Regulatory requirements applicable to long-term operation, cont.

- Requirements for inspections and control of mechanical components are presented in YVL Guides.
- STUK controls by periodical inspections the implementation of condition monitoring and preventive maintenance programs of piping and reviews the examination results.
- The safety class of components is taken into account when determining the scope of control and setting the requirements.
- Only organizations granted approval by STUK are allowed to perform licensed inspection and expert duties relating to mechanical components.

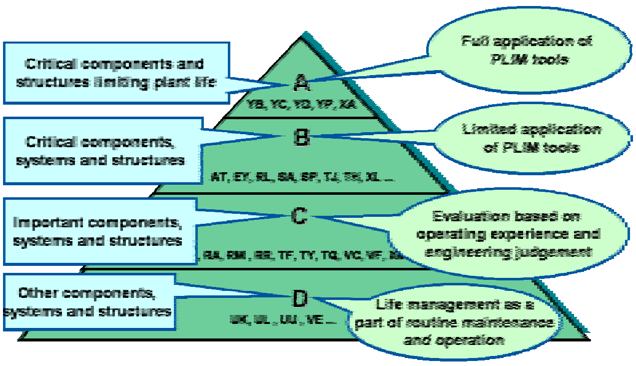
2.0 Regulatory requirements applicable to long-term operation, cont.

- Availability of safety significant SSCs can be upgraded by analysing their surveillance testing periods and procedures.
  - PSA for identifying SSCs, potential and common cause failures
- The testing program of these SSCs shall be regularly evaluated on risk basis during the operation of the plant.
- An estimate of risk significance of preventive maintenance during operation of safety significant SSCs
- The results of PSA shall be used in drawing up and development of the inspection programs of piping.
  - Combining the information from PSA and the damage mechanisms of pipes and the secondary impacts of damages, risk informed inspection program is drawn up.

Fortum

2.1 Process used to develop scope of systems, structures and components within the long-term operation review (in Loviisa)

Definition and classification of critical systems in Loviisa NPP



Fortum

2.2 Configuration control practices used to control design basis

•When renewing the operating licence the plant construction and safety are assessed to comply with the current requirements.

•Design data shall be given on all situations at the NPP for which requirements concerning the operability of equipment are set.

•Ageing during normal operation caused by ambient conditions and other ageing phenomena shall be considered in design.

2.2 Configuration control practices used to control design basis, cont.

•The construction plan of SSCs shall state design bases.

•The design bases shall be so extensively presented that acceptability of SSCs to the system, regarding e.g. overall planning, operational prerequisites, structural requirements, functioning, control of materials, strength calculations, inservice inspections, condition monitoring and maintenance operations, can be evaluated.

•The prerequisites for design bases of SSCs are provided in appropriate YVL Guides.

2.2 Configuration control practices used to control design basis, cont.

•Design bases for e.g. piping shall include

- safety class
- process and instrumentation charts
- a description of the operation of piping, incl. operating and design parameters, their range of variation and the number of load cycles
- information on the contents of piping and external conditions
- information on pressure tests and piping accessories.
- In addition harmful dynamic and fatigue-inducing loads shall be prevented
  - e.g. vibrations, pressure shocks, restriction of thermal expansion, temperature fluctuations in thermal mixing locations and thermal stratification of the medium
- susceptibility to erosion corrosion shall be limited
- phase transitions of the flowing medium and accumulation of uncondensable gases shall be considered.

2.2 Configuration control practices used to control design basis, cont.

•In a system's design tested and proven design methods, appropriate regulations, guidelines and standards, and the quality requirements of the safety class shall be followed.

•VVER-440 type reactors have initially been designed in accordance with Russian rules and codes.

•Materials and manufacturing procedures have been defined in Russian standards and rules.

2.2 Configuration control practices used to control design basis, cont.

•The Loviisa NPP has been designed to meet the Finnish requirements.

•The periodic inspections programs are in accordance with ASME Code.

•Other important standards

- the standardization of ANSI (American National Standards Institute),
- ASME (American Society of Mechanical Engineers),
- DIN (Deutsches Institut für Normung),
- ISO (International Organization for Standardization),
- SFS (Finnish Standards Association) and
- EN (European Committee for Standardization).

2.3 Ageing management programs

•From start up of operation 1977 Lo1, 1981 Lo2:

- The safety improvement program guided by PSA has been permanently carried out.
- Deteriorating or obsolete equipment have been replaced periodically or when necessary.
- Extensive procedures have been implemented related to degradation management of active components.

•Long-term investment plan

2.3 Ageing management programs, cont.

•A plant ageing management program has been developed at the Loviisa plant to co-ordinate different programs and projects related to ageing.

- Continuous ageing management processes of critical SSCs for the residual plant lifetime are carried out or established.
- The PLIM program comprises research of the ageing phenomena, operational experience, routine repairs and preventive maintenance.
- All maintenance and inspection activities should support plant life management.
- The ageing management related data of the selected critical SSCs is collected into a SSC specific database.
- Instructions related to plant life management processes have been prepared.

2.3 Ageing management programs, cont.

•A new ageing management group has been established in 2002 at the Loviisa NPP.

- A nominated PLIM engineer is in charge of ageing management of a system group (SSC) co-ordinating ageing analyses, updating ageing information into the ageing specific database and making proposals for changes to the preventive maintenance program, for appropriate repair or modification work.
- Specialists from the plant and technical support organization give feedback and pass information to the responsible PLIM engineers on operating, maintenance and inspection experiences of the Loviisa plant or other plants and the results of ageing research.
- Yearly seminars are organized for exchanging of information.

2.3 Ageing management programs, cont.

•The activities of ageing management program include

- The actions to be performed and assessed yearly (preventive maintenance during operation and annual outages, timing and evaluation of the scope of the changes in operation routines).
- Evaluation work and activities during the licensing period (investigation and research programs for components, optimal scope of the preventive maintenance practices, a long-term modification program including safety improvements required e.g. do to new regulation guides).
- Activities to be considered and realized during the residual plant lifetime (correct economical timing of the large modifications, ageing management regarding comprehensive inspection, maintenance, research and data collection programs for economically irreplaceable components).

2.3 Ageing management programs, cont.

•In connection with the operating licence renewal STUK supervises and evaluates

- whether a systematic plant life management program has been established to ensure adequate safety functions of the SSCs during the applied operating period
- effectiveness of the plant life management procedures and functioning of the program
- technical aspects of ageing management activities
- that a plant has inspection and control programs which monitor the effects of ageing on the intended function of the SSCs, instructions to ensure required integrity and functional capability of SSCs, and maintenance programs to prevent degradation to maintain the ability to perform the intended function.

2.3.1 Research/ Process Providing Basis for Applicable Aging Effects on Structure and Component Intended Function(s)

- The research programs on plant ageing phenomena and life management methods as well as the program on nuclear plant safety concentrating on ageing, accidents and risk issues are carried out in co-operation with the Technical research centre of Finland (VTT).
- STUK carries out nuclear safety research including
  - development of the regulatory control of nuclear power plants
  - structural safety of nuclear power plants
  - operational safety of nuclear power plants
- Power companies carry out their own research activities.

2.3.2 Aging Mitigation Measures Applied in Nuclear Power Plants

- Section 3

3.0 Operational Approaches Applicable to Long-Term Operation

•The most important plant life management issues at Loviisa NPP

3.1 Normal Operational Practice/Programs Applicable to Aging Management

•Description of key areas of the Loviisa plant life management

- management of the reactor pressure vessel embrittlement,
- erosion corrosion of secondary system piping
- fatigue in piping systems,
- steam generators,
- austenitic castings / main coolant pumps,
- main gate valves,
- reactor internals embrittlement,
- control rod drive mechanisms,
- I&C and electrical equipment, electrical cables,
- building structures

3.1 Normal Operational Practice/Programs Applicable to Aging Management, cont.

3.1.1 In-service Inspection Practices for Passive Components

- Section 2.

3.1.1.1 Augmented Inspection Programs That Address Issues Such As Erosion/Corrosion

- Subsection 3.1.

3.1.1.2 Augmented Inspection of Steam Generator Tubing

- Subsection 3.1.

3.1.1.3 Augmented Inspection for Specific Degradation Mechanisms Such As Intergranular Stress Corrosion Cracking

Fortum

3.1 Normal Operational Practice/Programs Applicable to Aging Management, cont.

3.1.2 Maintenance Codes or Practices for Active Components

- Section 2 and Subsection 3.1.

3.1.3 Equipment Qualification Practices

- Section 2 and Subsection 3.1.

3.1.4 Component function tests

3.1.5 Applied diagnostic systems

3.1.6 Surveillance specimen programmes (irradiation damage, corrosion loops)

- Subsection 3.1.

Fortum

3.1 Normal Operational Practice/Programs Applicable to Aging Management, cont.

3.1.7 Nondestructive Material Properties Tests (hardness measurement etc.)

3.1.8 Destructive Tests and Material Research Carried out During NPP Operation

3.1.9 Special Loading Measurement Systems (temperature, deformation etc.) Combined with Damage Calculation (e.g., on-line and off-line fatigue monitoring)

- **Subsection 3.1.**

3.1.10 Chemical regimes monitoring

3.2 Plant-Specific Safety Analyses Which May Have Been Based on an Explicitly Assumed Plant Life or Operating Period

- **The original design lifetime specified for the Loviisa power plant was 30 years.**
- **The amount of transients of each type was specified as the design basis.**
- **Analysis was provided to prove that the main equipment would tolerate all of the load cycles with an adequate safety margin during the plant lifetime.**
- **30 years of operation was only an estimate, the issues affecting the engineering lifetime of the plant have to be identified and assessed for the extended operation of the plant.**

3.2 Plant-Specific Safety Analyses Which May Have Been Based on an Explicitly Assumed Plant Life or Operating Period, cont.

- **The modifications performed as well as new assessments based on the analysed operating data and more accurate calculations have provided basis on current evaluation of the plant residual lifetime.**
  - **The importance of gentle treatment of all equipment during normal operation and under test conditions has been emphasized.**
- **For re-licensing a thorough safety review is required, which also includes the assessment of ageing.**
  - **PLIM will make an important element of the operating license renewal in 2007.**

4.0 Compilation of a list of reference documents from which the above information was collected

- **Nuclear Energy Act (990/1987)**
- **Nuclear Energy Decree (161/1988)**
- **The Council of State Decision on the General Regulations for the Safety of Nuclear Power Plants,** 14 Feb. 1991/395
- **Guide YVL 1.0 Safety criteria for design of nuclear power plants, 12 Jan. 1996 (Finnish Centre for Radiation and Nuclear Safety Guide)**
- **Guide YVL 1.1 Finnish Centre for Radiation and Nuclear Safety as the regulatory, 27 Jan. 1992**
- **Periodic Safety Review of Nuclear Power Plants. Safety Guide No. NS-G-2.10, IAEA, August 2003.**
- **Guide YVL 1.3 Mechanical components and structures of nuclear facilities. Approval of testing and inspection companies, 17 March 2003 (in Finnish)**
- **...**

HUNGARY

Country Information Report

Chapter 1-2

Rátkai Sándor

Wien, 16-18. November, 2004.

1. Applicable Laws Specific to Materials and Components for LTO

Features:

- US NRC License Renewal approach was adapted
- Focusing on long-lived passive SSC
- Environmental Impact Study has to be performed
- Two step process

Most important guides:

4.15 Life extension programs during operation;
1.28 Requirements related to the content of application for long term operation license;
3.13 Considering the aging processes in nuclear power plants designing;
4.12 Considering the aging processes during operation of nuclear power plants;
2.15 Quality Assurance in managing the aging of NPP equipment and components;
1.26 Regulatory inspection of aging management program;
1.26.S1 The scope of aging management;
4x In-service inspection of civil engineering structures and structural components. (Draft guide).
4.18 Maintenance programs;
1.27 Regulatory reviews during controlling the maintenance efficiency;
1.21 Construction Authority inspection in nuclear power plants (under revision);
4.6 Maintenance of NPP equipment (under revision).

2. Regulatory Requirements Applicable to LTO (1)

2.1. The process used in developing the scope of systems, structures and components (SSCs) that are within the long term operation (LTO) review

- scoping + screening

2.2. Configuration Control Practices used to Control Design Basis

- Updated FSAR has to be submitted (shows the as is condition after every reconstruction)
- (PSR performed every 10 years)
- (EQ)

2. Regulatory Requirements Applicable to LTO (2)

2.3. Ageing Management Programs

2.3.1. Research…

- In the last ten years comprehensive aging management studies have been performed
- The studies were based on IAEA methodology (TR:338)

2.3.2. Ageing mitigation measures applied in NPP…

RPV – low leakage core (30% flux reduction)

SSC(SC1) – operational cycle counting (+redesign)

Erosion/C.– introduction of high pH (up to 9.6)

….

Thank you for you attention!

HUNGARY

Country Information Report

Chapter 3

Rátkai Sándor

Wien, 16-18. November, 2004.

3. Operational approaches applicable to LTO (1)

3.1. Normal Operational Practice/Programs Applicable to Aging Management and Inspection Practices for passive Components

Major operational aging management programs:

- in-service inspection of high safety significant components,
- periodic maintenance programs,
- systematic technical safety inspection plans (TSIP).

3. Operational approaches applicable to LTO (2)

3.1.1. In-Service Inspection

The in-service inspection program is based on:

- regulatory requirements,
- design requirement,
- manufacturers instructions, (New development: ASME XI)
- operational experience

Features:

- obligatory to develop and execute (periodically) the ISI program
- based on the basis of safety classification of the SSCs
- evaluation criteria determined in accordance with the design requirements, technical standards
- the program include all examination techniques (visual, volumetric, destructive, non-destructive, etc..)
- strength and tightness pressure tests
- helps the decision if any necessary corrective or preventive measures shall be conducted
- the records shall be kept up to the time of the decommissioning

3. Operational approaches applicable to LTO (3)

3.1.1.1. Operators'documents connected to ISI

1. Material testing frame programs (K1-K10)
   - frame program validity data;
   - list of relevant documents;
   - subject of inspection;
   - inspection table with information pertaining to cycle time of given inspection;
   - reference concerning inspection technology and criterion;
   - accurate technical drawing of equipment with indication of test places
2. Criterion collection
3. Test technologies
   - validity area of technology;
   - description of test system, test equipment;
   - requirement system for test personnel;
   - records of technologies qualification
   - description of coordinate system applied;
   - description of test process;
   - determination of corrections applicable at evaluation;
   - data recording and evaluation levels;
   - content and formal requirements for documents recording test results;
   - safety engineering, radiation protection and fire protection regulations

3. Operational approaches applicable to LTO (4)

3.1.1.2. Technical safety inspection plan (TSIP)

TSIP is a means of in-service verification of the technical conditions of pressure vessels and piping (including tanks and piping under hydrostatic pressure).

