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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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Unclassified

TECHNICAL ASPECTS OF AGEING FOR LONG-TERM OPERATION

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996), Korea (12th December 1996) and the Slovak Republic (14th December 2000). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of 27 OECD Member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries.

CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus in different projects and International Standard Problems, and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups and organisation of conferences and specialist meeting.

The greater part of CSNI's current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

In implementing its programme, CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

FOREWORD

The CSNI Integrity and Aging (IAGE) Working Group deals with the integrity of structures and components, and has three sub-groups, dealing with the integrity of metal structures and components, the aging of concrete structures, and the seismic behaviour of structures. Ageing is also a primary consideration of the group.

This document restricts itself to the technical aspects of the basis for long-term operation. Its objective is the characterisation of technical key items. Regulatory aspects are being discussed within CNRA. The report referenced NEA/CNRA/R(1999)1 on "Regulatory Aspects of Ageing Reactors" -CNRA Special Issue Meeting June 1998 provides information on this issue.

ACKNOWLEDGEMENT

Gratitude is expressed to Mr. Helmut Schulz with GRS, Germany for writing and updating this report. Thanks are also expressed to all IAGE Working Group members for reviewing and providing information.

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Background

The aspects of plant ageing management gained increasing attention over the last ten years. Numerous technical studies have been performed to study the impact of ageing mechanisms on the safe and reliable operation of nuclear power plants. National research activities have been initiated or are in progress to provide the technical basis for decision making processes.

The long-term operation of nuclear power plants is influenced by economic considerations, the socio-economic environment including public acceptance, developments in research and the regulatory framework, the availability of technical infrastructure to maintain and service the systems, structures and components as well as qualified personnel.

Objective of the work

Besides national activities there are a number of international activities in particular under the umbrella of the IAEA, the OECD and the EU. For example, ageing research plays an important role in the 4th and 5th Framework Programme of the European Commission. A bibliography on major ageing related EU research activities under the 4th and 5th Framework Programme is given. International working groups and networks dealing with ageing related issues are summarised. Relevant guidance documents on plant ageing management, which could be used for orientation, are listed.

This document restricts itself to the technical aspects of the basis for long-term operation. Its objective is the characterisation of technical key items. Regulatory aspects are being discussed within CNRA.

What was done (including key assumptions and limitations, etc.)

The light water reactor (LWR) nuclear power plants in operation are generically of pressurised water reactor (PWR) or boiling water reactor (BWR) type. The design principles of the nuclear steam supply system (NSSS) are proven in long-term operation as well as the principles being applied to the balance of plant (BOP), although a rather large variety of technical solutions do exist. The steps in the developments were tremendous in the 60s and 70s up to a power output of 1200 MWe followed by a period of further optimisation in the 80s and 90s, which raised it close to 1500 MWe.

The NPPs in operation can be grouped into different generations of design in order to characterise similarities in technical principles and manufacturing standards. In the area of civil and mechanical structures and components there has been continuous development to optimise the design, the materials used as well as the manufacturing technology. The majority of these structures and components is still in use. In principle there are no major obstacles to further use of these components and structures even for a longer time of operation. However, it may be difficult to find technical expertise to support certain equipment in the future. In other areas like instrumentation and control (I&C) more drastic changes have taken place in the whole technology. A large number of nuclear power plants have already or will change their equipment to more modern I&C technology.

A plant specific ageing management programme is much more challenging in the cases where licensees are the main source of information (because the original designer and manufacturers are already out of business) in comparison to larger series plants built with a reliable technical infrastructure still active on the market. Exceptions may exist where licensees have developed in-house capability in place of the original designer. The necessary elements for plant ageing management are:

- Knowledge of ageing related damage mechanisms including benchmarking of the consequences of damage mechanisms into macroscopic behaviour of materials and structures under applicable conditions
- Predictive models to extrapolate behaviour of systems, structures or components up to a defined time
- Qualified methods for detection and surveillance of ageing degradation
- Qualified mitigation measures
- Reliable plant documentation
- Availability of technical service and knowledge base.

