Federal Ministry for the
Environment, Nature Conservation,
Building and Nuclear Safety

Report by the Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety (BMUB) on Topical Peer Review Ageing Management of Nuclear Power Plants and Research Reactors

Publisher:	Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety (BMUB)
	Division RS I 5 · PO box 12 06 29 · 53048 Bonn · Germany
	E-Mail: RSI5@bmub.bund.de
	Website: http://www.bmub.bund.de
Editors:	Division RS I 5 (General and Fundamental Aspects of Reactor Safety, Nuclear Safety Codes and Standards, Multilateral Regulatory Cooperation)
Date:	28th December 2017

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Abbreviations

AMP	Ageing Management Programme
AKR	Ausbildungskernreaktor Training reactor
AtG	Atomgesetz Atomic Energy Act
AtSMV	Atomrechtliche Sicherheitsbeauftragten- und Meldeverordnung Nuclear Safety Officer and Reporting Ordinance
BER II	Berliner Experimentier-Reaktor II Berlin experimental reactor II
BHB	Betriebshandbuch des FRM II Operating manual of the FRM II
BMUB	Bundesministerium für Umwelt, Naturschutz, Bau und Reaktorsicherheit Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety
BWR	Boiling Water Reactor
COMOS	Condition Monitoring System
DIN	Deutsche Institut für Normung e.V. German Institute for Standardization
DN	Nominal diameter (nominal size according to EN ISO 6708, approximate internal diameter in mm)
EnKK	EnBW Kernkraft GmbH
EnBW	EnBW Energie Baden-Württemberg AG
ENSREG	European Nuclear Safety Regulators Group
EOL	End-of-Life
ESK	Entsorgungskommission Nuclear Waste Management Commission
ET	Eddy current testing
ETFE	Ethylene Tetrafluorethylene Copolymer
EU	European Union
FAMOS	Fatigue Monitoring System
FRM II	Forschungs-Neutronenquelle Heinz Maier-Leibnitz Heinz Maier-Leibnitz research neutron source
FRMZ	Forschungsreaktor Mainz Mainz research reactor
GG	Grundgesetz Basic Law for the Federal Republic of Germany
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit gGmbH
GTRRN	Global TRIGA Research Reactor Network
HZB	Helmholtz-Zentrum Berlin für Materialien und Energie Helmholtz-Zentrum Berlin for materials and energy
IAEA	International Atomic Energy Agency
IMAS	Integrity Management System
IMS	Integrated Management System

INPO	Institute of Nuclear Power Operations
ISI	In-Service Inspection
KKS	Kraftwerk-Kennzeichensystem Identification system for power plants
KTA	Kerntechnischer Ausschuss Nuclear Safety Standards Commission
KÜS	Körperschallüberwachungssystem Structure-borne noise monitoring system
KWU	Kraftwerk Union AG
LOCA	Loss-of-Coolant Accident
LÜS	Leckageüberwachungssystem Leakage monitoring system
LWR	Light Water Reactor (reactor moderated and cooled with normal water)
MUEEF	Ministerium für Umwelt, Energie, Ernährung und Forsten Rheinland-Pfalz Ministry for the Environment, Agriculture, Nutrition, Viticulture and Forestry Rhine-land-Palatinate
MPA	Materialprüfungsanstalt/Materialprüfamt Materials testing institute/materials testing office
n	Neutron(s)
NAR	National Assessment Report
NDT	Non-Destructive Testing
OSART	Operational Safety Review Team (IAEA programme)
PA	Prüfanweisung Test instruction
PDCA	Plan Do Check Act Cycle
PHB	Prüfhandbuch (FRM II or BER II) Testing manual
PVC	Polyvinyl Chloride
PWR	Pressurised Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
RHWG	Reactor Harmonisation Working Group
RPV(s)	Reactor Pressure Vessel(s)
RSK	Reaktor-Sicherheitskommission Reactor Safety Commission
RWE	RWE Power AG
SENUVK	Senatsverwaltung für Umwelt, Verkehr und Klimaschutz Berlin Senate Department for the Environment, Transport and Climate Protection
SIR	Silicone Rubber
SSK	Strahlenschutzkommission Commission on Radiological Protection
StMUV	Bayerisches Staatsministerium für Umwelt und Verbraucherschutz Bavarian State Ministry of the Environment and Consumer Protection
SUR	Siemensunterrichtsreaktor Siemens research reactor designed for training purposes
SÜS	Schwingungsüberwachungssystem Vibration monitoring system

TPR	Topical Peer Review
TSCC	Transgranular Stress Corrosion Cracking
UT	Ultrasonic Testing
VENE	Vattenfall Europe Nuclear Energy GmbH
VGB	VGB PowerTech e.V.
VT	Visual Testing
WANO	World Association of Nuclear Operators
WENRA	Western European Nuclear Regulator’s Association
WLN	Weiterleitungsnachricht der GRS (nicht öffentlich) GRS information notice (not public)
w/c ratio	Water-cement ratio
ZMA	Zentrale Melde- und Auswertestelle Central incident reporting and evaluation office
XLPE	Cross-linked polyethylene

Executive Summary

The necessity of considering ageing effects in nuclear power plants/research reactors has already been recognised in Germany at an early stage. As a consequence, aspects of ageing have been taken into account in the design of German nuclear power plants. These include, for example, the careful and appropriate design, manufacturing and commissioning of the plants, including their components and systems, as well as the high quality of the materials used.

Structures, systems and components (SSCs) are monitored for possible ageing effects within the framework of in-service inspections, maintenance and servicing measures. Possible problems are identified in advance and preventive measures are taken in due time. By means of evaluating national and international operating experience, findings from nuclear installations worldwide are continuously incorporated into the measures to control ageing effects at the nuclear power plants. In addition, the state of the art in science and technology is evaluated on a regular basis for each plant in order to be able to take into account new findings on ageing where necessary, and thus to be able to continuously maintain or improve the safety level of the plants.

Within the framework of the nuclear rules and regulations, which provide the assessment criteria for the work of the nuclear supervisory authorities in Germany, a specific standard on ageing management in nuclear power plants was developed. This safety standard of the Nuclear Safety Standards Commission (KTA) specifies requirements for ageing management that encompass the technical and organisational measures for an early detection of ageing phenomena relevant for the safety of a nuclear power plant and maintenance of the required quality of the SSCs. The German nuclear rules and regulations are regularly updated, also on the basis of international standards and findings.

The operators have set up integrated management systems at the nuclear power plants, which also take into account findings on ageing effects. This ensures that ageing management is integrated into the operational processes and that all information required for safe operation is available. The German operators discuss the topic of ageing effects and exchange information and experience in their own working groups and expert committees.

The knowledge required for effective ageing management is summarised in a knowledge base and regularly updated so that the identification of safety-related degradation mechanisms is ensured and appropriate measures are derived.

The German nuclear power plants are continuously adapted to the state of the art in science and technology as regards ageing management. The annual evaluation of the results of the ageing management programme for German nuclear power plants confirms the effectiveness of ageing management in German nuclear power plants. The practised procedure in the context of the ageing management programme described in the following ensures that for German nuclear power plants and research reactors the high level of safety during operation is maintained.

Preamble

Article 8e (2) of Council Directive 2014/87/Euratom amending Directive 2009/71/Euratom establishing a community framework for the nuclear safety of nuclear installations stipulates that all Member States of the European Union shall ensure that a national assessment is performed and that, according to Article 8e (3), arrangements are in place to allow for a so-called topical peer review to start in 2017, and for subsequent topical peer reviews to take place at least every six years thereafter. The first peer review will be carried out in several steps, based on a specific technical topic related to nuclear safety of the relevant nuclear installation.

- Member States of the EU shall perform a national self-assessment, based on a specific topic related to nuclear safety of the relevant nuclear installations on their territory and prepare a national report.
- All other Member States, and the European Commission as observer, are invited to peer review the national self-assessments.
- Follow-up measures are agreed if necessary.

The relevant results will then be published.

Through the European Nuclear Safety Regulators Group (ENSREG), the Member States of the European Union have selected the topic of “ageing management” for the first peer review. This is to be carried out for all nuclear power plants that will be operating on 31 December 2017 as well as research reactors with a power equal to 1 MWth or more. In addition to general aspects of ageing management, the assessment report also has to address specific topics. These are

- electrical cables,
- concealed pipework,
- reactor pressure vessels, and
- concrete containment structures.

To ensure the most uniform structure possible of all national reports, WENRA developed the technical specification for the peer review at the request of ENSREG /WEN 16/.

Structure and contents of the German national assessment report are based on the technical specification /WEN 16/. It applies to the following nuclear power plants (from north to south)

- Brokdorf nuclear power plant,
- Emsland nuclear power plant,
- Grohnde nuclear power plant,
- Philippsburg nuclear power plant, Unit 2,
- Neckarwestheim nuclear power plant, Unit 2,
- Isar nuclear power plant, Unit 2,
- Gundremmingen nuclear power plant, Unit B and C,

as well as to the research reactors Heinz Maier-Leibnitz research neutron source (FRM II) of the Technical University of Munich, the Berlin experimental reactor II (BER II) and – on a voluntary basis - the Mainz research reactor (FRMZ).

1 General information

1.1 Nuclear installations identification

In Germany, there are currently eight nuclear power plants in power operation at seven sites. Six are pressurised water reactors (PWRs), three of which are of the Konvoi type and three of construction line 3. Two power plants are boiling water reactors (BWRs) of construction line 72. Table 1-1 presents the key data of the nuclear power plants. As defined in the Atomic Energy Act (AtG), the authorisation for power operation of the Gundremmingen nuclear power plant, Unit B, will expire on 31 December 2017. This means that it would formally no longer belong to the nuclear power plants to be considered in the peer review. Due to the identical design with the Gundremmingen C nuclear power plant, it was decided to include this nuclear power plant in the report. The shut down nuclear power plants as well as the nuclear power plants which are already being dismantled are not the subject of this report.

Table 1-1 Nuclear power plants in operation

	Nuclear power plant in operation Site	**a) Licensee** **b) Manufacturer** **c) Major shareholder**	**Type** **Power output (gross capacity) MWe**	**Construc-tion line**	**a) Date of first partial licence** **b) First criticality** **c) Shutdown date acc. to AtG**
1	**Neckarwestheim 2 (GKN II)** Neckarwestheim Baden-Württemberg	a) EnBW Kernkraft (EnKK) b) KWU c) EnKK 100%	PWR 1400	4 Konvoi	a) 09.11.1982 b) 29.12.1988 c) 31.12.2022
2	**Philippsburg 2 (KKP 2)** Philippsburg Baden-Württemberg	a) EnBW Kernkraft (EnKK) b) KWU c) EnKK 100%	PWR 1468	3	a) 06.07.1977 b) 13.12.1984 c) 31.12.2019
3	**Isar 2 (KKI 2)** Essenbach Bavaria	a) PreussenElektra b) KWU c) PreussenElektra 75%, Stadtwerke München 25%	PWR 1485	4 Konvoi	a) 12.07.1982 b) 15.01.1988 c) 31.12.2022
4	**Gundremmingen B (KRB B)** Gundremmingen Bavaria	a) Kernkraftwerk Gundremmingen b) KWU c) RWE 75%, PreussenElektra 25%	BWR 1344	72	a) 16.07.1976 b) 09.03.1984 c) 31.12.2017
5	**Gundremmingen C (KRB C)** Gundremmingen Bavaria	a) Kernkraftwerk Gundremmingen b) KWU c) RWE 75%, PreussenElektra 25%	BWR 1344	72	a) 16.07.1976 b) 26.10.1984 c) 31.12.2021
6	**Grohnde (KWG)** Grohnde Lower Saxony	a) PreussenElektra b) KWU c) PreussenElektra 83.3%, Stadtwerke Bielefeld 16.7%	PWR 1430	3	a) 08.06.1976 b) 01.09.1984 c) 31.12.2021
7	**Emsland (KKE)** Lingen Lower Saxony	a) Kernkraftwerke Lippe-Ems b) KWU c) RWE 87.5%, PreussenElektra 12.5%	PWR 1400	4 Konvoi	a) 04.08.1982 b) 14.04.1988 c) 31.12.2022
8	**Brokdorf (KBR)** Brokdorf Schleswig-Holstein	a) PreussenElektra b) KWU c) PreussenElektra 80%, VENE 20%	PWR 1480	3	a) 25.10.1976 b) 08.10.1986 c) 31.12.2021

Figure 1-1 Nuclear power plant sites in Germany

Germany will terminate the use of nuclear energy to generate electricity by the end of 2022. According to AtG §7(1), no further licences will be issued for the construction of nuclear power plants.

In Germany, seven research reactors are operated with a thermal power between 100 mW and 20 MW (see Table 1-2). The licensees are public or state-sponsored universities or research centres. Three of these reactors with a thermal power between 100 kW and 20 MW are operated as neutron sources for research. The other four research reactors are training reactors with a thermal power of 100 mW (SUR, Siemens research reactor designed for training purposes) and 2 W, respectively (AKR-2, training reactor of the TU Dresden), for the purpose of practical training in the fields of reactor physics and radiation protection at the universities of Furtwangen, Stuttgart, Ulm and the TU Dresden.

According to the technical specification /WEN 16/, only research reactors with a power equal to 1 MW_{th} or more have to be considered for the peer reviews and thus the national assessment report. The research reactor of the Johannes Gutenberg University Mainz participates on a voluntary basis.

Table 1-2 Research reactors in operation

Research reactor Site		**Licensee**	**Reactor type** **thermal power [MWth]** **th. n-flux [$cm^{-2}s^{-1}$]**	**First criticality**
1	**SUR-FW** Furtwangen Baden-Württemberg	Hochschule Furtwangen	SUR-100 $1 \cdot 10^{-7}$ $6 \cdot 10^{6}$	28.06.1973
2	**SUR-S** Stuttgart Baden-Württemberg	Universität Stuttgart Institut für Kernenergetik und Energiesysteme	SUR-100 $1 \cdot 10^{-7}$ $6 \cdot 10^{6}$	24.08.1964
3	**SUR-U** Ulm Baden-Württemberg	Hochschule Ulm Labor für Strahlenmesstechnik und Reaktortechnik	SUR-100 $1 \cdot 10^{-7}$ $5 \cdot 10^{6}$	01.12.1965
4	**FRM II** Garching Bavaria	Freistaat Bayern, Staatsministerium für Bildung und Kultur, Wissenschaft und Kunst, Technische Universität München	Swimming pool/ compact core 20 $8 \cdot 10^{14}$	02.03.2004
5	**BER II** Berlin	Helmholtz-Zentrum Berlin für Materialien und Energie GmbH (HZB)	Swimming pool/MTR 10 $1 \cdot 10^{14}$	09.12.1973
6	**FRMZ** Mainz Rhineland-Palatinate	Universität Mainz Institut für Kernchemie	Swimming pool/ TRIGA Mark II 0.1 $4 \cdot 10^{12}$	03.08.1965
7	**AKR-2** Dresden Saxony	Technische Universität Dresden Institut für Energietechnik	SUR-type $2 \cdot 10^{-6}$ $3 \cdot 10^{7}$	22.03.2005

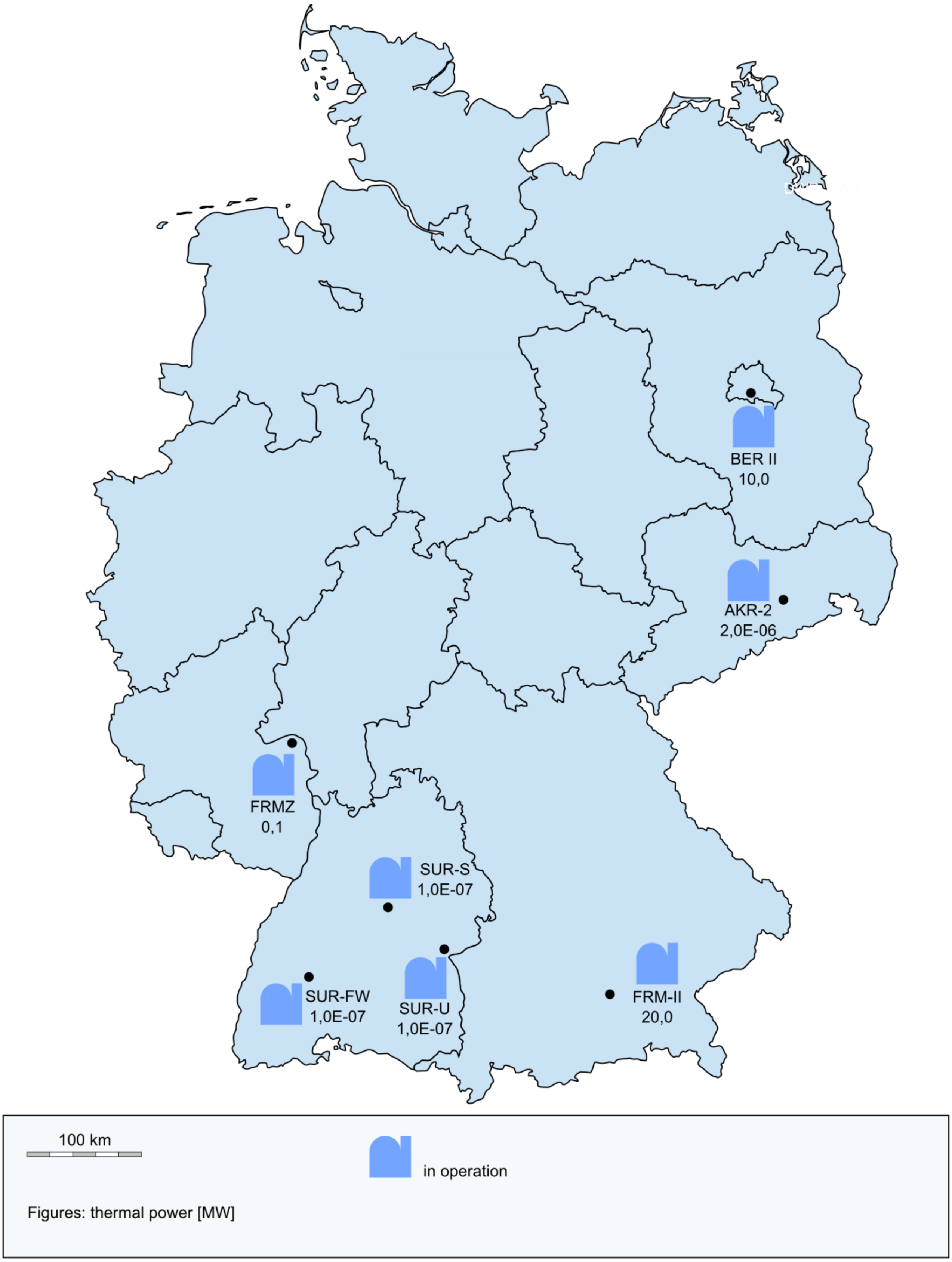

Figure 1-2 Research reactors in Germany

1.2 Process to develop the national assessment report

The process to develop the national assessment report (NAR) on the basis of the technical specification /WEN 16/ and the ENSREG Terms of References (ToR) was launched by the BMUB and the nuclear supervisory authorities of the *Länder* at the end of January 2017. The process consisted of preparing the reports for the individual chapters, the self-assessment by the licensees (nuclear power plants and research reactors), the review by the competent nuclear supervisory authorities of the *Länder* and the preparation of the overall report by the BMUB.

For the national assessment report, the nuclear supervisory authorities of the *Länder*, the operators of the nuclear power plants and the research reactors have prepared parts of the report. The parts of the reports of the plant operators were prepared under the umbrella of the VGB (international technical association for generation and storage of power and heat).

In the period from June to September, the reports prepared by the operators were reviewed by the nuclear supervisory authorities of the *Länder* and supplemented where required. In September, the BMUB began to merge the contents of the reports of the *Länder* into the overall report. From October to December 2017, the national assessment report has been agreed upon by all parties involved and translated. It was published on the BMUB Website in December 2017.

The report is published on the BMUB website www.bmub.bund.de in German and English.

The objective of the German report was to provide sufficient detail to identify both positive and negative experiences with existing ageing management programmes and to identify improvement measures. The description of the licensees' ageing management programmes focused on programmatic issues, the implementation of the programme for selected systems and components and experiences made. Furthermore, common features of the plants and research reactors as well as differences – due to different designs – were shown. For this purpose, the corresponding chapters were subdivided into nuclear power plants and research reactors (see example Table 1-3).

Table 1-3 Substructure to show differences and communalities in the German NAR using the example of Chapters 2.3, 2.6 and 3.1.1

Chapter	Title	Operator	Authority
2.3	**Description of the overall ageing management programme**	**x**	
2.3.a	Nuclear power plants	EnBW, PreussenElektra, RWE	
2.3.b	Research reactors (FRM II, BER II, FRMZ)	TUM, HZB, Uni Mainz	
2.6	**Regulatory oversight process**		**x**
3.1.1	**Description of ageing management programmes for electrical cables**		
3.1.1.a	Nuclear power plants	EnBW, PreussenElektra, RWE	
3.1.1.b	Research reactors (FRM II, BER II, FRMZ)	TUM, HZB, Uni Mainz	

2 Overall ageing management programme requirements and their implementation

2.1 German national regulatory framework

In Germany, the requirements for the quality of safety-related installations (hereinafter referred to as structures, systems and components – SSCs) are laid down in the Atomic Energy Act (AtG), the Safety Requirements for Nuclear Power Plants (SiAnf) and the nuclear safety standards of the Nuclear Safety Standards Commission (Kerntechnischer Ausschuss – KTA) and technical specifications for components and systems.

Essential elements of ageing management were already practiced in German nuclear power plants at an early stage. The possible impairment of SSCs due to degradation mechanisms was already taken into account in the plant construction and the specifications used. For best possible prevention of ageing effects, the following measures were taken, among other things,

- use of high-quality materials,
- consideration of loads on SSCs in the design, construction and operation, and
- design of safety-related components with large safety margins so as to cover the effects of ageing during operation.

The monitoring of possible ageing degradation mechanisms was also planned and practiced from the outset. This was realised, among other things, by

- ensuring the testability of the SSCs by means of appropriate design requirements, e.g. grinding on inner and outer surfaces of welds, accessibility, irradiation of materials during operation,
- the monitoring of loads, e.g. continuous recording of plant transients and temperature changes,
- extensive in-service inspections (ISIs), and
- preventive maintenance, such as the early replacement of wearing parts.

These measures of ageing management have been incorporated into the nuclear regulatory framework at an early stage and have been continuously developed on the basis of new findings in science and technology as well as the evaluation of internal and external operating experience.

The framework for national nuclear legislation and drafting of national nuclear rules and regulation is represented by the regulatory pyramid shown in Figure 2-1. At the upper hierarchy levels of the regulatory pyramid, there are generally binding laws and ordinances. In Germany these are the Basic Law (GG) /GRU 14/, the AtG /AtG 16/ and the ordinances issued on the basis of the AtG.

The GG lays down fundamental principles which also apply to nuclear law. The basic rights laid down in the GG, in particular the basic right to life and physical integrity, form the basis for the standard to be applied to the protective and preventive measures for nuclear power plants, which is further specified in the hierarchy levels of the regulatory pyramid (see Figure 2-1). In addition, the GG contains regulations on the responsibilities of the Federation and the *Länder* for law making and law enforcement.

The AtG contains the basic national regulations on protective and preventive measures, radiation protection and the management of radioactive waste and spent fuel in Germany and is the basis for the associated ordinances.

Statutory ordinances may contain additional authorisations for the promulgation of the general administrative provisions. General administrative provisions regulate the actions of the authorities and thus only have a direct binding effect for the regulatory authorities. However, they have a direct external effect if they are used as a basis for regulatory decision making.

After having consulted the *Länder*, the Federation publishes announcements (in the form of requirements, guidelines, criteria and recommendations). In general, these are regulations adopted in consensus with the competent licensing and supervisory authorities of the *Länder* on the uniform application of the AtG. The announcements of the BMUB describe the view of the federal supervisor on general issues relating to nuclear safety and the regulatory practice, and they provide orientation for the *Land* authorities regarding the enforcement of the AtG. They are referred to by the competent *Land* authorities within the framework of licensing procedures or their supervisory action under their own responsibility. This ensures that the implementation in the different *Länder* takes place according to comparable standards. In relation to the licensees, these become binding by taking them into account in nuclear licences or orders of the nuclear supervisory body.

Safety requirements are defined in the nuclear rules and regulations, which comprise the

- Safety Requirements for Nuclear Power Plants,
- announcements and guidelines of the BMUB,
- guidelines of the RSK,
- recommendations of the RSK, ESK and SSK,
- nuclear safety standards of the KTA,
- technical specifications for components and systems, and
- organisation and operating manuals.

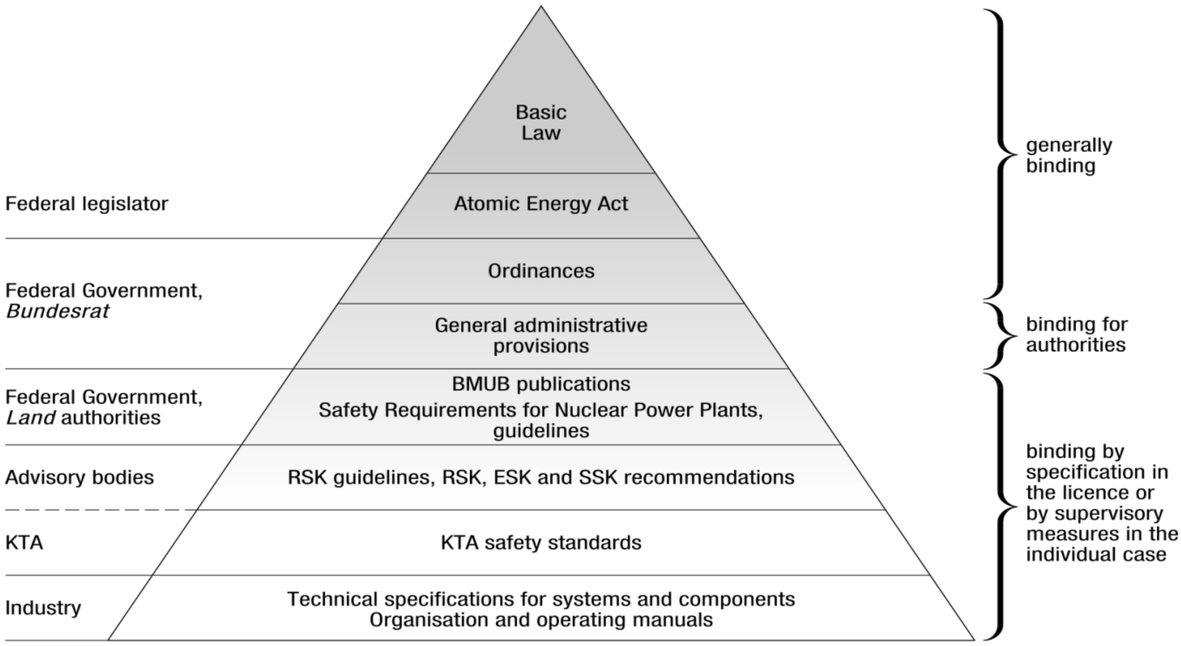

Figure 2-1 National regulatory pyramid

The operational practice relating to aspects of ageing management is based on the legal provisions and existing rules and regulations (AtG, Nuclear Safety Officer and Reporting Ordinance (AtSMV), Safety Requirements for Nuclear Power Plants, RSK recommendations, KTA safety standards, DIN standards and other conventional rules as well as plant-specific specifications).

The measures for long-term maintenance of the required quality (ageing management) are an integral part of the quality requirements specified in the German nuclear rules and regulations.

The Safety Requirements for Nuclear Power Plants /SIC 15/ include requirements for an integrated management system (IMS) that must also take into account objectives and requirements in terms of ageing. A specific requirement for levels 1 to 4 of the defence-in-depth concept is to draw up a monitoring concept for the detection of ageing degradation. Furthermore, it is required that precautions are to be taken against failure due to fatigue, corrosion and other ageing mechanisms, as far as ISIs cannot be performed to the required extent.

The Interpretations of the Safety Requirements for Nuclear Power Plants /INT 15/ also include requirements for ageing management. Accordingly, they stipulate to use operating experience for ageing management. The central requirement is laid down in Section 2.5.1 (9) of the Interpretation I-2 of the Safety Requirements for Nuclear Power Plants: "An ageing management system shall be implemented for the systematic detection, monitoring or prevention of ageing effects on the integrity of the components."

The safety standards of the Nuclear Safety Standards Commission (KTA) have the task of specifying safety requirements in order to achieve the protection goals laid down in the AtG and in the Radiation Protection Ordinance (StrlSchV) and further concretised in the Safety Requirements for Nuclear Power Plants to be able to prove that the necessary precautions have been taken in the light of the state of the art in science and technology to prevent damage resulting from the construction and operation of the facility (§ 7(2)(3) AtG). The KTA safety standards are reviewed to ensure that they are up to date and revised if necessary. Here, in particular, the further development of the international rules and regulations is taken into account.

The general requirements for the integrated management system from the safety requirements are concretised in safety standard KTA 1402 /KTA 12/ The special requirements for ageing management were defined in safety standard KTA 1403 /KTA 17/.

KTA 1403 applies to the safety-related SSCs, including the respective auxiliary and operating supplies, specified in the plant-specific licensing documents and operating procedures, of nuclear power plants still in operation. It deals with physical ageing taking into account new findings with regard to ageing processes. This safety standard applies, furthermore, to the procedures of ageing-management regarding the basic qualification and maintenance of competence and know-how of the personnel and, also, to the documentation and the data from information and operation management systems

Accordingly, the licensees have to set up a systematic and knowledge-based ageing management system as part of the integrated management system, which shall be organised, documented, assessed and updated. Ageing management shall be implemented in a process-oriented manner and integrated into the operational processes. For this purpose, the following basic requirements shall be implemented by the operator:

- The extent of ageing-related observations shall be defined and documented. The observations shall include
  - ageing of the auxiliary and operating supplies of the respective SSCs, and
  - ageing-related influences on the data from information and operation management systems including documentation.
- The procedures of ageing management shall ensure that safety-related degradation mechanisms are identified. The causes and/or consequences of these degradation mechanisms shall be controlled by appropriate measures.
- The further development of the state of the art in science and technology shall be monitored and assessed.
- The measures taken with respect to ageing management and the results achieved shall be documented and assessed. Corresponding reports shall be drawn up at regular intervals. Ageing-management shall be continuously optimised based on the assessments carried out. Impermissible deviations from the required quality shall be eliminated.

- Ageing management is part of an integrated management system. It shall be implemented in a process-oriented manner and integrated into the operational processes. The processes involved (e.g. servicing, maintenance), the interrelated activities as well as their interactions shall be identified, directed and controlled. This overall process shall be designed according to the principles of a PDCA cycle (Plan-Do-Check-Act) (see Figure 2-2).
- Ageing management shall be performed on the basis of a structured knowledge base. In particular, this knowledge base shall contain sufficient information on the respective design concept, ageing-related requirements from the rules and regulations, on the design and manufacture as well as the operating history of the SSCs, on the potential degradation mechanisms and, with respect to the relevant degradation mechanisms, the designated and possible monitoring, testing and corrective measures, including assessment of the results.