TSIPs are elaborated for each safety class (Class-1, 2, 3) piping and vessels including their supports.

TSIPs determine the requirements for the technical safety inspection by visual examination and by hydraulic pressure test.

Visual inspections conducted by technical safety inspection of piping and vessels are in line with VT-1 examination of ASME XI. IWA-2211 and are to detect discontinuities and imperfections on the surfaces of components including such conditions as cracks, wear, corrosion or erosion. For some of components (mainly Class-2, 3 piping and vessels and supports) a VT-3 examination is foreseen (ASME XI. IWA-2213). VT-3 inspections are to verify clearances, settings, and physical displacements, and to detect discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear or erosion.

For the detection of the evidence of leakage from pressure retaining components hydraulic pressure tests are conducted at a pressure 1,25xPlicensed, and are followed by VT-2 examination (see ASME XI. IWA-2212).

Visual inspection, non destructive examination and pressure test of Classes-1, 2 vessels and Class-1 piping are conducted every 4 years, while other inspections are on an 8-year frequency.

TSIPs contain a detailed description of vessels and piping with special emphasis to degradation mechanisms to be focused on during the inspection.

Results of the inspections are documented in the passport of the vessel or piping, which are completed for each vessel and for each pipeline with common tag-number.

3. Operational approaches applicable to LTO (5)

Periodic Maintenance Programs:

- **Maintenance and repair programs are elaborated in accordance with the design and manufacturing requirements.**
- **The requirements concerning the level and frequency of the preventive maintenance is reviewed during operation, on the basis of the safety classification, manufacturers' requirements, operational experiences and the results of periodical inspections, and modified if necessary.**
- **During development of the preventive maintenance strategy, the planned and anticipated lifetime of the different SSCs are taken into account.**
- **Maintenance instructions used for preservation of safety and good technical conditions of SSCs:**
- The scope and the aim of maintenance activities, Maintenance manual , Generic maintenance technology,
- Non-series repair technology, Construction and welding procedure, Welding procedure specification (WPS),
- Manufacturing procedure, Heat-treatment list, Lifting technology,
- The necessary conditions to start maintenance, The procedure and detailed instructions for operation of maintenance,
- The inspection activities and criteria, The measures to be taken in case of non compliance,
- The personnel necessary for maintenance, The necessary tools and maintenance materials,
- The QA requirements according to safety classification of SSC to be maintained or repaired
- Documentation requirements,
- The documents relating to the maintenance of Class-1, 2, 3 SSCs shall be preserved during the total lifetime of SSC.

3. Operational approaches applicable to LTO (6)

3.1.1.3. Inspection programs (erosion/corrosion)

- Systematic ultrasonic wall thickness measurements are performed regularly in the E/C affected components' locations. For the systematic control of the E/C process expert system COMSY(WATHEC) is in extensive use.

3.1.1.4. Inspection of SG tubing

- 100 % of the steam generator tubes have been detected by using Eddy-current method. The majority of the indications have been found under the tube support plates.
- Destructive testing of a sample of pulled out tubes was used for validation of the Eddy current inspection results. All the indications are originating from the secondary side of the SG.

3.1.1.5. Inspection for ISSC

- conventional ultrasonic ISI methods
- Eddy Current technique

3. Operational approaches applicable to LTO (7)

3.1.2. Maintenance Codes or practices for Active Components

- USNRC Maintenance rule has been adopted
- SC1-2 component level, SC3-4 system level

3.1.3. Equipment Qualification Practises

- mainly for I and C components
- in the case of high safety significance special function tests for mechanical components

3.1.4. Component function tests

Thank you for you attention!

PRACTICAL STEPS FOR AGEING MANAGEMENT

- Identification the scope of the ageing management
- Definition of dominant ageing mechanism and stressors
- Identification of sensitive locations
- Characterisation of dominant ageing mechanism (calculations and analyses)
- Determination of monitoring parameters
  - installed technological measurements
  - installation of new measurements (example:surge line)
  - review of ISI operation and ISI plans
- Development of data base for ageing monitoring
- Data acquisition
- Fill the monitoring system
- Analyse, feed back (maintenance, operation, conditions keeping

LIST OF COMPONENTS *NEEDS AGEING MONITORING*

- REACTOR PRESSURE VESSEL
- REACTOR VESSEL INTERNALS
- REACTOR VESSEL SUPPORTS
- CONTROL ROD DRIVE MECHANISM
- REACTOR COOLING SYSTEM
- PIPING CONNECTED TO RCS
- STEAM GENERATOR
- MAIN CIRCULATING PUMP
- PRESSURIZER
- MAIN GATE VALVE
- HYDROACCUMULATOR
- HIGH SAFETY SIGNIFICANCE PUMPS, VALVES AND CONNECTING PIPING
- EMERGENCY DIESELGENERATOR
- CABLES
- CONTAINMENT STRUCTURE
- CONTAINMENT PENETRATIONS (MECHANICAL AND ELECTRICAL)
- CONTAINMENT ISOLATION VALVES
- CONTAINMENT LINERS
- FEED WATER PIPING, PUMPS,VALVES

- IN THE NEAR FUTURE ADDITIONAL COMPONENTS WILL BE ANALISED !
  - SAFETY RELATED HEAT EXCHANGERS
  - PIPING SUPPORTS
  - SPENT FUEL POOLS
  - CONTAINMENT VENTILLATION SYSTEM

TYPICAL CONTENT OF THE AMP REVIEW REPORTS ASSESSMENT OF THE AGING PROCESSES IN THE DESIGN

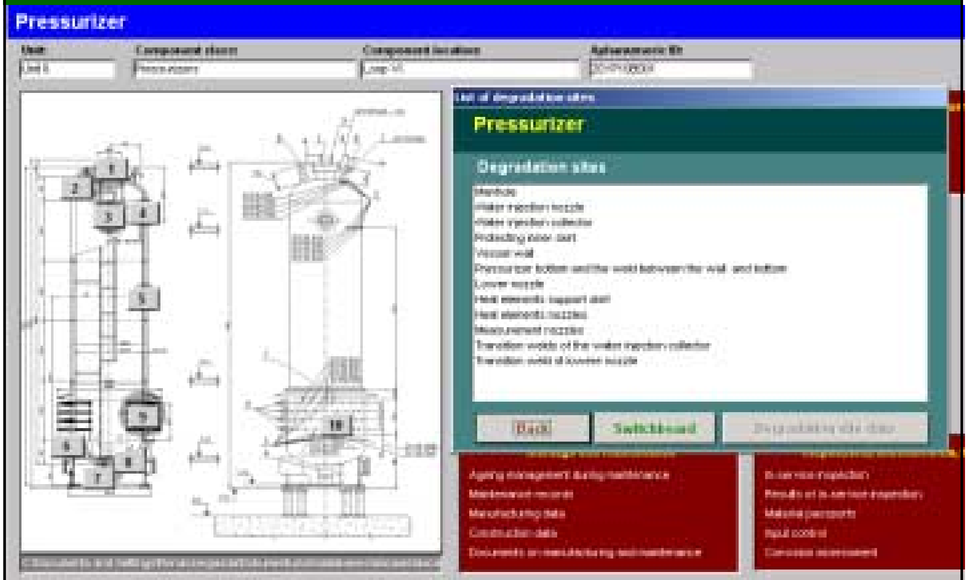
1. IDENTIFY AGING RELATED OPERATIONAL PROCESSES AND ENVIRONMENTAL CONDITIONS
   - 1.1. OPERATIONAL PARAMETERS OF THE SYSTEMS
   - 1.2. NORMAL OPERATIONAL CYCLES.
   - 1.3. SPECIAL OPERATIONAL MODES
   - 1.4. WATER QUALITY NORMS - CORROSION - STRESSORS
   - 1.5. ENVIRONMENTAL DATA AND PROVISIONS
   - 1.6. EQUIPMENT QUALIFICATION PROVISION
2. COMPONENT DESIGN
   - 2.1. LOCATIONS OF STRESS CONCENTRATION
   - 2.2. LOCATIONS SUSCEPTIBLE TO CORROSION
   - 2.3. POSSIBLE STRESSORS AND LOCATIONS OF OTHER AGING MECHANISMS
   - 2.4. THE INSERVICE INSPECTION CONDITIONS
   - 2.5. MONITORING CONDITIONS
   - 2.6. MAINTENANCE, REPLACEMENT, RENEWAL CONDITIONS
3. APPLIED MATERIALS
   - 3.1. STRENGTH AND CORROSION RESISTANCE CHARACTERISTICS
   - 3.2. LONG TERM STABILITY OF MATERIAL CHARACTERISTIC
   - 3.3. DAMAGE PROPAGATION RESISTANCE
   - 3.4. COMPATIBILITY OF THE MATERIALS
   - 3.5. NPP EXPERIENCE OF THE APPLIED MATERIALS
4. ANALYSIS OF THE AGING PROCESSES OF THE SYSTEM COMPONENTS
   - 4.1. IDENTIFY POSSIBLE DEGRADATION MECHANISMS
   - 4.2. DESCRIBE THE DEGRADATION PROCESS
   - 4.3. IDENTIFY DEGRADATION LOCATIONS
   - 4.4. ESTIMATE THE EXPECTED IMPACT OF THE AGING PROCESSES- RESIDUAL LIFE
   - 4.5. SENSITIVITY ANALYSIS OF THE AGING PROCESSES- MITIGATION POSSIBILITIES
5. DESIGN SPECIFICATION OF THE AGING MANAGEMENT
   - 5.1. OPERABILITY SPECIFICATION
   - 5.2. MAINTENANCE MODE SPECIFICATION
   - 5.3. PERIODIC MATERIAL INSPECTION AND FUNCTIONAL TESTING SPECIFICATION
   - 5.4. MONITORING
   - 5.5. QUANTITATIVE CHARACTERISTICS OF THE POSSIBLE AGING PROCESSES -

     CRITERIAL VALUES

TYPICAL CONTENT OF THE AMP REVIEW REPORTS: AGING MANAGEMENT DURING THE OPERATION

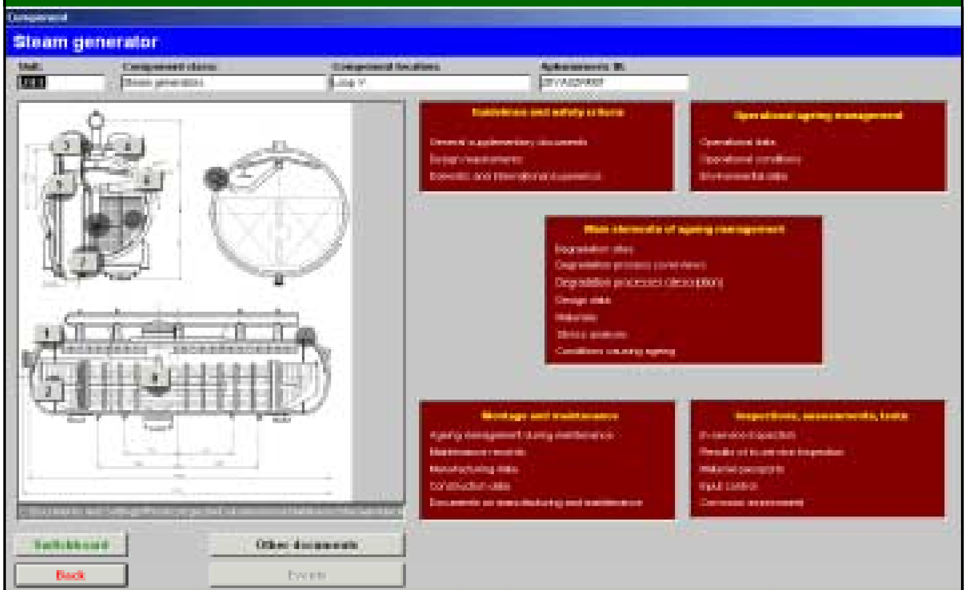
1. AGING MANAGEMENT PROGRAM
   - 1.1. DEFINITION OF THE EXTENT
   - 1.2. ORGANIZATIONAL BACKGROUND
   - 1.3. SYSTEM OF THE PROCEDURES AND REGULATIONS RELATED TO THE AGING MANAGEMENT OF SSC
2. OPERATIONAL AGING MANAGEMENT PROCESS
   - 2.1. DATA COLLECTION AND RECORD KEEPING OF AGING RELATED CHARACTERISTICS
   - 2.2. RECORD KEEPING ABOUT THE OPERATIONAL INSPECTIONS, FAILURES, MALFUNCTIONS E.T.C.
   - 2.3. OPERATION OF THE MONITORING SYSTEMS
3. AGING MANAGEMENT IN THE MAINTENANCE PROCESS
   - 3.1. PREDICTIVE MAINTENANCE PROGRAMS
   - 3.2. MAINTENANCE AND REPAIR INSTRUCTION MANUALS
   - 3.3. RECORD KEEPING OF THE RESULTS OF ROOT CAUSE ANALYSIS OF THE FAILURES
   - 3.4. RECORD KEEPING ABOUT THE SPARE PART RESERVING
4. ANALYSIS OF THE AGING PROCESSES OF THE SYSTEM COMPONENTS
   - 4.1. IDENTIFY POSSIBLE DEGRADATION MECHANISMS
   - 4.2. DESCRIBE THE DEGRADATION PROCESS
   - 4.3. IDENTIFY DEGRADATION LOCATIONS
   - 4.4. ESTIMATE THE EXPECTED IMPACT OF THE AGING PROCESSES- RESIDUAL LIFE
   - 4.5. SENSITIVITY ANALYSIS OF THE AGING PROCESSES- MITIGATION POSSIBILITIES
5. DESIGN SPECIFICATION OF THE AGING MANAGEMENT
   - 5.1. OPERABILITY SPECIFICATION
   - 5.2. MAINTENANCE MODE SPECIFICATION
   - 5.3. PERIODIC MATERIAL INSPECTION AND FUNCTIONAL TESTING SPECIFICATION
   - 5.4. MONITORING
   - 5.5. QUANTITATIVE CHARACTERISTICS OF THE POSSIBLE AGING PROCESSES - CRITERIAL VALUES