Those items are detailed in the report

Conclusions and recommendations

The safe and reliable operation of light water reactors has been proven for almost 30 years. It has been demonstrated that even for components with high field of radiation the access for repair or replacement methods is satisfactory. Although some serious damage mechanisms have limited the lifetime of some components, repair or replacement could be performed without excessive consequence on plants availability. In view of the general technical experience the factor of extrapolation with respect to time or usage are a factor of two or less if one considers operating times between 40 and 60 years.

Due to the drastic changes in industry, influenced by the declining demand for new nuclear power plants, it is to be expected that in most countries the industrial infrastructure will be reduced to match the requested level of service. In many technical areas continuous development will take place driven by other industrial developments than nuclear. This is certainly to be expected for instrumentation and control but also in the areas of civil engineering, material production and welding technology along with surveillance, testing and inspection technologies. However, there has been a tradition for nuclear industry to be the driving force for engineering development and it is not certain that non-nuclear industries will fully replace this driving force. For non-nuclear industry the level of in-depth analyses and safety demonstration is less demanding up to now.

To provide a technical basis for long-term operation of nuclear power plants it is necessary to:

- invest in research methods and strategies related to plant life management
- promote computer-aided modelling to predict degradation at a quantitative level
- update the individual plant documentation to avoid gaps in knowledge caused by the reorientation of industry and by the retirement of experienced people
- initiate, develop and promote clubs of users of similar technology internationally

- establish a system of information retrieval to bridge gaps between today's and previous design and manufacturing standards

- increase the flexibility of the quality assurance system to qualify products manufactured to other standards for plant specific use.

TECHNICAL ASPECTS OF AGEING FOR LONG-TERM OPERATION

1. INTRODUCTION

The aspects of plant ageing management gained increasing attention over the last ten years. Numerous technical studies have been performed to study the impact of ageing mechanisms on the safe and reliable operation of nuclear power plants. National research activities have been initiated or are in progress to provide the technical basis for decision making processes.

Besides national activities there are a number of international activities in particular under the umbrella of the IAEA, the OECD and the EU. For example, ageing research plays an important role in the 4th and 5th Framework Programme of the European Commission. **Appendix 1** contains a bibliography on major ageing related EU research activities under the 4th and 5th Framework Programme. International working groups and networks dealing with ageing related issues are summarised in **appendix 2**. Relevant guidance documents on plant ageing management, which could be used for orientation, are listed in **appendix 3**.

The long-term operation of nuclear power plants is influenced by economic considerations, the socio-economic environment including public acceptance, developments in research and the regulatory framework, the availability of technical infrastructure to maintain and service the systems, structures and components as well as qualified personnel.

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2. STATUS OF LWR NUCLEAR POWER PLANTS IN OPERATION

The light water reactor (LWR) nuclear power plants in operation are generically of pressurised water reactor (PWR) or boiling water reactor (BWR) type. The design principles of the nuclear steam supply system (NSSS) are proven in long-term operation as well as the principles being applied to the balance of plant (BOP), although a rather large variety of technical solutions do exist. The steps in the developments were tremendous in the 60s and 70s up to a power output of 1200 MWe followed by a period of further optimisation in the 80s and 90s, which raised it close to 1500 MWe.

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- Qualified mitigation measures
- Reliable plant documentation
- Availability of technical service and knowledge base.

2.1. Knowledge of ageing related damage mechanisms

This issue is discussed with respect to the safety-related barriers and the internals of the pressure boundary. Relevant ageing mechanisms for passive metallic components of LWRs and their predictability are summarised in **table 1**. The specified extrapolation factor represents the ratio of the total value of the scheduled operating time or use of a component to the existing knowledge about the corresponding degradation mechanism based on the elapsed time of operation or use.

Degradation factors that can impact the performance of safety-related concrete structures are compiled in /NEA 98/. Also, ageing of organic components such as cables and its management is discussed in a separate OECD paper /NEA 99/. The management of ageing of active components such as motor operated isolating valves is considered as a maintenance issue.