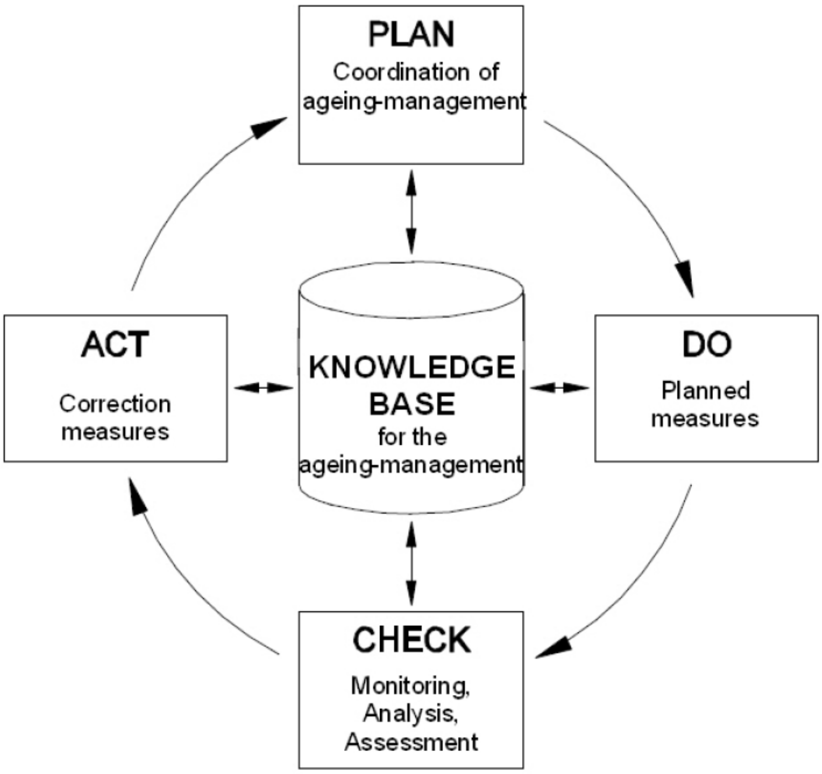

Figure 2-2 PDCA cycle of ageing management /KTA 17/

Process orientation and relevant aspects of a PDCA cycle are dealt with in safety standard KTA 1402 “Integrated Management Systems for the Safe Operation of Nuclear Power Plants”.

If ageing effects are detected on non-safety-related technical installations that are applicable to similar SSCs considered within ageing management, these findings shall be integrated into the ageing management.

KTA 1403 contains further specific requirements for various groups of SSCs, such as mechanical systems and components, components of electrical and instrumentation and control (I&C) systems, structural elements and auxiliary and operating supplies.

Further detailed requirements for maintaining the quality of SSCs that meets the specified requirements (hereinafter referred to as required quality) are also part of component-specific KTA safety standards, such as KTA 3201.4 or KTA 3211.4. Monitoring of compliance with the requirements with respect to the required quality is carried out by operational monitoring, maintenance and the ISIs. For this purpose, there are clear plant-specific specifications that determine the technical content and the procedure for implementation. These specifications and their implementation are reviewed by the nuclear supervisory authority and the authorised experts consulted.

In addition to ageing management, the periodic safety review, which has been carried out in all German nuclear power plants since the mid-nineties, comprehensively assesses the aspects of conceptual ageing.

2.2 International standards

The following gives an overview of the international requirements that have been taken into account in the national regulatory framework.

At the European level, Issue I "Ageing Management" of the WENRA Safety Reference Levels /WEN 14/ represents the basic requirements for ageing management.

The relevant requirements of the IAEA are contained in Specific Safety Requirements SSR 2/1 "Safety of Nuclear Power Plants: Design" /IAE 16a/, SSR 2/2 "Safety of Nuclear Power Plants: Commissioning and Operation" /IAE 16b/ and SSR 3 "Safety of Research Reactors" /IAE 16c/. Recommendations for the implementation of ageing management are given for nuclear power plants in Safety Guide NS-G-2.12 "Ageing Management for Nuclear Power Plants" /IAE 09a/ and for research reactors in Safety Guide SSG-10 "Ageing Management for Research Reactors" /IAE 10/.

Furthermore, representatives of German institutions (operators, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS)) have actively been participating in the IAEA Extra Budget Programme on International Generic Ageing Lessons Learned (IGALL) from the very beginning to contribute their experience in ageing management of German nuclear power plants and to be able to follow relevant new findings and developments in ageing management in nuclear power plants abroad.

European Council Directive 2009/71 EURATOM /EUR 14/ requires a periodic safety review every ten years. Safety Guide SSG-25 "Periodic Safety Review for Nuclear Power Plants" /IAE 13/ of the IAEA safety standards requires the review of ageing management in the section "Safety factor 4: Ageing".

2.3 Description of the overall ageing management programme

2.3.1 Scope of the overall AMP

2.3.1.a Nuclear power plants

According to safety standard KTA 1403, ageing management comprises the entirety of measures taken to control any ageing phenomena that could be detrimental to the safety of a nuclear power plant. KTA 1403 therefore also specifies, in addition to the general requirements for ageing management, requirements for the ageing management of SSCs and the respective auxiliary and operating supplies. In KTA 1403, the term technical facility (which, according to a note therein, corresponds to SSCs as understood in the context of this report) refers to mechanical components and systems, to electrical and instrumentation and control (I&C) equipment and components as well as to structures including structural elements (e.g. buildings, partial structures, structural systems and components).

In addition, KTA 1403 deals with non-technical aspects. These include the basic qualification and maintenance of competence and know-how of the personnel and, ageing of the documentation and the data from information and operation management systems. These non-technical aspects are not included in the technical specification /WEN 16/ and therefore not considered in this report.

Even before KTA 1403 came into force, extensive measures were carried out in the German nuclear power plants with the aim of ensuring the necessary functional features of the SSCs in the required quality meeting the requirements in the long term, i.e. taking into account the effects of degradation mechanisms. In terms of the technological aspects, the measures were already applied during the design of the nuclear power plants and the manufacturing of their components and continued by regulations for the operation of nuclear power plants.

In order to further develop the knowledge base for ensuring the required quality in the long-term, the German operators established an intensive exchange of information on technical as well as organisational issues at an early stage under the umbrella of the VGB together with the manufacturing companies.

Information on and experiences with degradation mechanisms are exchanged in numerous working panels. This cooperation among the operators ensures that all relevant findings and experiences are available in a timely manner.

A common, standard compendium on ageing management /VGB 97/ was established due to the work of the VGB already in the 1990s, long before the entry into force of KTA 1403. The contents of ageing management of German nuclear power plants existing at that time had been summarised in this compendium.

KTA 1403 defines measures to control ageing-related degradation mechanisms. In Germany, different concepts for the SSCs of mechanical engineering, electrical and I&C systems as well as civil engineering had already been developed the early phase of ageing management. The procedures of the German operators existing then had already been incorporated in component-specific KTA safety standards in the 1990s. With the introduction of KTA 1403 /KTA 10/ in 2010, the essential technical requirements were summarised in a generally formulated, higher-level management process. Overall, KTA 1403 represents a catalogue of requirements that enables and ensures a continuous assessment of the functional features of the SSCs and their ageing-related degradation mechanisms in a closed cycle. All measures are implemented in a process-oriented manner and are organisationally integrated into the operational procedures.

The process orientation of ageing management ensures defined responsibilities for SSCs also across different technical departments. This allows for effective action and leads to continuous improvement of ageing management quality through the recurrent assessment of the effectiveness of the process.

Since the entry into force of KTA 1403, the processes of ageing management have been uniformly implemented in German nuclear power plants. If necessary, the processes are adapted or further developed and implemented in the organisational process and workflow. The extent of ageing-related observation of the SSCs has been adapted to the respective technical requirements. In addition, the degradation mechanisms that are to be considered for the individual nuclear power plant according to KTA 1403 have been catalogued and systematically characterised in terms of their relevance for the SSCs.

The non-technical requirements contained in KTA 1403 have also been integrated into the ageing management process so that ageing-related impairments can also be identified and controlled in these areas in a timely manner. In accordance with the specifications of KTA 1403, these concepts were summarised in basic reports for each plant and their effectiveness is assessed annually in the form of status reports. For the structures and structural elements, this is done in a separate structure condition report, which is prepared at intervals of ten years.

The central incident reporting and evaluation office (ZMA) of the VGB organises the nationwide exchange of information on reportable events according to the AtSMV /ATS 92/ and other incidents in German plants and abroad. These events also include events of relevance in terms of ageing management. For these events, the ZMA organises any necessary statements of the manufacturers and distributes them to the operators, for example statements on findings on springs of flow restrictor assemblies and on the durability and functionality of electrolytic capacitors on I&C modules. The power plant operators analyse the information and findings and check their applicability to their own plant. If required, the necessary measures are taken. Examples of measures include whisker tests on solder joints and removal or the programme for the preventive replacement of electrolytic capacitors on I&C modules. Further examples are given in Chapters 2.3.4.a and 2.4.

Reportable events of German nuclear power plants or events of foreign plants of major importance are additionally examined by GRS on behalf of the Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety (BMUB) and, if necessary, the results distributed to the operators with recommendations of GRS within the framework of so-called information notices (WLN).

Overall, the German operators have a comprehensive system for the exchange of experience. In combination with the process of national and international exchange of experience of the nuclear regulatory authorities, this leads to an intensive exchange of experience on ageing-related degradation mechanisms and their assessment.

In safety standard KTA 1403, the specific extent of safety-related SSCs is divided as follows:

- mechanical systems and components
- SSCs of electrical and I&C systems
- structures including structural elements
- auxiliary and operating supplies of the SSCs

For some of these different SSCs, KTA 1403 further divides into groups to take into account the defence-in-depth concept. This means that for some groups, higher ageing management requirements apply than for others. The exact extent of ageing-related observations of SSCs depends on the respective design and the corresponding safety classifications in the written operating procedures.

The periodic safety review, which is to be conducted every ten years in Germany, ensures that the safety concept of the nuclear power plants and thus of the SSCs is also reviewed and, if necessary, updated based on the state of the art in science and technology.

The plant-specific scope of the SSCs that are subject to systematic ageing management has also been described in basic reports on ageing management in accordance with the requirements of KTA 1403. The plant operators have assigned a plant-specific programme of measures to control ageing-related degradation mechanisms to the correspondingly defined categories or groups of SSCs. The quality requirements have also been already available in the required form in the German nuclear power plants. The related specifications are contained in the operating manuals and other plant-specific documents on the issue-related and administrative procedures.

The maintenance and ISI measures are recorded in the operation management systems or in the accompanying documents. The operating data of operational monitoring are kept for several years and transferred into an appropriate long-term storage.

2.3.1.b Research reactors

An independent ageing management programme, as implemented in nuclear power plants, does not exist for German research reactors. In the German research reactors FRM II, BER II and FRMZ, ageing management takes place within the framework of maintenance (safety standards KTA 3301 and 3501). By appropriate application of KTA 1403, the mentioned aspects of ageing management are used as a basis for identifying a possible need for modification and for ensuring the compliance with the state of the art in science and technology.

Monitoring of ageing is dealt with in the ISI programme and regular plant inspections. The results of the inspections provide essential information for plant maintenance in terms of ageing management. The programme of ISIs (see e.g. /FRM 14) and of plant walkdowns is specified in the licences of the research reactors (e.g. third partial licence of the FRM II, third partial licence of the

BER II). Here, manufacturer specifications, operating records, internal and external exchange of experience as well as information notices are considered.

In addition, for the FRMZ, the ISI programme was revised in 2010 and a testing manual (PHB) was drawn up in consultation with the supervisory and licensing authority responsible for the FRMZ.

On the part of the operators, the responsibilities are regulated in the personnel-related operational organisation (see e.g. FRM II BHB Chapter 1, Part 1 on personnel-related operational organisation and FRMZ BHB Part 1, Chapter 1 on staff regulations).

The safety-related components are also listed in the operating manual (e.g. FRM II BHB Part 2, Chapter 3 on reporting criteria). Their monitoring takes place by using comprehensive instrumentation and within the framework of the ISI concept. Components with the same function (e.g. the shutdown rods) or maybe the same design (e.g. the secondary and tertiary pumps) are treated equally in this respect as far as possible.

As a result of the ISIs, the tested components are exchanged, if necessary, or replaced by improved ones. Another result of the inspections are statements about the expected further operating times of the components. Prominent example for the FRM II is the ISI related to material specimen irradiation during operation near-core and far-core.

The results of the ISIs are stored in paper form and to some extent also electronically.

In all relevant areas, inspections and tests are carried out that explicitly investigate long-term trends (control of operating records). Should a parameter show a trend in the long-term behaviour or should it be beyond the specified range, adapted measures are taken. Such measures include e.g. recalibration, shortening of the test interval, modification of the operating mode or replacement of the component in accordance with the maintenance rules (see e.g. FRM II BHB Part 1, Chapter 3).

2.3.2 Ageing assessment

2.3.2.a Nuclear power plants

Effective management of ageing of SSCs requires a comprehensive knowledge base. It forms the basis of ageing management.

The main quality features of the SSCs relevant for the knowledge base and the handling of deviations were already defined with the commissioning of the power plants. This was written down in the licensing documents, the operating manuals and the derived ISI requirements for the different SSCs and supplemented where necessary.

The knowledge base also has to include knowledge of the relevant degradation mechanisms to which the SSCs are subject to. On this basis, the necessary monitoring and testing measures were derived.

Another part of the knowledge base of ageing management are the information and findings from the operating history of the SSCs of the respective nuclear power plant.

As relevant information, the following documents and sources are used to build up the knowledge base:

- quality certificates from the manufacturers
- documents from the design and design assessments
- results from monitoring and their assessment
- results from ISIs and their assessment

- results from maintenance measures
- fault/defect reports and their assessment
- events from the operator’s plant and from other German plants
- GRS information notices
- event reports from nuclear power plants outside Germany
- national and international research projects (e.g. VGB/MPA projects)
- evaluation of experience by the manufacturers
- exchange of experience among the operators (e.g. VGB)
- contractor reports (VGB system for the assessment of contractors)

This information is collected with the use of databases and assessed.

The criteria for compliance with the required quality regarding ISIs of SSCs are described and subject to regulatory supervision. Maintenance measures for SSCs are exclusively carried out by qualified personnel. In addition to the expertise of the operators, the expertise of the service staff ensures continuous updating and monitoring of the required quality.

The relevant degradation mechanisms of the individual SSCs are compiled from the respective operating history, the operating experience of the other German nuclear power plants, the exchange of experience among the operators and from keeping track of the state of the art in science and technology.

Working panels within the VGB deal with issues relating to degradation mechanisms in the context of SSCs of mechanical engineering, electrical and I&C systems as well as civil engineering. The aim of these operator activities is to integrate the existing information in a common database, which comprises the known degradation mechanisms and serves to identify the degradation mechanisms that are relevant during operation and potentially effective. This database is part of the knowledge base for ageing management.

Issues relating to degradation mechanisms due to ageing processes and corresponding ageing management measures in civil engineering are dealt with in the VGB working panel on impacts on civil structures.

The VGB working panel on electrical and I&C engineering deals with ageing management of electrical and I&C systems and the provision of a module for the knowledge base for processing ageing management of electrical and I&C systems in accordance with KTA 1403. A database was created for safety-related components, which includes their function-relevant parts, the relevant degradation mechanisms and the respective diagnostic options. This part of the knowledge base has been integrated into the database of the VGB power plant information system, which is used today in all German nuclear power plants. The linking of the components to type classes (group of components with comparable technical and ageing-relevant properties, such as rectifiers, high-voltage converters, magnetic drives, etc.) in a central catalogue of the database allows direct assignment of the degradation mechanisms to plant-specific component types. This makes it possible to check in the nuclear power plants whether the servicing and maintenance measures carried out reveal specific ageing-related degradation mechanisms.

Topics of the competence field of mechanical engineering are dealt with by the VGB working panels on mechanical and process engineering and on component integrity. The procedure is similar to that in the field of electrical and I&C engineering and initially involves the identification of all degradation mechanisms occurring in the context of mechanical engineering components with a subsequent classification with regard to the criteria “relevant”, “possible” and “effective”. On this basis, a component-specific assignment of the degradation mechanisms is facilitated (for example in the operation management system).

In addition, research projects carried out by the operators under the umbrella of the VGB have led to a comprehensive expansion of knowledge about relevant degradation mechanisms. Where necessary, research projects were carried out by and for the nuclear power plant operators to build and expand the knowledge base on the required quality and relevant ageing mechanisms of the SSCs, and to qualify test methods that enable the detection of relevant phenomena. Basic findings on ageing were also obtained in publicly funded research projects. The findings and recommendations from these have meanwhile been taken into account by the operators. Where appropriate, measures have been implemented (see Chapter 3.1.3.a in conjunction with Chapter 3.1.4 and 5.1.2).

On the basis of the existing knowledge base, the relevant degradation mechanisms and measures for their control were assigned to the SSCs. These assignments were described by the operators of the nuclear power plants in a plant-specific manner in accordance with KTA 1403 and are assessed within the framework of the annual status reports and updated where necessary.

2.3.2.b Research reactors

In the German research reactors FRM II, BER II and FRMZ, ageing of SSCs is assessed within the framework of the regular plant inspections and comprehensive ISIs. For the assessment of the ageing behaviour, e.g. when determining the permissible operating times, the manufacturer's documentation is also used. Furthermore, operations monitoring and the consideration of operating records serve to identify and assess ageing effects. For instance, the relevant fluence levels for the respective components are recorded within the framework of ISIs for the assessment of ageing processes induced by neutron irradiation. At the FRM II, for example, embrittlement of near-core and far-core main components made of AlMg3 alloy (EN AW-5754) is monitored and assessed by means of a comprehensive irradiation programme with regular experimental analysis of the irradiation specimen. These investigations, in particular of the components near the core, are expanded and supplemented by a detailed simulation and calculation program.

Likewise, operating experience at the plant and of other operators is taken into account in the assessment of the ageing of the facility. This is done by internal communication (see e.g. FRM II BHB Part 1, Chapter 1 on communication elements), by checking the applicability of GRS information notices to the respective plant by the operator, and by exchange of experience with other operators, for example at international conferences or within the framework of the research reactor working group (AFR). In the case of the FRMZ, exchange of experience also takes place through the Global TRIGA Research Reactor Network (GTRRN) established by the IAEA.

Here, the assessment criteria are based on nuclear and conventional rules and regulations to be applied mutatis mutandis. Applied criteria for ageing assessment are e.g. change in material properties of components due to neutron irradiation, change in insulation resistance or operational irregularities.

Ageing-related changes are accepted if they remain within the specified range. For example, it may be advantageous to assess the quality of lubricants by chemical analysis. As long as use-related changes remain within the specified limits, there is no objective need to replace them.

Ageing-related changes are also detected and assessed in the framework of own research programmes (e.g. behaviour of plastics or electronic components under irradiation). In addition, membership of the relevant organisations (e.g. the Kerntechnische Gesellschaft e.V. (KTG), VGB), provides access to the latest research results. Furthermore, in individual cases, access is provided to the component databases of the operators of power plants (in particular of nuclear power plants). Own investigations of the FRM II comprise e.g. the ageing behaviour of the aluminium alloy AlMg3 (EN AW-5754), used in the FRM II, under neutron irradiation, which is monitored in the framework of the ISI programme and supplemented by comprehensive calculations.

2.3.3 Monitoring, testing, sampling and inspection activities

2.3.3.a Nuclear power plants

The essential elements of ageing management measures for monitoring of SSCs and control of the relevant degradation mechanisms are

- operational monitoring,
- preventive maintenance, and
- ISIs including functional tests.

Operational monitoring of the SSCs is carried out by automatic or direct evaluation of physical, chemical and biological parameters from operating instrumentation, sampling or, where applicable, task-specific special instrumentation with respect to target values (such as pressure, temperature, transients, vibrations).

In operational monitoring, different levels are effective:

- monitoring by the shift personnel and technical staff
- integrated automatic monitoring
- independent automatic monitoring

The shift personnel and technical staff permanently monitor operation and the plant-related data. Measurement data and findings from operation are recorded and evaluated by the shift personnel and technical staff. In addition, walkdowns and visual controls are carried out. In this way, ageing-related changes in the required quality of the plant and its SCCs are identified at an early stage by comparison with operating parameters and on-site experience.

Integrated automatic monitoring takes place by means of automatic monitoring systems (particularly applicable to electrical and I&C systems), which are designed to be self-monitoring. This monitoring is integrated in the reporting concept of the nuclear power plant. The reports show deviations at an early stage before impermissible changes in the quality of the systems might occur.

The independent automatic monitoring takes place by means of independent monitoring systems. The monitoring results are evaluated systematically. These systems are mainly integrated in the reporting concept of the nuclear power plant and show deviations from the operating parameters at the control room. Here are some examples:

- long-term monitoring systems (e.g. fatigue monitoring system FAMOS and integrity management system IMAS)
- structure-borne noise monitoring system of the reactor pressure vessel (KÜS)
- vibration monitoring system of the primary circuit (SÜS)
- vibration monitoring system of the main coolant pump shafts (condition monitoring system COMOS)
- leakage monitoring system (LÜS)

The maintenance measures maintain and monitor the required quality of the SSCs or restore it. These include, in particular,

- servicing (keeping the specified state),
- inspection (diagnosis of the current state), and
- repair (restoration of the specified state).

The maintenance measures of the ageing management are carried out preventively. This is done either as

- predetermined maintenance, or
- condition-based maintenance.

Predetermined maintenance takes place at specified intervals. It is the most frequently used type of maintenance of SSCs. For condition-based maintenance, inspection and diagnostic procedures are used at regular intervals, which make it possible to make a statement about the component condition. The inspection intervals are then adjusted individually depending on the condition. Inspections of active safety-related components are generally carried out my means of maintenance instructions with component-specific specifications such as test and inspection plan, dimensional record sheet, specifications for threaded joints and lessons learned and other relevant documents. This ensures that inspections of the required quality are carried out.

ISIs include, inter alia,

- walkdowns,
- non-destructive testing (NDT),
- functional tests, and
- measurements and calibrations.

For the implementation of the above-mentioned measures of monitoring of SSCs in the context of ageing management, which are subject to regulatory supervision, there are corresponding clear instructions in the German plants. The basic requirements to carry out maintenance measures and ISIs are specified in the operating licences of the plant, the operating manual and the testing manual. The boundary conditions both in terms of the technical issues and the procedures for an effective ageing management are thus clearly laid down in the licensing documents of the plants.

The data of the operational monitoring are stored for long periods. The results of all maintenance measures and the ISIs are documented. Deviations from the required quality detected within the framework of monitoring are also recorded via fault/defect reports.

In the case of findings (failures, faults or deviations from the specified condition) measures are generally taken – such as repair, replacement, etc. – to restore the required condition (quality). In the case of relevant findings, appropriate measures (servicing, assessment, inspection, repair) are also carried out on comparable components in order to exclude common-mode failures. Since ageing degradation mechanisms often develop slowly over time, trend analyses on the long-term behaviour can be conducted based on the evaluation of the measured data and possible degradation developments on SSCs can be identified at an early stage.

As part of the ageing management, the maintenance reports, repair reports, fault and defect reports of the SSCs are regularly reviewed with regard to relevant ageing phenomena. In addition, the maintenance reports, repair reports, fault and defect reports of all other non-safety-related components and systems are evaluated.

For ISIs, but also in the context of preventive maintenance, different non-destructive testing methods are used. The classic non-destructive test methods on site are:

- visual inspection
- surface crack testing (dye penetration test, magnetic particle test)
- ultrasonic testing
- eddy current testing

- radiographic testing
- potential probe testing

All classic test methods are only effective if materials are separated. They are used to describe the current state of the SSC, but they cannot provide a forecast. To forecast the further development, either trend analyses of the flaw growth over several years or fracture mechanics crack growth calculations are carried out.

2.3.3.b Research reactors

In the German research reactors FRM II, BER II and FRMZ, comprehensive monitoring of ageing processes is ensured by operational monitoring of the required safety-related conditions of components, regular plant inspections as well as comprehensive in-service inspections. Measurements and inspections of the technical components are carried out in accordance with the testing manuals (e.g. at the FRM II PHB, manual for conventional tests and manual for internal tests). The inspection records to be completed and the test instructions describe the test methods and the test equipment to be used in detail. For each test, the target specifications and tolerances acceptable for measured values are listed. This simplifies implementation for the tester and deviations from normal operation are easily detected and reliably documented.

To identify long-term effects, the results of ISIs are evaluated. In the ISI programme and with regard to I&C parameters, there are tests and parameters that reflect states and trends and thus serve to identify long-term effects. Tests in which trends are specifically examined in long-term recordings are conducted e.g. at the FRM II for the primary pumps, the emergency cooling system and the cold neutron source.

Additional inspections are carried out in accordance with manufacturer's documentation. Furthermore, the operators of the research reactors have staff who, due to their education (including relevant apprenticeship professions or studies, through training at external organisations or manufacturers), are able to assess in special cases whether unscheduled inspections are necessary. An example of inspections at the FRMZ beyond the PHB is the identification of coating defects in the cooling tower by the technical services of the Johannes Gutenberg University Mainz.

Experience feedback from external sources, such as the GRS information notices, is also taken into account after having been forwarded to the operator. For example, an inspection of the emergency power diesel for ageing effects (here: mounting of the stator and testing of the insulation resistance) was initiated at the FRMZ in the context of GRS information notice WLN 2014/11. Although no findings were made, the PHB was updated and an ISI interval specified in agreement with the regulatory supervisory authority.

Unexpected mechanisms are taken into account by the redundant and diverse design of SSCs, by the great care taken in the procurement and installation of such components (as regulated in the applicable specifications and maintenance rules of the BHB) and by the comprehensive instrumentation and operational monitoring, also by plant inspections. In addition, servicing of SSCs is carried out by qualified own personnel or specialist companies and manufacturers. If unexpected ageing effects should occur, measures are taken.

2.3.4 Preventive and remedial actions

2.3.4.a Nuclear power plants

During the operating lives of the German nuclear power plants, several preventive and remedial actions have been carried out within the framework of ageing management for the control of relevant degradation mechanisms. For degradation mechanisms of generic significance, concepts for their control are jointly developed by the operators within the VGB.

For I&C components, there are several certified workshops in Germany that are allowed to repair discontinued safety-related modules. The experiences and repair reports are exchanged via the VGB. This often resulted in comprehensive remediation projects, such as the exchange of wet capacitors of whole module families.

For electrotechnical components, measures include the definition of warning levels in project groups of the VGB to ensure that components are replaced or remediated before age-related failure occurs. An example of this are chemical values in oil samples from oil-cooled transformers.

In the following, some major mechanical engineering measures are presented by way of example.

The phenomenon of “transgranular stress corrosion cracking” (TGSCC) occurred in a number of cases preferentially in the area of the outer and inner surfaces of small-bore pipes and was identified as “chloride-induced TGSCC”. Due to the demonstrably chloride-free coolant used, there were no findings on the inner surfaces in pipes with through-flow, but only in areas where fortification is possible. This mechanism can be initiated e.g. by chloride-containing plastic support elements or plastic adhesive tapes for marking or fixation purposes or chloride-containing spiral-wound asbestos gaskets. Therefore, apart from the introduction of low-chloride adhesive tapes, in particular chloride-containing gaskets which are in contact with the medium were replaced in many areas. In addition, in certain areas, there was a change to higher-alloyed steels which have higher corrosion resistance.

Wastage corrosion, which is of special importance with regard to the steam generator tubes made of Incoloy 800 installed in German pressurised water reactors, has decreased since the change from phosphate treatment to high AVT (all volatile treatment – pH> 9.7) in all plants. Residual degradation potentials are prevented by regular tests and inspections as well as the removal of deposits on the steam generator tube sheets. In the plants, the transition to AVT, inevitably led to the installation of stainless steel or titanium tubes in the turbine condensers and thus a very high coolant circuit tightness. The resulting minimised entry of chlorides from the cooling water also precludes the occurrence of stress corrosion cracking and pitting on steam generator tubes. In addition, erosion corrosion on main feed water and main condensate lines can be effectively prevented in PWRs with high AVT.

Within the framework of fatigue analyses for components of the reactor coolant pressure boundary, the analyses available have recently been revised on the basis of newly acquired findings from operational measurements of plants in Germany and abroad. In some system areas, formations of fluid layers of different temperatures as well as intermittent piston flows with fatigue relevance were confirmed which had originally not been considered in the design. In order to determine these additional loads, thermocouples were installed distributed over the circumference in the affected or possibly affected pipeline sections of the German light water reactors at one or more measurement levels. On this basis, the transient temperature distributions are recorded for all relevant load cases and documented in transient reports. Together with the load case frequencies specified in the load specification, they form the basis for the revision of the fatigue analyses. Occasionally, strain gauges are also used temporarily to check the stress level and the safety of local deformations. The revision of the fatigue analyses, partly together with a reconstructed high utilisation already during the commissioning of the nuclear power plant, led in some plants to the renewal of certain system sections. At the same time, a constructive improvement was made with the aim of achieving the lowest possible stresses. Another consequence of the temperature measurements is the establishment of modes of operation that mitigate the fatigue-related loads.

2.3.4.b Research reactors

Extensive measures of monitoring, testing and inspection already described elsewhere in this report (see Chapters 2.3.1.b, 2.3.2.b and 2.3.3.b) have been taken in order to identify age-related effects in a timely manner. Operating manual and testing manual (e.g. in the context of the maintenance rules) regulate in detail how to proceed in these cases.

Measures of monitoring, testing and inspection on SSCs are carried out according to the PHB in the presence of an authorised expert consulted by the nuclear supervisory and licensing authority according to § 20 of the Atomic Energy Act (AtG). In the event of deviations from the set value, the nuclear supervisory and licensing authority determines what actions are to be taken based on the report of the authorised expert, specifying deadlines depending on the significance. The operator implements corrective actions in a timely manner and provides proof to the nuclear supervisory authority according to a standardised procedure.

Technical tests and servicing without involvement of authorised experts according to § 20 AtG are assessed in terms of their safety significance for reactor operation by the operator. If necessary, qualified contractors are commissioned to remedy the deficiencies. In case of major measures with implications for reactor operation or safety, these will be reported to the nuclear supervisory authority and, if necessary, corrected in the modification procedure subject to approval.

An example of the recent past at the FRMZ was the replacement of the cooling tower system due to coating defects on the water collecting reservoir and the supporting structure of the cooling tower accompanied by TÜV Rheinland and the State Office for the Environment Rhineland-Palatinate.

Qualified staff are used to carry out the work and tests specified in the PHB. If necessary, qualified contractors are commissioned, for example at the FRMZ for inspection of the cooling circuit pumps and cleaning circuit pumps. The organisational and personnel requirements for remediation of defects identified are set out in accordance with the maintenance rules of the BHB.

Based on the measures described in Chapters 2.3.1.b, 2.3.2.b and 2.3.3.b, preventive actions are also taken. This includes the replacement of components before reaching the wear limit.

2.4 Review and update of the overall AMP

2.4.a Nuclear power plants

As part of the integrated management system, ageing management is also subject to a continuous improvement process in the form of a closed PDCA cycle. In particular, the process-related results themselves are assessed by it, used to improve the process steps and, with the knowledge gained, are reintegrated into the process.

In the case of relevant deviations in terms of quality, component-related fault signals are generated with a corresponding prioritisation and these are processed systematically and consistently. By systematically assessing the findings gained during the preparation of the annual status reports on ageing management of the individual plants, generic effects are identified and measures taken to control them. The findings of self-assessments in the form of audits or reviews in accordance with safety standard KTA 1402 /KTA 12/ are also reintegrated into the processes for improving them.

Repairs and ISIs are carried out by adequately qualified and experienced personnel. In addition to an internal independent assessment, additional assessment of repairs and ISIs of safety-related SSCs is carried out by the authorised experts of the nuclear supervisory authority.

The evaluation and analysis of experience feedback on reportable events from the operator’s plant and from other power plants as well as on information notices also take place in German nuclear power plants on the basis of a structured process.

The entire ageing management process, including the evaluation of experience feedback, repairs and ISIs of safety-related SSCs, is carried out, reviewed and adapted on the basis of a structured process.

In the case of modifications to the nuclear power plant or its operation, all affected organisational units will be involved in accordance with safety standard KTA 1402 /KTA 12/. In addition, modifications to SSCs and the operation of safety systems are generally subject to approval in Germany and are thus assessed by the nuclear supervisory authority and by any authorised experts consulted by them prior to implementation. The entirety of the measures ensures that all safety-related aspects are taken into account in planned modifications. In this respect impacts from plant modifications on the ageing management are also considered.