CRITICAL COMPONENTS OF THE PRESSURIZER



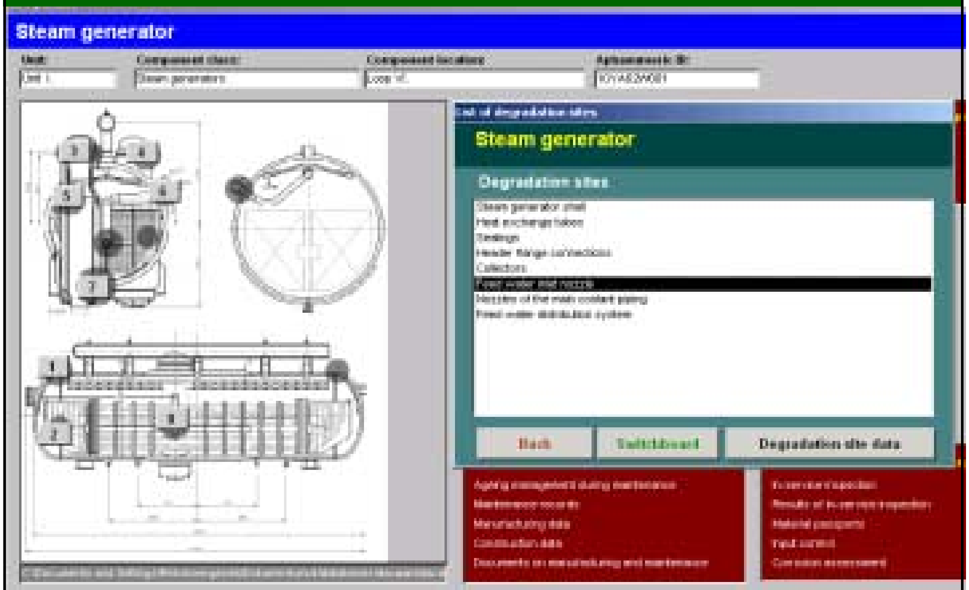
5

Data recovery on component/critical component level



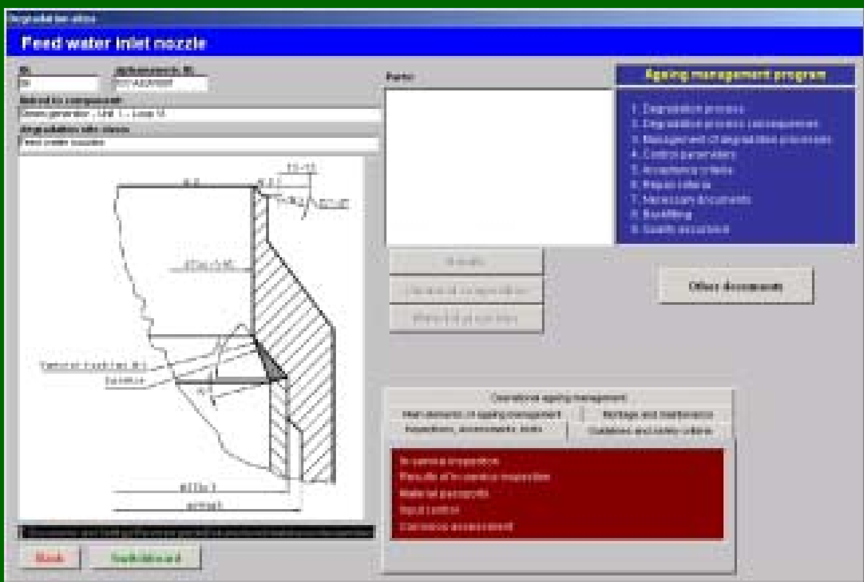
6

PAKS NPP AGING MANAGEMENT PC DATABASE: CRITICAL COMPONENTS OF THE STEAM GENERATOR



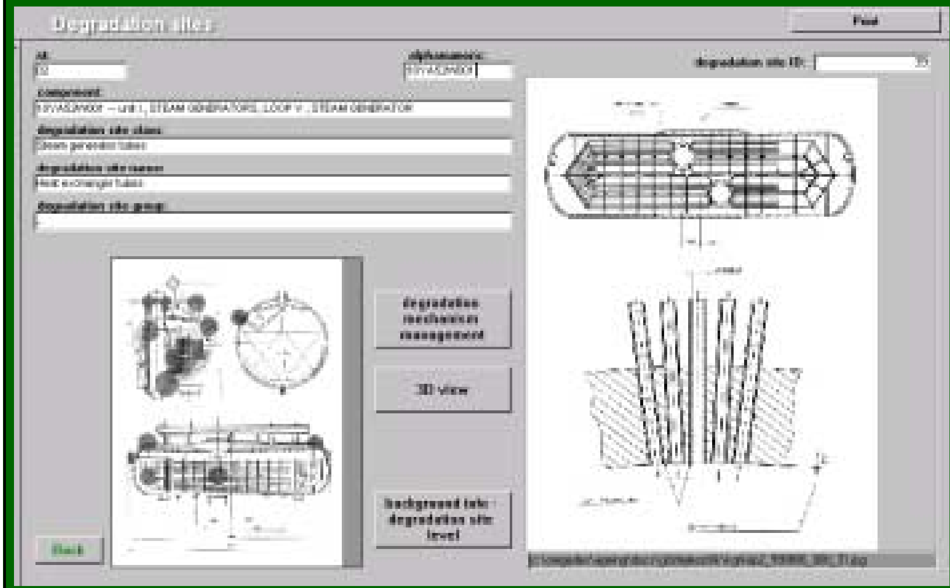
7

PAKS NPP AGING MANAGEMENT PC DATABASE: 2. CRITICAL COMPONENT - MAIN FEED WATER NOZZLE

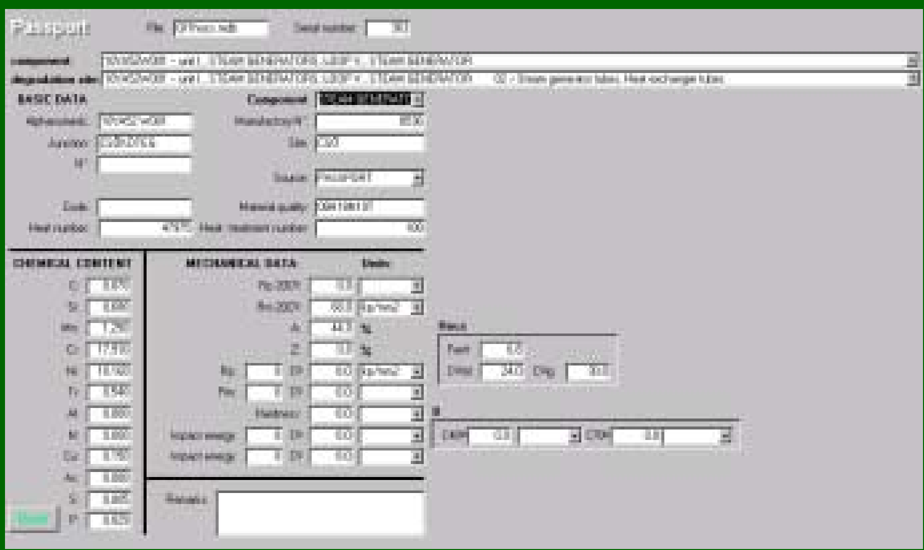


8

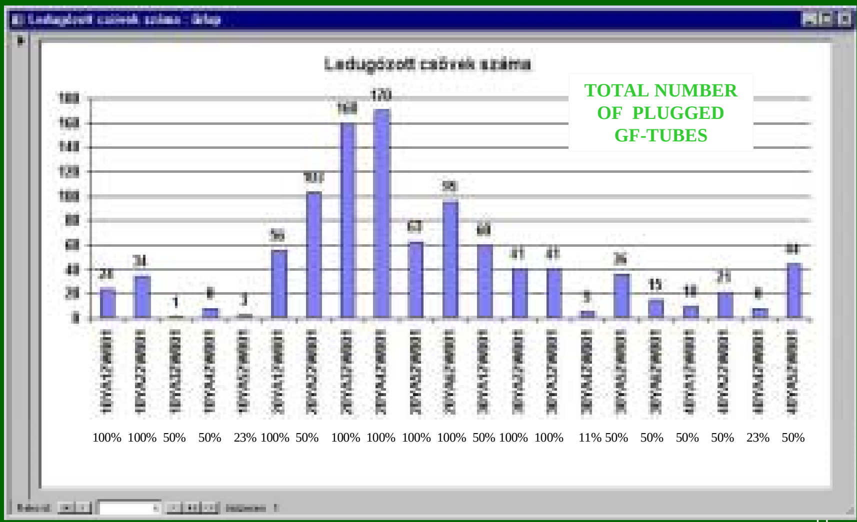
AGEING MONITORING SYSTEM



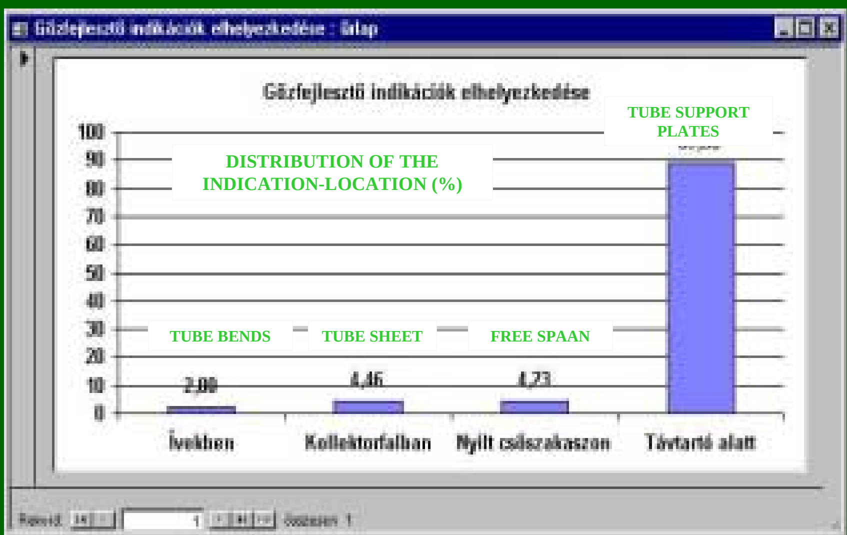
AGEING MONITORING SYSTEM



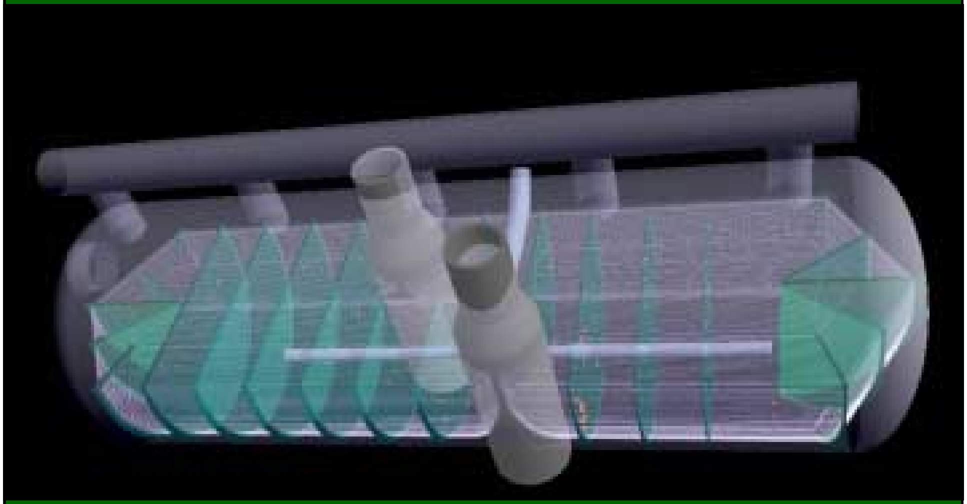
PLUGGED TUBES IN THE STEAM GENERATOR



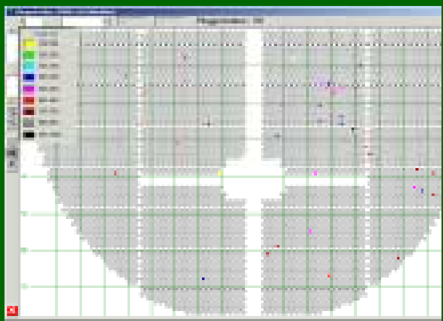
INDICATIONS IN THE STEAM GENERATOR HEAT EXCHANGE TUBES

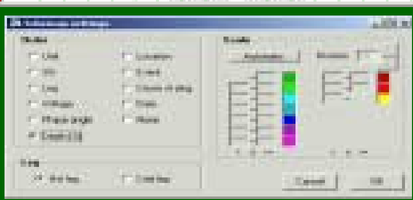


PLUGGED TUBES IN THE STEAM GENERATOR

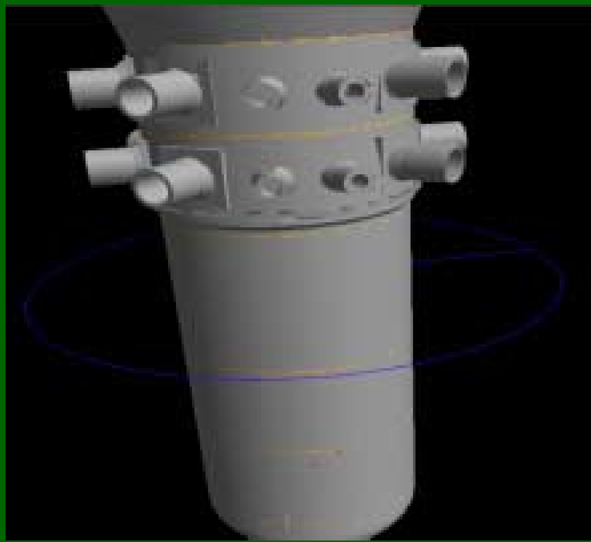


RECOVERY OF SG TUBE DEGRADATION DATA





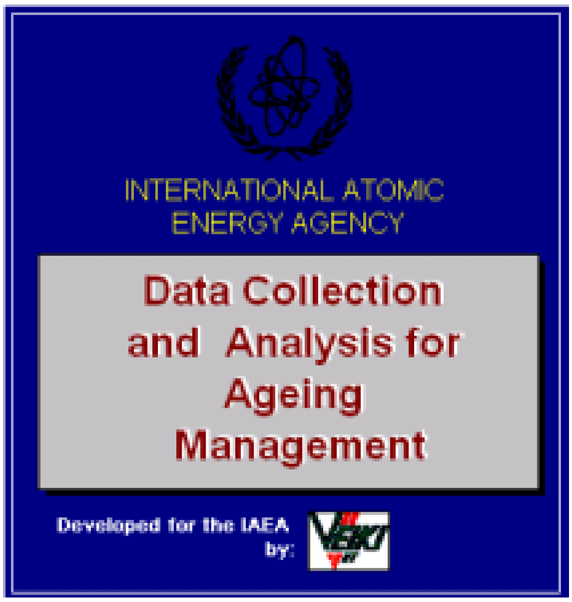