- Fuel cladding

The fuel elements are exchanged after limited time. Although the usage of the fuel is being optimised to achieve very high burn-ups the damage mechanisms acting on the fuel cladding are known in principle and have no direct impact on the plant life.

- Pressure boundary

Acting damage mechanisms are well known for the pressure boundary. Based on the available experience extrapolation to operating times is about a factor of 2.

With respect to the „usage factor„ fatigue criteria extrapolation factor is less than 2. For wet conditions more accurate calculation models are needed.

Regarding irradiation embrittlement major portion of material damage occurs in the early stage of operating life. Relating to the corrosion behaviour there is still a lack of predictability of the likelihood of different mechanisms and their kinetics.

Limited knowledge is available regarding the long-term behaviour of certain component support structures because of limitations in the access for routine inspection.

- Internals of the pressure boundary

Acting damage mechanisms are well known for the internals of the pressure boundary. Based on the available experience extrapolation to operating times is about a factor of 2.

Based on operating experience the internals of the pressure boundary may see more or less severe damages over the operating period. PWR-internals have seen a number of failures of bolted connections. For BWR-internals cracking of structures by intergranular stress corrosion cracking mechanisms have been observed.

- Containment

The containment structures see basically only very small loads during regular service. Depending on the regulatory requirements the highest loads arise during testing conditions. Damage mechanisms are well known. In steel containments and steel liners of concrete containments limited local corrosion attack has been observed. Regarding concrete containments the long-term behaviour of pre-stressed concrete containments is a matter of concern.

- Balance of plant

The damage mechanisms acting on systems, structures or components of „balance of plant„ are well known. A considerable fraction of components are or will be replaced during the lifetime of the plant. With respect to operating time or usage factor the extrapolation to future operation is less than a factor of 2.

2.2 Predictive models to extrapolate behaviour of systems, structures or components up to a defined time

Most of the available predictive models for the different damage mechanisms are semi-empirical models. The restrictions for application are not always very clearly described. Because of the semi-empirical character transferability to other materials conditions or usage is difficult. Within the ongoing and planned research programmes considerable effort is directed to improve predictive models from the semi-empirical state to physically based models.

2.3 Qualified methods for detection and surveillance and control of ageing degradation

Effective methods for detection, surveillance and control of degradation are available and in place for most of the known damage mechanisms. Table 2 summarises the goals and methodologies used for monitoring systems, which are mostly related to the evaluation of residual lifetime of components and systems. Adequate tools are available and are successfully implemented into the overall plant life and ageing management programmes to inspect and monitor stressors relevant to ageing process, to develop new materials and to allow early detection of any materials and components degradations (see table 3). Nevertheless there is a need to verify in an independent way the capability of these techniques against the detection and sizing objectives. Improving these methods towards detection of alteration in physical parameters, which could be used more effectively for predictive calculations, is a continuing challenge.

2.4 Qualified mitigation measures

For the damage mechanisms acting in the systems within the range of operating conditions of nuclear power plants proven methods are available to repair systems, structures or components if they are damaged. For areas where systematic degradation occurs due to design deficiencies, mitigation measures have been developed to delay or even to stop degradation processes, e.g. application of compressive stresses to delay or to reduce crack growth in case of intergranular stress corrosion cracking.

- Pressure boundary

The reactor pressure vessel (RPV) as part of the pressure boundary is seen as the key component because it is very difficult to replace although local damages can be repaired quite easily and loss of ductility by neutron radiation in the belt line region can in principle be recovered almost totally by heat treatment as shown in case of soviet designed VVER-440 RPVs. However, further work is needed to evaluate degrading effects on RPV materials.

Steam generators replacements are performed without a major impact on availability.