Relevant findings and open issues in the context of ageing management are implemented and dealt with by the operator as long as limitations in the control of ageing-related degradation mechanisms on SSCs cannot be excluded, so that any limitation with regard to the required quality is prevented.

The German nuclear power plant operators keep track of the state of the art in science and technology on the one hand directly by monitoring of legal provisions and legislative initiatives and, on the other, by participation in technical committees, such as in the framework of the working panels of the VGB (see also Chapter 2.3.2.a). The work of the VGB working panels not only includes monitoring of the state of the art in science and technology but also an intensive exchange on operating experience and other occurrences directly related to the nuclear power plants. Information about events at nuclear power plants outside Germany is distributed to the German nuclear power plant operators by GRS, INPO and WANO using the information system VGB-ZMA.

The VGB also initiates central research projects with relevance for ageing management. The results are available to all German nuclear power plant operators as members of the VGB.

The rules of ageing management are regularly reviewed to ensure that they are up to date. Due to the participation of the operators in the KTA working panels, necessary modifications to the ageing management are incorporated into the regulations also from the point of view of the operators. The known ageing-related degradation mechanisms of the SSCs are understood, are identified by appropriate test programmes and controlled by proven measures. Should new ageing-related degradation mechanisms be identified in the future, a research project can be initiated at short notice via the VGB if required (see also Chapter 2.3.2.a). From the operators' point of view, the VGB has established itself as an effective body for the effective handling of relevant issues by pooling the interests of the operators.

The process-based design of the ageing management and its anchoring in the management systems of the operators ensures the effectiveness of ageing management. The annual status reports and the structure condition reports to be prepared according to safety standard KTA 1403 provide a summary assessment of the effectiveness of ageing management in the nuclear power plants.

New findings on ageing-related degradation mechanisms are assessed by the operators and, if necessary, corrective actions for their control initiated. In the context of maintenance measures and ISIs of SSCs as well as in the course of the assessment of reportable events or findings relating to SSCs, this process is for the most part subject to nuclear supervision.

2.4.b Research reactors

The maintenance programmes of FRM II, BER II and FRMZ are subject to continuous improvement processes and are reviewed by the nuclear supervisory authorities and their authorised experts consulted.

The results of quality assurance and the operating experience from the operator’s plant and from other plants – e.g. from GRS information notices, from participation in expert committees, from participation in international conferences, from regular exchange of experience with other operators – are incorporated into his improvement process. The servicing and maintenance programme is adjusted as needed. If e.g. compared to the past, a higher degree of monitoring of individual components in terms of their ageing is identified, this is taken into account by a revision of the test specification with a possibly shortened test interval.

Furthermore, results of audits in the context of the QA programme of the FRM II, for instance, inspections (either by FRM II staff alone or in the presence of the authorised expert consulted according to § 20 AtG) and plant walkdowns are communicated within the framework of the communication elements specified in the BHB, and processed and implemented by the competent organisational unit with the involvement of the competent departments and, where appropriate, the operating management.

Plant modifications are carried out within the framework of the modification procedure defined in the BHB (maintenance rules), also taking into account impacts regarding the required ISI.

If these should cause any developments that necessitate a modification to the maintenance programme or the licence conditions, these will be taken into account and appropriately implemented. Currently, for example, the FRM II plans to change the monitoring concept of the core-near main components made of the aluminium alloy AlMg3 (EN AW-5754) from an exclusively time-based to a fluence-based concept. This would also affect the ancillary provisions of the operating licence.

The ageing management of the operators of the research reactors is considered in the context of the periodic safety review in accordance with the technical specification quoting WENRA Safety Reference Level I2.5 (/WEN 16/, Section 02.4), adapted to current requirements and also compared with the requirements of the nuclear rules and regulations (e.g. KTA safety standards).

The necessity of an ISI programme is regulated in the licence conditions. The nuclear supervisory authority approves the ISI programme. Should new findings emerge, they will be incorporated into the ISI concept too.

Should own operating experience, GRS information notices, participation in expert committees, participation in international conferences or regular exchange of experience with other operators lead to findings that cannot conclusively be clarified on the basis of the knowledge available to the operators of the research reactors, further research projects are initiated together with partner institutes/operators or are undertaken by themselves. This was the case e.g. in the context of the analysis of the deposits in the reactor pool of the FRM II, which was carried out in collaboration with the Radiochemistry Munich (RCM) of the TU Munich and the European Institute for Transuranium Elements (ITU) in Karlsruhe.

2.5 Licensee's experience of application of the overall AMP

2.5.a Nuclear power plants

Ageing management in German nuclear power plants is already carried out beginning with the design, construction, commissioning and operation. It makes a significant contribution to the safe operation of German nuclear power plants. Furthermore, due to the measures of ageing management, possibilities of plant improvement could be identified and implemented.

With the introduction of safety standard KTA 1403, a standardised assessment basis was created with regard to the terms related to ageing and ageing management. Thus, the implementation of the requirements of KTA 1403 has contributed to the further systematisation and structuring of ageing management.

The entirety of measures and processes (ISIs, fault detection systems, etc.) of the operators of German nuclear power plants, which are used in the context of ageing management, are suitable to gain ageing-related knowledge in the sense of the KTA 1403 as well as to identify and control ageing degradation mechanisms in good time before a damage occurs.

Key measures, which are to be assigned to ageing management, were already comprehensively established before the introduction of KTA 1403, at the latest with the commissioning of the nuclear power plants. The measures have been further developed by the work of the VGB for the expansion and consolidation of the knowledge base. The introduction of KTA 1403 formalised the processes and contents of ageing management. Recent years have shown that the ageing management of KTA 1403, based on the already proven processes and measures of the operators of the nuclear power plants for controlling ageing degradation mechanisms, is suitable for maintaining the required quality of the SSCs and thus for making an important contribution to plant safety.

2.5.b Research reactors

From the point of view of the operator of the FRM II, the programme for managing ageing effects introduced at the FRM II is robust and suitable for the monitoring of ageing-related changes in properties, in particular also of the safety-relevant components. Changes are detected in such a timely manner that there are sufficient opportunities for an appropriate response.

From the point of view of the operator of the BER II, the procedure for ageing management developed at the BER II over many years of operation has proven itself. This is reflected in the high availability of the facility. Due to targeted measures to adapt the facility to the state of the art in science and technology, in particular in electrical, electronic and I&C components, trouble-free operation and a secure supply of spare parts can be expected to continue.

From the point of view of the FRMZ operator, the definition of test intervals and the coverage of the FRMZ infrastructure within the framework of the PHB, the implementation of the test specifications by qualified and competent personnel and the level of inspections by authorised experts commissioned by the authorities derived from the PHB have been very effective in the last few years. Thus, ageing-related changes of components can be detected at an early stage, so that in recent years there have been no acute repair measures on safety-related components of the FRMZ.

2.6 Regulatory oversight process

Germany is a republic with a federal structure and is composed of 16 federal states, referred to as the *Länder*. Unless otherwise specified, the execution of federal laws generally lies within the responsibility of the *Länder*. The “regulatory body” is therefore composed of the nuclear licensing and supervisory authorities of the Federation and the *Länder* (see Figure 2-3).

The BMUB carries overall state responsibility towards the interior of Germany as well as towards the international community. It ensures that those in charge of the applicants and licence holders, federal and *Land* authorities, and of the technical safety organisations ensure effective protection of man and the environment against the hazards of nuclear energy and the harmful effects of ionising radiation at any times.

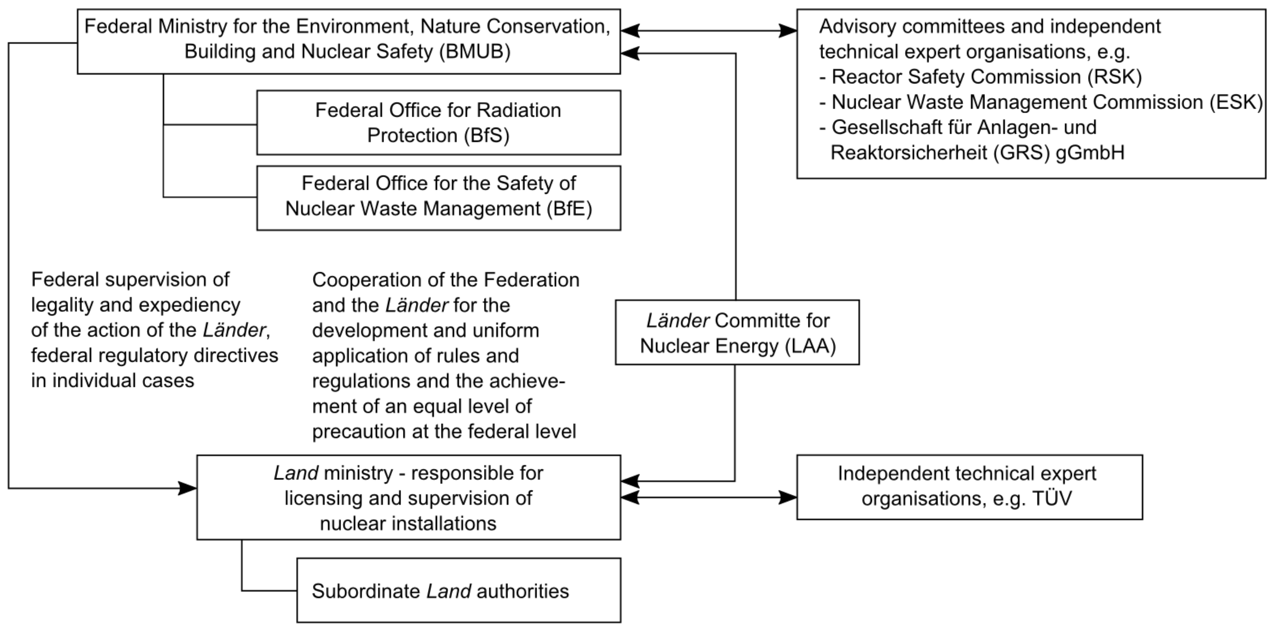

Figure 2-3 Structure of the regulatory body

According to Article 85 and 87c GG in conjunction with § 24(1) AtG, the *Länder* shall be responsible for supervision and thus monitoring of the safety of nuclear power plants and research reactors. Supervision is the responsibility of the nuclear supervisory authority in the *Land* where the nuclear power plant or research reactor is located. According to § 24 AtG, the supreme *Land* authorities (ministries) designated by the *Land* governments shall be responsible for nuclear licensing and supervision. In individual cases, subordinate authorities may also be tasked with supervisory functions. Within the ministries, the tasks of the nuclear licensing and supervisory authority are fulfilled by ministerial directorates. The structure of such directorates depends on the kind and scope of the nuclear activities and installations in the individual *Land*. These directorates are in turn subdivided into divisions for the execution of the licensing and supervisory procedures for the nuclear installations and are supported, where necessary, by additional divisions dealing with radiation protection and environmental radioactivity, waste management, fundamental issues and legal affairs. According to § 20 AtG, authorised experts may be consulted in the nuclear administrative procedure by the nuclear supervisory authorities of the *Länder*. The nuclear licensing and supervisory authorities of the *Länder* make use of this option regularly and extensively due to the large extent of the inspections and the associated wide range of different scientific and technical disciplines required as well as the special technical equipment needed. The work routines and processes of the nuclear licensing and supervisory authorities of the *Länder* are largely defined and regulated uniformly by the established organisational procedures for *Land* ministries. However, individual aspects of these management systems are also adapted specifically in the various authorities on a continuous basis, taking into account changing requirements. Here, the activities are focused on the description and analysis of process sequences in nuclear licensing and supervisory procedures. The *Länder* are supervised by the Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety (BMUB) to ensure that they carry out the tasks assigned to them in accordance with the principle of legality and appropriateness (Articles 85 and 87c GG).

The requirements related to ageing management for nuclear power plants are defined in safety standard KTA 1403 /KTA 16/, which is used as a reference in terms of the assessment criteria. As laid down in KTA 1403 /KTA 16/, the operators prepare a plant-specific basic report on ageing management and report on ageing-related activities and measures as well as findings and results from plant monitoring to the competent nuclear regulatory authority of the *Land* in the form of annual status reports. Should new knowledge about ageing processes or methods emerge, the basic report will be updated. In addition, a structure condition report is to be prepared for structures and structural elements, which must be updated at the latest after ten years. The status reports contain a summary assessment of the effectiveness of ageing management and the quality or change in the quality of the SSCs. Where potential for improvement is identified, appropriate measures must be taken to improve the effectiveness of ageing management and the quality of the SSCs.

In German research reactors, there is no independent ageing management programme as it is implemented in the nuclear power plants. However, the aspects of ageing management referred to in KTA 1403 are implemented within the framework maintenance. Monitoring of ageing is dealt with within the framework of the programme of the ISIs, the plant walkdowns by the authorised experts according to § 20 AtG, the internal and external exchange of experience and regular plant inspections. The necessity of an ISI programme is regulated in the licence conditions. The ISI programme is reviewed with the involvement of authorised experts and approved by the supervisory authority. Should new findings emerge, they too will be incorporated into the ISI concept.

The objective of regulatory supervision of ageing management is to verify that the organisational measures provided for by the operator cover all safety-relevant areas in the context of ageing. These include i.a. the areas of quality assurance, maintenance, modifications, qualification of the personnel, evaluation of special events, fulfilment of certain requirements, periodic safety review and operational monitoring.

In addition, the effectiveness of ageing management measures is checked by random controls of individual SSCs.

In particular, the following tasks are performed by the nuclear supervisory authorities of the *Länder*:

- evaluation of the annual status reports of the nuclear power plant operators on ageing management, also taking into account opinions of nuclear experts consulted (e.g. TÜV or for special issues in the field of materials science e.g. the MPA)
- review of the procedure and regulations of the operator's ageing monitoring in the different organisational areas (electrical and I&C systems, mechanical engineering, civil engineering, auxiliary materials) on the basis of presentations and explanations provided by the heads of department or heads of section.
- examination of documents and records, e.g. in the context of ISIs and maintenance or factory acceptance tests,
- interviewing the persons who implement ageing management measures (status meetings, accompanying supervisions)
- random review of individual measures of ageing management, e.g. in the context of ISIs, maintenance and, plant modifications, if any, or factory acceptance tests
- discussion with the power plant management about objectives, strategies, major projects etc.
- evaluation and discussion of the results of the entire ageing management process, in particular the annual status reports

Figure 2-4 shows, by way of example, the process of regulatory supervision of ageing management. The respective competent nuclear supervisory authority of the *Land*

- supervises the operator’s ageing monitoring,
- reviews the processes and organisational regulations of the operator, and
- evaluates the annual status reports on ageing management.

In the case of ageing-related findings, the authority examines applicability to other SSCs of the affected plant and the measures taken with regard to appropriateness and completeness. It also examines the applicability to other plants within its area of competence. If the nuclear supervisory authority of the *Land* recognises a fundamental significance with regard to such applicability, it informs the Federal Ministry for the Environment and, where appropriate, the other supervisory authorities of the *Länder*, or initiates referral to Federation-*Länder* committees. The implementation of the measures is monitored and documented within the framework of regulatory supervision. In order to detect non-technical ageing effects, the personnel and organisational regulations are reviewed to identify any need for improvement or updating. Identified need for improvement in the regulations on monitoring of ageing effects and the documentation is accompanied by the authority and finally assessed. One of the activities in this respect are the technical discussions between the authority and the licensee. If the measures have been initiated to the required extent or if no need for improvement has been identified, the authority and the licensee will conduct the annual status meeting. This completes and documents the process for the reporting year to be assessed.

Reportable events from German nuclear power plants or events from plants abroad of general significance are also investigated by GRS on behalf of the BMUB and, where appropriate, the results distributed to the nuclear supervisory authorities of the *Länder*, the technical expert organisations and the operators within the framework of information notices (WLN) with recommendations of GRS. GRS also evaluates annual status reports on ageing management of selected years with regard to generic findings for all plants. The results are incorporated into the WLN process (see "Handbuch über die Zusammenarbeit zwischen Bund und Ländern im Atomrecht" (Handbook on Cooperation between the Federal Government and the *Länder* in Nuclear Law) Process 6 on information notices).

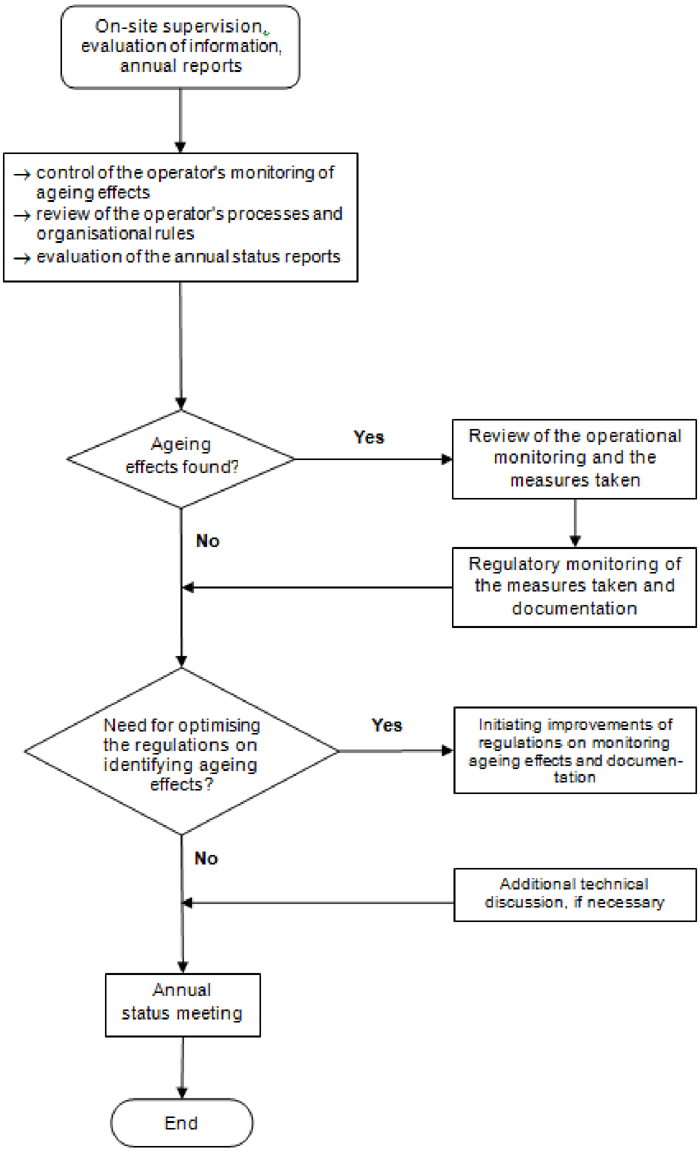

Figure 2-4 Process of supervision of the operator's ageing management by the supervisory authorities of the *Länder*

The supervisory activities of the *Land* authorities in the context of ageing management are defined in *Land*-specific regulations. This is described using the example of the *Land* of Baden-Württemberg /MS 16/.

For the organisation of supervision of nuclear power plants in the *Land* of Baden-Württemberg, the Ministry of the Environment, Climate Protection and Energy Sector Baden Württemberg developed an supervision concept /AK 15/, which describes i.a. the objects, tasks, standards and explicit inspection methods of regulatory supervision. Specific regulations and specifications of processes are contained in the supervision manual (AHB) /AH 17/. Different aids and process descriptions (mostly in the form of flow charts) are assigned to these regulations and specifications. Supervision concept and supervision manual thus specify the *Land*-specific regulations for the regulatory supervisory process.

For the contents of nuclear regulatory supervision, ageing management was integrated into the supervision manual as part of the basic supervision. The chapters on on-site inspections, inspection areas and ageing management include requirements for regulatory supervision and its implementation as well as various documents (regulations, process descriptions) to assist inspectors of the nuclear supervisory authority.

2.7 Regulator's assessment of the overall ageing management programme and conclusions

2.7.a Nuclear power plants

Assessment of the overall AMP

The overall ageing management programme (AMP) is correctly described in Chapter 2. The details of the structure and scope of the overall AMP of the German nuclear power plants correspond to the realised state.

The process-based implementation of the AMP is fully and correctly described in Chapter 2.3.1. The knowledge required for effective ageing management is summarised in a knowledge base and continuously updated.

The technical and organisational measures of ageing monitoring used to control ageing phenomena correspond to the requirements of the relevant nuclear safety standards (KTA). They are listed in full in Chapter 2.3.3.

The measures carried within the framework of the AMP are designed to ensure the required quality of the SSCs.

The AMP documented by the operators in Chapter 2 "Overall ageing management programme requirements and their implementation" for nuclear power plants in power operation fully corresponds to the ageing management practised in German plants.

While maintaining the currently practised procedure within the framework of the overall AMP, it is ensured for German plants that ageing phenomena do not unduly impair the level of safety of German plants during operation.

The AMP of the German plants complies with the international specifications of the WENRA Safety Reference Level Issue I "Ageing Management" and the IAEA Safety Guide NS-G-2.12 "Ageing Management for Nuclear Power Plants".

Experience with the application of the overall AMP

A process-based ageing management system had been implemented by the respective nuclear supervisory procedure for the German nuclear power plants, accompanied by the supervisory authorities, already before the entry into force of safety standard KTA 1403.

Ageing management is part of the integrated management system for the safe operation of nuclear power plants according to safety standard KTA 1402. This ensures that the Ageing Management is integrated into the operational processes and that all information required for safe operation is available.

The processes of ageing management are designed according to DIN EN ISO 9001 and according to the principles of a PDCA cycle (Plan-Do-Check-Act).

If classified as safety-related SSCs, mechanical components, equipment and components of the electrical and I&C systems, structures including structural elements as well as the auxiliary and operating supplies of the SSCs are subject to ageing monitoring. For these SSCs, plant-specific basic reports were prepared in accordance with KTA 1403 and updated as necessary. In addition, the state of knowledge on ageing management is updated in annual status reports.

Safety standard KTA 1403 provides for a graded approach in ageing monitoring of the SSCs according to their safety relevance. For SSCs with the highest safety relevance (e.g. Group M1, Group B1), comprehensive monitoring and maintenance measures are specified so that the quality required to comply with the design requirements is ensured throughout the entire operating lifetime.

In addition, there is a statutory requirement to conduct a periodic safety review, which must be carried out by each operator for the respective plant on the basis of a guideline at an interval of ten years. In this respect, it is to be determined whether there are safety-related deficits compared to an advanced state of science and technology. Here, the nuclear supervisory authority reviews the operator’s assessment with the support of the authorised experts consulted.

Experiences with the application of the overall AMP in the German nuclear power plants show that ageing phenomena at the safety-related SSCs can be identified at an early stage. Remedial actions were implemented in a timely manner.

Effectiveness assessment of the existing ageing management programme

In order to demonstrate compliance with the requirements for an ageing management programme in accordance with safety standard KTA 1403, the operators report to the nuclear supervisory authority annually on the activities carried out and irregularities identified as well as on the results of the evaluation of the AMP for the reporting period under review.

In the German plants, ageing management relevant processes, events and measures are documented using a suitable IT software application (e.g. operation management system). This ensures that ageing management relevant processes from plant operation are fully and comprehensively taken into account. An assessment in terms of ageing management relevance is conducted at least once a year.

External events (e.g. VGB reports, events from plants in Germany and abroad, GRS information notices with AMP relevance) are checked by the operators for applicability to their own plants. Furthermore, an event is also assessed in terms of its AMP relevance. The results of this assessment are reported and documented within the status report.

The annual evaluation of the results of ageing management by the nuclear supervisory authorities of the *Länder* confirms the effectiveness of the ageing monitoring programmes.

Main strengths

Ageing management in the German nuclear power plants takes place under consideration of the established processes in the power plants. The measures in the context of these processes (e.g. maintenance measures) are managed using the operation management system and comprehensively and systematically evaluated for ageing management relevance (component, degradation mechanism).

Through the annual reporting, the assessment processes of the nuclear power plant operators are presented in a transparent and comprehensible manner. The ageing management documentation regarding the SSCs is updated continuously. The current ageing management results are assessed in a timely manner.

Weaknesses identified

Ageing management in the German nuclear power plants is carried out in accordance with KTA 1403 in terms of a continuous improvement process (PDCA cycle) with an updated knowledge base.

There are no conceptual weaknesses in the ageing management process for overall ageing management.

2.7.b Research reactors

A graded approach regarding the overall requirements of the German regulations presented in Chapter 2.1 is permissible for the research reactors due to their lower hazard potential and varies depending on the individual research reactor.

The competent nuclear licensing and supervisory authorities Bavarian State Ministry of the Environment and Consumer Protection (StMUV), Ministry for the Environment, Agriculture, Nutrition, Viticulture and Forestry Rhineland-Palatinate (MUEEF) and the Senate Department for the Environment, Transport and Climate Protection (SENUVK) deem the entirety of the measures presented by the operators of the FRM II, BER II and FRMZ appropriate to identify and address ageing-related degradation mechanisms at an early stage.

3 Electrical cables

3.1 Description of ageing management programmes for electrical cables

3.1.1 Scope of ageing management for electrical cables

3.1.1.a Nuclear power plants

According to KTA 1403, all safety-related structures, systems and components (SSCs) of electrical and instrumentation and control (I&C) systems have to be included in the ageing management programme. This also includes cables that supply and connect these safety-related structures, systems and components.

As for all electrical and instrumentation and control (I&C) systems, the redundancy and single-failure concept also applies to the cables. By choosing suitable and reliable cable types and cable materials, a high level of operational safety and availability is achieved. The failure of individual cables is covered by the above-mentioned redundancy and single-failure concept. Thus, any ageing degradation mechanisms leading to systematic effects/impairments must be addressed in the context of ageing management for cables.

In German nuclear power plants, cables are divided into the following categories:

- High- and medium-voltage cables (> 1 kV)
  For the supply of large consumers, for the establishment of connections to transformers and within the plant on-site electrical systems. This category includes the cables defined by WENRA in the technical specification /WEN 16/ as high-voltage cables > 3 kV.
- Low-voltage cables (< 1 kV)
  For the supply of electrical consumers such as motors, heaters, actuators.
- I&C cables
  For the transmission of analogue and binary signals. This category includes neutron flux instrumentation cables in accordance with the technical specification /WEN 16/.
- Special cables
  Cables for special applications, e.g. coaxial cables for small and frequency signals.

In German nuclear power plants, there are no buried cables with safety significance in the voltage range between 380 V and 3 kV[1]. Cables of this voltage level that have safety significance are either mainly routed through accessible cable ducts or, in some few cases, in cable conduits. Here, the ageing-relevant loads do not differ, regardless of the implemented routing. The cables are each installed according to the manufacturer's specification.

For this reason, deviating from the requirements of the technical specification /WEN 16/, this report does not address this category.

The ageing management of electrical cables is applied to the cables associated with safety-related functions. In order to determine findings with relevance for the ageing management of cables, operating experience in connection with the use of cables in the operational area (without safety relevance) is also used.

[1] In accordance with “Medium voltage cables buried or in trenches. For the purpose of the national assessment report, medium voltage cables are those in the approximate range of 380 V to 3 kV.” of the technical specification /WEN 16/.

The measures for the ageing management of cables do not differ in principle between the types of cable to be addressed within the scope of the national assessment report. Therefore, the measures for the ageing management of the different cable types are presented together in the following – only in a few cases where the measures differ is a distinction made between individual cable types. These measures can be divided into

- monitoring of cables for ageing phenomena by means of in-service inspection/measurement of the insulation resistance of cables with trending based on measured values previously determined by the plant personnel,
- cables as a subset when checking and measuring electrical and I&C components, devices or measuring circuits,
- visual checks for damage or changes,
- ongoing demonstration that the loss-of-coolant-accident resistance according to KTA 3706 /KTA 00/ for cables with LOCA requirement is maintained,
- evaluation of operating experience feedback, and
- special measures derived from insights gained from the above aspects.

In the context of ageing management for cables, ageing mechanisms are considered that result from the operation of the cables themselves as well as from the environmental influences impacting on the cables (from outside).

The aim of ageing management is to ensure the long-term functioning of the cables both under normal operating conditions and under the influence of postulated accidents. Here, particular attention is paid to the electrical loads acting on the cables in the event of different accident scenarios and to the loads caused by the accident atmosphere in a LOCA. Mechanical loads from external hazard events need not be considered for the cables as these loads are absorbed by the associated cable support structure.

For the ageing management of the electrical and I&C SCCs and thus also of the cables, plant-specific basic reports for the implementation of the ageing management were prepared. These basic reports describe the ageing management processes as well as the interfaces with the maintenance, in-service inspection, and plant monitoring processes.

3.1.1.b Research reactors

All safety-relevant cables of the research reactors FRM II, BER II and FRMZ are an integral part of the ageing management programme. For the purposes of this report, only the ageing management of neutron flux instrumentation cables is considered as there are neither any inaccessible safety-significant cables in the voltage range between 380 V and 3 kV at FRM II, BER II and FRMZ (for example, all cables at BER II are installed on accessible cable racks throughout the facility) nor any cables with safety significance in the voltage range above 3 kV.

In integral inspections, the

- conductors,
- insulation,
- armouring/shielding,
- jacket/sheath and
- termination arrangements

of the cables to be considered are checked in terms of their condition meeting the requirements.

3.1.2 Ageing assessment of electrical cables

3.1.2.a Nuclear power plants

According to KTA 1403, the relevant degradation mechanisms that may affect the specified required functional characteristics are determined for all safety-related SSCs of electrical and I&C systems. This is done on the basis of the corresponding manufacturer's specifications, which in turn are based on national and international regulations.

Other sources of knowledge regarding the ageing management for cables are:

- results from monitoring and their assessment
- results from ISIs and their assessment
- results from maintenance measures
- fault/defect reports and their assessment
- overall maintenance inspection outage reports
- GRS information notices (WLN)
- reportable events from the operator's plant and from other German plants
- event reports from nuclear power plants outside Germany
- national and international research projects
- evaluation of experience by the manufacturers
- exchange of experience among the operators
- contractor reports (VGB system for the assessment of contractors)

In the evaluation of the operating experience for cables, findings from the plant itself as well as information and reports from other nuclear power plants, information from the industrial application of the cables, and lessons learned by the cable manufacturers themselves are referred to.

In the area of safety-related SSCs of the German nuclear power plants, only electrical cables with polymer insulation are used. The insulating materials mainly used are:

- polyvinyl chloride (PVC)
- cross-linked polyethylene (XLPE)
- ethylene propylene rubber (EPR)
- silicone rubber (SIR)

<u>Cable insulation</u>

Regarding the electrical cables, it is mainly the ageing of these polymer materials that is of interest. The electrical properties of the cables are determined by the insulation materials of the cable wires/conductors and by the properties of the conductor material itself. The cable sheath has no electrical function for the cable; it serves primarily for the mechanical protection of the wires/conductors.

The ageing of the insulating materials can lead to a change in the insulation capacity of the respective cable conductor. This is monitored by measures from the ageing management programme that are described below.

Conductor materials

Changes or corrosion on the metallic cable conductors is relevant only for the areas of interfaces of the cables with components and equipment. Corrosion phenomena in these areas have an effect on the conductivity (line resistance) or the signal transmission behaviour.

Load variables for cables

To assess the relevant ageing phenomena, the relevant loads for the cables are recorded and their impact on the required cable characteristics is evaluated. This evaluation of the load parameters already took place within the framework of the design of the cables and during the selection of the cable types together with the materials used and the construction of the cables. The aim was and still is to use preferably ageing-resistant cables that can be used throughout the entire planned operating lifetime of the nuclear power plants.