DETECTED INDICATIONS IN THE REACTOR PRESSURE VESSEL

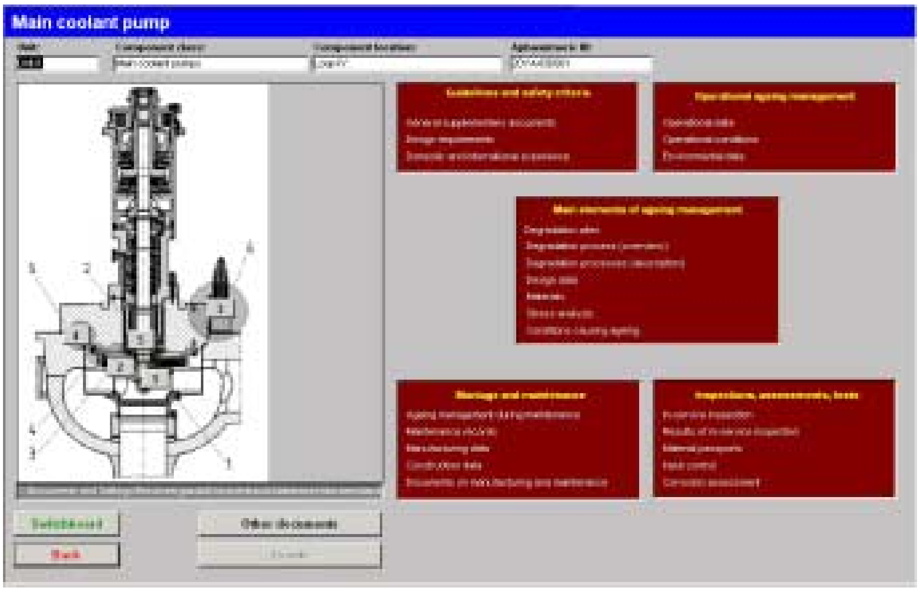


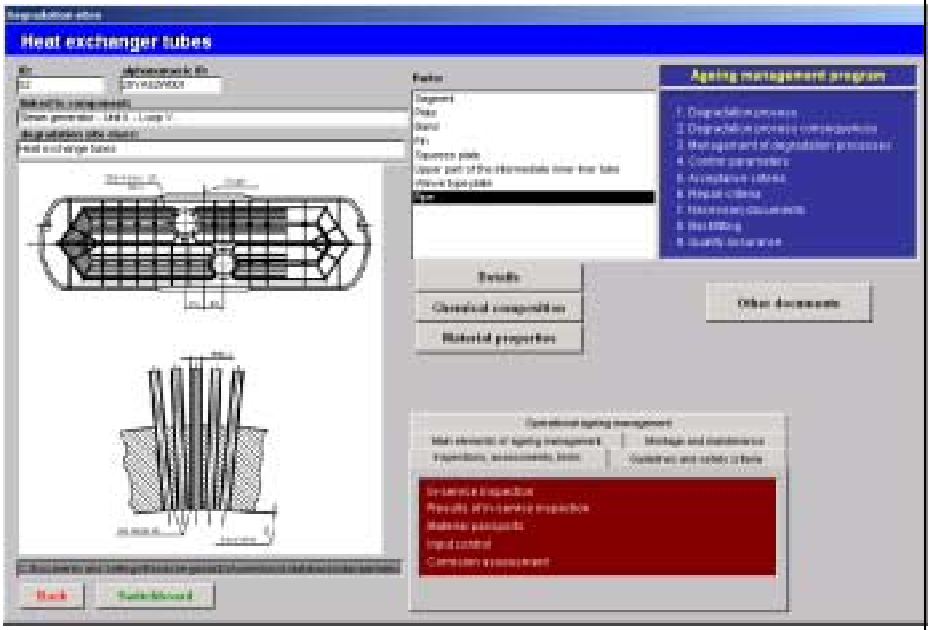
15

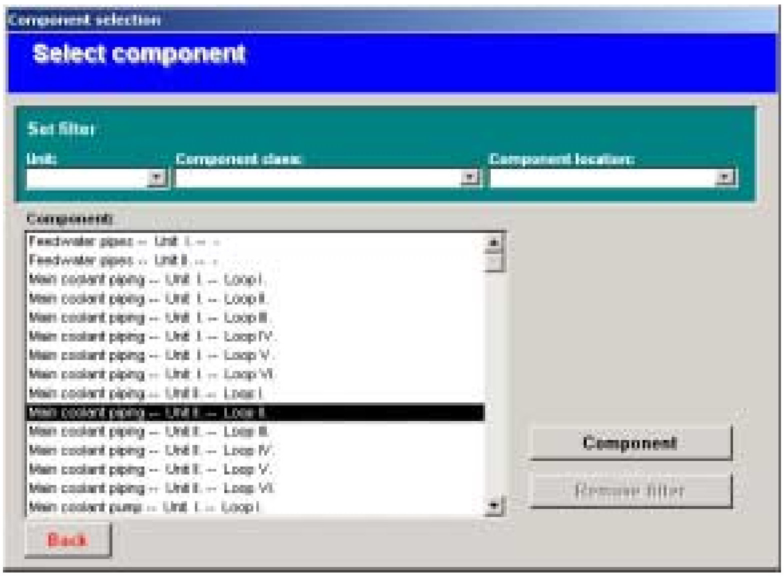
DETECTED INDICATIONS IN THE REACTOR PRESSURE VESSEL (ZOOM)

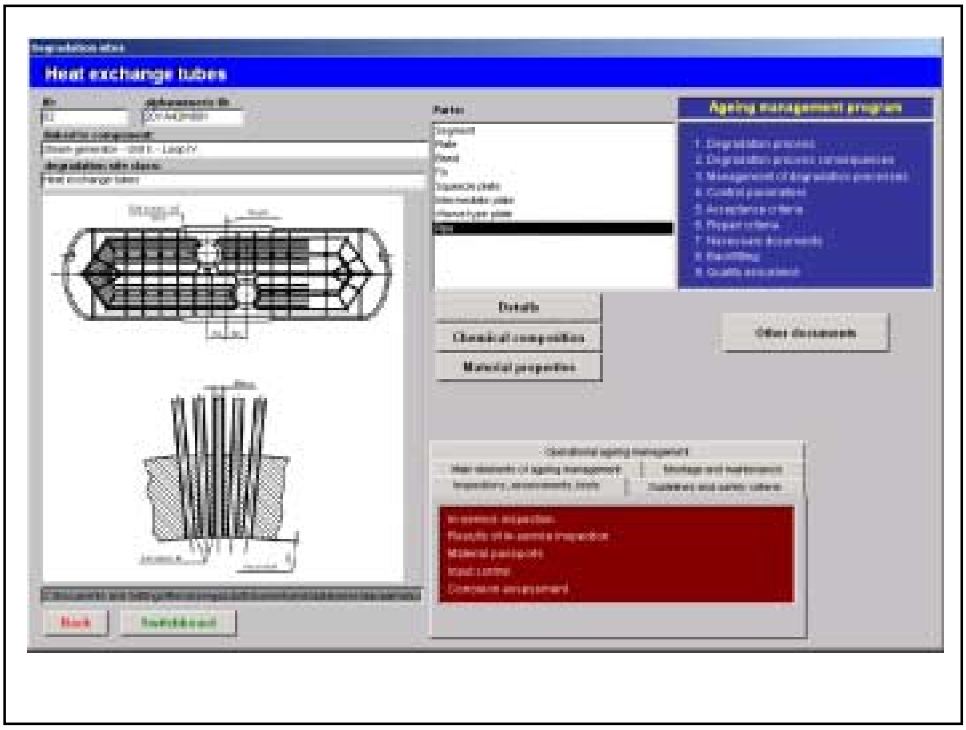
16

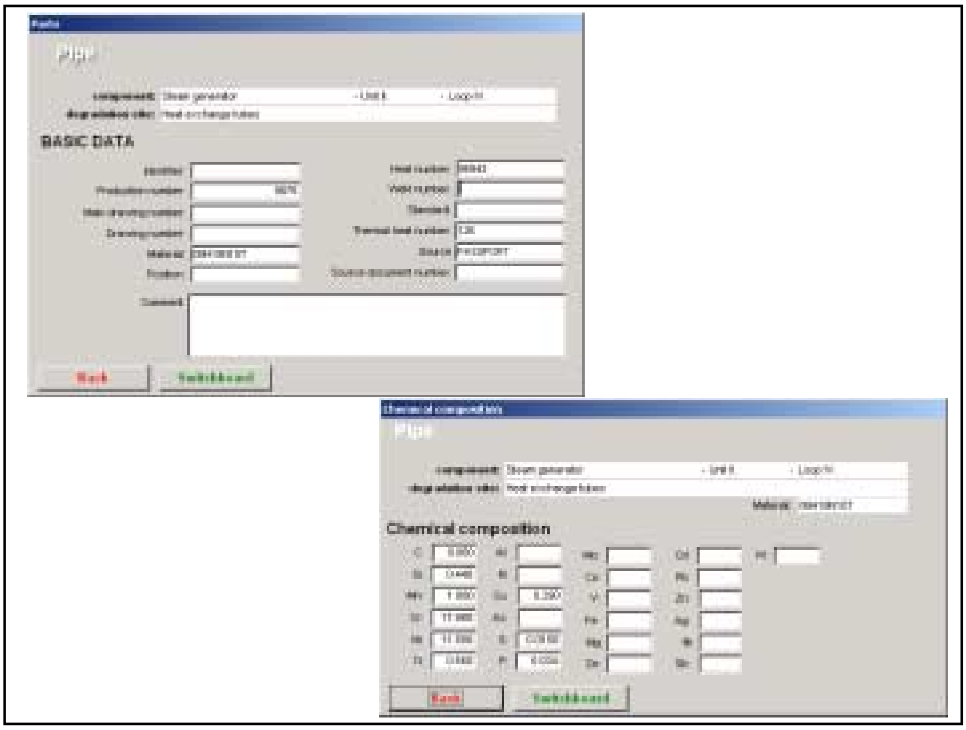

INTERNATIONAL ATOMIC
ENERGY AGENCY
Data Collection
and Analysis for
Ageing
Management
Developed for the IAEA
by:

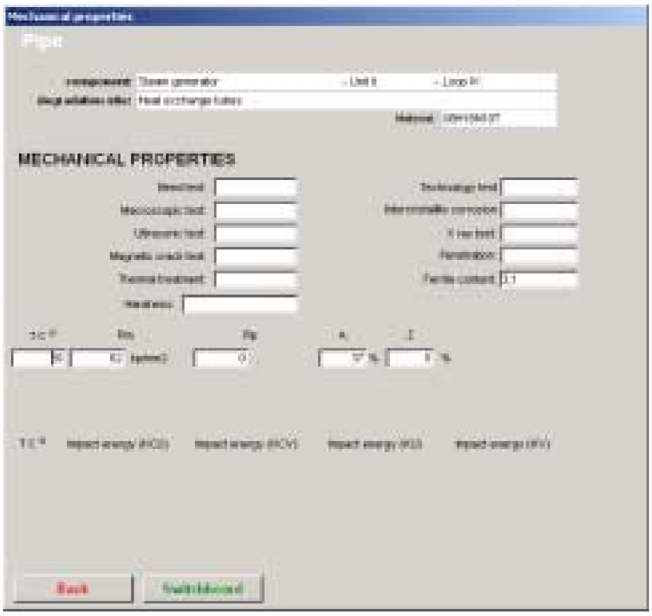

Main coolant pump
Other documents


Heat exchanger tubes
Ageing management program
1. Degradation process
2. Degradation process consequences
3. Management of degradation processes
4. Control parameters
5. Acceptance criteria
6. Repair criteria
7. Necessary documents
8. Retrofitting
9. Quality assurance
Details
Chemical composition
Material properties
Other documents
In-service inspection
Results of in-service inspection
Material passports
Input control
Corrosion assessment
Back