- Internals of the pressure boundary

Generally, the main internals of PWR RPV can be completely removed, implying that repair or replacement strategies may be feasible. Repair technologies developed for bolted connections of PWRs have proven to be effective. Regarding core internals of BWRs, repair methods have been developed but further optimisation seems to be necessary. Replacement of larger sections of reactor pressure vessel internals for BWR may be used as an alternative to local repair methods.

- Containment

For local corrosion attack in steel containments and liners proven repair technologies are available. Regarding concrete containment designs, which use ungrouted cables, the cables can be replaced if the loss of prestress may endanger the function of leaktightness. For designs that use grouted cables additional mitigation measures might be necessary to sustain the function of leak tightness.

2.5 Reliable plant documentation

Any plant specific analysis with the intention to identify principal damage mechanisms, which may occur in the systems, depends on the information available on systems, structures and components specific documentation. The documentation of the technical details of plants, which have been built before the 80s may be not sufficient with respect to information required for predictive models today. For this reason methods which are able to measure in-situ the material properties of the components and structures are of considerable advantage.

2.6 Availability of technical service and knowledge base

As already pointed out, enormous steps in development have taken place in the 60s and 70s. During this early period codes and standards used in the design and fabrication of components were not as prescriptive as later on. To characterise the technical expertise applied in the design and fabrication of the components there is a strong dependence on plant documentation. As in many other industries with rapid development it may be difficult to trace back the details of technology being used at the early stage. Although all of this knowledge is to a certain extent available in the technical literature it might still be a challenge to obtain the right interpretation of previous codes and standards without the practitioners being available anymore. For this reason the uniqueness of components and structures may be a critical criteria with respect to availability of technical support.

3. CONCLUSIONS AND RECOMMENDATIONS FOR FUTURE ACTIVITIES

The safe and reliable operation of light water reactors has been proven for almost 30 years. It has been demonstrated that even for components with high field of radiation the access for repair or replacement methods is satisfactory. Although some serious damage mechanisms have limited the lifetime of some components, repair or replacement could be performed without excessive consequence on plants availability. In view of the general technical experience the factor of extrapolation with respect to time or usage are a factor of two or less if one considers operating times between 40 and 60 years.

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More detailed information on future ageing R&D issues are listed in /OEC 99/ for the OECD countries as well as /CEC 00/ for future European R&D work under the umbrella of the 5th Framework Programme of the EU.

References

/CEC 00/ Integrity Assessment of Ageing Components (INTACT). Final Report, produced for the Commission of the European Communities under Contract FI4S-CT 98-0049, May 2000

/NEA 98/OECD/NEA: Development priorities for Non-Destructive Examinations of concrete structures in nuclear plant. NEA/CSNI/R(98)6, October 1998

/NEA 99/Survey on Organic Components in NPPs. NEA/CSNI/R(98)7, January 1999

/OEC 99/OECD Senior Expert Group Report on Facilities and Programmes (SESAR/FAP). Major Facilities and Programmes at Risk. Draft 18.8.1999

Table 1 Relevant ageing mechanisms for passive metallic components of LWRs and their predictability

ageing mechanism		material affected	incubation time	kinetics	governing variables	controllable variables	extrapolation factor	examples for components affected
em-brittle-ment	neutron irradiation activated	carbon steels, low alloy steels	limited	degres-sive	neutron flux, material composition and impurity content, temperature	fluence	< 2	RPV belt line, RPV supports
	thermal activated	cast duplex stainless steels, carbon steels	limited	degres-sive	temperature, ferrite content	temperature	< 2	piping, housings
corro-sion	IGSCC	stainless steels	big scatter	linear	carbon content, microstructure, tensile stress, water impurities, ECP, temperature	water impurities, ECP	~2	piping, core internals
		nickel-base alloys	big scatter	linear	Cr-content, micro-structure, tensile stress, ECP, temperature	ECP	~2	SG-tubes, penetrations
	pitting +TGSCC	stainless steels	big scatter	linear	tensile stress, temperature, chloride concentration	chloride concentration	2 ?	Piping, valve housings
	SICC	carbon steels, low alloy steels	limited	linear f(cycle)	strain rate, tensile stress, number of transients, temperature, O2	strain rate, number of transients, O2	~2	piping, nozzles
	FAC	carbon steels, low alloy steels	limited	linear	Cr-content, flow, temperature, pH-value, (water chemistry)	pH-value, (water chemistry)	< 2	piping of water-steam circuit
	boric acid corrosion	carbon steels, low alloy steels	limited	linear	leak rate	leak rate	< 2	reactor vessel head, closure studs
	MIC	carbon steels, stainless steels	big scatter	?	composition, flow, che-mistry, micro-organisms	flow, water chemistry	> 2	piping of service water systems