In this respect, the following load parameters have been and are still considered:

- thermal loading:
  - self-heating of the cable from the current load
  - heating of the cable from the ambient temperature
- radiological loading:
  - change in the properties of the insulating materials due to the radiation-induced degradation of the molecular structure of the polymers used
- exposure to UV radiation (sunlight)
- voltage/frequency/electric fields:
  - voltage as a load parameter for cables with design voltage > 1 kV[2] for insulation and conductive layer
  - voltage/frequency as a load parameter acting on dielectrics in I&C cables (with coax signal cables)
- mechanical loading through field forces
- water influences/humidity
- contact with chemicals (e.g. oil vapour, acids/bases, intumescent coatings)

Functional characteristics of electrical cables

The function-determining characteristics are considered in terms of age-related change over time. The following are function-determining characteristics of electrical cables:

- insulation capacity (conductor-to-conductor and conductor-to-ground)
- current carrying capacity and conductivity
- signal transmission behaviour (for I&C and special cables)

[2] In accordance with “High voltage cables subject to adverse environment (environment limited to the immediate vicinity that is hostile to the component material. This can be due to moisture, radiation, temperature etc.). For the purpose of the national assessment report, high voltage cables are those above about 3 kV.” of the technical specification /WEN 16/.

The measurands listed below are used to monitor and assess the effect of ageing on the function-determining characteristics of the cables.

Measurands

The following measurands are used to assess the condition of the cables:

- insulation resistance (cables with design voltages > 1 kV[2])
- line resistance (all cable types)
- partial discharge measurement (cables with design voltages > 1 kV[2])
- checking of the signal transmission behaviour (I&C cables[3])
- visual inspection (all cable types)
- thermography for the identification of hot spots/locations of defects (all cable types)
- Elongation at break of the insulation and sheath materials (all cable types)
- electrical operability under LOCA conditions (all cable types)

Applicable acceptance criteria are derived both from the relevant industry standards (VDE, DIN etc.) and from the manufacturer's specifications. The frequency and cycles of subsequent measurements and examinations are specified in the respective written operational regulations of the plants. Based on the manufacturers' recommendations, the cycles may be adapted on the basis of the operating experience in the respective plants. The use of the listed measurands in the ageing management of cables is explained below.

Insulation resistance measurements are carried out for several reasons. On the one hand, compliance with the maximum leakage currents during normal operation of the cable prescribed by the existing conventional regulations is demonstrated by the insulation resistance measurement. On the other hand, the insulation resistance measurement on cables was established as a test method in order to detect and monitor ageing effects on cables (usually cables with a design voltage > 1 kV). The measured values are compared on the one hand with the maximum leakage currents derived from the relevant regulations and industry standards (KTA, VDE, DIN, etc.) or from the manufacturer's specifications; on the other hand, trends can be tracked from the individual results achieved over longer periods of time in order to assess the ageing behaviour of the cable insulation materials.

Measurements of the line resistivity are carried out in addition to the insulation resistance measurements, accompanying maintenance activities on electrical machines and devices, and within the framework of the (in-service) inspection of I&C circuits. Here, the examination of the cable and the associated termination technique are implicitly part of the measurements made.

Partial discharge measurements serve for the assessment of the state of the insulation characteristics of high-voltage machines. Partial discharges can occur in emerging local defects, as a result of which the insulating material will be damaged during further operation. Partial discharge measurements are carried out on the electrical machines themselves (motors and transformers) as well as on cables with design voltages > 1 kV.

The measurements can be performed with the machine stopped (offline) and/or while the machine is running (online). To detect the partial discharge pulses, coupling capacitors are connected to the terminals of the machine. The partial discharge signal and the phase reference of the machine voltage are fed via a measuring cable to the measuring device, which records and evaluates the

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[3] In accordance with “Neutron flux instrumentation cables.” of the technical specification /WEN 16/.

partial discharge pulses. For analysis purposes, the partial discharge pulses are sorted by phase angle, amplitude value, and frequency of occurrence. The distribution functions thus obtained are also referred to as partial discharge patterns whose appearance and characteristics (e.g. average discharge current, average impulse charge, etc.) are typical of the various partial discharge types. Statements about the ageing condition of the insulating materials (both the insulating materials of the electrical machine itself and the connected cables) in the monitored area can then be derived from these patterns.

The inspection of the signal transmission behaviour of I&C cables for the transmission of analogue measuring signals is usually carried out in connection with the inspection of analogue measuring chains, which also contain the relevant cables. In the case of deviations of the measuring results from the expected values, the individual links of the measurement chain are examined with respect to the influence; this way, changes in the cables involved can also be detected and appropriate measures derived. As part of these measures, the neutron flux instrumentation cables are also looked at.

Visual inspections of cables are carried out during routine walk-downs as well as specifically during special visual inspections in the context of inspection and maintenance activities. Changes in the visible parts of the cables (e.g. local discoloration of the sheath surface) are assessed and, depending on the result, further measures may be initiated, such as additional tests and studies to assess the condition of the cables concerned.

Thermography as a method for the non-contact measurement of the surface temperature of objects is also used to monitor cables. In particular, cables with higher current loading can be examined by means of thermography. In this case, both the absolute temperature of the cable under consideration and possible local hot spots, which are usually caused by contact resistance in the area of connection points, can be identified. By means of the thermographic results, statements can then also be made about the actual thermal load acting on the cables in the respective installation situation and thus about their predicted ageing behaviour. This procedure is used additionally in the assessment of abnormalities. If the cables are designed according to the rules, hot spots are not to be expected, which is why they may be an indication of ageing.

Elongation at break tests on insulating and sheath materials of cables serve to monitor and assess the (ageing-related) structural changes of the materials used. As a reference for the ageing of cable polymers, the relative elongation at break is used. The elongation at break value of (aged) samples taken from operation is set in relation to samples of the same material without any ageing stress. A reduction of the ductility of the material allows the conclusion that there is an ageing-related reduction of the average chain length of the polymer molecules. If a threshold specified in the test specifications is then undercut, this reduced ductility is considered as an indicator that further investigations have to be initiated with respect to the insulation properties and the further usability of the cables that have correspondingly aged.

The **electrical operability under LOCA conditions** is a special aspect of the ageing management of cables. With the above-mentioned measurands and associated monitoring methods, it is only possible to detect and asses the ageing effects from the loading of the cables by normal system operation. Cables needed for the electricity supply of electrical components or for the transmission of instrumentation and control signals under the aggravated ambient conditions of a LOCA must have the corresponding characteristics even after advanced ageing. These properties are checked during tests as part of the qualification of the cables. The temperature- and/or radiation-induced ageing of the cables caused by the loads from plant operation is already taken into account in the corresponding test specifications.

In addition, cable samples stored in so-called cable depositories in areas of a Konvoi-type PWR (representative for the nuclear power plants) with high radiological and thermal loads (reactor coolant line) are monitored for their ageing behaviour. At regular intervals (approximately every three years), individual cable samples are taken and the change in the relative elongation at break is used as a guide to assess the change in the polymer material. In addition, such pre-aged cable samples are subjected to electrical tests under LOCA conditions (high-pressure and high-temperature steam atmosphere). The electrical operability under LOCA conditions is specified as an acceptance criterion for each cable type; this is derived from the typical applications of the cable type. The results obtained can be applied to the cables that are used in the nuclear power plants and which must fulfil the corresponding requirements.

The test results are summarised in so-called service life curves. The graph of a service life curve can be used to determine the proven lifetime (relative to the cumulative radiological loading of the cable in its installation position). For this purpose, material-specific service life curves are generated from the results of the above-mentioned LOCA tests and from the ageing parameters of the specimens that have come out of the tests positive.

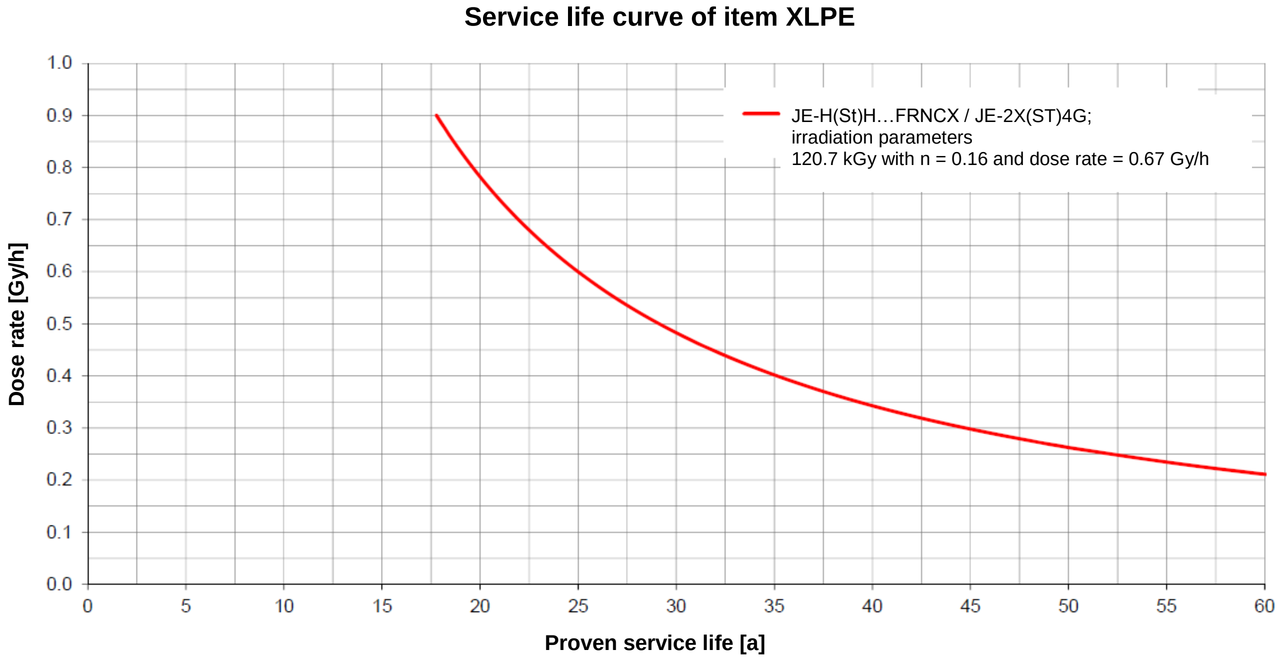

Figure 3-1 Example of a service life curve for cables with XLPE insulation with LOCA requirement

3.1.2.b Research reactors

The focus in connection with the ageing management of cables connecting the neutron flux detector is on the functionality of the cables. This could be limited by effects such as

- corrosion on plugs and contacts,
- short circuit due to incorrect manufacture of the cables,
- breakages due to wrong handling of the cables,
- age-related insulation loss between cable conductor and external insulation in coaxial cables,
- age-related loss of insulation against water ingress in submerged cables.

In addition, due to the radioactive radiation to which the cables in the immediate vicinity of the reactor core are exposed, accelerated embrittlement with an alteration of the insulation may occur. It can be assumed that the probability of such radiation damage scales with the neutron flux and the reactor power in the location where the cables are installed.

The cables to be considered are checked for being in conditions meeting the requirements.

If necessary, the manufacturer's specifications will be used for the assessment in addition to the test objectives specified in the ISIs.

3.1.3 Monitoring, testing, sampling and inspection activities for electrical cables

3.1.3.a Nuclear power plants

For the monitoring of the ageing condition of the cables used in the nuclear power plants, the test, measuring and demonstration methods mentioned in the previous section are used. Since the electrical cables are simple and robust components for which extensive operating experience also exists from the non-nuclear field, a comprehensive application of the methods is not required. The selection of the appropriate procedures as well as the reason for the test and the test intervals is carried out according to the safety requirements, the availability requirements, and the cable technology of the respective cable. The procedure is as follows:

- insulation resistance measurements on representative installed cables with design voltages > 1 kV[4] in the context of ISIs and special tests
- permanent measurement or measurements at certain intervals of the partial discharge on electrical machines and transformers as well as the associated cables to assess the ageing condition of the insulating materials used
- visual inspections to assess the condition of the cables used, if necessary supplemented by thermography
- checking of cables during inspection and maintenance work on electrical consumers or I&C equipment
- cable depository to assess the state of ageing and to demonstrate that cables maintain LOCA resistance. Storage of cable samples on the reactor coolant line of a Konvoi-type PWR, representative for the German nuclear power plants

Figure 3-2 Cable sample depository next to a PWR loop line (left); detailed view of cable samples (right)

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4 In accordance with "High voltage cables subject to adverse environment (environment limited to the immediate vicinity that is hostile to the component material. This can be due to moisture, radiation, temperature etc.). For the purpose of the national assessment report, high voltage cables are those above about 3 kV." of the technical specification /WEN 16/.

- evaluation of findings or cable failures by cable types and, if applicable, materials, if necessary with an assessment of the individual cable failures and their identified causes. Here, in accordance with the single-failure concept mentioned above, an evaluation is carried out with regard to any indications of systematic effects, i.e. also ageing effects
- monitoring and evaluation of the internal and external feedback of experience regarding damage and events involving cables
- initiation of examination programmes/studies relating to cables and cable materials to identify causes of known cable defects in cooperation with suppliers and cable manufacturers. The measures derived from such examination programmes/studies are described in Chapter 3.1.4.a
- keeping track of the state of the art in science and technology

3.1.3.b Research reactors

The required condition of the affected cables of the neutron flux instrumentation is ensured by ISIs according to the testing manual (PHB) and plant walkdowns.

For the FRM II, the examinations to be carried out according to the testing manual (PHB) /FRM 14/ include, in particular, the following examinations:

- “Testing of the impulse path of the wide-range measuring channels with a neutron source (reactor neutrons) by recording the characteristic curve (functional test)”,
- “Recording of the Characteristic curve of the detectors of the power range measuring channels (functional test)”, and
- “Measurement of the insulation resistance of the detectors (measurement, calibration)”.

These are carried out each year in the presence of the authorised expert consulted by the nuclear supervisory authority. In addition, the plant walkdown “neutron flux measuring equipment” by the regulating authority's authorised expert takes place annually.

In the case of the BER II, inspection of the cabling is carried out integrally within the subsequent ISI:

- neutron flux source range: insulation resistance test of all coaxial cables between detector and preamplifier and between preamplifier and cabinet
- neutron flux intermediate range: insulation resistance test of coaxial cable with connected ionization chamber
- neutron flux power range: insulation resistance test of coaxial cable with connected ionization chamber
- fire protection material for cables: determination of the compatibility between cable insulation and Unitherm fire protection bandage

These examinations take place annually with the participation of the authorised expert consulted by the nuclear supervisory authority. The insulation resistance tests are additionally carried out every six months within the scope of the testing manual.

In the case of the FRMZ, the ionization chambers used for monitoring reactor power (the ionisation chamber of the source range channel, the compensated measuring chamber of the logarithmic power measurement channel, and the compensated measuring chamber of the linear power channel) are checked for their operability in installed condition with cables attached and cable positions unchanged for years. At the beginning of each day of operation (average 200-220 days per year), a test neutron source is positioned by the operators next to the chambers in a given measuring position. The resulting signal (background count rate and the count rate) is recorded at the control panel of the reactor, compared with the specified nominal values, and documented on the daily start-up checklist. If there are any deviations, the shift supervisor of the FRMZ will initiate investigations into the cause (damage to the measuring chamber, defective amplifier in the measuring circuit, cable defect).

At the FRMZ, the test with start-up test source as part of the start-up checklist serves as the primary test function in connection with ageing management. As an additional measure, a test within the framework of ISI is embedded in the testing manual. In this test, the condition of the cables is examined by measuring the insulation resistance at normal operating voltages.

3.1.4 Preventive and remedial actions for electrical cables

3.1.4.a Nuclear power plants

In the following, examples of active measures from the ageing management of cables are described that have been implemented in German nuclear power plants. Applicable acceptance criteria are derived both from the relevant industry standards (VDE, DIN, etc.) and from the manufacturers' specifications.

- *Preventive replacement of PVC-insulated medium-voltage cables (6 kV/10 kV)*[5] *in the on-site electrical system and for the supply of safety-relevant consumers with cables with XLPE insulation:* Here, the testing of insulation resistances of medium-voltage cables, mentioned under 3.1.3, was also established to assess the ageing condition of the cables. The motive for this was the short-circuit in a PVC-insulated medium-voltage cable (10 kV) in the on-site electrical system of a German nuclear power plant in 2004 and the investigations carried out in this context.
- *Replacement of older-type cables with SIR (silicone rubber) insulation in high-dose-rate areas in the PWRs:* In tests on representative test specimens, cables with advanced radiological and thermal ageing showed a reduction in the insulation resistance under LOCA conditions. The cables with corresponding requirements were replaced in the 1990s for cables with other insulating materials or cables with highly cross-linked silicone insulation materials.
- *Replacement of electrical and I&C cables with ethylene tetrafluoroethylene (ETFE) insulation (Tefzel cable) required to withstand LOCA conditions in areas with high dose rates:* At an advanced radiological and thermal ageing of the cables, the insulation materials showed an increased sensitivity to accident-induced moisture and steam. The cables with corresponding requirements were replaced in the 1990s with cables with other insulating materials.
- Use of service life curves (see Chapter 3.1.2.a) to determine new demonstration steps for cables required to meet LOCA demands or to replace the corresponding cables if the end of the specified service life has been reached.
- Replacement of individual cable sections in areas where high dose rates (> 2.5 kGy/year) are reached if no sufficient lead time of ageing demonstration can be achieved by the mentioned depository method and the associated service life curves.

For cables where deviations are found when the methods described in Chapter 3.1.3 for the monitoring of cable ageing are applied, corresponding remedial measures are taken. In the case of findings or failures without any indication of newly recognised systematically occurring ageing phe-

[5] In accordance with "High voltage cables subject to adverse environment (environment limited to the immediate vicinity that is hostile to the component material. This can be due to moisture, radiation, temperature etc.). For the purpose of the national assessment report, high voltage cables are those above about 3 kV." of the technical specification /WEN 16/.

nomena, such measures may be the replacement of complete sections or subsections of the cable connection concerned.

3.1.4.b Research reactors

Preventive measures and maintenance measures of the cables of the neutron flux instrumentation to be considered are i.a. carried out within the framework of the ISIs and walkdowns listed under Chapter 3.1.3.b and on the basis of operating experience from other facilities.

Should there be any findings, these will be assessed. If findings and their assessment should result in a need for action, the specified condition of the component concerned will be restored by repair or replacement in accordance with the maintenance rules (e.g. FRM II operating manual Part 1 Chapter 3).

At the FRMZ, every time the measuring chambers are replaced, the connected cables in the area of the reactor pool are replaced at the same time by new cables of the same specification. This is done by qualified electronics personnel of the FRMZ.

3.2 Licensee’s experience of the application of AMPs for electrical cables

3.2.a Nuclear power plants

In accordance with KTA 1403, an evaluation of the effectiveness of the ageing management programme for electrical and I&C equipment is carried out. The effectiveness of the existing and possibly additional measures taken to detect and control relevant degradation mechanisms is assessed at defined intervals. The length of the intervals is determined according to the expected ageing behaviour.

At regular intervals, the basic reports named in Section 3.1.1 are supplemented by the status reports on the ageing management, in which the current status and the knowledge gain in accordance with the PDCA cycle as scheduled are described.

Experience with the ageing management for cables as practiced in the German nuclear power plants is positive. Considering the high number of cables used and the total length of cables installed as well as the number of different cable types, the number of indications and failures identified is low and ranges in the area of statistically distributed random failures. This suggests that the selection and specification of the cable materials and cable types used in the design and construction of the nuclear power plants and the associated qualification of the cables were sufficiently conservative.

Already known systematic ageing mechanisms, such as the loss of plasticisers and embrittlement of the insulating material, and newly recognised systematic ageing mechanisms are recorded and monitored. If necessary, suitable measures are taken to counteract this e.g. by adapting or expanding appropriate procedures or by replacing cables that have aged to a degree that is impermissible so that no effect on their operability need be assumed. In this context, recent findings are taken into account and currently available cable types with the appropriate qualification level are used as successor parts.

As stated in Chapter 3.1.4.a, as part of ageing management, various cables were replaced as a precautionary measure in order to be able to reliably ensure fulfilment of the demands under LOCA conditions. These included cables with SIR or ETFE insulation in areas with high dose rates. In connection with the replacement of PVC-insulated cables, testing of the insulation resistance has been established for medium voltage cables.

3.2.b Research reactors

At the FRM II, the cables of the neutron flux instrumentation have not yet shown any findings that would have required measures according to Chapter 3.1.4.b.

At the BER II, the ISIs and operating experience show the following:

- The fire protection bandages have no influence on the cable sheathing.
- No ageing effects have been detected on the cabling outside the reactor pool area.
- No ageing effects have been observed on the mineral-insulated metal-jacket coaxial cabling of the power range instrumentation within the reactor pool.
- No ageing effects have been observed on the mineral-insulated metal-jacket coaxial cabling of the intermediate-range instrumentation within the reactor pool.
- The specified cabling between the start-up chamber and the terminal box on the reactor bridge consists of a non-mineral-insulated, polymer-coated coaxial cable. This cable showed insulation problems after a longer service life.

The cause of the insulation problems described was embrittlement of the cable in the connector area on the chamber due to radiation damage. Residual moisture was able to penetrate into these embrittled areas and negatively affected the insulation resistance. The following measures were taken:

- shortening or replacement of the cable and installation of new plugs
- Periodic drying of the free volume in the chamber guide linkage and flushing with nitrogen.
- Improvement of the operating regime to minimise the residence time of the chambers in areas of high radiation

As a result of the measures described, it was possible to reduce these ageing effects and ensure the specified condition in the long run.

At the FRMZ, the operator has no knowledge that any short-circuits or other obvious defects in the cables were found in past cable replacements, even in cases where cables had been subjected to loading over several decades. This is due to the ambient conditions of the cables at the FRMZ and the comparatively low neutron flux of the system (max 10^{12} neutrons $cm^{-2}s^{-1}$ in the core centre).

3.3 Regulator’s assessment and conclusions on ageing management of electrical cables

3.3.a Nuclear power plants

Assessment of the ageing management process for cables

The ageing management programme for cables is correctly described in Chapter 3.

A differentiation of the different cable types was made according to their respective applications. The most important cable materials still in use today are properly named, as are the test methods used in practice to determine the operational or ageing condition of the various cables. Degradation mechanisms that may affect the specified normal function of the cables during their service lives are described in the individual plants' basic reports on their ageing management. The environmental influences to which the various cable types may be exposed at the installation positions in the nuclear power plants and which may cause damage in the long term are described in full in Chapter 3.1.2 and have been correctly assessed in accordance with the current state of knowledge.

During the operation of the plants so far, the measures taken in the context of the ageing management of cables have shown themselves to be suitable for ensuring a quality of the cables that meets the requirements and allows their continued use. Whenever in individual cases unexpected ageing phenomena occurred or were detected, an exchange of information among the different plants was carried out according to best practices, if necessary with the involvement of or under the auspices of the VGB, so that in each case an applicability assessment and, if necessary, plant-specific measures could be derived. Examples of cases with a corresponding knowledge gain are mentioned in Chapter 3.1.4. This way it was possible to counteract, to the extent needed, any systematic ageing phenomena that could have led to an impairment of safety-relevant SSCs in case of demand without prior countermeasures.

Experience with the application of ageing management programmes for cables

Already prior to KTA 1403 taking effect in November 2010, a formal process-oriented ageing management system was implemented under the supervision of the competent authority in the respective nuclear regulatory procedures. In KTA 1403, enveloping requirements for ageing management were laid down, i.a. with the aim of harmonising the different plant-specific procedures. The SSCs of the electrical and I&C systems, also including the cables, are dealt with in Section 4.2 of safety standard KTA 1403.

The holistic ageing management of the electrical and I&C systems is based on the following elements:

- the documentation pertaining to the components (with the quality certificates and engineering documentation, referring to the construction)
- the basic report according to KTA 1403, describing the general regulations relating to the ageing management of a plant
- the operating documentation (i.a. life-cycle records, special databases for keeping track of and assessing whether correspondingly designed components maintain their LOCA resistance)
- the annual status reports according to KTA 1403, which deal with changes in the fundamental regulations regarding ageing management as well as with peculiarities of the past observation period and in which a qualitative and quantitative effectiveness assessment is made

Ageing management in the area of electrical and I&C systems is applied according to the requirements of KTA 1403 to safety-related electrical and I&C system components. In the context of the creation and expansion of the knowledge base regarding ageing management, however, operating experience with cables not important for safety is also used. Regardless of that, it is common practice to include availability-relevant components not important for safety in ageing management. Examples are buried cables of the standby grid connection, if such a type of connection exists at a plant.

Cables are considered to be components with indirect safety relevance whose failure will have controllable consequences due to the plant concept (single-failure concept) as long as there is no reason to suspect any consequences of systematic degradation mechanisms in case of a challenge.

In the various plants, the condition of the cables is usually assessed as part of integral functional tests. Cables that are susceptible to certain ageing phenomena or where changes in material properties may cause unexpected or accelerated functional impairments are subject to separate tests in such cases, such as insulation resistance and loop resistance measurements. In this case, the interfaces to adjacent components (e.g. plug or clamp connections) are usually also assessed. Examples are the 10-kV medium voltage cables for the on-site power system and emergency power supply or the in-core instrumentation cables.

A special group is formed by the cables required to be LOCA-proof. For these cables and lines, an extended type-specific in-service verification procedure in accordance with the requirements of KTA 3706 /KTA 00/ and coordinated by VGB is conducted to demonstrate that the functional properties under LOCA conditions are maintained. The underlying purpose of this procedure is to update qualification data for the remaining operating time, using a more up-to-date knowledge of material data of individual cable types or an extended verification by testing the accident behaviour of cable sections that have already been subject to operational loads.

Furthermore, coordinated by VGB, cable sample depositories are operated for German nuclear power plants during plant operation in areas with high radiation exposure (near the loop line). The cable sample depositories contain an array of samples of cables that are widely used in the plants that are currently still in operation. This array of samples is used i.a. for the in-service verification that the cables maintain their LOCA resistance. The arrangement of a cable sample depository next to reactor coolant line is shown in Chapter 3.1.3.

The cable samples kept in such a depository thus age radiologically at a quicker rate compared with the cables installed in the plant. These pre-aged samples are subjected to further lifetime tests (e.g. applying the criterion of elongation at break and its reduction over time as a consequence of embrittlement by radioactive radiation and the influence of temperature) and serve for the determination of so-called service life curves, from which the remaining useful service life at a given dose rate in the place of installation can be extrapolated. An example of such a service life curve is shown in the illustration in Chapter 3.1.2.

The results of these investigations are included in the Ageing Management programmes of the nuclear power plants within the framework of nuclear regulatory supervision.

Another important element of ageing management is the applicability assessment of findings from other domestic or foreign plants, resulting e.g. from unexpected results of ISIs with indications of systematic phenomena or reportable events. Such results are then assessed in the respective plant-specific supervisory procedures, taking into account the relevant experience and findings, and the measures that may then be deemed necessary are taken. Relevant examples of such cases are mentioned in Chapter 3.1.4.

Effectiveness assessment of the existing ageing management programme

To comply with regulatory requirements, the regulatory authority is informed annually by the licence holder about the results of ageing monitoring for the previous operating cycle. The use of the test methods described in Chapter 3.1.2 allows – as can be seen e.g. from the service life curves – a statement on the remaining useful service life in a particular application. In other cases, trending of specific measuring results is possible if needed, e.g. via trending of insulation resistances of PVC medium-voltage cables.

The annual evaluation of the results of the plants' ageing management programmes confirms the effectiveness of the plant-specific ageing monitoring programme for cables and lines.

The ageing management programme documented in Chapter 3 “Electrical Cables” by the licence holders for nuclear power plants in power operation corresponds fully to the ageing management implemented in the plants. Through operational regulations and the clear definition of responsibilities, the ageing management is permanently integrated into the work processes of the power plants. At the plants, processes, events and measures relevant in connection with ageing management are documented, using suitable EDP software (including the plant management system or life cycle records). This ensures that the relevant processes from plant operation are fully and comprehensively taken into account, and that consequently the component documentation is updated as well. An assessment regarding the relevance of inspection results is carried out at least once a year, irrespective of the assessment of any more recently made findings.

External events (e.g. VGB reports, events from domestic and foreign plants, GRS information notices) are checked by the operators for applicability. Here, the relevance of an event to the ageing management programme is also assessed. As soon as a need for action is identified from this assessment, the measures to be taken are implemented in accordance with the safety requirements.

Main strengths

Ageing management at the plants takes place with consideration of the established processes at the respective nuclear power plant. The measures in the context of these processes (e.g. fault signals, maintenance measures) are managed via the plant management system and, with regard to their relevance for ageing management, are comprehensively and systematically evaluated individually for components or mechanisms. This takes into account the importance of ageing management and ensures that the interactions of the ageing management processes with other processes are recognised and controlled.

The experience feedback e.g. from the evaluation of information notices prepared by Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), from VGB working groups and from national and international technical literature as well as the knowledge gained from the assessment of ageing effects observed in non-safety-relevant components and systems is also included into the knowledge base and assessed as part of the ageing management process.

Annual reporting (e.g. in status reports) makes it possible to comprehend the nuclear power plant operator's results and assessment processes. Taking into account findings from the respective supervisory procedures, it is thus possible to assess the effectiveness of the existing ageing management, irrespective of the operators' statements and remarks.

Through continual improvement based on the PDCA (Plan-Do-Check-Act) cycle and the feedback of experience, the knowledge base can provide the information that is needed for effective ageing management.

Weaknesses identified

The ageing management in the plants is carried out in accordance with KTA 1403 in the sense of a continual improvement process (PDCA cycle) with an updated knowledge base.

There were no weaknesses found in the ageing management process for the ageing management of cables.

3.3.b Research reactors

Within the framework of the operation as licensed for the FRM II, BER II and FRMZ, the competent nuclear licensing and supervisory authorities check all measures specified by the licence holder, applying the graded approach described in Chapter 2.7. After examination by the competent authorities, the above-mentioned measures described by the licence holders are considered to be suitable for the early detection of ageing effects.

4 Concealed pipework

4.1 Description of ageing management programmes for concealed pipework

4.1.1 Scope of ageing management for concealed pipework

4.1.1.a Nuclear power plants

The concealed pipework in German nuclear power plants differ in terms of their laying which are steel pipes

- buried in soil or
- installed in covered trenches, usually made of concrete.

In accordance with KTA 1403, mechanical components are divided into groups M1 to M3 within the framework of ageing management.

Group M1 includes components and parts of the reactor coolant pressure boundary and the external systems the failures of which are specified as being impermissible, as well as plant-specific components and parts the failures of which are not covered by the design of the nuclear power plant.

Group M2 includes safety-relevant mechanical components and parts which do not belong to Group M1. These are, in particular, components and parts that are used to demonstrate the control of accidents and for which the failure of a redundancy based on the redundancy concept is covered by the design concept of the nuclear power plant in the event of the required function. These also include other radioactivity-retaining systems.

Group M3 includes other mechanical components and parts which only have operational functions and are therefore not included in Groups M1 and M2. They are not subject to ageing management in accordance with KTA 1403. Within the framework of the ageing management of German nuclear power plants, concealed pipework are classified with regard to their laying as follows:

- There are no non-accessible systems in Group M1.
- Group M2 systems buried in soil. These are only lines that are required for the control of accidents and residual heat removal in normal operation. In PWR and BWR plants, these are the pipework of the secured service water system. The pipework is partially concealed and buried in soil and partially installed in buildings and accessible from outside.
- Group M2 systems with safety-relevant functions installed in covered trenches, usually made of concrete. In PWR plants, these are the emergency feedwater system pipework of the steam generator. In German PWRs, these pipework in trenches are generally accessible from the outside and passable. Thus, they do not differ from pipework installed in buildings.
- Group M2 systems, radioactivity-retaining or having other safety-relevant functions. These are plant-specifically the chilled water system and radioactive waste water pipework and, if necessary, fire mains for fire protection reasons. These pipework are partially buried in soil not accessible from the outside and partially installed in buildings accessible from the outside.
- Group M3 systems having operational functions. These are in particular the pipework of the main cooling water system and the (non-secured) service water system.