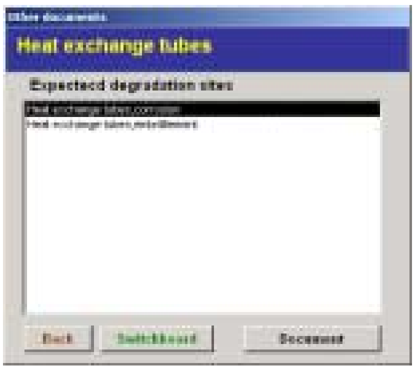

Component selection
Select component
Set filter
Unit:
Component class:
Component location:
Component:
Feedwater pipes -- Unit I -- -
Feedwater pipes -- Unit II -- -
Main coolant piping -- Unit I -- Loop I
Main coolant piping -- Unit I -- Loop II
Main coolant piping -- Unit I -- Loop III
Main coolant piping -- Unit I -- Loop IV
Main coolant piping -- Unit I -- Loop V
Main coolant piping -- Unit I -- Loop VI
Main coolant piping -- Unit II -- Loop I
Main coolant piping -- Unit II -- Loop II
Main coolant piping -- Unit II -- Loop III
Main coolant piping -- Unit II -- Loop IV
Main coolant piping -- Unit II -- Loop V
Main coolant piping -- Unit II -- Loop VI
Main coolant pump -- Unit I -- Loop I
Component
Remove filter
Back

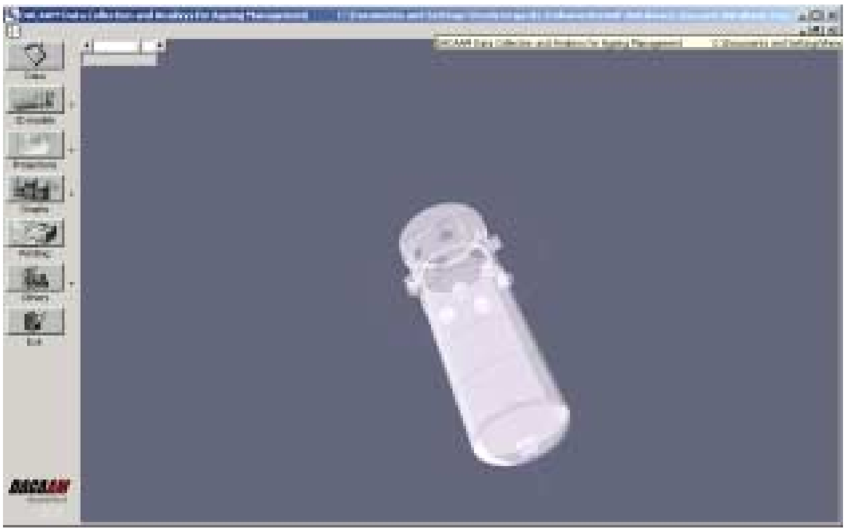

Heat exchange tubes
Ageing management program
1. Degradation process
2. Degradation process consequences
3. Management of degradation processes
4. Control parameters
5. Acceptance criteria
6. Repair criteria
7. Necessary documents
8. Backfitting
9. Quality assurance
Parts:
Details
Chemical composition
Material properties
Other documents
Operational ageing management
Main elements of ageing management
Montage and maintenance
Inspections, assessments, tests
Guidelines and safety criteria
In-service inspection
Results of in-service inspection
Material passports
Input control
Corrosion assessment
Back
Switchboard


Parts
Pipe
component:
Steam generator
degradation site:
Heat exchange tubes
BASIC DATA
Identifier
Production number
Main drawing number
Drawing number
Material
Position
Comment
Heat number
Weld number
Standard
Thermal heat number
Source
PASSPORT
Source document number
Back
Switchboard
Chemical composition
Pipe
component:
Steam generator
degradation site:
Heat exchange tubes
Material
Chemical composition
C
Si
Mn
Cr
Ni
Ti
Cu
S
P
V
Back
Switchboard

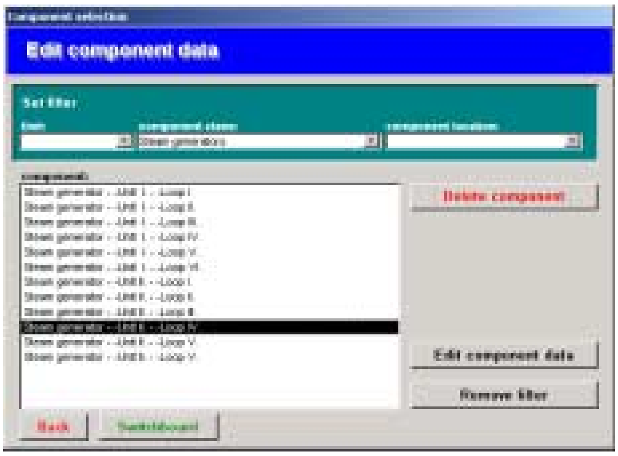

Pipe
MECHANICAL PROPERTIES
Back
Switchboard


Heat exchange tubes
Document type
Back
Switchboard
Documents

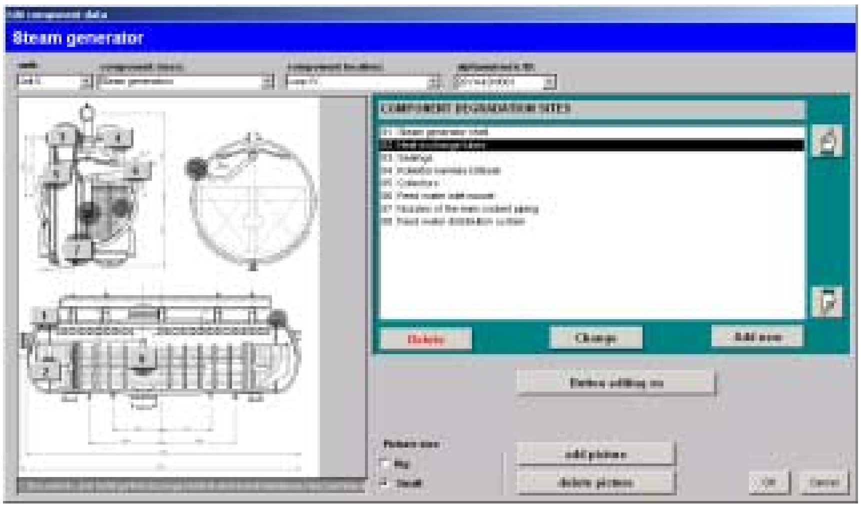

Heat exchange tubes
Expectecd degradation sites
Back
Switchboard
Document



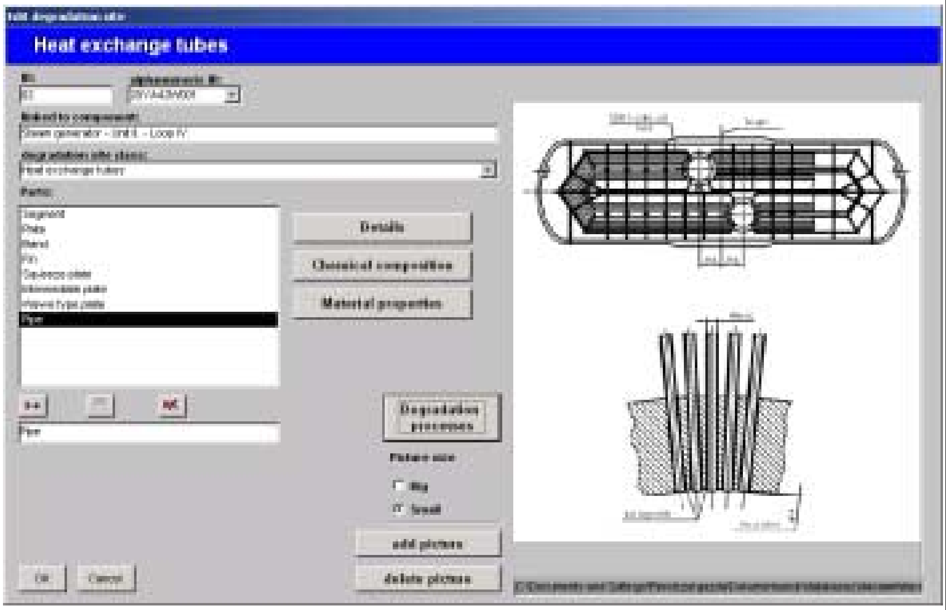

Edit documents
Other documents
Back
Document data



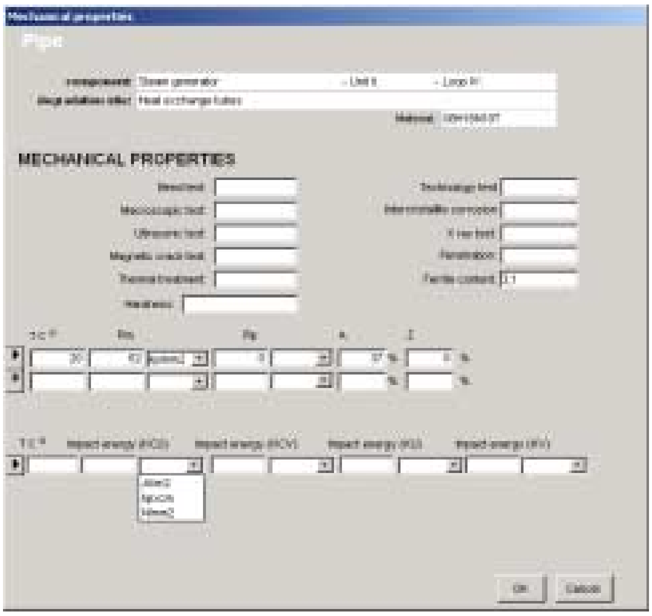
Edit component data
Set filter
Delete component
Edit component data
Remove filter
Back
Switchboard

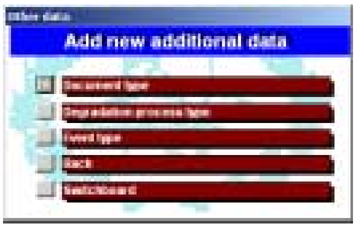


Steam generator
COMPONENT DEGRADATION SITES
Delete
Change
Add new
add picture
delete picture

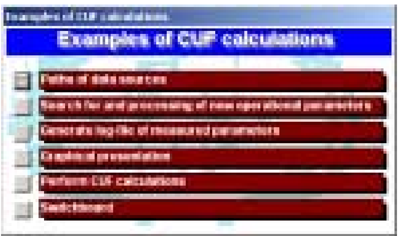

Heat exchange tubes
Details
Chemical composition
Material properties
Degradation processes
add picture
delete picture

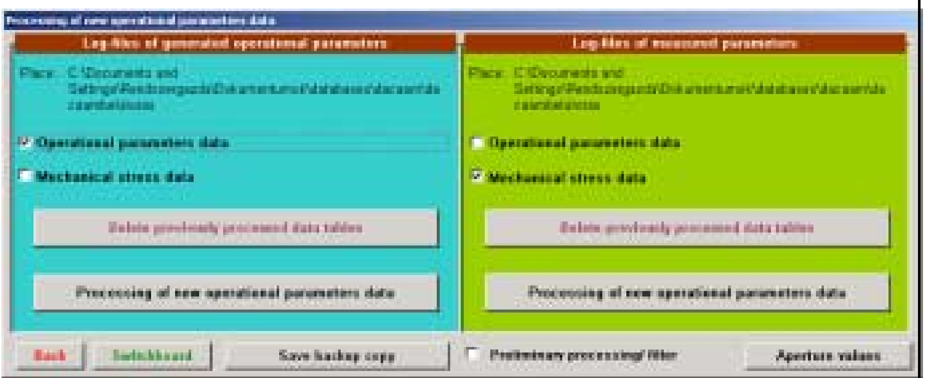

Heat exchange tubes
Back
Switchboard
Add
Delete


MECHANICAL PROPERTIES


Add new additional data


Degradation process types
Back
Switchboard


Examples of CUF calculations


Back
Switchboard
Save backup copy
Aperture values

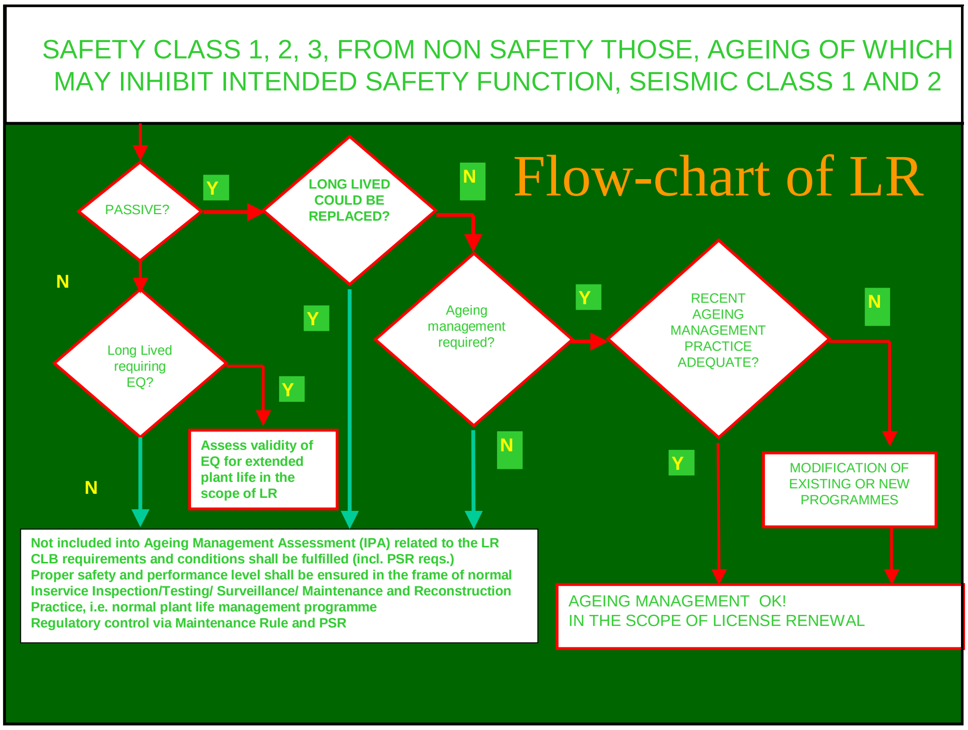
REVIEW ON THE CURRENT ISI PROGRAM

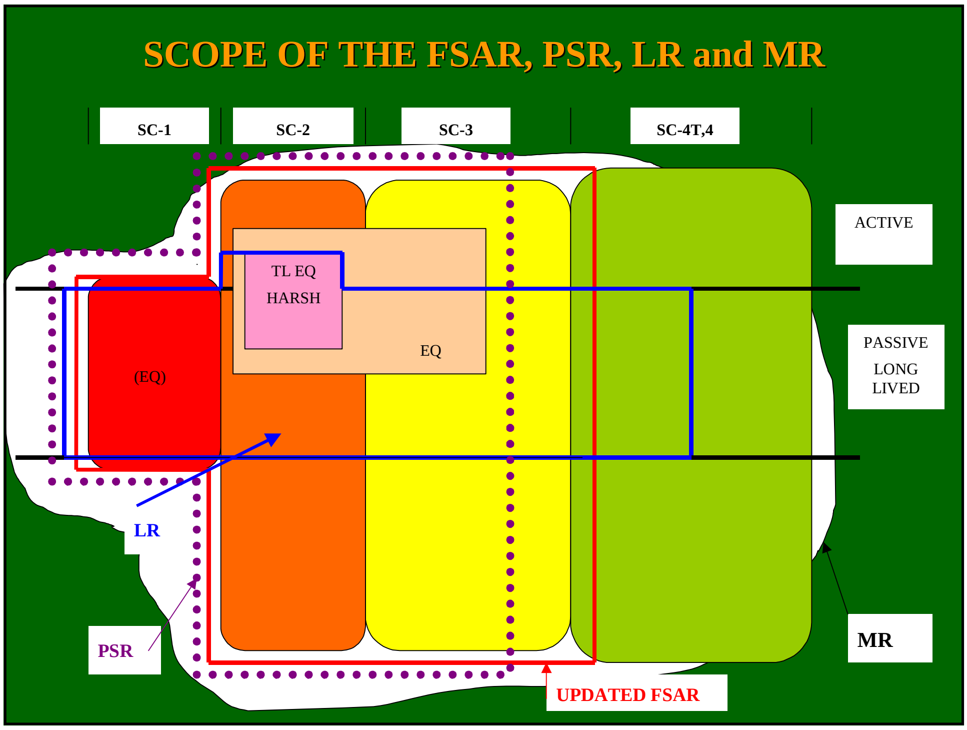
GOALS:

- supporting the LR activities
- introduction of widely accepted ISI methodology
- maintaining the current outage period

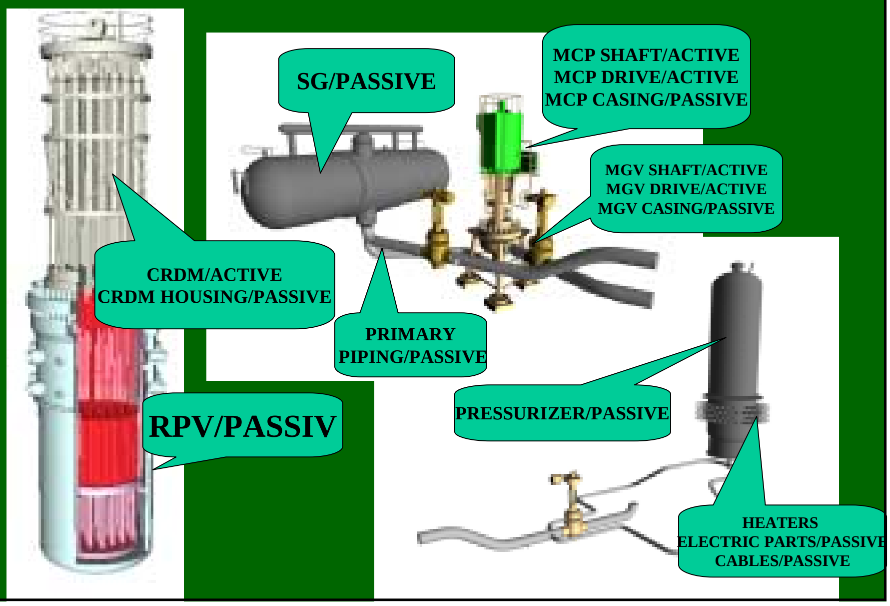
NEW FEATURES:

- Adopting the ASME XI code requirements
- Inspection qualifications
- Changing the inspection interval (4⇔8 years)
- Introducing risk based approaches (not accepted yet)

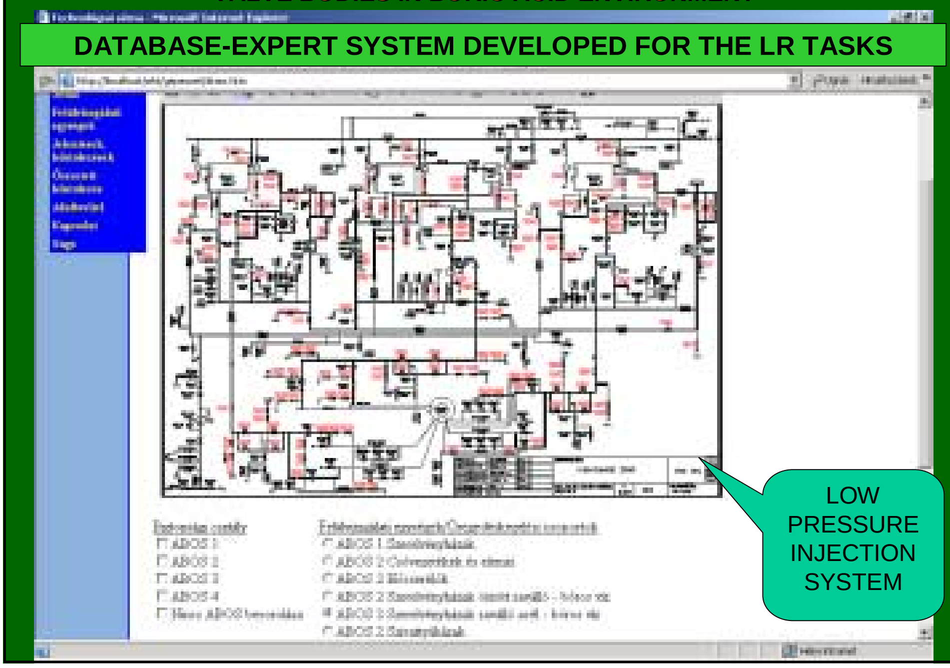

SAFETY CLASS 1, 2, 3, FROM NON SAFETY THOSE, AGEING OF WHICH MAY INHIBIT INTENDED SAFETY FUNCTION, SEISMIC CLASS 1 AND 2
Flow-chart of LR
PASSIVE?
Y
LONG LIVED COULD BE REPLACED?
N
N
Long Lived requiring EQ?
Y
Y
Assess validity of EQ for extended plant life in the scope of LR
N
Ageing management required?
Y
RECENT AGEING MANAGEMENT PRACTICE ADEQUATE?
N
N
Y
MODIFICATION OF EXISTING OR NEW PROGRAMMES
Not included into Ageing Management Assessment (IPA) related to the LR
CLB requirements and conditions shall be fulfilled (incl. PSR reqs.)
Proper safety and performance level shall be ensured in the frame of normal Inservice Inspection/Testing/ Surveillance/ Maintenance and Reconstruction Practice, i.e. normal plant life management programme
Regulatory control via Maintenance Rule and PSR
AGEING MANAGEMENT OK!
IN THE SCOPE OF LICENSE RENEWAL


SCOPE OF THE FSAR, PSR, LR and MR
SC-1
SC-2
SC-3
SC-4T,4
TL EQ
HARSH
EQ
(EQ)
ACTIVE
PASSIVE
LONG
LIVED
LR
PSR
MR
UPDATED FSAR

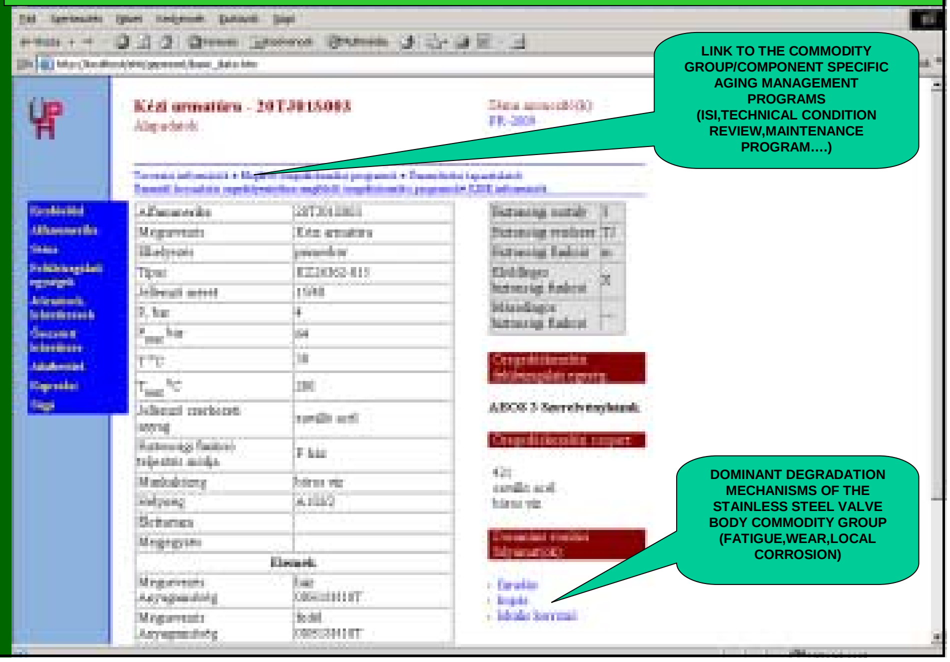
TYPICAL MAIN PASSIVE/ACTIVE VVER 440/213 COMPONENTS



EXAMPLE OF AGING MANAGEMENT COMMODITY GROUPS:STAINLESS STEEL VALVE BODIES IN BORIC ACID ENVIRONMENT



EXAMPLE FROM THE DATABASE-EXPERT SYSTEM DEVELOPED FOR THE LR TASKS: LR AMP REVIEW BASIC DATA OF A SC3 VALVE BODY



LINK TO THE COMMODITY GROUP/COMPONENT SPECIFIC AGING MANAGEMENT PROGRAMS (ISI,TECHNICAL CONDITION REVIEW,MAINTENANCE PROGRAM….)

DOMINANT DEGRADATION MECHANISMS OF THE STAINLESS STEEL VALVE BODY COMMODITY GROUP (FATIGUE,WEAR,LOCAL CORROSION)

MAJOR TIME LIMITED AGEING ASSESSMENTS

FATIGUE CALCULATIONS:

- PARTIAL RE-DESIGN IS NEEDED FOR THE MAIN SAFETY RELATED COMPONENTS
- ASME III METHODOLOGY
- RUSSIAN MATERIAL DATA

PTS CALCULATIONS:

- RE-DESIGN IS NEEDED
- METHODOLOGY IS BASED ON THE INTERNATIONAL PRACTICE (IAEA,VERLIFE,ASME,PNAE)
- POSTULATED DEFECT INITIATION/ARREST,WPS,RESIDUAL STRESSES,CLADDING
- EOL FLUENS CALCULATIONS TO BE VALIDATED BY DOSIMETRY (SURVEILLANCE,CAVITY)

TIME LIMITED AGEING ASSESSMENTS (FATIQUE)

- Review of the load cycles and the required number of cycles assuming 60 (50+10) year operating time.
- Determine the “re-design“ load curves, which represent uniform conservatism for fatigue calculations of equipment with safety class 1-2
- Perform fatigue analysis (+in limited scope redesign) of the equipment with safety class 1-2 for 60-year operating time.
- Review of the current cycle monitoring activities, the monitored CUF affecting parameters, the required additional measurements relating to the completeness of monitoring

IAEA SALTO PWR EBP
SECOND MEETING OF THE WORKING GROUP 2 ON
MECHANICAL COMPONENTS AND MATERIALS,
IAEA HEADQUARTERS, VIENNA , 16-18 NOVEMBER 2004

COUNTRY INFORMATION REPORT ON LONG TERM OPERATION OF MECHANICAL COMPONENTS AND MATERIALS AT UKRAINIAN NPPs

Sergiy Kostenko
Office of Safety Assessment,
State Nuclear Regulatory Committee of Ukraine

NATIONAL STRATEGY RELATING TO LTO OF PWRs

At the present time operation of the most of NPP Units has been exceed half of design term which will be exhausted in 2025 for all Units with the exception of Rovno-4 and Khmelnitski-2.

Therefore taking in account presence of deficiency of alternative kinds of fuel, capacity substitution problem in Ukraine will be resolved by two directions:
- new units commissioning ;
- life time extension of operating units.

Ukrainian NPP lifetime extension over design term operation is the part of " Ukrainian Energy Strategy up to 2030 and further perspective ".

2

REGULATION BASIS FOR LTO OF PWRs MECANICAL COMPONETS

Principal opportunity of Utility to perform LTO works are foreseen in par. 3.18 RD 306.1.02./1.034-2000 "General Provisions of safety assurance of NPPs":
An Utility within the terms stated by governmental nuclear and radiation safety regulatory agency but not less than once in 10 years shall re-assess nuclear power units safety and report to government nuclear and radiation safety regulatory agency. The decision on LTO is decided on results of nuclear power unit safety re-assessment.

The margins and conditions of significant for safety SSCs over design term operation (including mechanical components) are defined in PL-D.0.08.126-04 "Provision of procedure for Lifetime extension of significant safety systems".

This Provision installs common requirements to order and matter of works on LTO of SSC related to 2 and 3 safety class (in accordance with RD 306.1.02./1.034-2000) or B and C group (in accordance with PNAE G-7-008-89).

3

PROCEDURES FOR LTO OF MECHANICAL COMPONENTS AND MATERIALS (MCM)

Procedures for LTO consist works as follows:
- MCM technical status definition;
- renewal of technical conditions of MCM.

Procedure for LTO of active mechanical components (except moving part of valves) performs during regular main repair which precedes to finishing of design term of operation and LTO is drawn up by Technical Decision (TD) with grounded documents as follows:
- statement of deficiencies;
- list of performed works;
- acceptance report of repair works;
- trail report that verify design data;
- conclusion about technical capability to operate over-design term.