Table 1 Relevant ageing mechanisms for passive metallic components of LWRs and their predictability (continued)

ageing mechanism		material affected	incubation time	kinetics	governing variables	controllable variables	extrapolation factor	examples for components affected
fa-tigue	low-cycle high-cycle	carbon steels, stainless steels	limited	linear f(cycle)	number of cycles, strain amplitude, T	number of cycles, strain amplitude	< 2	piping, nozzles, elbows, bolts
syner-gism	corrosion fatigue	carbon steels, low alloy steels, stainless steels	big scatter	linear f(cycle)	number of cycles, strain rate + amplitude, sulphur cont. (material), O2, temperature, water impurities	number of cycles, strain rate + amplitude, water impurities	~2	piping, nozzles
	IASCC	stainless steels, nickel-base alloys	?	progres-sive ?	neutron flux, composition, microstructure	?	> 2	core internals

IGSCC: Intergranular Stress Corrosion Cracking; TGSCC: Transgranular Stress Corrosion Cracking;

IASCC: Irradiation-Assisted Stress Corrosion Cracking; SICC: Strain-Induced Corrosion Cracking;

FAC: Flow-Accelerated Corrosion; MIC: Microbiological-Induced Corrosion

ECP: Electrochemical Potential

Table 2 Monitoring systems; general goals and methodologies, /CEC 00/

Goals		Methodology
Prevention from failures and damages	Load (height, frequency) Verification of specified load collective	On-line measurement of global plant data (p, T) Transient book-keeping Local temperature distributions Calculations related to the load collective
	Stresses (height, frequency) Verification of calculated design limits	On-line measurement of global plant data (p, T) Local measurement of temperatures, strains, displacements Calculation of fatigue usage
	Environment (oxygen content, pH, conductivity) Control of electrochemical potential Influence on protective oxide layers and fatigue strength	Measurement of plant data by sampling (sampling lines) Direct measurement at operating parameters, e. g. oxygen content (partially under development) Direct measurement of redox- and electrochemical potential
Early detection of damage	Loose parts monitoring Loose parts, cracking, damage Crack growth (during pressure test) Leakage monitoring system	On-line measurement and analysis of structure-borne acoustic signals (impact of loose parts), using e. g. piezoelectric accelerometers On-line measurement and analysis of vibrational behaviour (shifts in certain natural frequencies and amplitudes) using signals of the following categories: displacement (absolute, relative), pressure fluctuations, ex-core neutron flux noise On-line measurement and analysis of acoustic emission signals Visual inspection during operation Acoustic monitoring systems, using the noise generated by a leakage flow, detected by piezoelectric resonant acoustic emission probes Localisation of leaks Humidity measurement systems
Control of damage, Crack propagation	Measurement of crack depth and ligament	On-line measurement of crack growth by direct instrumentation of the affected component Potential probe Ultrasonic measurement

Table 3 Procedures for the early detection of ageing mechanisms

Ageing mechanisms	Embrittlement		Corrosion				Fatigue	Synergism	
Measuring-/ Test-procedure	Neutron irradiation activated	Thermal activated	Stress Corrosion Cracking	Strain-Induced Corrosion Cracking	Flow-Accelerated Corrosion	Local corrosion attack	High-cycle, Low-cycle	Corrosion fatigue	Irradiation-Assisted Stress Corrosion Cracking
NDE (UT, RT, MP)	○		●	●	●	●	●	●	●
Hardness (dyn.)	○	○							
EPR			●						○
Barkhausen-Rauschen (noise)							○		
TEM-Analysis	○	●	●				○		○
Global parameters (T, p)							●		