The buried pipework systems of the secured service water system with steel pipes (tag number VE according to the AKZ system (plant and equipment identification system)), Group M2, required for the removal of residual heat after an accident, are thus the most important concealed pipework according to their safety-related significance with regard to the ageing management in German nuclear power plants. In a few cases, the service water system pipework is made from prestressed concrete.

The measures for the ageing management of concealed pipework are described below using the example of the secured service water pipework made of steel. Only the discharge pipes for the controlled discharge of radioactive waste water are designed as buried activity-retaining pipework. The contamination of waste water in the area of these pipes is already below the permissible limit values. In addition, they are designed as guarded pipes/pipe-in-pipe system with leakage monitoring. Furthermore, the transported media (demineralised water) is less corrosive than the secured service water. Due to the guarded pipes/pipe-in-pipe system with leakage monitoring and the transported media, the discharge pipes are of minor significance with respect to their ageing management that is covered by the management of the secured service water pipework as described below.

In German nuclear power plants, the diesel lines of the emergency diesel generator fuel supply required for the control of accidents are not laid in trenches or concrete structures and thus accessible at all times.

Description and task of the secured service water systems

In German nuclear power plants, the secured service water system (VE or PE system) basically has several redundant trains. Its task is to remove heat from the secured intermediate coolant circuit and nuclear intermediate coolant circuit. Conditioned river water serves as coolant of the service water system (see also /VGB 00/). In some nuclear power plants, cell coolers serve train-wise as alternative heat sinks, here, purified raw water is used as coolant.

The operating pressure of the secured service water system is up to 6 bar. The maximum temperature in the inlet line during normal operation is around 30 °C; the temperature in the return line is slightly higher. Normally, the systems are designed for pressures of up to 10 bar and temperatures ranging from 50 °C to 80 °C.

Due to the described task of the secured service water, a large break of the multiple-redundant pipes shall be avoided by the ageing management. The functional capability and the compliance with the protective goals are not questioned due to local leaks.

4.1.1.b Research reactors

In the FRM II, the following two lines were identified, which – with restrictions due to their safety-related significance – were included in the analysis:

- diesel engine plant (fuel line from the tank)
- well water supply (pipes from the well to the buffer tank)

The wastewater pipes to the Isar River are not considered here, because they only contain sampled and cleared water and their leakage would be without safety significance. Both of the above mentioned lines are buried in soil. External visual testing is not possible or only possible with considerable effort. Nevertheless, both lines have a certain safety significance: The well water supply is important for the fire water supply in case of unavailability of drinking and service water and the diesel engine for emergency power supply in case of failure of the external power supply (note: the FRM II is designed in such a way that diesel engines are not required to comply with the protective goals).

In the research reactor plants BER II and FRMZ, no lines have been identified that are considered in the context of this TPR Ageing Management. For this reason, the following subchapters only focus on the FRM II.

4.1.2 Ageing assessment of concealed pipework

4.1.2.a Nuclear power plants

Ageing management of the secured service water

The basis for the definition of ageing management measures in German nuclear power plants is the systematic analysis and assessment of degradation mechanisms. In accordance with KTA 1403, this includes the following:

a) knowledge of the required and current quality state (design, materials and manufacturing)

b) knowledge of the operation to date, including commissioning

c) knowledge of the relevant degradation mechanisms and their prevention

d) protection against degradation mechanisms (design, operation to date)

e) operational monitoring measures

f) considering the current state of knowledge on degradation mechanisms

The aim of ageing management is to prevent large leaks in secured service water systems as a result of ageing mechanisms. The procedure applied for components of Group M2 in accordance with KTA 1403 for the points c) to f) includes in particular the following:

- monitoring the consequences of operational degradation mechanisms at representative positions (e.g. in-service inspections, operational monitoring or laboratory tests)
- considering the findings from the operation of other nuclear power plants
- following the state of knowledge on possible degradation mechanisms

The following, partly plant-specific sources, are used to evaluate the points a) to f):

- manufacturing specifications and standards referred to
- quality certificates
- documents from the design and design assessments
- results from monitoring and their assessment
- results from in-service inspections and their assessment (see Chapter 4.1.3.a)
- results from maintenance measures
- fault/defect reports and their assessment
- overall maintenance inspection reports
- GRS information notices
- reportable events from the operator’s plant and from other German plants
- events from nuclear power plants outside Germany
- national and international research projects (see Chapter 4.1.3.a)
- evaluation of experience by the manufacturers

- exchange of experience among the operators
- contractor reports (VGB system for the assessment of contractors)
- VGB Cooling Water Guideline /VGB 00/

Knowledge of the required and assessment of the current quality condition (a):

The task of the secured service water, as described in Chapter 4.1.1, is the heat transport from the connected intermediate coolant circuits to the heat sink (river or cell cooler) by means of appropriately conditioned river water or, if applicable, raw water as coolant. A sufficient resistance to the corrosive damaging mechanisms described below with adequate margins was ensured in the design of German nuclear power plants for an operating life of 40 years.

The basis for this was provided by decades of experience with cooling water systems in fossil power plants, other industries like e.g. chemical industry and in particular water supply.

This state-of-the-art technology, which has been tried and tested in practice, and the relevant standards and guidelines were applied in the design, manufacturing and laying of the secured service water systems.

This includes the choice of a unalloyed steel pipe with additional material thickness with regard to corrosion, the choice of a suitable exterior coating of buried system areas as passive corrosion protection and, if necessary, the additional use of a cathodic corrosion protection system in the buried area as active corrosion protection. Furthermore, depending on the nominal diameter, this preferably includes a suitable interior coating to minimise corrosion; in the nuclear power plants still in operation, a cement mortar lining was preferably used for the larger nominal sizes (see also DIN 2614 and DIN 2880).

Lines with a cement mortar lining have been in use for more than 100 years and have been increasingly used in Germany since the mid-1960s, especially in the water supply, and have a very good operational reliability. The secured service water pipes were laid in pipe trenches and beddings according to the proven state of the art in order to minimise settlement effects (see also DIN 4142, DIN 2614 and DIN 19630 and DIN 4033). This method of laying includes, among other things, a trenching under soil-specific choice of slope angles or sheet piling walls and the insertion of a water-permeable geotextile film at the bottom of the trench to prevent local settlements. Then a layer by layer filling with fine sand as pipe bedding and filling material in the lower area of the near-pipe area of the trench such that damage to the exterior layer of the corrosion protection of the pipes is safely prevented. Above this, a filling with non-sharp-edged filling material of sand or soil free of stones. Furthermore, in order to minimise later settlements during the entire filling process, a high degree of compaction is carried out layer by layer by means of sludges or mechanically with a surface vibrator, especially repeated in the area of the excavation bottom.

On this basis, the service water systems of the German nuclear power plants were designed and approved site-specifically. The construction specifications of the plant manufacturer Siemens/KWU for the secured service water pipework of the German nuclear power plants have been established on the basis of the above mentioned rules and regulations, assessed during the erection phase, approved by the supervisory authority, the proper implementation of the specifications is monitored by authorised experts.

Thus, the standard of quality of the secured service water pipework implemented in the German nuclear power plants was established according to long established rules and regulations and, according to the current knowledge, complies with the required quality for the remaining operating time. This is explained further in the following.

Materials and dimensions:

The secured service water pipework is made of unalloyed steels. The nominal diameters typically extend up to DN 1200.

Classification of integrity:

Due to their low operating pressures (max. 6 bar) and operating temperatures (max. approx. 30 °C to 40 °C), the secured service water pipes are low-energy systems. This means, that there are no mechanical stresses that can cause a failure of the system on their own e.g. caused by material fatigue and fatigue crack growth. In the event of failure due to local corrosive mechanisms like e.g. shallow pit corrosion (see below), a leak-before-break behaviour is always to be assumed in a low-energy system. This means that local small leaks may occur, but not a spontaneous break.

Coatings and corrosion protection systems:

For the interior and exterior surfaces of the pipework, different coatings or coating systems are used plant-specifically and depending on nominal diameter.

Interior surfaces:

- rubber coating
- galvanising
- plastic coating based on epoxy resin
- plastic coating based on tar epoxy resin
- ceramic epoxy resin
- polyethylene inliner
- cement mortar lining (if necessary with gaps in weld areas, DIN 2614)

Exterior surfaces:

- bitumen coating, partly with fibreglass inlay
- plastic coating based on tar epoxy resin
- embedded in concrete

As already mentioned above, in the case of the German nuclear power plants in operation, interior cement mortar linings (DIN 2614) were preferred for the larger diameter ranges.

In addition to the coatings, systems for cathodic corrosion protection in the buried area are used in some nuclear power plants (see also DIN 30676 and the current DIN EN 12954). Such systems have been in use in Germany since 1906.

<u>Evaluation of the experience gained from the previous operation, including commissioning (b):</u>

There are no new findings on the operational impact on the secured service water systems from previous operation. This includes both, with regard to the mechanical aspect, the laying of pipelines to avoid settlements and with regard to the media aspect, the coolant used.

The operating experience with regard to the ageing mechanisms is described below under “experience feedback”.

<u>Results of the analysis of relevant degradation mechanisms and assessment of precautions to avoid them (c):</u>

Ageing phenomena:

The degradation mechanisms of buried cooling water pipes made of unalloyed steel without or with coatings are common in German and foreign nuclear power plants as well as in fossil power plants and other industrial plants, e.g. in the chemical industry and from water supply for many years.

Due to the design as a low-energy system with low operating temperature, the degradation mechanisms such as material fatigue, creep and thermal ageing that are possible in high-energy systems with high operating temperatures are excluded. There is also no impact of irradiation.

Due to the limitation of flow rates, wear due to erosion as well as erosion corrosion are excluded due to low temperatures and high oxygen content.

Stress corrosion cracking and pitting corrosion are also avoided by material selection and low temperatures.

This leaves the pure corrosion mechanisms. Surface corrosion is reliably controlled by additional material thickness of the unalloyed steel pipes. The following essential mechanisms remain:

Degradation mechanisms of the unalloyed pipelines:

The main degradation mechanisms of unalloyed pipelines are the following two local corrosion types:

- shallow pit corrosion from inside
- possible increase of shallow pit corrosion from inside by microbiologically induced corrosion (MIC)

In contrast to stainless chromium-nickel steels, MIC is not an independent mechanism for unalloyed pipelines, but rather increases the mechanism of shallow pit corrosion. Microbiologically induced corrosion occurs especially when bacterial films can form in relevant quantities in areas of pipelines. As a rule, this is limited to system areas with smaller nominal diameters and, in particular, to branches with small flow rates or stagnant medium.

Local leaks due to shallow pit corrosion of non-coated areas of pipelines are in principle distributed stochastically and do not imply a reduction of material thickness of the pipe cross-section. Thus, it can be assumed that even if individual leaks occur locally in a system section, the integral load-bearing capacity of the line is not affected.

Damage of the coatings:

The main reasons for damage of the interior and exterior coatings of unalloyed pipelines are:

- mechanical damage from outside or inside (no ageing)
- damages of plastic coating from the inside: limited adhesion, blistering, locally exposed surfaces
- damages of cement coatings from the inside, cracking and chipping

Mechanical damage to coatings is not an ageing phenomenon. The consequence of mechanical damage or age-related damage of the coating is local shallow pit corrosion of the pipeline areas exposed to the medium, i.e. either by direct contact or ingress of the medium under the coating.

Damage of the coatings can be attributed to incorrect coating application during the manufacturing phase, mechanical effects during operation and ageing influences.

In addition to inadequate or incomplete coatings, a typical damage sequence in the event of production-related coating defects can be described as follows: The coating material is applied incorrectly during the manufacturing process. As a result, it peels from the pipe wall in the course of the operating lifetime, rust forms underneath the coating thus pushing it further away from the pipe wall. Leaks in the coating cause the cooling water to reach the surface of the base metal and initiate the further corrosion process.

<u>Experience feedback: operating experience (b) and Consideration of the current state of knowledge on the degradation mechanisms (f):</u>

The operating experience with the buried secured service water pipework in German nuclear power plants is very good, especially with regard to nuclear power plants which are still in operation.

In several older German nuclear power plants, which are no longer in operation today, repeated damage in the form of small leaks or drip leaks in ferritic service water pipes was found. These were non-coated, tar epoxy resin coated, rubberised or galvanised pipes. The damage was caused at the interior surface of a pipe. The main findings were presented in the GRS information notice 2007/02. Here, information from the GRS information notice 2005/06 on microbiologically induced corrosion of components of service water systems was also taken into account.

Based on the GRS information notice 2007/02, a comprehensive inventory and status assessment of the secured service water systems in the German nuclear power plants was carried out by the operators and evaluated by authorised experts consulted by the nuclear supervisory authority. In the course of these measures, representative inspections and visual testing were carried out on the buried systems by means of pigging systems from the inside and, digging up the pipes from the outside, with positive results.

It was shown that the properly laid pipelines do not show any external damage relevant for the tightness even after many years of use. Precondition is that in the course of current earthworks, there is no external damage to the pipelines and the associated trenches and pipe beds. This is the subject of appropriate administrative measures, but not of the ageing management.

Leakage due to damage from the inside of buried pipelines with larger dimensions (DN ≥ 400) occurred only in older nuclear power plants, which are no longer in operation today. In the nuclear power plants still in operation, buried pipelines are designed only with cement mortar linings. Leakages of buried piping with cement mortar linings have not yet occurred.

In an older German PWR, specific follow-up examinations of the removed old pipes were carried out as part of a preventive replacement measure (see Chapter 4.1). Based on the positive results achieved for the existing quality, local wall thickness measurements were also carried out on the remaining lines from outside after digging up some parts at random. All in all, these measures showed that the actual damage to the non-coated pipes, which had been in operation for more than 30 years without cathodic corrosion protection, was significantly lower than previously assumed and that an integral failure of the pipes was not to be expected. The replacement measure was then terminated and the redundancies that had not yet been replaced were left.

4.1.2.b Research reactors

In the FRM II, the systems concerned are tested by means of integral visual testing (see Chapter 4.1.3.b) and chemical analysis of the content (diesel fuel). During visual inspections, the following must be considered /FRM 14/:

- mechanical damages (friction points, visible deformations, breaks)

- bolt connections (loosening, screw locking)
- connections of measuring points and instrument lines
- insulations (joints, incrustations)
- component displacements (pipe displacements, foundations, anchorages, structural attachment)
- supports, spring and constant hangers, sufficient freedom of displacement
- free displacements of component support structures, taking into account the operating deflection cold/warm, if necessary, the dynamic deflection and the related safety margins required by KTA safety standards
- leakages, leakage indications in the surroundings, contamination
- residues of test equipment or adhesive residues
- corrosion of non-insulated components

The following are also applied:

- standards: for well water pipes according to specification KS D 7350/FRM II among others DIN 1626, DIN 2460, DIN 30670; for diesel pipes according to specification KS D 7350/FRM II among others DIN 6608 Part 2 and Technical Rules for Flammable Liquids
- manufacturer's documentation and specification
- internal and external operating experience

4.1.3 Monitoring, testing, sampling and inspection activities for concealed pipework

4.1.3.a Nuclear power plants

In addition to the operational monitoring measures of the secured service water systems with detection of

- operational parameters, as pressure, temperature and flow rates and
- media parameters of the conditioned cooling water, as pH-value and essential water constituents from laboratory analyses,

monitoring of ageing effects of degradation mechanisms in German nuclear power plants by means of

- inspections e.g. by walkdowns of accessible areas of pipework installed in buildings representative for buried pipelines and
- in-service inspections (one to two years inspection lot at least once in eight years per redundancy), including non-destructive testing (among others, visual inspection from inside by means of pig) see below, and functional testing.

If necessary, additional measures include e.g.

- integral leakage monitoring of the systems by means of differential pressure measurements and
- visual inspections of the buried areas for signs of large leaks.

The non-destructive testing and the definition of the appropriate testing techniques are described in more detail below.

Non-destructive testing:

The objective of the non-destructive testing is to ensure the required quality in further operation (based on KTA 3211.4 /KTA 13a/, see Figure 3-1).

Coating systems used in the respective nuclear power plant depend on the nominal diameter of the piping. Depending on the coating system and considering the results of the GRS information notice 2007/02, see Chapter 4.1.2, in-service inspections to control the effects of the ageing phenomena described in Chapter 4.1.2 for the German nuclear power plants were established on a case-by-case basis in the corresponding test manuals. The following selection of suitable testing techniques is based on the testing techniques described in the KTA safety standards (e.g. KTA 3211.4 /KTA 13a/, Tab. 2-1).

Test locations:

In principle, it is not possible to carry out in-service inspections of the pipes in the buried area from the outside and, due to the risk of mechanical damage by the earthworks required for this purpose, cannot be recommended. It is also not necessary, as described in Chapter 4.1.2, due to the lack of an active ageing mechanism and under consideration of the positive operating experience. The only thing to be avoided here is mechanical damage to the coatings during earthworks; this shall be ensured by administrative measures.

Thus, the effects of ageing degradation of the service water pipework are monitored from the inside.

For systems where the buried area is protected by a corrosion protection system, representative tests are carried out in pipework areas installed in buildings.

Test procedures:

The following test procedures for the detection of the effects of internal degradation mechanisms were evaluated with regard to their suitability for representative tests within the framework of ageing management:

- visual inspection of pipework installed in buildings from the outside
- visual inspection from the inside with devices depending on coating and accessibility (e.g. videoscope, endoscope, pipe pig, camera)
- ultrasonic wall thickness measurement from the outside, pipework areas installed in buildings
- ultrasonic wall thickness measurement from the inside depending on coating and accessibility
- eddy current testing from the outside, pipework areas installed in buildings
- eddy current testing from the inside depending on coating and accessibility (pipe pig)
- radiographic testing from the outside using shadowgraph techniques
- functional testing and pressure tests of the systems

Further information on the testing techniques listed here can also be found in the final report on the BMU research project SR 2521 /BMU 07/.

Suitability of the test procedures and test techniques:

In the assessment of these test procedures and test techniques with regard to their suitability for testing buried pipelines and obtaining reliable test results, the following aspects were considered:

a) Visual inspection from the inside:

Under consideration of the technical test conditions, a basic statement on the general condition of the interior surface can be made by means of the visual inspection. Damage that would affect the integrity of the pipeline can be detected even if the surfaces are not cleaned (e.g. extensive corrosion or damage of the coating due to spalling or chipping.

In case of mechanical cleaning of the internal surface, it is important to note that

- damage to the coating cannot be excluded,
- loosened deposits can be transported into the heat exchanger and
- damage to the naturally developed protective layer can occur in non-coated areas.

b) Ultrasonic wall thickness measurement:

Wall thickness measurements are suitable for the detection of systematic damage (e.g. reduction of non-coated pipes or non-coated weld areas). For detecting local damages that can lead to a small leakage, the use of this procedure is limited.

Testing from the outside, does not require the removal of the exterior coating when using suitable probes and should even be avoided if the coating is adherent and sound-permeable.

c) Eddy current testing:

Eddy current testing from the inside can only be carried out by means of pigging systems. The ferritic pipe materials require pre-magnetisation, which leads to an increase in the total weight of the test equipment. The use of such test equipment is prohibited due to the risk of damage to the coating.

The process-related limitations of eddy current technology in connection with the real geometry of the lines lead to restrictions of the test statement/result. Welds cannot be evaluated due to the differences in permeability.

Eddy current testing can thus be excluded.

d) Radiographic testing:

Radiographic testing in non-buried areas is suitable for the detection of systematic damages. For detecting local damages that can lead to a small leakage, the use of this procedure is limited.

Implementation in nuclear power plants

Based on the results of the plant-specific inventories assessment within the frame of the GRS information notice 2007/02, see also Chapter 4.1.2, in 2015, the VGB carried out a joint evaluation of the non-destructive testing procedures for secured service water systems on the basis of the current state of knowledge and including the experiences gained in nuclear power plants of the manufacturer Siemens/KWU in other European countries. As a result of this evaluation, the VGB recommends two alternative procedures for in-service inspections depending on the coating system with the aim to prevent extensive failure of the pipe walls:

Case 1: Pipelines or pipework areas non-coated inside:

For non-coated pipelines or pipework areas, ultrasonic wall thickness measurement of test sections accessible from the outside is a suitable test method to obtain a representative statement about the presence of a local damage. The test results can also be transferred to buried pipework areas with the same design of the coating system.

This procedure is particularly recommended for non-coated weld areas in lines with cement mortar coatings, as an alternative to visual inspection from the inside. The non-coated areas cover the coated areas with respect to degradation. This ensures that the global failure of the lines can be excluded.

Case 2: Pipelines coated inside:

In principle, visual inspection from the inside is a suitable test method for excluding extensive wall failure. The procedures to be used (inspection of the removable pieces, limited inspection of large nominal diameter, visual inspection with a borescope, videoscope or videoscope pigging system) and the extent of the inspection shall be specified plant-specifically and depending on nominal diameter.

Mechanised cleaning of the interior surface must be avoided due to the risk of damage to the coating and the problem of sludge removal; local manual cleaning may be required for direct visual inspection in individual cases.

These recommendations for suitable periodic non-destructive testing have currently been implemented, evaluated and approved in the plant specific test instructions of German nuclear power plants. This means that in the nuclear power plants, one of the following two non-destructive testing procedures is currently being used system-specifically:

- ultrasonic wall thickness measurement (each redundancy at least once in eight years) of test section accessible from the outside or
- visual inspection (spot checks each revision; each redundancy at least once in eight years) from the inside by inspection of removable pieces, limited inspection of large nominal diameters, visual inspection with a borescope, videoscope or videoscope pigging system

These periodic non-destructive testing measures implemented in the nuclear power plants together with operational monitoring measures and periodic functional and leak testing ensure the early detection of the effects of the degradation mechanisms. The documentation is provided in the form of inspection records which are evaluated by the respective employee responsible for the system. Thus, a trend analysis is ensured.

4.1.3.b Research reactors

In the case of FRM II, the following in-service inspections are relevant in accordance with the testing manual (PHB):

- in-service inspection “visual container inspection” (diesel tank), five-years inspection intervals with authorised experts participation (Manual for conventional tests, implementation with standard test instruction 0-SP.04.0 and container log-book),
- in-service inspection “quality of diesel fuel, determination of density, flash point and water content according to DIN 51750”, semi-annual inspection intervals (Manual for internal tests, implementation according to /DIN 91/)
- in-service inspection “integral visual inspection of the well water supply”, annual inspection intervals (Manual for internal tests, no separate test instruction, implementation with standard test instruction 0-SP.04.0) and event-related (no recurrent inspections) e.g. camera inspection of pipework from the horizontal well to the buffer tank

4.1.4 Preventive and remedial actions for concealed pipework

4.1.4.a Nuclear power plants

Preventive measures:

As an active preventive measure in German nuclear power plants, the use of cathodic corrosion protection systems in the buried area since the construction of the plant, as already referred to in Chapter 4.1.2, should be mentioned.

Remedial actions:

Remedial actions to restore the required quality level are carried out in the event of detected leaks.

The detection of local leaks is usually limited to the non-buried areas of the secured service water system. As described in Chapter 4.1.3.a, such local leaks are limited to system areas with smaller nominal diameters and, in particular, to branches with low flow rates or stagnant medium.

In relation to the larger, buried nominal diameters, industrial practice has shown that, in particular, non-coated pipelines made of unalloyed steels have a generally limited operating life due to ageing-related mechanisms such as shallow pit corrosion. The secured service water systems are designed for an operating life time of about 40 years. In some older German nuclear power plants, condition-oriented replacement measures of complete buried lines have already been carried out. Here, the non-coated pipes of the plant installation were replaced by new pipes coated with cement mortar lining on the inside which, as already described in Chapter 4.1.2.a, ensure the best possible resistance to ageing mechanisms according to today's knowledge.

The basis for the decision on such condition-oriented replacement measures is the occurrence of local leaks during visual inspection of the interior surface for condition assessment. The unsightly “rust-pitted” surface, especially of non-coated pipelines or pipework areas, can lead to over-interpretations of the real integral damage condition of the remaining metallic wall. Since the rust layers growing up in oxygen-containing water are of various thicknesses and have uneven surfaces, but with increasing thickness form an effective oxygen diffusion barrier. Thus, the initially high corrosion rate decreases asymptotically to very small values due to the strong decrease in oxygen transport through the top layer as a rate-determining step.

See also the positive operating experience described in Chapter 4.1.2.a with regard to the subsequent inspections of removed old pipes.

4.1.4.b Research reactors

In the case of FRM II, the concealed pipework is monitored by means of the measures described in Chapter 4.1.3.b. If there are any findings, these will be evaluated. So far, there have been no findings related to the components considered here. If there are any findings and the evaluation requires a need for action, the specified condition of the affected component is restored by repair or replacement according to e.g. the maintenance rules (Operating manual of the FRM II Part 1 Chapter 3).

4.2 Licensee's experience of the application of AMPs for concealed pipework

4.2.a Nuclear power plants

The ageing management of concealed pipework in German nuclear power plants has already started within the framework of the design, the selection of materials and coating systems, the manufacturing processes as well as the production and installation.

The basis for this was provided by long-term operating experience with cooling water systems in fossil power plants, other industries like e.g. chemical industry and in particular water supply. With regard to their operational stresses, the medium transported (conditioned river water or raw water) and the possible ageing mechanisms, the secured service water pipework of the German nuclear power plants basically does not differ from cooling water pipework in other areas of application described.

By implementing well-established material and coating concepts, sufficient precautions have already been taken during the construction of the plant to control the essential ageing degradation mechanisms of the buried, secured service water system pipework.

Since commissioning of the nuclear power plants, the known degradation mechanisms have been systematically monitored and evaluated by means of operational monitoring and appropriate in-service inspections. This enables the early detection of changes in the systems and the reliable prevention of failure due to a large break in the pipes. Within the scope of the condition-based maintenance, appropriate replacement measures are carried out to eliminate leaks in the event of occurrence of locally limited leakages.

The entirety of the measures implemented in the nuclear power plants ensures the required quality of the secured service water pipework and its maintenance. The consistent monitoring and assessment of the general knowledge level provides protection against potential degradation mechanisms. The effectiveness of the measures to ensure the required quality during operation is subject to regular review. This includes

- methods and scope of monitoring the causes and consequences of possible operational degradation mechanisms,
- fault/defect reports ("internal event evaluation"),
- national and international operating experience (evaluation of events, exchange of information through VGB and WANO) and
- other new findings.

Due to damages in older German nuclear power plants, which are no longer in operation today in the form of smaller leakages or drip leaks, a comprehensive inventory and status assessment of these systems has been carried out in recent years (see also Chapter 4.1.2.a). Based on the information exchange of the operators within the VGB, recommendations for suitable in-service inspections in the nuclear power plants currently in operation were derived and implemented in the test manuals (see also Chapter 4.1.3.a).

To date, no relevant additional information on the known or new degradation mechanisms in these systems has been obtained from the plant-specific acquisition of information and evaluation of potentially ageing-related events as part of the implementation of an ageing management in accordance with KTA 1403.

In all cases, the detected locally limited leaks were of little safety significance. There was no extensive damage that could cause an integral failure of the pipework. Thus, the damage did not have a significant impact on the safe operation and the availability of the secured service water systems required for the control of accidents.

According to estimates by the German operators, there is no need for additional action, justified from a safety point of view, beyond the currently implemented ageing management today. Furthermore, based on the positive operating experience to date, it is currently not possible to deduce any need for technical action for pipework with cement mortar coating.

In general, from the point of view of the German nuclear power plant operators, it should be noted that due to the overall concept within the framework of the ageing management of concealed pipework in German nuclear power plants, sufficient precautions have been implemented for the early detection of known and potential degradation mechanisms.

4.2.b Research reactors

So far, there have been no findings on the concealed pipework of the FRM II. No further measures were therefore required.

4.3 Regulator's assessment and conclusions on ageing management of concealed pipework

4.3.a Nuclear Power plants

Assessment of the ageing management process of concealed pipework

The ageing management of concealed pipework is correctly described in Chapter 4. The information on the structure of the pipelines of the German nuclear power plants corresponds to the actual condition. In the context of planning and design, it was correctly referred to the many years of operating experience with cooling water pipes in conventional power plants or in the chemical industry. As a result, it was possible to take precautionary measures against the significant ageing degradation mechanisms already during the construction of the German nuclear power plants by means of well-established material and corrosion protection concepts.

The degradation mechanisms affecting the concealed pipework, which have been identified in the context of ageing management, are fully described in Chapter 4.1.2 and have been correctly evaluated according to the current state of knowledge.

The operational monitoring measures and in-service inspections used for ageing monitoring of the concealed pipework are carried out on the basis of KTA 3211.4; and are fully listed in Chapter 4.1.3.

The measures carried out within the framework of the ageing management of the concealed pipework are appropriate to ensure the required quality of the concealed pipework.

The ageing management described by the operators in Chapter 4 "Concealed pipework" corresponds in its entirety to the ageing management practiced in the German nuclear power plants in operation.

By maintaining the current practice of ageing management in German nuclear power plants, the compliance with the required quality of the concealed pipework can also be ensured in future.

Experience with the application of ageing management for concealed pipework

The ageing management for concealed pipework of German nuclear power plants is based on the requirements of KTA 1403 for mechanical systems and components. The safety-related concealed pipework is assigned to Group M2.

The buried systems with safety-related significance include the secured service water system for the control of accidents and residual heat removal, and other plant-specific systems with safety-related tasks (e.g. fire mains, waste water pipes). The pipes are made of steel and, in some cases, of prestressed concrete. The pipework of the secured service water is multiple-redundant.

The buried systems are operated at low energy levels (p<20 bar, T<100°C). Due to the design concept (materials, strength, manufacture) and measures within the framework of ageing management, large leaks are excluded. Within the supervisory procedure, it was demonstrated that locally limited leaks do not affect the functional capability and the compliance with the protective goal.

For the purposes of corrosion protection, steel pipes are coated with corrosion protection (e.g. plastic coating, cement mortar lining, bitumen) on their interior and exterior surfaces. In some plants buried steel pipes are additionally provided with cathodic corrosion protection. In order to monitor the effects of degradation mechanisms, periodic functional and leak tests are carried out. Visual inspections of the interior surface or, in some German nuclear power plants, inspections of the exterior surface, as well as ultrasonic wall thickness measurements of externally accessible test sections are carried out. The results of the ultrasonic wall thickness measurements taken at the test sections accessible from the outside can be used to determine the wall thickness of the non-accessible areas.

The periodic inspections and the inspection intervals are specified in site-specific inspection instructions.

Shallow pit corrosion of the interior surface facilitated by microbiologically induced corrosion and mechanical damage to the coating of the interior surface were identified as relevant degradation mechanisms in the German nuclear power plants. These were locally limited damages; no extensive damage has been detected so far.

The damages were recorded, assessed and repaired according to specified repair concepts.

The evaluation of the periodic testing and examination confirms that the known degradation mechanisms are recognised in good time (no catastrophic failure) and that appropriate preventive measures can be initiated.

Effectiveness assessment of the ageing management

In order to meet regulatory requirements, the operator submits annual reports to the supervisory authority on the results of the ageing management for the preceding operating cycle.

In the German plants, suitable IT software (e.g. operation management system) is used to record all safety-relevant components and systems, processes, events and measures. The relevance for ageing management is assessed at least once a year. This ensures that relevant processes from plant operation are fully and comprehensively taken into account.

External events (e.g. VGB reports, events in national and international nuclear power plants, GRS information notices relevant for the ageing management programme) are reviewed by the operators for applicability to their own plants. The operators submit annual reports to the respective regulatory body on the activities carried out in the reporting year, including findings and their evaluation. The assessment of transferability of events and measures that may have been initiated in other plants is also part of the annual reports.

The annual assessment of the ageing management results for German nuclear power plants confirms the effectiveness of the ageing monitoring programmes for concealed pipework.

Main strengths

Ageing management in German nuclear power plants is based on the implementation of the established processes in power plants. The measures within the framework of these processes (e.g. maintenance measures) are controlled via the operation management system and are comprehensively and systematically evaluated in relation to ageing management on a component or mechanism-specific basis.