Procedure for LTO of internals of collapsible valves based on results of Main Repairs (MR), Routine Repairs, (RR) or Maintenance according to belonging to safety significant group. If MR, RR or Maintenance terms do not install in manufactory documentation their periodically as follows:

4

PROCEDURES FOR LTO OF MECHANICAL COMPONENTS AND MATERIALS (MCM)

	Valves «B» group		Valves «C» group	
	Du < 15	Du > 15	Du < 32	Du > 32
Maintenance	once per year	once per year	once per year	once per year
RR	-	-	-	once per 4 years
MR	-	once per 4 years	-	once per 8 years

5

PROCEDURES FOR LTO OF MECHANICAL COMPONENTS AND MATERIALS (MCM)

Procedure for LTO of internals of collapsible valves "B" group (Du > 15) and "C" group (Du > 32) based on MR of each specific valve unit.

It is acceptably to perform procedure for LTO of moving part of collapsible valves "B" group (Du < 15) and "C" group (Du < 32) based on RR results and reliability analysis.

Procedure for LTO of internals of not collapsible valves "B" group Maintenance performs once per year, and MR once per 10 years taking into account the following:
- for valves not exhausted of their Life time by switching cycles necessity to perform TR is depended their technical conditions which defined by "Standard program on Life time extension of moving part of not collapsible valves 05.09.81.03.03";
- valves exhausted of their Life time by switching cycles are subject for MR performing or replacement.

Regulations for MR performing of not collapsible valves "C" group is not set on the assumption of execution of annual Maintenance and package plan in accordance with "Standard program on Life time extension of moving part of not collapsible valves 05.09.81.03.03". Life time extension of internals of not collapsible valves is being performed in accordance with "Standard program on Life time extension of internals of not collapsible valves 05.09.81.03.03".

6

STANDARD AGEING MANAGMENT PROGRAM

NNEGC "Energoatom" has developed the first edition of the document "Standard ageing management programme of NPP components". The document AMP is the basic branch regulating technical-organizational document about the long-term operation of NPP power-units.

The objects of the AMP Document are as followed:
- certainment of general requirements to organizing and ordering of application of ageing management system of NPP components;
- definition of volume and consecution of taking technical measures to provide a systematic and effective ageing management of NPP components.

At the present time the Document AMP was handed over to IAEA experts for analyzing and will be discussed on the work-meeting in the frame of the project UKR/4/013 "Action plan to issues of operation term management" in July 2004.

After elimination of reviews of IAEA experts the Document AMP will be sent to the Regulating authority for approval, and then to NPP for application.

7

AGEING MANAGMENT OF ACTIVE COMPONENTS

AMPs for separate kinds of MCM will be developed on base of "Standard ageing management programme of NPP components" after its commission.Now ageing management of all kinds of MCM is being realized in frame Maintenance, MR, RR performing except internals of not collapsible valves for which ageing management is being realized in accordance with "Standard program on Life time extension of internals of not collapsible valves 05.09.81.03.03".

The main provision of "Standard program on Life time extension of internals of not collapsible valves 05.09.81.03.03" relevant to ageing management are following.For valves survived in cycles till overhaul life or for the ones with revealed mismatches between technical conditions and design documentation the major repair or replacement is being performed.

8

AGEING MANAGMENT OF ACTIVE COMPONENTS

While performing general overhaul the corresponding parameters of technical conditions for valve internals are being repaired. The valve in cycle operation before overhaul life with no any claims with respect to their technical condition are being distributed into "observation groups". It is recommended to have in the same observation group the valves:

- of the same type, the same model, but different conventional diameter;
- of the same type, different models, different conventional diameters, but the same manufacturer;
- of the same type, different models but similar design, the same conventional diameters or near the same dimensions, but different manufacturers.

9

PROCEDURES FOR LTO OF MECHANICAL COMPONENTS AND MATERIALS (MCM)

For evaluation of technical conditions of valve internals the control sample is being selected.
The criteria for control sample selection is a follows: the longest cyclic actual operation life with account of the maximal parameters of working medium (pressure, temperature, pressure drop at shutter; presence of two phase working mediums; velocity of heating up and cooling down).

The control sample is being dismantled from pipe and technical observation of valve internals is being performed by maintenance subdivision or manufacturer.

For satisfactory technical condition of control sample selected the technical state of all valves internals is being considered as satisfactory for all valves units of the same observation group.

For non satisfactory technical conditions of control sample selected the observation group is divided into two subgroups based on actual cycle life time. In a subgroup (assigned as No. 2) with less operation life time another control sample is being selected and it's technical condition assessment is performed as it was described above.

10

PROCEDURES FOR LTO OF MECHANICAL COMPONENTS AND MATERIALS (MCM)

In case of satisfactory technical condition of control sample selected from subgroup No. 2, the technical state of valve internals within this subgroup is considered as satisfactory as well and then subgroup No. 1 must be assessed with respect to further technical evaluation with control sample.

In case of non satisfactory technical condition of control sample selected from subgroup No. 2 the dismantling and technical evaluation of all valve units in subgroup No. 1 is being performed by maintenance division. Then subgroup No. 2 is divided into two parts and so on.

To generalize and optimize the efforts under the "Program .." the observation and technical assessment results for valve internals of dismantled items of one-piece valves may be used as justification for life time extension of similar valves for other NPP units.

11

AGEING MANAGMENT OF PASSIVE COMPONENTS

General regulatory requirements to ISI are set in "Regulation for structure and safety operation of NPP equipment and pipelines. PNAE G-7-008-89".
In accordance with this document ISI is being realized by destructive and non-destructive methods.
It is exist as general document for non-destructive testing realisation and results estimation (PNAE G-7-008-89) and separate documents for each method of inspection except eddy current testing.
Non-destructive testing:
- **Visual and instrumental;**
- **Radiographic;**
- **Ultrasonic;**
- **Magnetic;**
- **Eddy current;**
- **Penetration;**
- **Eddy current;**
- **Pneumatic- hydraulic;**
- **Impenetrability**

12

AGEING MANAGMENT OF PASSIVE COMPONENTS

Methods, scope, periodicity and results estimation of ISI are being defined in "Standard program of ISI base metal, welds and cladding for NPP equipment and pipelines with reactor WWER" (AIEU-9-04 for WWER-1000 and AIEU-10-03 for WWER-440) which is a base for working program of each units.

ISI by destructive methods execute:
-for equipment – by means of specimens surveillance testing which installed in this equipment;
-for pipelines – by means of pieces which cut from pipelines.

In PNAE G-7-010-89 it is foreseen to perform inspection of mechanical properties modification of "A" and "B" groups equipment and pipelines every 100000 hours of operation. For this kind of inspection it was developed. "Standard program of periodical inspection of equipment and pipelines metal. TPKM-10-01" and respective working program for each unit.

13

AGEING MANAGMENT OF PASSIVE COMPONENTS

Mechanical properties are being determined by the following methods:
- **kinetic hardness measurement with re-count it to mechanical properties;**
- **specimens surveillance testing installed in equipment;**
- **testing of pieces cut out of earlier replaced pipelines;**
- **testing of micro-pieces cut out of equipment and pipelines surface in according with "Procedure of NPP equipment and pipelines mechanical properties determination by means of micro-pieces. RD 00. EK.HF.VO.M.09".**

Utility NAEK "Energoatom" does not have Standard ISI program on erosion-corrosion wear therefor monitoring and diagnostic system of erosion-corrosion wear of equipment and pipelines is being organized at every NPP.

In a general way it may be describe as follows:
- **taking in account experience of passive components operation for each unit it is defined zones for mandatory inspection of thickness walls of equipment and pipelines;**
- **on thickness measurement results and comparison with allowable its value it is produced conclusion concerning capacity of passive component for work up to following inspection.**

14

STEAM GENERATOR HEAT EXCHANG TUBES AGEING MANAGEMENT

Since SG operation start it was found their design lacks, especially it is concern to PGV-1000 type SG installed at WWER-1000 units and not only in Ukraine but also in other state.

In ninetieth years several SGs were replaced because of SG collector ligament cracking at South Ukraine and Zaporozhye NPPs.
At the beginning of this century it was arisen problem of mass corrosion damage of SG heat exchange tubes (HET) and their in-service inspection by eddy-current testing (ECT).

The main goal of SG HET eddy-current testing is defects detecting on early stage of their appearance and further development of monitoring that enable to forecast life time of HET and SG on the whole. Moreover ECT on a par with pneumatic-hydraulic aquarium method (PHA) permit to perform a rejection and plugging of HET.

15

STEAM GENERATOR HEAT EXCHANG TUBES AGEING MANAGEMENT

In 1999 Regulatory Body to put in claims to Utility as follows:

- **to realize 100% ECT of HET all SG up to 2006 year;**
- **to fix periodicity so-called critical zone (zone with the biggest slag pollution located between 85-110 rows of right half of SG collector) - once per 4 year ;**
- **to perform annual ECT of all not plugged HET with thickness walls more than 20%;**
- **optimal scopes, periodicity of IS, HET plugging criteria set after realizing 100% ECT of HET all SG .**

In process of 100% ECT of HET performing it were accepted temporary plugging criteria of HET as following which for different units vary from 60 to 85 %.
In frame of SG Ageing Management it was set up SG data base, realized arrangements on reconstruction and modernization feed and blow systems, chemistry coolant monitoring, chemistry cleaning technology.
It is being planed to change ECT periodicity, scopes for different zones of SG HET, make more precise plugging criteria, critical zones for inspection. Such works have been already started by SNRSU initiative.

16

AGEING MANAGEMENT OF WWER-1000 TYPE REACTOR PRESURE VESSEL

For RPV integrity providing it is being performed destructive (specimens surveillance testing) and non-destructive (In-Service) inspection.

The goal of specimens surveillance (SS) testing is confirmation RPV metal resistance to embrittlement destruction.

Two SS Program are being realized in Ukraine at the same time: regular (SS are in cylindrical container that not provide their uniform irradiation) and modernized (SS are in flat container that not provide their more uniform irradiation).

For SS Program performing the following data is being used:

- Properties of base metal and welds are located opposite of active zone:
- Content of Cu, P, Ni, Mn;
- Value of initial brittle-toughness temperature transition Tко;
- Physics-mechanical properties;
- Design calculated value of marginal brittle-toughness temperature transition $T\kappa^a$

17

AGEING MANAGEMENT OF WWER-1000 TYPE REACTOR PRESURE VESSEL

- Results of SS testing;
  - data of SS loading and modification in regular SS Program;
  - information on SS Program modernization, SS loading-unloading terms;
  - SS testing results and their estimation for RPV LTO;
  - data of SS for next unloading.
- Results of irradiation loading monitoring:
  - maximum value of neutron fluence in RPV critical zones;
  - value of neutron fluenc in SS set into container.
- Operational loads on RPV:
  - RPV mode of operation list;
  - information about emergency mode connected with RPV cooling.

In-Service Inspection.

The goal of ISI is detecting and fixing of RPV base metal and welds. Methods, scope, periodicity and results estimation of ISI are defined in "Standard program of ISI base metal, welds and cladding for NPP equipment and pipelines with reactor WWER. ISI data are being used for brittle-toughness temperature transition calculations.

18

AGEING MANAGEMENT OF WWER-1000 TYPE REACTOR PRESURE VESSEL

Additional measures directed on RPV integrity providing:

- **water heating in active part of SAOZ and in TANK;**
- **RPV irradiation loading by forming fuel charging with neutron slight drain.**
- **RPV safety substantiation for design term operation.**

The recalculation of a RPV residual lifetime is being performing for each unit with allowance of actual condition of reactor block systems and initial performances of materials, embrittlement data and new normative requests. By results of recalculation the list of necessary measures on maintenance of a design lifetime is being corrected. The methodology of the substantiation of a RPV integrity now is being under developing. RPV metal resistance to brittle-toughness destruction in critical zones owing to emergency operating mode has fixed in basis of this methodology.

19

U. S. CIR
Sections 1.0 & 2.0

T.T. Taylor
Pacific Northwest National Laboratory
SALTO WG2 Meeting
November 16-18, 2004

Battelle

Pacific Northwest National Laboratory
Operated by Battelle for the U.S. Department of Energy

Presentation Contents

- Background Information on Regulation Development
- Section 1.0 Summary
- Section 2.0 Summary

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Background Information

- Atomic Energy Act of 1954
  - 40-year license to operate
  - Allows for renewal
- License Renewal Rule allows new license to be issued to operate for up to 20 years beyond the current term
- Application submittal requirements
  - Not earlier than 20 years before expiration of current license
  - Not later than 5 years before expiration of current license for timely renewal provisions

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Background Information

- In 1982, the U.S. NRC held a workshop on nuclear power plant aging, in anticipation of license renewal.
  - Workshop resulted in comprehensive plan for Nuclear Plant Aging Research (NPAR). Based on that research, a technical review group concluded that many aging phenomena are readily manageable and do not pose technical issues that would preclude life extension for nuclear power plants.
  - Work under NPAR provided basis for GALL Report
- In 1986, the NRC published a request for comment on a policy statement that would address major policy, technical, and procedural issues related to life extension.

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Section 1.0
Applicable Laws Specific to Materials and Components for Long Term Operation

- The NRC published the license renewal rule in the *U.S. Code of Federal Regulations* as 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" in 1991 – This is the law that governs license renewal
  - **Next, NRC sponsored a demonstration program to apply the rule to pilot plants and develop experience to establish implementation guidance for license renewal.**
    - Initially scope of review for license renewal, defined age-related degradation unique to license renewal.
    - The demonstration program, showed NRC that many aging mechanisms occur and are managed during the period of the initial license.
    - NRC also found that the scope of the review did not allow sufficient credit for existing programs, particularly the implementation of the maintenance rule, which also manages plant aging phenomena.

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Section 1.0
Applicable Laws Specific to Materials and Components for Long Term Operation

- As a result, in 1995, NRC amended the license renewal rule to ***focus on managing the adverse effects of aging rather than on the identification of all aging mechanisms***.
  - Also, the integrated plant assessment (IPA) process was clarified and simplified to be consistent with the revised focus on passive, long-lived structures and components.