● Adequate tools available
o Needs for future R&D

NDE: Non-Destructive Examination
UT: Ultrasonic Examination
RT: Radiographic Examination
MP: Magnetic Particle Examination

EPR: Electrochemical Potentiokinetic Reactivation Method
TEM: Transmission Electron Microscope

Appendix 1

Relevant EU Research Programmes

4th Framework Programme

AMES, the *Dosimetry and Irradiation Programmes of Ames European Network*, deals with the establishment of the dosimetry and irradiation programmes for the projects AMES1 and AMES2 as well as the improvement and harmonisation of the neutron dosimetry within the AMES network.

BIMET, *Structural Integrity of Bi-metallic Components.* Two benchmark 4-point bend pipe tests have been conducted on a nominal 6’’ piping assembly containing a notched and pre-cracked ferritic to stainless steel bi-metallic weld. The crack growth and path of each crack will be followed by a range of analysis methodologies.

DISWEC, *Evaluation of Techniques for Assessing Corrosion Cracking in Dissimilar Metal Welds*. The objectives of this project are the identification and evaluation of techniques suitable for assessing environmentally assisted cracking of dissimilar welds as well as the generation of quantitative data on crack growth rates in dissimilar metal welds.

REFEREE, *Relation Between Different Measures of Exposure – Induced Shifts in Ductile-Brittle Transition Temperatures – Validation of Surveillance Practice & Mitigation Methods for Ageing Reactor Materials*, issued the experimental determination of the differences between quasi-static fracture toughness, dynamic fracture toughness and Charpy impact transition curves as well as micro-mechanical and micro-structural modelling to obtain a better understanding of the differences between the different measures of the ductile-to-brittle transition temperature shift.

RESQUE, *Reconstitution Techniques Qualification & Evaluation to Study Ageing Phenomena of Nuclear Pressure Vessel Materials*. The objectives of this project are the qualification, verification and comparison of the two different reconstitution techniques stud-welding and electron-beam welding as well as a proposal for a European Code of Practice for the reconstitution of irradiated CV impact and fracture toughness specimens.

VORSAC, *Variation of Residual Stresses in Aged Components*, issued the better understanding and knowledge of the evolution of residual stresses and related material phenomena in nuclear components during manufacture and service life as well as the development and validation of improved methods for modelling and measurement of residual stresses in ageing nuclear components.

Concerted Actions

AMES-NDT, *Ageing Materials Evaluation and Studies by Non-Destructive Tech niques*, is aiming to undertake the intermediate needed step of listing all known non-destructive techniques, their application and limits and reporting on sample demonstration tests before a proper systematic qualification of techniques.

EURIS, *European Network of Risk-Informed In-Service Inspection (RI-ISI)*, see major working groups and networks acting on ageing issues, **appendix 2**.

INTACT, *Integrity assessment of Aged Components*. The aim of this concerted action is to review the current state on understanding and managing ageing of different LWR components such as metallic components, civil engineering structures, motor operated valves, electrical equipment and data acquisition systems as well as cables. Furthermore, a review of European laboratories shall give information of available installations for a more effective organisation of future European R&D work.

MADAM, *Conversion Table of Material Neutron Damage Indexation for All Different European Reactor Types*, deals with the review of the actual situation in the field of neutron damage indexation, neutronix codes and data libraries for PWR, BWR, WWER, and MAGNOX reactors. Further issues are the preparation of the necessary databases, the determination of the calculation methods that should be used and the identification of all the necessary cross-calculations needed to create a qualified conversation table of all the damage indices used for embrittlement assessment.