The annual reporting provides a transparent and comprehensible presentation of the assessment processes of the nuclear power plant operators. The documentation on ageing management related to the concealed pipework is updated continuously.

Weaknesses identified

The ageing management in German nuclear power plants is based on KTA 1403 as part of a continuous improvement process (PDCA cycle) with an updated knowledge basis.

No weaknesses have been identified in the ageing management process for the concealed pipework ageing management.

4.3.b Research reactors

Within the framework of the operation as licensed for the FRM II, the competent nuclear licensing and supervisory authority checks all measures specified by the licence holder, applying the graded approach described in Chapter 2.7. After examination by the competent authority, the above-mentioned measures described by the licence holder are considered to be suitable for the early detection of ageing effects.

5 Reactor pressure vessels

5.1 Description of ageing management programmes for RPVs

5.1.1 Scope of ageing management for RPVs

5.1.1.a Nuclear power plants

The design of the reactor pressure vessels (RPVs) was based on a design that is favourable to stress and meets the material, production and testing requirements.

Reactor pressure vessel design of a pressurised water reactor (PWR)

Figure 5-1 shows the actual RPV design of a PWR. The RPV is designed for the following loads:

- design pressure 175 bar
- design temperature 350 °C

The system pressure of the RPV is 157 bar, the coolant inlet temperature of the RPV is 292 °C and its outlet temperature is 326 °C.

The RPV of the Konvoi-Plants KKI-2 and GKN-2 is made from fine-grained steel 20 MnMoNi 5 5, all the others are made of fine-grained steel 22 NiMoCr 3 7. These two steels that have been improved in their chemical analysis basically correspond to the fine-grained steels ASTM A 508 Class 2 and Class 3 used internationally in PWRs and boiling water reactors (BWRs).

The RPV consist of the lower section, the RPV upper head removable for refuelling and the stud bolts (material 26 NiCrMo 14 6) and nuts (material 34 CrNiMo 6) which detachably connect the two parts together.

The lower section of the RPV consist of a forged vessel flange ring where the eight forged main coolant nozzles (four inlet and four outlet nozzles) are welded to, two cylindrical forged rings (upper and lower ring), a single-piece forged transition ring, and a forged bottom dome.

The nozzles are welded onto the reinforced integral vessel flange ring. As a result, the nozzles are relatively low stressed. Furthermore, this design also offers a good accessibility for the non-destructive test. The relatively rigid upper head and vessel flanges ensure a flexurally rigid construction.

The RPV bottom of PWRs is designed without penetrations.

The RPV upper head consists of a single-piece forged upper head closure flange and a single-piece forged upper head dome.

The RPV upper head nozzles (control rod and core instrumentation nozzles) of PWRs are not welded to the RPV upper head by means of dissimilar metal welds, as is the case with foreign nuclear power plants, but screwed into it and sealed inside it by means of an austenitic seal weld to the internal cladding.

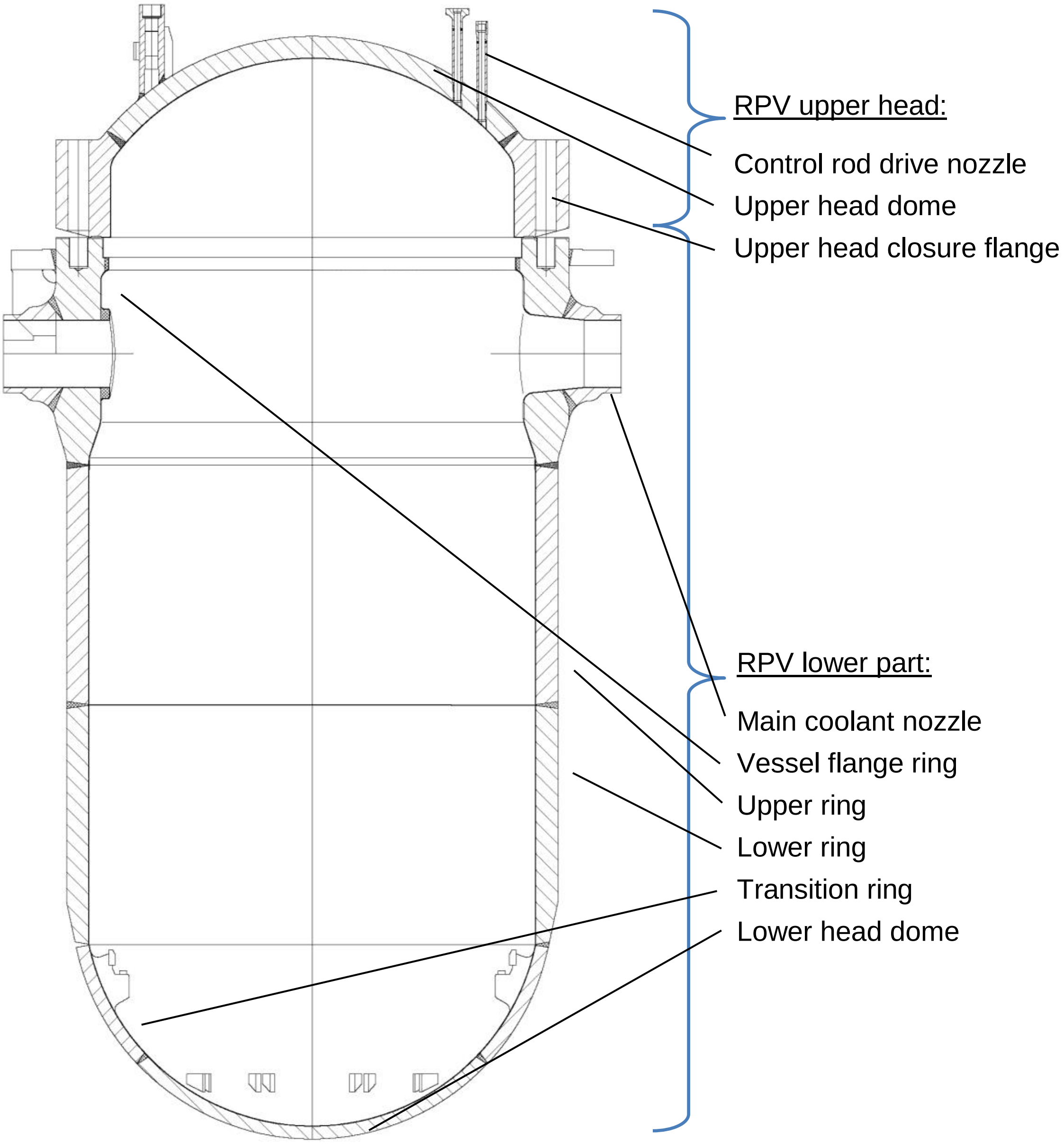


Figure 5-1 Cross section of a PWR reactor pressure vessel

Due to this design, there is no nickel-based dissimilar metal weld (J-groove dissimilar-metal weld). Moreover, thermal expansion of the upper head and nozzles therein does not affect the integrity of the nozzles during the start-up and shutdown.

All RPV large components are forgings that are connected by circumferential welds.

All surfaces of the RPV wetted by the primary medium have a two-layered austenitic cladding (material X10 CrNiNb 18 9). The cladding serves to avoid surface corrosion caused by the coolant and thus, to minimise mobilisable activation products in the primary circuit for radiation protection reasons.

The RPV supports itself by its support brackets against the building over a steel structure cast into the concrete.

The German PWRs are designed with a relatively large water gap between the core and the RPV wall, in order to keep the toughness reduction of ferritic beltline materials caused by neutron irradiation during operation as low as possible.

In PWRs and BWRs, two concentrically arranged metal O-ring seals (material Inconel 718 silver-plated) are used between the RPV upper head and the RPV lower section for sealing; after each opening of the upper head, the O-ring seals are replaced. The outer O-ring serves as an additional safety feature having a sealing function in case of an inner ring leakage. To control the tightness of the inner ring, the space between the two rings is monitored for any leakages.

The flange bolts are clamped and loosened hydraulically. A hydraulic clamping device allows a controlled simultaneous clamping of all bolts.

RPV design of a boiling water reactor (BWR)

Figure 5-2 shows the actual RPV design of a BWR. The RPV is designed for the following loads:

- design pressure 86.3 bar
- design temperature for the cylindrical part without the vessel flange and the reinforced bottom ring: 310 °C
- design temperature for the flange connection, RPV upper head and bottom: 300 °C

The operational pressure is 69.6 bar, the operating temperature is 286 °C, the feed water inlet temperature is 215 °C.

The RPV is designed as a cylindrical container with a spherical lower head and a hemispherical upper head. The upper head is flange connected to the vessel. To absorb the deformation forces, the cylindrical vessel part at the transition to the lower head is designed as a reinforced bottom ring.

The RPV lower head contains the nozzles for

- control rod drive (N),
- core flux measurement (O),
- blank nozzle (H),
- coolant recirculation pumps (P),
- core and pump instrumentation (K) and
- coolant temperature measurement (L).

The cylindrical part contains the nozzles for

- main steam (B),
- feedwater (C),
- high pressure feed-in (D),
- coolant extraction (E),
- shutdown pipe line (J) and
- pressure and level measurement (A11-A16).

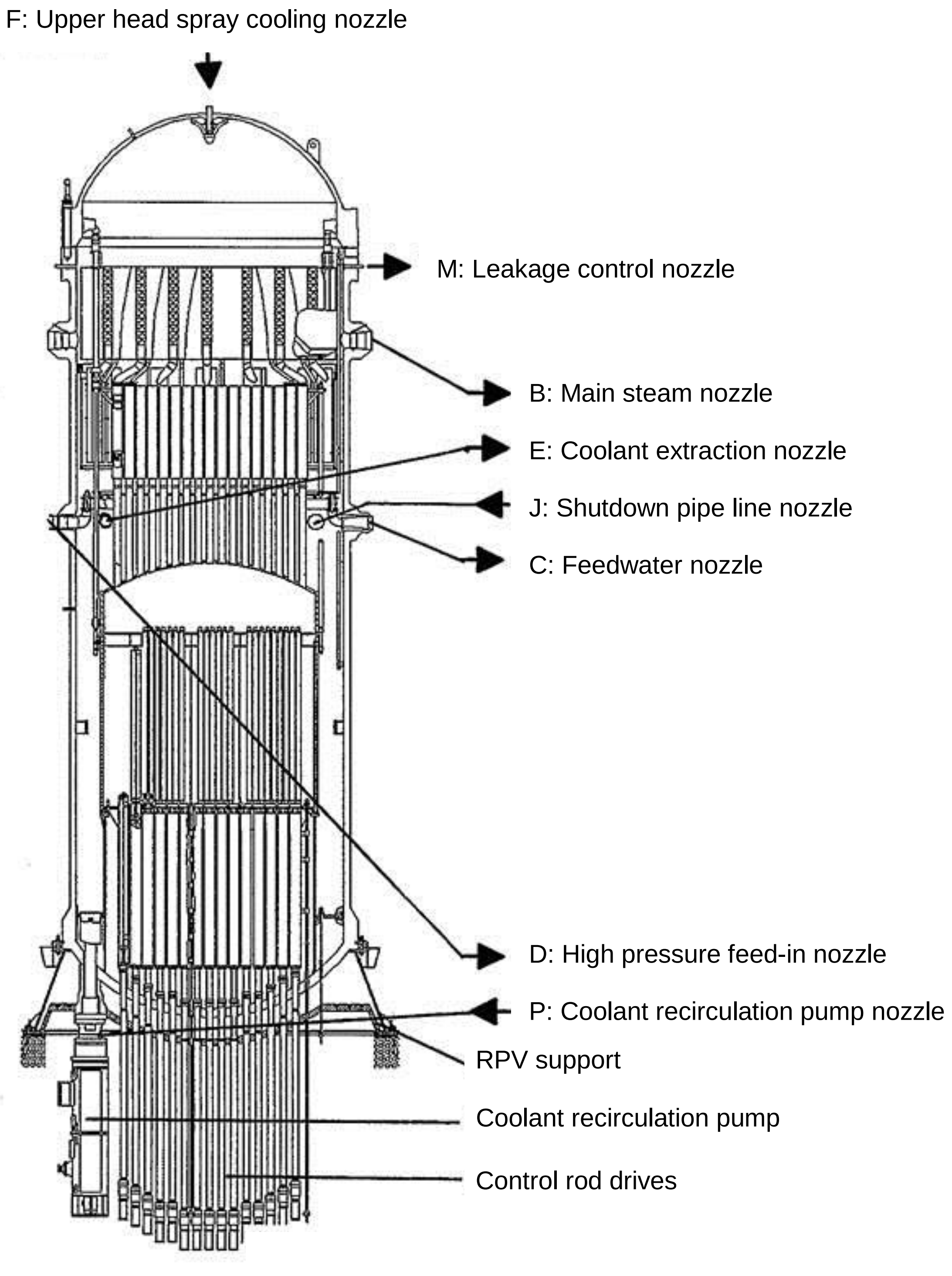


Figure 5-2 Cross section of a BWR reactor pressure vessel

The vessel flange ring contains the nozzles for the

- pressure and level measurement (A3/4/6/7/9),
- steam flow measurement (A2/5/8/10) and
- leakage control of the flange seal (M).

The nozzles welded into the upper head are required for

- wide-range level measurement (A1) and
- upper head spray cooling system (F).

All surfaces of the RPV wetted by the primary medium have a two-layered austenitic cladding (1.4551) unless the components themselves are made of austenitic materials. The cladding serves to avoid surface corrosion caused by the coolant and thus, to minimise mobilisable activation products in the primary circuit for radiation protection reasons.

The pressure-retaining RPV wall consists of the following product forms:

- Vessel
  The bottom consists of:
  - the lower head made of pressed forging segments connected by longitudinal welds,
  - the bottom ring forged as a seamless ring.

  The cylindrical part consists of:
  - 6 seamless forged rings and
  - a seamless forged vessel flange.
- Upper head
  The upper head consists of:
  - the seamless forged upper head closure flange,
  - the upper head segment ring made of dished plate segments and connected by longitudinal welds and
  - the upper head dome made of dished plate.

The RPV is made of heat-resistant fine grained steel 22 NiMoCr 3 7.

The large nozzles B, C, D, E, J in the cylinder part and the F nozzle in the upper head are forgings and are made of the same base material as the RPV. These nozzles are inserted into the vessel wall and welded to it by means of butt welds.

To avoid peak stresses, the transition radii to the RPV wall were designed in accordance with the rules and regulations. The required cut-out reinforcements are placed in the nozzles.

The nozzles C, D and F through which media which do not have reactor temperature flow during operation or during transient processes are provided with a thermo-sleeve.

The transition from the thermo-sleeve to the nozzle is located outside the pressure-retaining wall and outside the cut-out reinforcements at a point where the additional temperature stresses can easily be absorbed.

The nozzles for the coolant recirculation pump P are also forgings made of the same base material as the RPV. They are attached to the outside of the RPV bottom by means of a full penetration weld. The assemblies of the axial-flow-pumps located outside the RPV are flanged to the nozzle head. On the inside, the pump nozzles are lined with an austenitic steel sleeve.

The nozzles for the control rod penetrations N, the core flux measurement O, the temperature measurement L and the pump and core instrumentation K are made of austenitic steel 1.4550 and joined to the RPV bottom by means of a butt weld with full penetration. The housings of the control rod drives and the core flux measurement are made of the same material as the nozzles and are welded into the nozzles N and O. The same material was chosen to avoid temperature stresses due to different expansion coefficients.

The weld seam by means of which the control rod drive housings are welded into the nozzles is a butt weld.

The core flux measurement housings are welded to the RPV nozzles by means of a fillet weld.

All small nozzles for pressure and water level measurements and instrumentation A are designed with DN 50. These nozzles are made of austenitic steel inserted through the RPV wall and welded to it from the inside.

The RPV is supported by a support skirt. The weight of the RPV with installations, reactor core, water, reactor coolant pumps and control rod drives is transferred by the support skirt into the foundation of the reactor building.

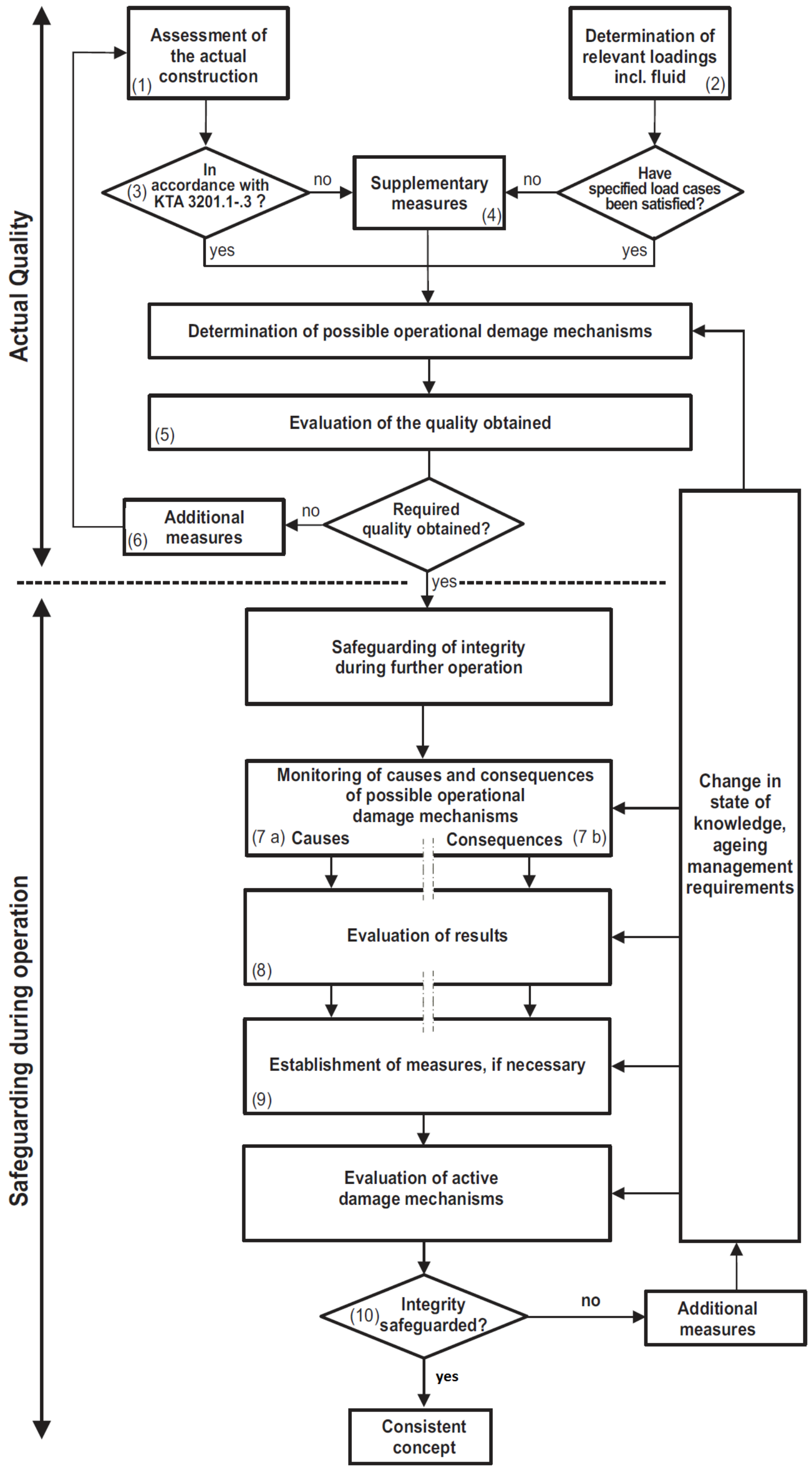
Owing to the design of the German BWRs, the water gaps are larger than that of the PWRs, thus, the RPV neutron irradiation and its toughness reduction in the beltline is even lower than that of the PWR.

<u>Ageing management for RPVs</u>

The RPV is a component, failures of which are specified as being inadmissible. For the RPV, the requirements for ageing management of the group M1 according to KTA 1430 are fulfilled. This includes

- knowledge of the required and actual quality state of the RPV (compliance with design, material and production requirements),
- knowledge of the operation-to-date including commissioning,
- knowledge of the relevant degradation mechanisms and their prevention,
- examination procedures to ensure protection against relevant degradation mechanisms (design, operation-to-date),
- a fracture-mechanical assessment of postulated failures (assessment against brittle fracture),
- operational monitoring measures and,
- consideration of the current state of knowledge.

For the German RPVs, this approach was implemented in an overall concept. The procedure shown in Figure 5-3 corresponds to KTA 3201.4 /KTA 16b/ and has been implemented in all nuclear power plants.



(1) Criteria
- material
- welding
- design
  - shape
  - weld seam
  - dimensions
- existing relevant indications

(2) Data
- operation
  - monitoring
  - state of knowledge
- Incidents
  - state of knowledge
- regarding
  - stress analysis
  - fatigue analysis
  - fracture-mechanic assessment

(3) This check is only possible for components > DN 50.
This test criterion is deemed to have been met if the deviations from the safety standards KTA 3201.1, KTA 3201.2 or KTA 3201.3 do not require additional stipulations for in-service inspections or operational monitoring.
For components ≤ DN 50 the same applies accordingly in which case concrete requirements for the actual construction shall be individually laid down for each plant.

(4) Additional measures
- more detailed analysis
- materials testing
- special tests and examinations
- monitoring
  - loads
  - chemical water quality
- adaptation of the in-service inspection concept

(5) Evaluation criteria
- stresses
- fatigue analysis usage factor
- safety margin to failure
- crack growth values
- corrosion

(6) Measures
- modification of
  - process technology
  - hardware

(7 a) Causes
- mechanical and thermal loadings
- water chemistry

Measures regarding monitoring of causes (examples)
- measuring sensor
- measuring point (local, global loads)
- recording (monitoring of data)

(7 b) Consequences
- cracking
- reduction of wall thickness

Measures regarding monitoring of consequences (examples)
- non-destructive examination
- monitoring for loose parts
- leakage monitoring
- destructive tests

(8) Evaluation criteria
- stresses
- fatigue analysis usage factor
- safety margin to failure
- crack growth values
- corrosion

(9) Measures
- modification of
  - process technology
  - hardware
- operational monitoring
- special tests and examinations
- adaptation of the in-service inspection concept

(10) Criteria
- possible operational damage mechanisms safeguarded
- new active damage mechanisms

Figure 5-3 Overall concept of safeguarding the component integrity during operation /KTA 16b/

Overall concept for continuous assurance of component integrity during operation

For the RPV, in addition to the plant-related experiences, the further development of the general knowledge with regard to ageing-related aspects is also pursued. The applicability of findings from these evaluations is subject to continuous and comprehensive analyses and, if necessary, additional measures are taken. For the RPV, this analysis and assessment has been carried out within the framework of the specified integrity concept since commissioning by involving experts and authorities. The evaluation of experience takes into account new findings from:

- the GRS information notices,
- the reportable events from the operator's plant and from other German plants,
- the events from foreign nuclear power plants,
- the national and international research projects,
- the evaluation of experience by the manufacturers,
- fault/defect reports and their assessment,
- the in-service inspections (ISIs) and other maintenance measures,
- the exchange of experience among the operators (VGB working panels and VGB working groups) and
- the quarterly reports on the research projects of the BMUB.

The monitoring measures and the in-service inspections to be defined for monitoring the causes and consequences of operational degradation mechanisms are derived from the above mentioned overall concept for the occurring operational loads. The areas subject to higher loads are taken into account. These are, in particular, those parts of the RPV which

- are subjected to higher stress as compared to the general stress intensity level, also taking into account the frequency (fatigue) or
- can be subjected to radiation-induced property changes of the RPV materials.

For the RPV, ageing management involves both comprehensive monitoring of the causes and comprehensive monitoring of the consequences of possible degradation mechanisms.

5.1.1.b Research reactors

This chapter is not applicable to German research reactors.

5.1.2 Ageing assessment of RPVs

5.1.2.a Nuclear power plants

The following, partly plant-specific documents, are used to evaluate the relevant degradation mechanisms and affected areas of the RPV:

- manufacturing specification
- applicable safety standards: KTA 1403 /KTA 17/, KTA 3201.1 /KTA 16c/, KTA 3201.2 /KTA 13b/, KTA 3201.3 /KTA 07/, KTA 3201.4 /KTA 16b/ and KTA 3203 /KTA 01/
- quality certificates
- documents from the design and design assessment (e.g. analysis of the mechanical behaviour of the RPV)

- design fluence
- safety analysis on resistance to brittle fracture/pressurised thermal shock analysis
- monitoring results and their assessment
- in-service inspection results and their assessment
- maintenance measures results
- fault/defect reports and their assessment
- overall maintenance inspection reports
- results from the irradiation surveillance programme
- GRS information notices
- reportable events from the operator's plant and from other German plants
- events from foreign nuclear power plants
- national and international research projects
- experience evaluations of the manufacturer
- exchange of experience among the operators
- contractor reports (VGB system on the assessment of the contractor)

Numerous research projects have been focusing on the RPV safety and the long-term behaviour within the framework of the design, selection of materials, production processes and the manufacturing of RPVs of German nuclear power plants.

Research programmes

At the time of the design and construction of the first German RPVs, research projects were initiated in order to achieve high-quality execution and a high quality for the long-term operation by selecting suitable materials and production processes. Already in 1971, an "action committee on underclad cracking" (AK UPR) dealt with the measures taken during production to prevent relaxation cracks in the RPV material 22 NiMoCr 3 7 /LAU 13/. Within the framework of the projects "research programme - RPV high priority programme 22°NiMoCr 3 7" and "immediate programme 20 MnMoNi 5 5", the materials 22 NiMoCr 3 7 and 20 MnMoNi 5 5 (ASTM A 508 Class 2 and 3) used in nuclear technology worldwide have been thoroughly analysed for the use in German RPVs and improved in their chemical composition and heat treatment /LAU 13/. Recommendations for the stress-relief heat treatment and the final annealing were derived from the extensive investigations.

A "Research project on Component Safety" was carried out in two phases. In the phase I from 1977 to 1983, the safety margins as well as optimisation possibilities in RPV production were investigated and demonstrated. In the subsequent phase II, which lasted until 1997 due to the long-term irradiation tests of the materials, tools were made available to assess the safety of the reactor components even after a long period of operation /LAU 13/.

Besides these investigations, the improvements of the non-destructive test methods (NDT) are also to be emphasised, as the performance of the NDE methods has been tested, improved and verified on 1:1 test objects under real test conditions and with realistic flaws; the findings have been incorporated into the production and acceptance tests to ensure a high level of flaw detectability.

With the extensive findings gained from these research programmes /LAU 13/ with regard to

- material selection and weld joints,
- test technology,

- material-mechanical investigations,
- long-term behaviour and
- non-destructive tests

for German RPVs

- more resistant chemical material compositions have been specified, the quality of the processing and welding safety has been improved, thus, ensuring the material's resistance to cracking in long-term operation,
- constructions avoiding weld seams with high operational stresses and complex residual stress conditions have been developed,
- changes in the material properties caused by neutron irradiation have been minimised by the design and selection of the base and weld metal alloys, and good accessibility for periodic NDEs has been created and
- manufacturing processes have been recommended to ensure a high quality of the components.

The results of these research projects have already been implemented into the manufacturing specifications of the RPVs in operation and in the KTA safety standards.

Further investigations on irradiation behaviour of German RPV materials were carried out by the plant manufacturer Siemens/KWU within the framework of the research project "CARISMA" and "CARINA" /VGB 13a/ funded by the German government. These research projects have contributed to the expansion of the experimental database for irradiated German RPV materials.

The results of the CARINA project are representative of the irradiated German RPV base and weld metals and also show very low ductile brittle transition temperatures RT_{T0j} < 0 °C in the fluence range, which is significantly higher than the assessment fluence of 1.5 to $8 \cdot 10^{18}$ cm^{-2} of the German BWRs and PWRs.

Ageing-relevant degradation mechanisms

The following degradation mechanisms are to be postulated in principle for the base metal, weld metal and heat-affected zone of the RPVs due to the experience feedback from the operator's plant and from external LWRs in foreign countries and are limited to their relevance in the following on the basis of the material selection, construction, design and manufacturing quality of the German RPVs.

1 Stress corrosion cracking

The corrosive degradation mechanisms under mechanical stress, which are described below, can generally cause crack formations in metallic materials. Intergranular stress corrosion cracking of austenitic chromium-nickel steels and primary water-induced stress corrosion cracking of nickel alloys are basically two similar types of crack formation that can emerge as a result of the simultaneous occurrence of a sensitive material condition, tensile stresses and oxygen-containing high-temperature water. These degradation mechanisms are controlled by suitable water chemistry, material selection, mechanical design and avoidance of peak stresses. The primary water stress corrosion cracking (PWSCC) mechanism of nickel alloys did not occur in the RPVs of German PWRs and BWRs. In German plants, no dissimilar metal welds were used in connections with large nominal diameter (DN) and wall thickness. Thus, unfavourably designed dissimilar metal welds with high manufacturing-related residual stresses were avoided. In foreign nuclear power plants, systematic damages were caused by PWSCC at such fluid-wetted nickel-based dissimilar metal welds. Furthermore, the dissimilar metal weld connections of the in-core instrumentation nozzles in German BWRs are welded with austenitic root passes. This reliably prevents the fluid from contacting the underlying nickel weld metal of the filler and final passes.

The intergranular stress corrosion cracking in the weld seam area of austenitic steels is not to be expected at RPVs of German PWRs due to the strongly reducing water chemistry, the selected material and the manufacturing process used. Cold formed components, depending on the degree of cold forming, shall be heat treated in accordance with the requirements of KTA 3201.1. Otherwise, it shall be demonstrated by means of a review of the cold forming process that the material properties specified in KTA 3201.1, if necessary, also in consideration of welding – with regard to the intended use of the part – are met. For austenitic components carrying reactor water that are subject to operating temperatures equal to or exceeding 200 °C in BWR plants, a qualification of the processing procedures is required; with the aim, in addition to the requirements to ensure low heat input into the component and to avoid the entry of impermissible impurities (e.g. halogens), to ensure only slight cold forming and only slight hardening in the surface-near area for the fluid-wetted surface.

In case of German BWRs, this degradation mechanism is controlled by the points listed above. Notwithstanding the above, in BWRs and PWRs, this degradation mechanism is periodically controlled by means of water chemistry monitoring and in-service inspections at weld seams and the adjacent base metal.

2 Fatigue

Material fatigue can occur as a result of periodically changing mechanical and/or thermal stresses. The occurring stresses in German BWRs and PWRs are known from the continuous monitoring of the relevant stresses (temperature and pressure changes) of the reactor coolant system. The occurring stresses are recorded and evaluated on a regular basis. With regard to fatigue, the main base metal areas in PRWs and BWRs are the RPV upper head studs, the ligaments of the RPV upper head of a PWR, the ligaments of the RPV bottom of a BWR and the nozzle inner edges of the RPV inlet and outlet nozzles for sizes > DN 250. These areas are included in the in-service inspections for both PWRs and BWRs. In addition, the weld seams of the RPV pressure-retaining wall are subject to in-service inspections for both reactor types.

3 Embrittlement (radiation-induced ageing)

Neutron irradiation of ferritic metals causes a change in the material properties leading to an increase in mechanical strength with simultaneous decrease in toughness. This effect must be taken into account for neutron fluences $> 10^{17}$ n/cm^2 (neutrons per square centimetre) (see also /KTA 01/). In the case of German RPVs, this only applies to the near-core area (“beltline”) for both types BWRs and PWRs and comprises the upper and lower cylindrical shell course with its circumferential weld in the PWR and the three middle shell courses with its circumferential welds in the BWR. In order to assess the influence of the irradiation on the material properties, for all German RPVs, there is a surveillance programme with accelerated irradiation of specimens made of base and weld metals from a production control test piece made in the course of manufacturing from cut-off overlengths of forging rings.