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Background Information for Section 2.0

Significant Assumptions in U.S. LTO Regulatory Requirements

- Existing regulatory process is adequate for ensuring safety of operating plants
- Current licensing basis (CLB) is adequate and carries forward into the period of extended operation
- Issues relevant to the current operation of plants will be addressed by the regulatory process, which will carry forward into the period of extended operation.
- Plant-specific licensing basis must be maintained during the renewal term in the same manner and to the same extent as during the original licensing term.

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Section 1.0 Regulatory Requirements Applicable to Long-Term Operation

- Currently, Federal Code of Regulations 10CFR50.24 describes the Regulatory Requirements for LTO and applies to:
  - Safety-related systems, structures and components relied upon to:
    - Maintain integrity of the reactor coolant pressure boundary
    - Ensure capability to shut down and maintain a safe shutdown condition
    - Prevent or mitigate offsite exposures comparable to 10 CFR part 100 offsite dose analyses for siting
  - Nonsafety-related systems, structures and components whose failure could prevent safety-related function as outlined above

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Section 2.1
Regulatory Requirements Applicable to Long-Term Operation

- 10CFR54.21 Requires an Integrated Plant Assessment (IPA)
  - Identify and list structures and components subject to an aging management review (AMR)
  - Describe/justify methods to identify structures and components subject to an AMR from those systems, structures, and components within the scope of the rule
  - Demonstrate effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Section 2.1
Regulatory Requirements Applicable to Long-Term Operation

- Basis for Review for Compliance with 10CFR50.24 is the Standard Review Plan for License Renewal - (SRP-LR)
- Includes guidance for the following review sections
  - Areas of Review
  - Acceptance Criteria
  - Review Procedures
  - Evaluation Findings
  - Implementation
  - References *GALL Report* for generic aging evaluation as Technical Basis document

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Section 2.1 Regulatory Requirements Applicable to Long-Term Operation

- Sections Gall Report Appropriate for WG2
  - Chapter I Application of ASME Code
  - Chapter III Structures and Component Supports
  - Chapter IV Reactor Vessel, Internals, and Reactor Coolant System
  - Chapter V Engineered Safety Features
  - Chapter VII Auxiliary Systems
  - Chapter VIII Steam and Power Conversion System
  - Chapter X Time-Limited Aging Analyses
  - Chapter XI Aging Management Programs

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Section 2.1 Regulatory Requirements Applicable to Long-Term Operation

- 10CFR54.21 also Requires an Evaluation of all Time Limited Aging Analysis as follows:
- Definition of Time-limited Aging Analyses (TLAAs)
  - Involve systems, structures and components within the scope of the rule (not limited to passive and long-lived),
  - Consider the effects of aging,
  - Involve time-limited assumptions defined in the current operating term,
  - Determined by licensee to be relevant in safety determination,
  - Involve conclusions related to performance of intended functions, and
  - Contained or incorporated by reference in CLB

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Section 2.1
Regulatory Requirements Applicable to Long-Term Operation

- For TLAAs10CFR54.21 Requires an Evaluation as follows:
  - Applicant must provide a list of time-limited aging analyses and shall demonstrate that:
    - The analyses remain valid for the period of extended operation;
    - The analyses have been projected to the end of the period of extended operation; or
    - The effects of aging on the intended function(s) will be adequately managed for the period of extended operation

Battelle

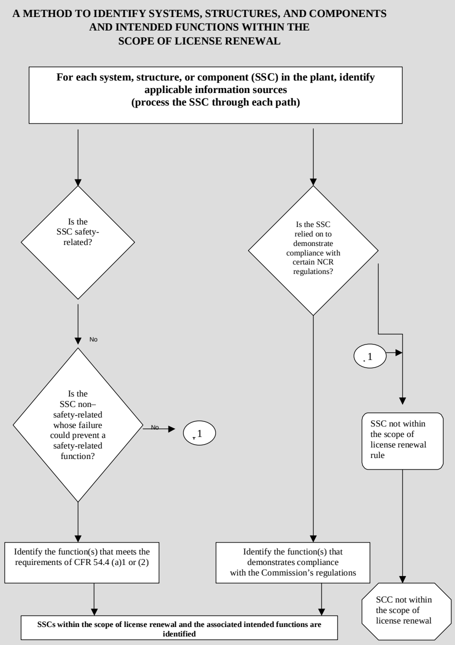
Pacific Northwest National Laboratory
U.S. Department of Energy

Section 2.1
Regulatory Requirements Applicable to Long-Term Operation

- Typical TLAAs include:
  - Reactor vessel neutron embrittlement
  - Concrete containment tendon prestress
  - Metal fatigue
  - Environmental qualification of electrical equipment
  - Metal corrosion allowance
  - In-service flaw growth analyses that demonstrate structure stability for 40 years
  - In-service local metal containment corrosion analyses
  - High-energy line-break postulation based on fatigue cumulative usage factor

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy



Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Section 2.2
Configuration Control Practices Used To Control Design Bases

- As defined in 10 CFR 50.2, *design bases* means that information which identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design.
  - These values may be:
    - 1) restraints derived from generally accepted "state-of-the-art" practices for achieving functional goals or
    - 2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Section 2.2
Configuration Control Practices Used To Control Design Bases

- In the United States, control of the design bases and design bases supporting documentation are based upon two legal requirements.
  - First, 10 CFR 50.2 design bases and supporting design information are subject to design control and
  - Other requirements of 10 CFR Part 50 Appendix B, as applicable according to the safety classification of particular SSCs.
- ***Since Design Basis is part of the Current Licensing Basis – for LTO it must be demonstrated the Design Basis is maintained during the period of LTO.***

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Section 2.3
Aging Management Programs

- See Presentation on Section 3.0

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

U. S. CIR
Section 3.0

T.T. Taylor
Pacific Northwest National Laboratory
SALTO WG2 Meeting
November 16-18, 2004

Battelle

Pacific Northwest National Laboratory
Operated by Battelle for the U.S. Department of Energy

Presentation Contents

- **In-service Inspection Practices for passive Components**
  - Augmented inspection programs that address issues such as erosion/corrosion,
  - Augmented inspection of steam generator tubing or
  - Augmented inspection for specific degradation mechanisms such as Intergranular stress corrosion cracking
- **Maintenance Codes or Practices for Active Components**
- **Equipment Qualification Practices**
- **Component function tests**
- **Applied diagnostic systems**
- **Surveillance Program (irradiation damage, corrosion loops)**
- **Nondestructive material properties tests (hardness measurement etc.)**
- **Destructive tests and material research carried out during NPP operation**
- **Special loading measurement systems (temperature, deformation etc.) combined with damage calculation (e.g. on-line and off-line fatigue monitoring)**
- **Chemical regimes monitoring**

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

In-service Inspection Practices for Passive Components

- **XI.M1 ASME Section XI In-service Inspection, Subsections IWB, IWC, and IWD**
  - Details the in-service inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, for Class 1, 2, and 3 pressure-retaining components and their integral attachments in light-water–cooled power plants

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

In-service Inspection Practices for Passive Components

- **Specific ISI Programs High lighted In GALL Report**
  - XI.M3 Reactor Head Closure Studs
  - XI.M4 BWR Vessel Inside Diameter Attachment Welds
  - XI.M5 BWR Feedwater Nozzle
  - XI.M6 BWR Control Rod Drive Return Line Nozzle
  - XI.M8 BWR Penetrations
  - XI.M9 BWR Vessel Internals
  - XI.M10 Boric Acid Corrosion
  - XI.M12 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel
  - XI.M13 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

In-service Inspection Practices for Passive Components

- **Specific ISI Programs High lighted In GALL Report**
  - XI.M14 Loose Part Monitoring
  - XI.M15 Neutron Noise Monitoring
  - XI.M16 PWR Vessel Internals
  - XI.M18 Bolting Integrity
  - XI.M25 BWR Reactor Water Cleanup System
  - XI.M26 Fire Protection
  - XI.M28 Buried Piping and Tanks Surveillance
  - XI.M29 Aboveground Carbon Steel Tanks
  - XI.M30 Fuel Oil Chemistry

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

In-service Inspection Practices for Passive Components

- Augmented inspection programs that address issues such as erosion/corrosion
  - XI.M17 Flow-Accelerated Corrosion
  - XI.M20 Open-Cycle Cooling Water System – MIC
  - XI.M21 Closed-Cycle Cooling Water System
- Augmented inspection of steam generator tubing
  - XI.M19 Steam Generator Tube Integrity
- Augmented inspection for specific degradation mechanisms such as Intergranular stress corrosion cracking
  - XI.M7 BWR Stress Corrosion Cracking

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

In-service Inspection Practices for Passive Components

- Augmented inspection for specific degradation mechanisms such as Intergranular stress corrosion cracking
  - XI.M7 BWR Stress Corrosion Cracking
  - XI.M11 Nickel-Alloy Nozzles and Penetrations
  - XI.M32 One-Time Inspection
  - XI.M34 Buried Piping and Tanks Inspection

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Maintenance Codes or Practices for Active Components

- The maintenance of active components in nuclear power plants in the United States is regulated by the requirements in 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants.”

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Equipment Qualification Practices

- Seismic Qualification
- XI.M23 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems
  - Normally, fewer than 10 cranes fall within the scope of 10 CFR 54.4. These cranes comply with the Maintenance Rule requirements provided in 10 CFR 50.65. The NRC Regulatory Guide 1.160 provides guidance for monitoring the effectiveness of maintenance at nuclear power plants

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Component Function Tests

- XI.M27 Fire Water System
  - Program applies to water-based fire protection systems that consist of sprinklers, nozzles, fittings, valves, hydrants, hose stations, standpipes, water storage tanks, and aboveground and underground piping and components that are tested in accordance with the applicable National Fire Protection Association (NFPA) codes and standards.
- Plants have Technical Specifications that require functional Tests of components and Systems
  - Example, monthly valve tests for flow
  - Monthly pump tests for operation

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Applied Diagnostic Systems

- XI.M22 Boraflex Monitoring
  - A monitoring program for the actual Boraflex panels is implemented in the spent fuel racks to ensure that no unexpected degradation of the Boraflex material would compromise the criticality analysis in support of the design of spent fuel storage racks.

Surveillance Specimen Program (irradiation damage, corrosion loops)

- XI.M31 Reactor Vessel Surveillance
  - 10 CFR Part 50, Appendix H, requires that peak neutron fluence at the end of the design life of the vessel not exceed 10^{17} n/cm^2 (E >1MeV), or that reactor vessel beltline materials be monitored by a surveillance program to meet the ASTM E 185 standard

Nondestructive Material Properties Tests (hardness measurement etc.)

- NDE Material Property Tests are done as part of the “One Time Inspection” aging management program required under LTO (License Renewal)

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Destructive Tests and Material Research carried out during NPP operation

- The U.S. requires surveillance capsules to be monitored during reactor operation
- U.S. NRC has sponsored several programs on Material Issues
  - Pressurized Thermal Shock
  - Fabrication Flaw Data Base
  - Steam Generator Reliability Program
  - Material Degradation research at PNNL and Argonne
  - Heavy Vessel Program at Oak Ridge National Laboratory
- EPRI has sponsored several research programs on materials to address issues during operation
  - Past Program Examples
    - Lead Industry research on IGSCC for BWRs
    - Lead research in E/C – CHECKMATE
  - Current Programs
    - BWR VIP (Vessel Internals Program)

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Special Loading Measurement Systems

(temperature, deformation etc.) combined with damage calculation (e.g. on-line and off-line fatigue monitoring)

- XI.M24 Compressed Air Monitoring
  - The program consists of inspection, monitoring, and testing of the entire system:
    - 1) frequent leak testing of valves, piping, and other system components, especially those made of carbon steel;
    - 2) preventive monitoring that checks air quality at various locations in the system to ensure that oil, water, rust, dirt, and other contaminants are kept within the specified limits.
- Fatigue Calculations and Monitoring
  - Fatigue analysis for the main steam supply lines to the turbine-driven auxiliary feedwater pumps
  - Fatigue analysis of the reactor coolant pump flywheel
  - Fatigue analysis of polar crane

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Chemical Regimes Monitoring

- XI.M2 Water Chemistry
  - The main objective of this program is to mitigate damage caused by corrosion and SCC. The water chemistry program relies on monitoring and control of primary and secondary system water chemistry based on EPRI guidelines
- XI.M33 Selective Leaching of Materials
  - The program for selective leaching of materials ensures the integrity of the components made of cast iron, bronze, brass, and other alloys exposed to a raw water, brackish water, treated water, or groundwater environment that may lead to selective leaching of one of the metal components.
    - The aging management program includes a one-time visual inspection and hardness measurement of selected components susceptible to selective leaching to determine whether loss of materials due to selective leaching is occurring, and whether the process will affect the ability of the components to perform their intended function for the period of extended operation.

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Plant-Specific Safety Analyses Based on an Explicitly Assumed Plant Life or Operating Period

- Time-limited aging analyses (TLAAs) are certain plant-specific safety analyses that are based on an explicitly assumed 40-year plant life
- Typical TLAAs include:
  - Reactor vessel neutron embrittlement
  - Concrete containment tendon prestress
  - Metal fatigue
  - Environmental qualification of electrical equipment
  - Metal corrosion allowance
  - In-service flaw growth analyses that demonstrate structure stability for 40 years
  - In-service local metal containment corrosion analyses
  - High-energy line-break postulation based on fatigue cumulative usage factor

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy

Plant-Specific Safety Analyses Based on an Explicitly Assumed Plant Life or Operating Period

- Typical TLAAs include:
  - Intergranular separation in the heat-affected zone (HAZ) of reactor vessel low-alloy steel under
  - Austenitic SS cladding. Low-temperature overpressure protection (LTOP) analyses
  - Fatigue analysis for the main steam supply lines to the turbine-driven auxiliary feedwater pumps
  - Fatigue analysis of the reactor coolant pump flywheel
  - Fatigue analysis of polar crane
  - Flow-induced vibration endurance limit, transient cycle count assumptions, and ductility reduction of
  - Fracture toughness for the reactor vessel internals
  - Leak before break
  - Fatigue analysis for the containment liner plate
  - Containment penetration pressurization cycles
  - Reactor vessel circumferential weld inspection relief (BWR)

Battelle

Pacific Northwest National Laboratory
U.S. Department of Energy