MODAGE, *Assessment of Modelling Requirements to follow Ageing Phenomena*. The objectives of this project are to establish what is the current level of understanding of plant corrosion and ageing mechanisms and what software is currently available to tackle these issues. In addition, a review of current plant status is underway to identify the main operator concerns in this area.

5th Framework Programme

CONMOD, Concrete containment management using the Finite Element technique combined with in-situ Non-Destructive Testing of conformity with respect to design and construction quality
MAECENAS, Modelling of ageing in concrete nuclear power plant structures
ATHENA, AMES thematic network on ageing
SMILE, Structural margin improvements in aged-embrittled RPV with load history effects
PISA, Phosphorus influence on steel ageing
FRAME, Fracture mechanics based embrittlement modelling
CASTOC, Crack growth behaviour of low alloy steel for pressure boundary components under transient light water reactor (LWR) operation conditions
THERFAT, Thermal fatigue evaluation of piping system "TEE"-connection
SPICKRACK, Signal processing for improved qualification for non-destructive testing of ageing reactors
LIRES, Development of light water reactor (LWR) reference electrodes
ENPOWER, Management of nuclear plant operation by optimising weld repairs
VOCALIST, Validation of constraint-based assessment methodology in structural integrity
NURBIM, Nuclear risk-based inspection methodology

Appendix 2

Major Working Groups And Networks Acting On Ageing Issues

AMES, *Ageing Materials Evaluation and Studies*, is the European Network on nuclear component Ageing Studies co-ordinated by JRC Petten and relies on irradiation and annealing data available by the members or to be generated. The main objectives are to provide information on neutron irradiation effects on RPV materials in support of designers, operators, regulators, researcher and to establish and execute projects in above areas.

ENDEF, the European Non Destructive Examination Forum was created and launched by the Directorate General DGXVII for Energy of the EC to establish an industrial co-operation framework together with the Central and Eastern European Countries as well as the New Independent States operating soviet designed reactors, considering the effectiveness and the adaptability of European NDE technology.

ENIQ, the *European Network for Inspection Qualification,* issued the first edition of the European Methodology for Inspection Qualification in 1995. Qualification of inspection procedures for the well known components of the nuclear reactor is effectively a way of harmonising inspection requirements with the objective of being equally open to any inspection technology.

EPERC, the *European Network of the Non-Nuclear Pressure Equipment Industry*, connects the fields of fatigue design, high strength steel for pressure equipment thickness reduction, harmonisation of inspection programming in Europe and flanges and gaskets.

EPLAF, the *European Plant Life Assessment Forum,* was created in 1997 under the auspices of the General Directorate for Energy of the European Commission (DG XVII), in order to foster communication and dialogue between different European industrialists involved in Eastern Europe, Russia and the Ukraine in bilateral or multilateral projects related to the assessment of ageing and remaining lifetime of WWER nuclear power plant components.

ESIS, the *European Structural Integrity Society*, has several technical committees dealing with several issues related to structural integrity.

EURIS, the *European Network of Risk-Informed In-Service Inspection*, deals with the Analysis of different approaches to the estimation of failure probabilities of components and of the failure consequences, gathering operator feedback; definition of effective ISI programmes and qualification strategies based on a risk-based approach.

ICG-EAC, the *International Co-operative Group on Environmentally Assisted Cracking of Light Water Reactor Materials***.** The general objective of the ICG-EAC is to co-ordinate the international efforts on EAC of structural materials in LWR service environments in order to develop the fundamental understanding and the relevant database for disposition/design criteria for safe operation and life extension. The ICG-EAC co-ordinates co-operative research programmes, develops recommended test procedures, advised on the interpretation of results in order to obtain data relevant to component design and flaw assessment and encourages transfer of this information by organising symposia, interacting with code committees.

IG-RDM, the *International Group on Radiation Damage Mechanism*, was established in late 1980 and is a forum of more than 100 scientists all over the world to discuss newest results, exchange information and initiate new projects in the field of irradiation damage mechanisms.