4 Thermal ageing

The fine grain steels 20 MnMoNi 5 5 and 22 NiMoCr 3 7 used in German RPVs are resistant to thermal ageing due to their chemical composition and microstructure. The operating temperatures are below the range in which thermal embrittlement can occur; this usually requires temperatures exceeding 350 °C. In addition, long-term ageing samples of older German RPVs were used to verify that thermal ageing is not relevant.

Thus, within the framework of the ageing management programme of the German RPVs, the following relevant degradation mechanisms are monitored

- stress corrosion
- fatigue
- irradiation-induced changes in material properties

The causes and consequences of stress corrosion cracking and fatigue are monitored and evaluated in accordance with KTA 3201.4 /KTA 16b/. If indications are determined by non-destructive tests reaching or exceeding the evaluation limit in accordance with KTA 3201.4 /KTA 16b/, they shall be referred to as findings. First of all, a comparison with the results of the previous test must be made. In case of indication changes, the results of previous tests must also be taken into account in order to be able to draw conclusions about the temporal course of the changes. If this comparison confirms that a new recordable indication was found or an old one has grown larger becoming a relevant indication, an investigation of causes and a subsequent safety analysis is required. The results of the cause investigation and the safety analysis are decisive for the determination of the permissible limit, i.e. decide whether the finding may be retained.

The consequences of the irradiation-induced changes in material properties are monitored using the irradiation surveillance programme in accordance with KTA 3203 /KTA 01/.

The following acceptance criteria for the degradation mechanisms listed above must be adhered to:

- fatigue: compliance with the admissible fatigue values in accordance with the rules and regulations must be demonstrated
- stress corrosion cracking: the design and manufacturing must ensure that operational crack corrosion cannot be assumed. In case of operational findings, the causes would have to be determined and eliminated. Furthermore, corrective measures, in particular water chemistry changes, should be taken
- irradiation-induced changes in the material properties: compliance with the assessment fluence of 5 to $8 \cdot 10^{18}$ n/cm^2 for PWR and $1.5 \cdot 10^{18}$ n/cm^2 for BWR, compliance with the “End of Life” (EOL) reference temperature RT_{NDT} or RT_{Limit} determined on the basis of the assessment against brittle fracture and the requirements for toughness must be demonstrated

All of the above-mentioned degradation mechanisms have already been taken into account in the design and manufacture of the German RPVs.

Experience feedback

On the basis of the experience feedback from the operator’s plant and from other LWR plants in Germany and abroad, the following essential ageing phenomena, which are relevant for the RPVs, have been observed in the last decades.

a) Primary water stress corrosion cracking (PWSCC) – e.g. Virgil-C-Summer event

In foreign BWR and PWR plants, some cases of intergranular stress corrosion cracking have occurred in dissimilar metal welds between the buttering and the ferritic base metal (e.g. reactor coolant line (RCL) nozzles in the Virgil-C-Summer and Ringhals nuclear power plants, see also GRS information notice 2001/05; and the RPV upper head nozzles in numerous PWR plants e.g. Bugey-3 1991 and Davis Besse, see also GRS information notice 2003/02). The cause is primarily an inappropriate completion of the dissimilar metal welds (site fabrication, multiple repairs).

As described above, there are no pressure-retaining dissimilar metal welds with large nominal diameters at RCL nozzles in PWRs and feed-water lines in BWRs in German plants, as well as no welds with large volumes and unfavourable stresses at RPV upper head nozzles in PWRs.

The results of in-service inspections of fluid-wetted dissimilar metal welds with nickel weld metal on components and systems adjacent to the RPV (e.g. pressure tubes of a PWR and safe ends of a BWR) do not reveal any operational changes.

The dissimilar metal welds at the control rod nozzles located above the RPV (compound pipe/flange) are not fluid-wetted as these are covered by austenitic cladding. The eddy current tests of the control rod nozzles confirm that there are no operation-related abnormalities at these dissimilar metal weld joints.

No further measures are required beyond the existing in-service test provisions in KTA 3201.4 /KTA 16b/.

b) Boric acid-induced surface or shallow pit corrosion

Boric acid-induced surface and shallow pit corrosion occurs on ferritic components of PWRs if these are boric acid-wetted as a result of leaks, for example. Such damages occurred as secondary damages resulting from primary damages of dissimilar metal welds at upper head nozzles due to the primary water stress corrosion cracking in US-American PWRs (see also GRS information notice 2003/02). The event in Davis Besse is not directly applicable to German PWRs due to the different design of the upper head nozzles. Nevertheless, mechanically performed visual in-service tests of the RPV upper heads from the outside were implemented in all German PWR plants. Prior, to the Davis Besse event, the inner side of the RPV upper head was already subject to visual in-service tests in all German nuclear power plants. Visual tests of the RPV upper heads on the inner side and outside and visual tests of the RPV bottom on the outside are carried out in all BWR plants.

In BWR plants, no boric acid is used within the RPV.

There are no indications of boric acid-induced damage at RPVs of German PWR and BWR plants.

c) Volume displays with nearly “laminar” indications in the ferritic base metal

In the Belgian PWR plant Doel 3, an ultrasonic straight beam testing of the base metal areas of the cylindrical shell course was carried out during the Revision 2012 for the first time since commissioning as part of the in-service ultrasonic testing of the RPV. During this ultrasonic testing, a large number of nearly “laminar” indications were detected in the volume of the ferritic base material of the cylinder shell courses, the so-called hydrogen flakes. Hydrogen flaking can occur when cooling large forgings. This phenomenon has already been investigated within the framework of the research project on component safety. The experience gained from this project has been incorporated into the German rules and regulations and implemented in the manufacturing of the German RPVs /LAU 13/.

According to the current state of knowledge, the nearly laminar indications within the volume are manufacturing defects that have not changed during operation and therefore cannot be classified as ageing phenomena.

Based on the findings of the research project on component safety, the non-destructive production tests have already been adapted within the framework of the German RPV production. Within the scope of the German production tests, for example, the test positions and their number, the angles of incidence used and the recording level were significantly increased compared to the requirements of the ASME code valid at that time. Figure 5-4 shows a comparison of the differences in ultrasonic testing according to ASME and the ultrasonic testing procedure in Germany. Table 5-1 shows a comparison of the recording level.

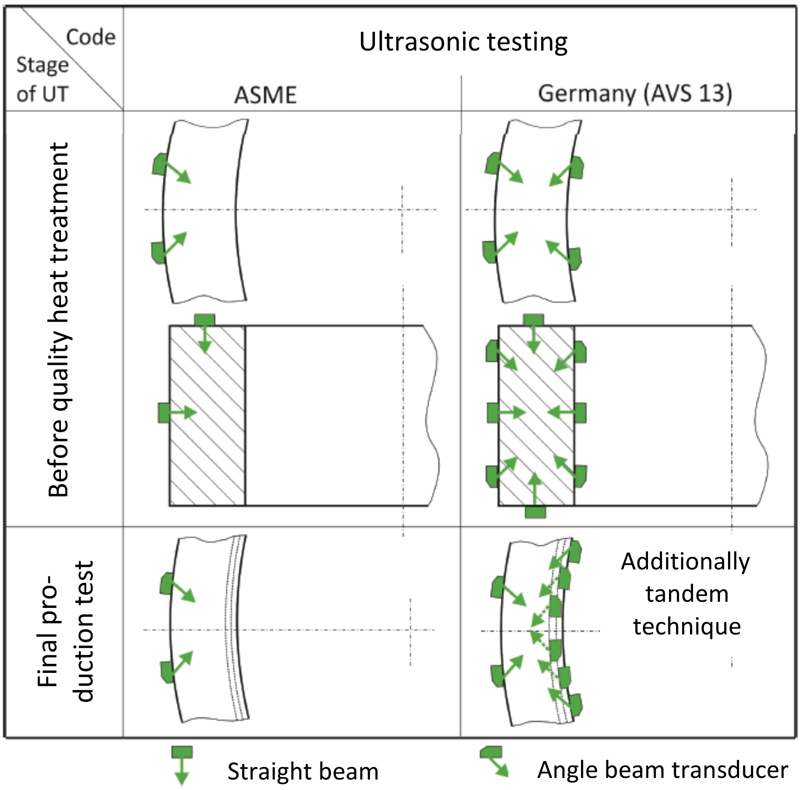


Figure 5-4 Comparison of the requirements of German production tests (ultrasonic testing according to manufacturer's specification AVS 13) with the requirements of the ASME code valid at that time /ERH 15/

Table 5-1 Comparison of the recording limits /ERH 15/

<table>
<tr><th colspan="2">Product</th><th colspan="6">Flanges and tube sheets</th></tr>
<tr><td colspan="2">Detection limit
(flaw size or equivalent diameter at 2 MHz)</td><td colspan="6">0.5 mm at 200 mm
0.8 mm at 400 mm
1 mm at 800 mm</td></tr>
<tr><td rowspan="3">Recording limit</td><td>ASME</td><td colspan="6">• flaw echo ≥ 10% of noise level
• interconnected flaws ≥ 2 times diameter of transducer
• laminar flaw ≥ 25 mm long
• alterable indications ≥ 5% of noise level, 25 mm long
• accumulation of flaws ≥ 5 in cube of 50 mm side length</td></tr>
<tr><td rowspan="2">AVS 13</td><td>Wall thickness [mm]</td><td>15 – 30</td><td>30 – 60</td><td>60 – 120</td><td>120 – 250</td><td>> 250</td></tr>
<tr><td>Flat Bottom Hole [mm]</td><td>1.5</td><td>2</td><td>3</td><td>4</td><td>6</td></tr>
</table>

The review of the manufacturing documentation of the German RPVs has shown that no additional measures are required. In order to confirm this, however, one-off representative special tests were carried out on the cylindrical shell courses of the RPV in all operating German nuclear power plants. In the course of this, the cylinder shell courses were tested in a representative selected area by using ultrasonic systems (straight beam search unit).

No changes were detected compared to the results of the manufacturing tests. No further measures are required.

5.1.2.b Research reactors

This chapter is not applicable to German research reactors.

5.1.3 Monitoring, testing, sampling and inspection activities for RPVs

5.1.3.a Nuclear power plants

Operational monitoring measures and in-service inspections are carried out to monitor the causes and consequences of possible operational degradation mechanisms. These are tests specified in KTA 3201.4 /KTA 16b/, such as

- non-destructive tests – ultrasonic testing (UT), eddy current tests (ET),
- visual testing (VT),
- pressure and leak tests,
- vibration monitoring,
- loose parts monitoring and
- leakage monitoring.

Monitoring of the medium conditions and stresses of the RPV is carried out by

- monitoring of the water quality,
- monitoring of the stresses (e.g. pressure, temperature),
- determination of the actual cycle-dependent neutron fluences, and an
- irradiation surveillance programme according to KTA 3203 /KTA 01/.

The details of the implementation and evaluation of these measures, as well as the involvement of the supervisory authority and experts, are laid down in the testing manuals (including checklists and related test instructions). Details on the individual tests (test scopes, test intervals, preparation and performance of tests, evaluation of the test results) are specified in KTA 3201.4 /KTA 16b/ and are implemented in the nuclear power plants accordingly. These measures are summarised below.

As an additional measure within the framework of the German ageing management, findings from maintenance measures, repair reports, fault/defect reports and their assessment, etc. are regularly evaluated with regard to possible ageing effects, which could also affect the Group M1 components.

Non-destructive tests (UT/ET)

<u>Ultrasonic (UT) and eddy current testing (ET)</u>

For monitoring the consequences of degradation mechanisms, non-destructive tests are carried out at weld seams and selected base metal areas of the pressure-retaining wall of the RPV in accordance with the plant-specific test instructions and KTA 3201.4 /KTA 16b/. The test methods and techniques are chosen in such a way that operational failures and their potential orientations are recorded.

Nowadays, non-destructive tests of the RPV (bottom, upper head and connecting elements are basically performed as mechanised examinations. The test covers the inner and the outer surface with its near-surface areas and also the volume, if required by KTA 3201.4 /KTA 16b/. The requirements and performance of the UT/ET are specified in DIN 25435-1 /DIN 14a/ and KTA 3201.4 /KTA 16b/ and have been implemented accordingly in the nuclear power plants.

RPV bottom

In the PWR, all circumferential welds of the RPV bottom and all nozzle attachments are inspected for longitudinal and transverse flaws using mechanised ultrasonic systems on the RPV inner side. The test area comprises the weld seam, the heat-affected zone and the adjacent base metal area. The weld seams are inspected over the entire length and volume including the surfaces with their near-surface areas. In addition, all internal nozzle edges are tested by means of UT for radial defects and the adjacent area of the nozzle neck for longitudinal and transverse defects. The inspection of the base metal on the inside includes the surfaces with their near-surface areas to exclude cracks.

In the BWR, all circumferential and longitudinal welds in the bottom of the vessel, insertion welds of the vessel nozzles B, C, D, E, J and the internal nozzle edges, the weld seams of the nozzle P at the vessel bottom, the bottom ligaments of the nozzle area in the RPV bottom and all circumferential welds between the welding stubs and the nozzles (D, E and J) are each inspected from the outside.

The weld seams are inspected over the entire length and volume including the surfaces with their near-surface areas. The internal nozzle edges are inspected by means of UT for radial defects and the adjacent area of the nozzle neck for longitudinal and transverse defects. The inspection of the base metal on the inside includes the surfaces with their near-surface areas to exclude cracks.

RPV upper head

In PWRs, mechanised inspections are performed of the weld seam between the upper head dome and the flange outside for longitudinal and transverse flaws. The test area includes the weld seam, the heat-affected zone and the adjacent base metal area. The weld seam is inspected over the entire length and volume, including the surfaces with their near-surface areas. The ligaments in the nozzle areas of the RPV upper head are inspected by means of UT for radial defects.

In BWRs, the ultrasonic testing of the RPV upper head includes all circumferential and longitudinal welds, the weld seams of the Nozzle F and the attachment welds of the lifting lugs at the RPV upper head.

The in-service ultrasonic tests of the RPV bottom are performed every five years for all German RPVs designed, constructed and manufactured in accordance with the safety standards KTA 3201.1 /KTA 16c/ to KTA 3201.3 /KTA 07/ or for which a re-evaluation has shown that deviations from the above mentioned safety standards do not require any additional specifications for in-service inspections and operational monitoring. Otherwise, a test interval of four years is applied (see also Tables 5-2 and 5-3).

The dissimilar metal welds with a single-layer cladding at the upper head nozzles (core instrumentation and control rod nozzles) are inspected for longitudinal and transverse flaws by means of eddy current on the inside. Within five or four years, at least 10% of the seams at the upper head nozzles are inspected.

The RPV stud bolts and nuts are inspected for flaws transverse to the thread axle by means of eddy current. The RPV blind holes are inspected for flaws transverse to the thread axle by means of eddy current. For stud bolts, the test covers the entire length including the threaded section, for blind holes also the entire thread length and for nuts the threaded section and the stressed end face. The surfaces with their near-surface areas are inspected.

The test of the RPV bolted joint is carried out in accordance with the KTA 3201.4 (see also Tables 5-2 and 5-3).

The Tables 5-2 and 5-3 show the minimum scopes of testing, test procedures and test intervals on the pressure-retaining RPV wall in accordance with KTA 3201.4 /KTA 16b/.

Table 5-2 Non-destructive tests of the reactor pressure vessel according to KTA 3201.4 /KTA 16b/

Item to be inspected	Test procedure / Test technique	Flaw orientation	Extent of testing	Test interval [1]
Longitudinal and circumferential welds	UT [2]	I and t	all weld seams, entire length, entire volume as well as the surface areas with their near-surface regions	5 years (4 years)
Nozzle attachment and set-in welds of the following systems: PWR: reactor coolant line BWR: live steam line, feedwater pipe, deluge system, head spray cooling system, reactor water clean-up system, axial pumps	UT	I and t [8]		
Nozzle-to-fitting welds (dissimilar welds) in BWR plants	UT [3]	I and t [4]	surface areas with their near-surface regions	
Connecting areas of thermal sleeves in BWR plants	UT or selective VT [12]	I in the case of UT any in the case of VT	due to different designs the test extent shall be specified for each individual plant	
Nozzle inside edge ≥ DN 250 [5]	UT [2]	r	surface areas with their near-surface regions of the entire inside edges of all nozzles	
		I and t	adjacent area in nozzle pipe in the case of BWR plants	
	selective VT	any	surfaces of inside edges of representative nozzles	
Ligaments in nozzle fields	UT [7]	r	all ligaments with respect to surface areas with their near-surface regions as well as the centres of ligaments	
	selective VT	any	outside surface	
Inner surface	integral and selective VT [12]	any	representative locations, in particular of the - RPV cover - belt-line area of the RPV cylinder - nozzles - RPV bottom end the test extent shall be specified for each individual plant	
Screw bolts	UT or MT or ET	transverse to the bolt axis	surface areas with their near-surface regions of all bolts, entire tensioned length including the threaded regions [9]	Within 5 years (4 years) [6] at least 25 % of the bolts with the corresponding threaded blind holes, nuts and washers, however, at three successive test intervals of 5 years (4 years) 100 % shall be tested. Alternatively, the test may be performed at intervals of 10 years (8 years) [6] where 100 % each shall be tested.
	selective VT	any		
Threaded blind holes	UT or ET	transverse to the thread axis	surface areas with their near-surface regions of all blind holes, entire thread length	
	selective VT [10]	any		
Nuts	selective VT or ET or UT	- VT: any - ET and UT: transverse to the thread axis	threaded region and loaded end face (contact surface) of all nuts	
Washers	selective VT	any	both contact surfaces as well as the surface of the washer hole	

Item to be inspected	Test procedure / Test technique	Flaw orientation	Extent of testing	Test interval [1)]
Attachment welds	Agreements shall be made because of the differing design details. The type and extent of the tests shall be incorporated in the test instructions.			
Auxiliary welds	MT or UT	The requirements shall be specified in accordance with 5.2.1.1 (4).		
Outside surface	integral and selective VT [12)]	any	representative locations, the test extent shall be specified for each individual plant [13)]	5 years (4 years) [11)]

Abbreviations for the test procedures and techniques are explained in **Table 2-1**.
l : longitudinal flaw t : transverse flaw r : radial flaw (e.g. for nozzle inside edges or ligaments in nozzle fields)

1) See clause 5.3 (7) as regards the applicable inspection interval.
2) Where a confirmation of the cladding integrity is required in the brittle fracture analysis, the testing level for examining the cladding shall be adjusted to 4.2.3.3.3 (9).
3) Head spray nozzle: selective VT on fluid-wetted surface instead of UT.
4) In the case of welded joints provided with Ni-alloy weld metal on the fluid-wetted surface, an examination for transverse defects shall be performed from both sides additionally to the examination for longitudinal defects. This examination is also required if between the Ni-alloy weld metal and the fluid-wetted surface an austenitic root ≤ 3 mm is provided.
5) In the case of nominal diameters of the connecting pipe less than DN 250, the requirement for in-service inspections shall be reviewed from case to case.
6) VT of stud bolts (where accessible), nuts and washers after each unbolting of bolted joints.
7) The test requirements shall be laid down for PWR and BWR plants in dependence of the test objective.
8) On axial pumps only examination for longitudinal defects.
9) Selective VT only where accssible, if stud bolts are not disassembled.
10) For BWR plants: if stud bolts are disassembled for operational reasons.
11) PWR cover: integral VT for traces of leakage during each overhaul.
12) The test procedures / test techniques to be used shall be specified for each individual plant.
13) For pipe connections with nickel alloyed weld metal see cl. 5.2.1.6 (2) c).

Table 5-3 Non-destructive tests of the control rod nozzle at the pressure-retaining wall KTA 3201.4 /KTA 16b/

Item to be inspected	Test procedure / Test technique	Flaw orientation	Extent of testing	Test interval [1)]
Circumferential welds PWR [2)] Dissimilar weld on cover nozzle	ET	l and t	Inner surface of representative welds on 10 % of pipes in due consideration of accessibility	5 years (4 years)
Circumferential welds on pressure pipes CW no. 1 [3) 4)]	UT or RT or PT or ET	l and t	Inner surface	
CW no. 2 [3)], CW no. 3, CW no. 4	ET	l and t	Inner surface of representative welds on 10 % of pipes in due consideration of accessibility	
Circumferential welds BWR Connecting weld on control rod nozzle	UT	l	Connecting welds of 4 pipes of control rod drive housing [5) 6)]	10 years (8 years) [6)]
Circumferential welds of control rod drive housing	UT	l	Outside surface of the connecting welds of 4 control rod drive housings [5) 6)]	
	ET	l and t	Inner surface of the connecting welds of 4 control rod drive housings [5) 6)]	
	selective VT	any	Inner surface of the connecting welds of 4 control rod drive housings [5) 6)]	

Abbreviations for the test procedures and techniques are explained in **Table 2-1**. l : longitudinal flaw t : transverse flaw

1) See cl. 5.3 (7) as regards the applicable inspection interval.
2) These cover the welds with nickel-alloyed weld metal of the core instrumentation and control rod nozzles as well as of the venting nozzle.
3) dissimilar metal weld
4) Inspection only if pressure pipe is dismounted and latch units are pulled out.
5) Where more than 4 control rod drives are disassembled for operational reasons, the extent of testing on the welds of these control rod housings shall be laid down for each individual plant.
6) The extent of testing shall be laid down such that two welds each are examined at inspection intervals of 5 years (4 years).

Acceptance criteria

In the case of periodic non-destructive tests, only qualified testing techniques are used the suitability of which has been confirmed by the authorised expert. The adjustment of the sensitivity level for both ultrasonic and eddy current testing is performed at reference blocks. According to KTA, the depth extent of the reference defects depends on the wall thickness of the component. The sensitivity level of the ET and UT for the surface with its near-surface areas of the pressure-retaining wall is adjusted by means of a three millimetres deep reference defect in the base metal. For testing cladded RPV areas, a cladded reference block shall be used. For the evaluation of indications, the cladded reference block contains further reference defects in the cladding and in the base metal.

For the stud bolts and nuts, a three millimetres deep reference defect is used for the adjustment of the sensitivity level and for the weld seams of the upper head nozzles a two millimetres reference defect is used.

The tests are carried out according to the plant-specific test instructions and the corresponding test specifications.

Inspection history and trend analyses

Indications the test signal of which corresponds to that of the reference defect plus a sensitivity allowance of 6 dB shall be registered (recording level). In case of recordable indications, a comparison is made with regard to changes. For this purpose, a comparison of current test results and the results of the previous in-service inspection is made.

This comparison reveals changes below the acceptance level. The evaluation of the measuring results was carried out in accordance with KTA 3201.4 /KTA 16b/ and did not reveal any changes in the entire tested area within the scope of measuring tolerance.

Based on the results of the periodic non-destructive tests by means of UT and ET, no further measures were required. There are no integrity-relevant indications for the RPV.

Visual testing (VT)

Visual testing is carried out regularly at the RPV to verify the proper condition of the pressure-retaining wall surfaces and the bolt connections. The type and scope of the visual testing at the RPV are carried out in accordance with the requirements of KTA 3201.4 /KTA 16b/, DIN 25435-4 /DIN 14b/ and the plant-specific test instructions.

The VT is carried out in accordance with the requirements of KTA 3201.4 /KTA 16b/ as

- integral visual testing to evaluate the general condition of the components or
- selective visual testing, as a local visual testing for the unambiguous detection of specific characteristics.

Integral and selective visual testing is carried out at representative areas of the lower part from the inside. These tests are carried out by means of a remotely operated vehicle (ROV). At the RPV upper head, mechanised integral and selective visual testing is performed at the outside and inside surface. Selective visual testing is performed at RPV bolt connections. The test areas include all outer surfaces and the thread surfaces.

The following points must be observed during visual testing of the RPV:

- mechanical damage
- material separations (e.g. cracks)

- corrosion
- indications of leakage
- defects on bolt connections
- deposits and foreign matters

Visual testing of the internal surfaces of the RPV is carried out regularly at intervals of five years (or four years; see also Tables 5-2 and 5-3).

The visual testing of the RPV bolted connection is carried out plant-specifically in accordance with the requirements of KTA 3201.4 (see also Tables 5-2 and 5-3).

In addition, visual testing of the RPV upper head outer surface is carried out in accordance with the plant-specific test instructions to verify the proper condition of the pressure-retaining wall surface. Selective visual testing of the RPV upper head ligaments of PWR is carried out every five years (or four years; see also Tables 5-2 and 5-3). Integral testing of the external condition of the RPV upper head is performed annually during the revision.

Acceptance criteria

Only qualified testing techniques are used for remote-controlled visual testing. The sensitivity level is measured on testing patterns and on reference blocks.

Deviations of the recorded actual condition from the expected nominal condition are recorded as conspicuous indications. Any conspicuous indications requiring measures to restore the proper condition (e.g. leaks, cracks) are treated as relevant indication.

Inspection history and trend analyses

Conspicuous indications are evaluated; in case of indirect visual testing, conspicuous indications are compared with the image recordings of previous tests with regard to changes.

Within the framework of visual testing carried out to determine the proper condition of the internal and external surfaces of the pressure-retaining walls and bolt connections of the RPV, no integrity-relevant findings have so far been determined.

No relevant indications have been found at the German RPVs so far.

Pressure test

Pressure tests represent an integral stress test for the component.

There are plant-specific test instructions and specifications in accordance with KTA 3201.4 /KTA 16b/ for the pressure tests to provide the integrity of pressure-retaining walls. As these are non-isolatable, the pressure test also includes the components of the primary circuit and all connected pipes up to and including the first isolation valve.

The requirements and performance of the pressure test are specified in DIN 25475-3 /DIN 15/ and KTA 3201.4 /KTA 16b/ and have been implemented in the nuclear power plants accordingly.

Pressure tests of the reactor coolant pressure boundary are carried out at 1.3-fold design pressure (corresponding to 1.3x175 bar = 227.5 bar for the PWR, and 1.3x86.3bar = 112.2 bar for the BWR). In addition, the testing temperature is determined plant-specifically depending on the reference temperature RT_{NDT}.

Generally, non-destructive testing of higher stress areas at the RPV is carried out after the in-service pressure tests.

Pressure tests are performed every ten years (or eight years; see also Tables 5-2 and 5-3).

Acceptance criteria

The pressure test is deemed to have been passed successfully if the components have withstood the required pressure level over the entire holding period.

Inspection history and trend analyses

Resulting from the in-service pressure tests carried out so far, there were no integrity-related findings for the RPV.

Vibration monitoring

The vibration monitoring system is used for the early detection of possible operational damages and to allow sufficient time for the evaluation of the altered vibration condition, in order to identify possible emerging defects or damages and thus enable targeted inspections. In the PWR, this monitoring is focused on the main components of the reactor coolant system and thus also on the RPV. For this purpose, the vibration displacement at representative points of the primary circuit, the neutron flux oscillations and pressure fluctuations of the coolant are measured. In the BWR, the recirculation pumps (reactor coolant pumps) are monitored. Vibration sensors are permanently installed for this monitoring. All changes in vibration are monitored and evaluated.

The requirements and performance of vibration monitoring are specified in DIN 25475-2 /DIN 09/ and KTA 3201.4 /KTA 16b/ and have been implemented in nuclear power plants accordingly.

During commissioning of the nuclear power plants, the vibration behaviour of the primary circuit was measured and documented as a reference measurement. It has been demonstrated that the vibration monitoring system meets the requirements of the regulations.

According to the requirements of the regulations, operational measurements are carried out regularly in accordance with the plant-specific test instructions (at least after refuelling, at the middle of the operating cycle and before refuelling). A comparison of the reference and current operating spectra shows whether the vibration behaviour of the reactor coolant system has changed.

Acceptance criteria

On the basis of the reference measurements during commissioning, attention thresholds were defined for the characteristics to be monitored for each nuclear power plant beyond which further measures are required. Monitoring for changes in these characteristics, e.g. frequency peaks, magnitude and form, is done by comparing them with the reference signals. If the values exceed the specified attention thresholds or new peaks occur, further measures are taken to clarify the situation (trend analyses, consideration of further measurement information, targeted tests.

Changes in vibration behaviour are always evaluated together with the plant manufacturer.

The results of the vibration monitoring are compiled in a report.

Inspection history and trend analyses

In addition to the required operational measurements, further operational measurements can be carried out at any time if changes are detected or if changes due to potential damage development require special monitoring.

The results of the vibration monitoring measurements of German nuclear power plants do not reveal any impermissible mechanical changes in the monitored components (RPV and primary circuit).

Structure-borne noise monitoring/Loose parts monitoring system

The structure-borne noise monitoring system is used to locate loose parts within the reactor coolant pressure boundary of the primary circuit at an early stage, thus enabling to prevent (consequential) damages by appropriate measures. The requirements for the system, its commissioning, implementation and scope as well as documentation and the in-service inspections are specified in DIN 25475-1 /DIN 13/ and have been implemented accordingly in the nuclear power plants.

Continuously measuring structure-borne noise monitoring systems have been installed in nuclear power plants in accordance with the requirements of the safety standards (e.g. KTA 3201.4 /KTA 16b/). In accordance with the requirements, the scope of the monitoring and the in-service inspections are carried out in accordance with the plant-specific test instructions.

The signals of the sensors are monitored at regular intervals.

<u>Acceptance criteria and trend analyses</u>

The reference recordings include information on operational background noise, noise emitting events during operation and test strokes. Based on these reference recordings, plant-specific limit values are set for the monitored signal, the exceeding of which triggers an alarm and starts the structure-borne noise monitoring and analysis system recording the signals from all monitoring channels.

The cause of a single-event noise can be limited by comparison with reference recordings.

Any irregularities in structure-borne noise signals are generally assessed with the plant manufacturer.

The structure-borne noise monitoring system is inspected regularly (measurement and recording of background noises, test strokes, signal transmission testing and limit values), the results are recorded.

<u>Inspection history</u>

The results of all structure-borne noise monitoring measurements carried out in the previous operation do not reveal any findings relevant for the RPV integrity.

Leakage monitoring system

The task of the leakage monitoring system is to detect leaks in the reactor coolant pressure boundary during operation as early as possible and to enable a sufficiently precise localisation as required by KTA 3201.4 /KTA 16b/. This task is performed by the leakage monitoring system according to the plant-specific test instructions in conjunction with the activity measurement system.

With regard to leakages into the containment vessel, the following measurements are carried out by the leakage monitoring system:

- measurement of the humidity of the air (dew point temperature)
- measurement of the room temperature
- measurement of the condensate flow/volume at the recirculating cooler
- measurement of the water accumulation in the sumps

These measurements are completed by measurements of the

- air activity and
- pressure in the containment vessel.

All measured values of the leakage monitoring system and the activity monitoring are continuously stored on the process computer.

Acceptance criteria and trend analyses

In each nuclear power plant, limit values (e.g. for the dew point temperature) are specified for the measuring points. If the specified limit values are exceeded, fault alarms are triggered by the system. The measures to be taken by the operating personnel thereupon are specified in the operating manual.

The leakage monitoring focuses on those areas within the reactor containment vessel which are not accessible or only accessible to a limited extent during operation. Accessible areas within the reactor containment vessel are subject to global monitoring; here the metrological measures are completed by regular inspections.

Functional tests of the leakage monitoring system take place at regular intervals.

Inspection history

The results of leakage monitoring measurements in German nuclear power plants do not reveal any findings that are relevant for the integrity of the RPVs.

Water quality monitoring

In order to avoid systematic cracking caused by stress corrosion cracking or to be able to exclude or control corrosion at the RPV as a degradation mechanism, a water chemistry compatible with the RPV materials is specified and applied in all German PWR and BWR plants. The specifications for this are derived from the VGB guideline R 401 J /VGB 06/.

Acceptance criteria and trend analyses

The most important chemical and physical parameters that are monitored in German PWRs are lithium, hydrogen, oxygen, chlorides and sulphates; in BWRs, the conductivity, chlorides and sulphates are monitored during power operation. If the specified action levels are exceeded, measures shall be taken, depending on the nature, ranging from the identification of the deviation to the immediate shutdown of the nuclear power plant.