IWG-LMNPP (IAEA), *the International Working Group on Life Management of Nuclear Power Plants.* Its main objective is to provide information and guidance on design, materials, testing, maintenance, monitoring and mitigation of degradation with regard to major components, with the aim of assuring high availability and safe operation of NPPs. First phase of an international database on nuclear power plant life management has been established.

NESC, the *European Network for Evaluating Structural Components*, connects the NDT / NDE methods and material aspects with the structural integrity assessment. The tasks deals with structural integrity problems like Constraint, Cladding, Sensitivity Analyses, Crack Arrest, Residual Stresses, Probabilistic Approach, Fracture Criteria Consistency, transients, large scale projects.

OECD NEA/CSNI IAGE WG, *Working Group on Integrity and Ageing of Components and Structures.* Its main objective is to provide information and guidance on structural integrity and ageing issues like, fracture and damage mechanics modelling, fracture toughness measurements, neutron embrittlement of RPV steels, SSC of RPV steels and of stainless steels, fatigue of piping, NDT, NDE, residual stresses, long-term behaviour of concrete structures and containments, etc. The group co-ordinates Benchmarks and Specialists' Meetings.

Appendix 3

List Of Relevant Guidance Documents

IAEA-Documents

1. Safety Aspects of Nuclear Power Plant Ageing. IAEA-TECDOC-540, 1990

2. Data Collection and Record Keeping for the Management of Nuclear Power Plant Ageing. Safety Series No. 50-p-3, 1991

3. Methodology for Ageing Management of Nuclear Power Plant Component Important to Safety. Technical Reports Series No. 338, 1992

4. Pilot Studies on Management of Ageing of Nuclear Power Plant Components. TECDOC-670 1992

5. Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Steam Generators. TECDOC-981, 1997

6. Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Concrete Containment Buildings. TECDOC-1025, 1998

7. Assessment and management of ageing of major nuclear power plant components important to safety: PVR vessel internals. TECDOC-1119, 1999

8. Assessment and management of ageing of major nuclear power plant components important to safety: PVR pressure vessels. TECDOC-1120, 1999

9. Implementation and Review of Nuclear Power Plant Ageing Management Programme. Safety Report Series No. 15, 1999

10. AMAT guidelines - Reference document for the IAEA Ageing Management Assessment Teams (AMATs). IAEA Service Series No. 4, 1999

11. Safe Management of the Operating Lifetimes of Nuclear Power Plants. International Nuclear Safety Advisory Group, INSAG-14, 1999

12. Assessment and management of ageing of major nuclear power plant components important to safety: In-containment instrumentation and control cables. Vol. I and II, TECDOC-1188, 2000

OECD-Documents

1. Fatigue crack growth benchmark. NEA/CSNI/R(1997)8

2. CSNI Specialist Meeting on Advanced Instrumentation and Measurements Techniques: summary and conclusions (1997: Santa Barbara,Calif.), 1997. NEA/CSNI/R(1997)32 and NEA/CSNI/R(1997)33

3. Report on the Uncertainty Methods Study, 1998. Vol.1 and 2 NEA/CSNI/R(1997)35 and NEA/CSNI/R(1997)35/Vol.2

4. Development priorities for Non-Destructive Examinations of concrete structures in nuclear plant. NEA/CSNI/R(98)6, October 1998

5. Survey on Organic Components in NPPs. NEA/CSNI/R(98)7, January 1999

6. Glossary of Nuclear Power Plant Ageing. OECD/NEA, 1999

7. Experience with thermal fatigue in LWR piping caused by mixing and stratification: proceedings (1998: Paris,France), 1998. NEA/CSNI/R(1998)8

8. Proceedings of Workshop on FE Analysis of Degraded Concrete Structures (BNL New York/Oct 1998) NEA/CSNI/R(1999)1

9. Comparison Report of RPV Pressurised Thermal Shock International Comparative Assessment Study (PTS ICAS) NEA/CSNI/R(1999)3

10. Tendon Prestress Loss in NPP Containments NEA/CSNI/R(1999)11