The monitoring measures for water chemistry are specified plant-specifically in the operating manual or chemical manual. These are: type of medium to be monitored, measuring and sampling points, measured variables and frequencies, values to be observed and the measures required in case of non-compliance with specified values.

In the event of deviations from the specified chemical and physical values, an assessment of the deviations is carried out in accordance with the overall concept of the German ageing management for Group M1 components with regard to their impact to the ageing management.

The results of the water chemistry monitoring are compiled in the monthly technical reports as well as annual reports of the nuclear power plants.

Inspection history

There are currently no deviations relevant for the integrity of the RPV with regard to water chemistry in German PWR and BWR plants.

Stress/fatigue monitoring

The operating data relevant for the integrity such as pressure, temperature, flow rate and level as well as the temporal changes of these state variables are continuously measured and recorded by the standard instrumentation since the commissioning. At the main areas with regard to fatigue, local measuring points were selectively installed. Their measured values are continuously recorded and evaluated promptly. The requirements for the accompanying determination of thermal stresses, the selection of measuring points, the measurement system, the evaluation of the measurement results and the documentation for the purpose of the evaluation of the component's cumulative usage factor are specified in DIN 25475-3 /DIN 15/.

The actual and predicted EOL fatigue levels for the RPV are low. The permissible degrees of fatigue in accordance with KTA 3201.2 /KTA 13b/ are complied with. The main areas of the RPV with regard to fatigue are subject to periodic in-service inspections.

Irradiation surveillance programme

Changes in the properties of the RPV material are caused by neutron irradiation in PWRs and BWRs. The irradiation can cause varying degrees of changes in the material properties depending on the dose and material characteristics. At a neutron fluence greater than $1 \cdot 10^{17}$ n/cm^2 (neutron energy > 1 MeV) specimens of the original ferritic materials were subject to accelerated irradiation inside the reactor pressure vessel to experimentally verify the tensile and fracture toughness properties of the RPV material in German LWRs. The irradiation surveillance programme at German PWR and BWR plants covers the beltline materials of the RPV (base metal, weld metal and heat-affected zone). The irradiation surveillance programme in accordance with KTA 3203 /KTA 01/ is intended to determine the change in tensile and fracture toughness properties (base metal, welded joints of the beltline of the RPV) after a certain accelerated neutron irradiation. This ensures, in particular, a comparison of the actual state of the material condition with the assumptions on which the design is based. The irradiation specimens were taken from the original materials of the respective RPV and correspond in their production and manufacturing process (including weld filler materials, weld consumables, welding parameters, heat treatment) to the materials used in the beltline. The irradiation surveillance programme covers three specimen sets: one unirradiated specimen set and two irradiated specimen sets which were taken and tested at approx. 50% and 100% of the assessment fluence of the RPV. In BWRs, only one specimen set has been taken and tested until now. The reason for this is the low fluence of specimen sets and the RPV wall achieved due to significant improvements in fuel element loading.

The irradiation surveillance programme covers notched bar impact specimen and tensile specimen for the base metal of the beltline and notched bar impact specimen and tensile specimen of the circumferential beltline welds (one in PWRs, two in BWRs).

Acceptance criteria

On the basis of the specimen sets subject to accelerated irradiation it is shown that the ductile-brittle transition temperature defined as the adjusted reference temperature RT_{NDTj} is lower than the limit value RT_{Limit} = 40 °C /KTA 01/ for the assessment fluence, and that the upper-shelf energy characterised by a ductile percentage > 95% of the fracture area is not less than a value of absorbed energy of 68 J (individual value).

<u>Inspection history and trend analyses</u>

The plant-specific irradiation surveillance programmes show that the changes in fracture toughness properties of materials subject to accelerated irradiation are low and that the underlying assumptions (e.g. $RT_{NDT\_EOL\_Design}$ < 12°C) are conservative.

The reasons for the relatively low neutron fluence values and relatively low irradiation reactions in the German PWRs and BWRs compared to other countries are the advantageous design of the RPVs with a large water gap, especially in the PWR compared to foreign nuclear power plants, and the consistent limitation of the Cu, P and S content of the fine-grained steels used with their weld seams in the beltline (see also KTA 3201.1 /KTA 16c/).

Identification of unexpected degradation mechanisms

Already during commissioning of the German RPVs, German operators have implemented the overall concept, which has also been incorporated into the ageing management concept. The overall concept comprises the monitoring of causes of operational degradation mechanisms e.g. monitoring of water chemistry and stresses, monitoring of consequences of operational degradation mechanisms (in-service inspections of the weld seams and base metal of the RPV), structure-borne noise monitoring system and vibration monitoring system for the early detection of changes in the mechanical behaviour as well as the leakage monitoring system. Furthermore, the overall concept implies that experiences gained from the operation of other plants are taken into account and that the knowledge of possible degradation mechanisms is monitored according to the state of the art in science and technology. If deviations occur that are detected within the scope of the monitoring measures, they are assessed together with the plant manufacturer and independently of the authorised expert consulted by the authority and, if necessary, further measures are determined.

5.1.3.b Research reactors

This chapter is not applicable to German research reactors

5.1.4 Preventive measures and remedial actions for RPVs

5.1.4.a Nuclear power plants

The ageing behaviour of the materials was already taken into account in the design and manufacture of the German RPVs for PWR and BWR plants. The choice of materials, the extensive quality assurance measures within the framework of manufacturing and the comprehensive monitoring concept for the causes and consequences of operational degradation mechanisms, as well as the continuous evaluation of the experience of other nuclear power plants and the monitoring of the state of the art in science and technology within the framework of the German ageing management concept for the RPV have provided for an undisturbed operation of the RPVs until the “End of Life” (EOL). This is confirmed by the previous operating experience of the German PWRs and BWRs, so that no further preventive measures or repairs were required for the RPV pressure-retaining wall.

5.1.4.b Research reactors

This chapter is not applicable to German research reactors.

5.2 Licensee’s experience of the application of AMPs for RPVs

5.2.a Nuclear power plants

The ageing management of the RPVs of German nuclear power plants has already started within the framework of the design, the selection of materials, the manufacturing processes and the production of the RPVs. For this reason, the materials and manufacturing processes used by the industry (manufacturers, VGB) and by research institutions using funding from the Federal Republic of Germany were investigated in the Federal Republic of Germany in the early 1970s; furthermore, the possibilities for optimising the production and behaviour of materials and components, even after a longer operating life of reactor pressure vessels were investigated. These findings of the research programmes have been incorporated directly into the manufacturing specifications and the German nuclear safety standards (KTA).

In addition to the requirements for achieving high-quality materials, constructional requirements (e.g. large water gap, design favourable in terms of stress) have also been specified, so that extensive precautions have already been taken during production to control the essential ageing mechanisms of the RPVs.

Since the commissioning of the nuclear power plants, the known degradation mechanisms have been systematically monitored and evaluated by comprehensive non-destructive testing, the irradiation surveillance program and operational monitoring measures which enable the detection of changes in component behaviour at an early stage.

The entirety of the measures implemented in the nuclear power plants ensures that the RPVs have and can maintain the required quality with regard to the already known degradation mechanisms. The consistent monitoring and assessment of the general state of knowledge serves as an additional protection against potential new degradation mechanisms. The effectiveness of the measures to ensure the required quality is regularly reviewed.

From the point of view of the German nuclear power plant operators, it should be noted that due to the overall concept within the framework of the ageing management of the German RPVs, the known degradation mechanisms are controlled and sufficient precautions are provided for the early detection of potential new degradation mechanisms.

5.2.b Research reactors

This chapter is not applicable to German research reactors.

5.3 Regulator’s assessment and conclusions on ageing management for RPVs

5.3.a Nuclear power plants

Assessment of the ageing management processes for the RPVs

The ageing management for the reactor pressure vessels (RPVs) is correctly described in Chapter 5. The data on the structure of the RPVs of the German nuclear power plants correspond to the actual status.

The degradation mechanisms affecting the RPV are described in detail in Chapter 5.1.2 and have been evaluated correctly according to the current state of knowledge.

The operational monitoring measures and in-service inspections used for ageing monitoring of the RPV are specified in the nuclear safety standards KTA 3201.4 (in-service inspections and operational monitoring) and KTA 3203 (irradiation surveillance programme). These are listed in Chapter 5.1.3 in detail.

The measures carried out within the framework of the ageing management of the RPVs are suitable to ensure the required quality of the RPVs. The ageing management documented by the operators in Chapter 5 “Reactor pressure vessels” for the nuclear power plants in power operation is compliant with the ageing management practiced in German nuclear power plants.

If the currently applied operating mode is maintained, the fatigue values forecast for at least 40 years until the shutdown of the German plants fall below these values. The permissible usage factors are adhered to. The brittle fracture analyses carried out for the individual RPVs demonstrate that the safety against brittle fracture is ensured for an operating period of at least 40 years, including the accidents to be taken into account.

Experience with the application of ageing management for the RPVs

Even before the entry into force of KTA 1403, monitoring programmes based on safety standard KTA 3201.4 were introduced in the German nuclear power plants with the aim of ensuring operational integrity in operation, accompanied by the relevant nuclear supervisory authority. The assignment of the RPVs to the group of components for which integrity maintenance is required corresponds to the current assignment to the AM Group M1 of the now established KTA 1403.

The elements for implementing ageing management as described in the basic reports on the ageing management of the individual plants in accordance with KTA 1403 have been monitored in power plant operation since commissioning. Proof of the design, manufacture and quality of the RPVs as well as of the operational monitoring was prepared and evaluated.

The measures carried out within the framework of in-service inspections and operational monitoring confirm that the required quality of the RPVs is ensured. All integrity-relevant areas of the RPV are monitored. This includes the RPV wall with the weld seams, the RPV nozzles, RPV upper head as well as the RPV upper head screws and nuts.

All RPV weld seams on the pressure-retaining wall are subject to in-service inspections and assessments.

The water chemistry specified for the German nuclear power plants is monitored at regular intervals. Corrosive degradation mechanisms can thus be avoided.

Stress corrosion cracking, neutron embrittlement in the core area of the RPV and fatigue due to cyclic thermal and mechanical stress were identified as relevant degradation mechanisms.

In order to assess the effects of neutron embrittlement on the RPV material behaviour, material specimen sets were introduced into the irradiation channels provided for this purpose on the core barrel during commissioning.

The irradiation specimen sets of the individual RPVs of the PWR plants have already been assessed with regard to their tensile and fracture toughness properties. In case of the BWR-72 plants KRB II, units B and C, only one specimen set was taken and assessed as the desired fluence (100% assessment fluence) for the second specimen set has not yet been achieved due to measures to optimise the core loading (low leakage loadings).

The determined material characteristic values for the irradiated condition were taken into account plant-specifically in the brittle fracture analyses of the PWR-RPVs. The evaluation of the brittle fracture analyses for the respective RPVs shows that despite the changed tensile and fracture toughness values due to irradiation, the resistance to brittle fracture of the RPV is ensured. The brittle fracture analyses available for KRB II show clear safety margins on the basis of the adjusted reference temperatures extrapolated for the assessment fluence. Based on the current knowledge of the irradiation behaviour under BWR conditions, it can be assumed with certainty that the extrapolated adjusted reference temperatures will be confirmed by the results of the second irradiated specimen sets.

The areas relevant for the fatigue strength of the RPV (RPV upper head screws, the ligaments in the nozzle areas of the PWR-RPV upper head, the ligaments at the BWR-PWR bottom and the internal nozzle edges of the RPV inlet and outlet nozzles) have also been recorded and evaluated since commissioning by means of analyses of the operation.

All measuring points (internal pressure, temperature) used for fatigue monitoring are documented and evaluated in measuring point plans. The recordings are regularly evaluated by the operators. The results of these records are evaluated annually by the experts.

Effectiveness assessment of the ageing management

In order to meet regulatory requirements, the operator submits annual reports to the nuclear supervisory authority on the results of fatigue monitoring for the preceding operating cycle.

In the German nuclear power plants, suitable IT software (e.g. operation management system) is used to record all safety-relevant components and systems, processes, events and measures. The relevance for ageing management is assessed at least once a year. This ensures that relevant processes from plant operation are fully and comprehensively taken into account.

External events (e.g. VGB reports, events in national and international nuclear power plants, GRS information notices relevant for the ageing management programme) are reviewed by the operators for applicability to their own plants. The operators submit annual reports to the respective regulatory body on the activities carried out in the reporting year, including findings and their evaluation. The assessment of transferability of events in other plants and measures that may have been initiated is also part of the annual reports.

The annual assessment of the ageing management results for German nuclear power plants confirms the effectiveness of the ageing monitoring programmes for RPVs.

<u>Main strengths</u>

Ageing management in German nuclear power plants is based on the implementation of the established processes in power plants. The measures within the framework of these processes (e.g. maintenance measures) are controlled via the operation management system and are comprehensively and systematically evaluated in relation to ageing management on a component or mechanism-specific basis.

The annual reporting provides a transparent and comprehensible presentation of the assessment processes of the power plant operators. The documentation on ageing management related to the RPV is updated continuously. The current RPV usage factors are assessed annually.

<u>Weaknesses identified</u>

The ageing management in German plants is based on KTA 1403 as part of a continuous improvement process (PDCA cycle) with an updated knowledge basis.

No weaknesses have been identified in the ageing management process for the RPV ageing management.

5.3.b Research reactors

This chapter is not applicable to German research reactors.

6 Calandria/pressure tubes (CANDU)

CANDU reactors are not operated in Germany, thus there is no reporting on this topic.

7 Concrete containment structures

7.1 Description of ageing management programmes for concrete structures

7.1.1 Scope of ageing management for concrete structures

7.1.1.a Nuclear power plants

As described in Chapter 2.3.1.a, the structures are classified according to their safety relevance. The concrete containment structures are classified as safety-relevant civil structures/structural members according to KTA 2201.1 /KTA 11/ as B1. According to the definition of KTA 2201.1, safety-relevant structures and components are required to fulfil at least one of the following protective goals:

a) controlling reactivity

b) cooling fuel assemblies

c) confining radioactive substances

d) limiting radiation exposure

Accordingly, concrete containment structures are integrally taken into account in ageing management.

Safety-relevant structures of nuclear power plants are built as highly massive reinforced or prestressed concrete structures due to the fact that they are designed to withstand extraordinary external and internal hazards, such as blast waves, aircraft crashes or accidents, like e.g. loss-of-coolant accidents (LOCA).

However, the significance of the structures considered here, namely the reinforced-concrete shield building of the pressurised water reactor (PWR) and the containment and outer shell of the reactor building of the boiling water reactor (BWR), is different for the fulfilment of the protective goals. This is explained below.

<u>Pressurised water reactors</u>

The containment of the German pressurised water reactors is a spherical steel vessel, which is designed as a large dry containment (see Annex 1). This is not part of this report in line with the technical specifications. The steel containment is enclosed by the reinforced-concrete shield building, which is designed to withstand external hazards such as aircraft crash and blast wave. This reinforced-concrete shield building has the following functions in addition to the protection against external influences:

- accommodation of safety-related equipment (such as pumps, valves, vessels)
- retention of radioactive substances by staggered negative pressure (with filtering, if necessary)
- shielding of the radiation from the containment in case of events and accidents involving a release of radioactive substances in the containment (shield building)

A significant pressure build-up need not be assumed in the reinforced-concrete shield building, so that no airlocks or specially sealed penetrations are necessary.

Boiling water reactors

The construction line 72 of the KWU boiling water reactor has a prestressed, cylindrical concrete containment (see Annex 1). The horizontally and vertically prestressed cylindrical shell forms the lateral isolation. The upper end of the containment is formed by the non-prestressed reinforced concrete cover of the drywell, the lower end by the non-prestressed reinforced-concrete foundation slab of the reactor building.

This construction is enclosed by a cylindrical reactor building, so that it is protected from direct external weather-related and extraordinary effects (aircraft crash and blast wave). This outer shell of the reinforced-concrete reactor building is functionally comparable to the reinforced-concrete shield building of the PWR.

The containment is located on a common base plate with the outer shell of the reactor building. Constraint-free deformation of the containment is made possible by the structural separation from the outer shell of the reactor building, effected by joints in the rising structures.

The containment houses the pressure suppression system; its function is to protect the environment from a release of radioactivity into the event of an accident in the area of the reactor pressure vessel up to the first and second isolating valves of the connecting piping.

The leakage integrity function of the containment is ensured by a steel liner with a thickness of 8 mm on the inside of the cylinder, ceiling and floor surfaces. The support function is assumed by the concrete structure, in which the steel liner is anchored by welded-on shear studs.

The design pressure and temperature of the containment are 3.3 bar and 150 °C, respectively.

7.1.1.b Research reactors

The FRM II has no containment against a “pressure build-up after a significant coolant loss from reactor coolant loops” /WEN 16/. This is not necessary as amongst other things

- temperature and pressure in the primary system are low (about 50 °C/max 8 bar or unpressurised in the flow direction between fuel assembly and primary pumps) and
- the residual heat is insufficient to evaporate the pool water.

Nonetheless, some aspects are listed below which in the case of the FRM II are considered regarding the assessment of the ageing of concrete structures.

The reactor building, which mainly houses the safety-relevant equipment, operational facilities and test and experimental facilities, consists of a total of six floor levels with the following essential room areas:

- reactor pool and related structures including
  - reactor pool with its steel pipe penetrations,
  - storage pool,
  - primary cell,
  - neutron guide tunnels, and
  - hot cell with pre-cell and working cell,
- floor levels 05 and 06,
  in which i.a. the reactor protection system, the uninterrupted power supply and the ventilation system for the reactor building are installed,
- reactor hall,
  from where i.a. the reactor pool and the storage pool are directly accessible,

- experimental hall,
  in which a large number of experiments around the reactor and spent fuel pool are located,
- cellar area,
  in which together with the cellar area under the neutron guide hall west the main process engineering facilities are installed.

The reactor building has a floor area of approx. 42 m x 42 m and a height of approx. 30 m. The roof of the reactor building is arched in the central tract and designed as a flat roof in the area of the side wings. The reactor building is a reinforced-concrete structure. The outer walls and roof are 1.8 m thick. The reactor building has a watertight seal liner to protect against an intrusion of groundwater.

To avoid shock transmission, the reactor pool and related structures are decoupled from the outer walls by means of expansion joints at the connections to adjacent ceilings and is thus protected from undue effects due to induced vibrations. In addition, the ponds are dimensioned such that they remain leaktight even if at increased pool water temperature due to a loss of the main heat sink.

The BER II has no containment in the form of a pressure-resistant enclosure. The reactor building itself is a steel frame structure with walls of brick. This building accommodates the concrete reactor block, which is designed as a biological shield. This concrete surrounds the aluminium liners of the reactor pool and the nozzle pipes and represents the containment protecting against a loss of pool water. At the upper edge of the pool, this liner merges into the steel liner of the reactor hall, which represents the containment for airborne radioactive substances.

The FRMZ also has no containment in the form of a pressure-resistant enclosure in terms of the Topical Peer Review. Therefore, only the ageing management of the concrete structures of FRM II will be discussed in the following subchapters.

7.1.2 Ageing assessment of concrete structures

7.1.2.a Nuclear power plants

In the following, the ageing phenomena for concrete structures, i.e. concrete, reinforcement steel and prestressing steel /HOC 15/ are described in order to be able to explain their relevance in a comprehensible manner.

From the knowledge of the different impacts and the resulting possible degradation mechanisms, criteria are derived that allow the detection of the first signs of material deterioration or the identification of the type of degradation. The monitoring of these criteria for each material is the basis for defining the monitoring methods for building structures. Due to the factual context, these methods are presented in this chapter and not in Chapter 7.1.3.

For an evaluation of the ageing phenomena, the annual status reports on ageing management, which contain the results of the relevant ISIs in the reporting period, an evaluation of fault signals, and the experience feedback from domestic and foreign power plants (e.g. GRS information notices, reportable events, etc.) are considered. In accordance with KTA 1403, the structure condition report, which must be produced every ten years, proves that all safety-relevant structures, partial structures, systems and structural component parts have been assessed with regard to their ageing condition.

<u>Cracking in concrete</u>

Concrete is a brittle building material and has only low tensile strength. Cracking in the concrete is therefore not an extraordinary process and does not lead to a deterioration of the stability and serviceability when the structure is properly designed and executed. Under certain conditions, the occurrence of larger cracks may indicate initial damage to the component involved and reduce its durability. Therefore, cracking in concrete, in addition to other visible changes in the concrete surface (efflorescence, weathering, discoloration, moisture penetration) is an indicator of a possible deterioration of the building fabric due to ageing.

The causes of cracking in concrete are manifold. For example, cracks may already occur during hardening (“plastic shrinkage”) if the concrete is not protected from rapid dehydration by adequate post-treatment measures. Another possible cause of crack formation in young concrete results from the generation of heat during the hydration process and the subsequent cooling. This produces – especially in thick-walled and massive components – non-uniform temperature distributions across the cross-section, which can cause tension and consequent cracking unless appropriately counteracted by concrete technology measures (use of cement with low and slow heat generation) or by controlling the hydration temperature (post-treatment and additional cooling). An overview of the correlations of thermally induced cracking of concrete at a young age can be found in reference /EFN 95/.

Cracks in hardened concrete or in existing buildings made of reinforced or prestressed concrete may i.a. have the following causes:

- constructive defects, in particular insufficient minimum reinforcement
- building settlement
- adverse ambient influences (frost)
- corrosion of the reinforcement
- hindered heat expansion
- shrinkage
- chemical attack (sulphate attack, alkali reaction)

Cracks can have negative effects on the fatigue strength of reinforced concrete and prestressed concrete components. This has mainly to do with the corrosion protection of the reinforcement. For example, it is known from experiments that corrosion-initiating processes such as carbonation and chloride attack generally develop faster in cracks than in non-cracked regions (see also /DAF 89/). In this connection, an important influencing parameter for the intensity of the corrosion is the crack width. As long as crack widths are small (depending on the component and the ambient conditions: 0.2 mm to 0.5 mm), a corrosion-induced reduction of the planned service life of the relevant component is not expected. For larger crack widths, on the other hand, a reduction in durability is to be expected under conditions that facilitate corrosion, in particular if chlorides reach the reinforcement over a longer period of time.

Depending on the surface condition of the concrete, cracks are already visible to the naked eye from crack widths of approx. 0.2 mm, i.e. they are detected in the course of a visual inspection of the relevant component. What is essential for the further procedure after the detection of crack formation is an evaluation of the crack pattern (individual cracks, crack nets), the crack width and the crack length as well as the position of the crack. In a reinforced concrete girder, for example, vertical cracks in the centre of span (bending tensile stress) are to be assessed differently from inclined cracks in the bearing area (shear stress).

What is also essential for the assessment is whether identified cracks change over time. This can be measured qualitatively by applying plaster marks that are checked at regular intervals. For a quantitative determination of crack width changes, measuring devices such as crack monitors, mechanical extensometers and inductive displacement transducers are commonly used.

The crack depth can only be estimated in exceptional cases involving a large crack width. When it comes to assessing a crack by the crack depth, core drilling is required. An assessment of cracks in terms of the affected component's durability is only possible when the ambient conditions are taken into account. For example, individual cracks in reinforced concrete components have no influence on the durability under dry ambient conditions, even with crack widths of more than 0.5 mm.

Creep and shrinkage

Creep is understood as an increase in elongation over time. Creep is triggered by a constant external permanent load, with load-independent elongations (shrinkage, swelling, temperature) deducted.

The creep of the concrete is due to the changing of position of water molecules in the cement stone gel, caused by an external load as well as to the sliding and consolidation processes between the gel particles. Decisive influencing parameters for the extent of the creep deformations are therefore the water content in the concrete at the beginning of the loading and possible changes in the water content during the loading.

Creep deformation occurring during the same period is approximately proportional to the concrete compressive stress under service loads. If higher loads occur, however, the creep deformation increases disproportionately with the stress. This is due to a progression of micro-crack growth under permanent loads.

What is characteristic of the time history of creep deformations is that there is a disproportionate increase during the initial phase and a significant decrease in the long-term range. Over a period of a few decades, the creep rate under service load decreases to such an extent that no further increase in creep deformation occurs after further decades of continuous loading.

The extent of the creep deformations depends on the concrete composition, the ambient conditions, the age of the concrete at the beginning of the permanent loading, and the component dimensions. For example, creep is i.a. favoured in the case of concrete types with a high cement content, high w/c (water/cement ratio) or aggregates with a low modulus of elasticity. With decreasing relative humidity and rising temperature, creep deformation increases. At the beginning of the permanent loading at a young age, concrete creeps more than at a higher age (influence of the “degree of maturity”). Thin components creep faster than thick ones because they dry out faster.

The creep of concrete can have a positive or an unfavourable effect on the durability of components. The beneficial effects include the reduction of undesired restraint stresses, provided they build up slowly or act over a long period of time (in particular shrinkage tensions). Unfavourable effects are in particular losses of prestressing in prestressed structures, which must be taken into account during the design.

Shrinkage is understood as load-independent temporal deformation due to drying-out of the concrete. Significant influencing parameters for the extent of the shrinkage deformations are, in addition to the ambient conditions (relative humidity), the composition of the concrete, and the component dimensions. As the relative humidity of the surrounding air decreases, the shrinkage deformations increase. An increase in shrinkage deformations also ensues from an increasing w/c value, increasing cement content, and increasing grinding fineness of the cement. Conversely, the shrinkage deformations are the lower the greater the modulus of elasticity of the concrete aggregate, since stiff aggregates hinder the shrinkage of the cement stone more. Finally, thin-walled components shrink more than thick-walled components.

When concrete shrinkage happens – especially in thick-walled components – uneven deformations occur over the cross section. This leads to residual stresses (tension), which under unfavourable conditions can cause surface cracks. However, such residual stresses are at least partially reduced by creep. In prestressed structures, prestress losses occur as a result of the shrinkage, which must be taken into account in the design.

Generally, it should be noted that computational studies on creep and shrinkage are prone to variation due to the assumptions made. In damage analyses or ageing considerations, these ranges of variation have to be taken into account.

Swelling

Swelling is the time-dependent increase in volume of the concrete by water absorption at very high humidity or immersion in water. The increase in volume can cause damage in the form of cracks or spalling. This mechanism can be aggravated by aggressive substances (e.g. sulphates) that penetrate into the concrete with water absorption.

Swelling of concrete can also occur at a higher age and may cause further adverse effects, in particular corrosion of the reinforcement and restraint stresses in components connected with the concrete.

Carbonation

Carbonation is the chemical transformation of the alkaline constituents of the cement stone (in particular: calcium hydroxide) by carbon dioxide intruding from the environment into calcium carbonate. This changes the pore structure of the cement stone. Furthermore, the pH of the pore water, which is usually between 12 and 13 depending on the degree of hydration of the concrete and the type of cement, decreases to pH values < 9.

The rate of increasing carbonation from the concrete surface inwards is highly dependent on the moisture of the concrete. In extremely dry concrete, the rate of carbonation tends towards zero due to the insufficient pore water content. As regards very wet concrete (e.g. concrete surfaces exposed to rain), the rate of carbonation is also low because the intrusion of carbon dioxide into the pores of the cement stone decreases with increasing pore water content. A maximum of the carbonation rate occurs according to /HIL 98/ at a relative humidity of the concrete of 50% to 60%.

Further influencing parameters for the carbonation rate or the carbonation depth at a certain age of the concrete are the w/c value and the tightness of the near-surface edge zone, which is determined by the degree of hydration. Low w/z concretes are more resistant to carbonation than high w/c concretes because the proportion of capillary pores is lower and the concrete is thus tighter. Since the w/c value also influences the compressive strength of the concrete (the lower the w/c value for a given cement strength class, the higher the compressive strength of the concrete), the carbonation rate is lower for higher-strength concretes than for concretes with lower compressive strength. Finally, a high degree of hydration also reduces the porosity of the cement stone and thus slows down the carbonation progress. Therefore, sufficient curing of the concrete is essential, which in particular increases the resistance of the near-surface layer, which is first exposed to carbonation (see also /DAF 76/).

Carbonation changes the pore structure of the cement stone. This alone does not affect the durability of reinforced concrete. On the contrary, depending on the type of cement, a more or less marked reduction in capillary porosity ensues, which results i.a. in an increase in surface hardness.

Adverse effects in terms of durability of reinforced concrete structures may arise when carbonation penetrates as far as the embedded reinforcement, i.e. when the carbonation depth becomes deeper than the concrete cover of the reinforcement. In this case, the reduction of alkalinity associated with carbonation down to pH < 9 results in loss of the oxidic protective layer of the reinforcement steel, causing corrosion under adverse conditions (moisture, oxygen). This can also be of importance for interior components in which the concrete is carbonated up to the rear of the reinforcement and which later on come under the influence of moisture.

The carbonation depth can be visualised and measured on fresh concrete fracture surfaces created by lifting or on cleavage surfaces by means of drill cores. For this purpose, the respective fracture or cleavage surface is sprayed with an indicator liquid of 1% phenolphthalein in 70% alcohol, which changes at a pH of about 9. The non-carbonated area turns violet, while the carbonated area remains colourless (see DAfStb Issue 422 /DAF 91/).

If the concrete cover of the reinforcement is sufficient, a harmful influence due to carbonation can be excluded. If the effectiveness of the concrete cover is broken by cracks, this will be recognised first by visual checking. After a first technical assessment of possible crack patterns, further investigations may then be carried out if necessary.

Chloride attack

Chlorides only slightly affect the properties of the hardened concrete. However, chloride at the reinforcement may cause corrosion. If chloride ions reach the inserted reinforcement, the oxidic protective layer of the reinforcement steel is destroyed locally. In such places, an electrolytic corrosion process may be triggered in connection with water and oxygen (see section on “Corrosion”).

Visible signs of a chloride attack are crystallised chlorides that are produced on the concrete surface or on cracks due to changing humidity conditions by evaporation. Concrete spalling due to corrosion of the reinforcement, on the other hand, does not occur at all or only at a very advanced stage since chloride-induced corrosion generally occurs locally through deep pitting.

Chemical attack

In the case of chemical effects, a distinction is made between driving and solvent attacks. These can be caused by groundwater, waste water or exhaust gases, with the attacking media reacting with constituents of the concrete. The sulphate attack and the alkaline reaction are the driving forces. Acids cause a solvent attack.

Sulphate attack

In the case of sulphate attack, sulphates intrude from outside into the hardened concrete by diffusion or capillary suction and react chemically with constituents of the cement stone (aluminates). This results in the formation of ettringite (trisulphate). If there is sufficient water present, this chemical reaction is associated with a high volume increase, which can lead to severe cracking and spalling. Significant influences for damage through sulphate attack are:

- ambient conditions (e.g. groundwater containing sulphate)
- sensitivity of the concrete (sulphate resistance of the cement)
- tightness of the cement stone
- amount of water (w/c value)

In the event of a sulphate attack, the volume can increase to 8 times the original volume, involving the formation of ettringite. This leads to swelling of the concrete surface and net-like cracks and flaking, which will then be detected in the course of visual checking.

Alkaline reaction

An alkaline reaction takes place through the chemical reaction between the alkalis in the pore water and the concrete aggregates with soluble silica (e.g. opal sandstone, flint). The reaction product is an alkali silicate gel which occupies a larger volume than the original components. If the resulting pressure exceeds the acceptable tensile stress of the concrete, cracking, separation, and spalling may occur.

The kind and extent of the alkali reaction and possible damage to the concrete depend i.a. on the amount, type and grain size of the aggregate, the solubility of the silica, and the concentration of the alkali hydroxide solution in the concrete. A prerequisite for the formation of swellable alkali silicates is a sufficient moisture supply. In addition, there are influences from the strength, the tightness and the deformability of the concrete at the time of the alkali reaction. Finally, the extent of any possible damage also depends on the relevant structure and the type and arrangement of the reinforcement.

A damaging alkali-silica reaction leads to spider-web-like cracks, gelatinous precipitation, and efflorescence, which will be detected by visual checking. When exiting cracks, the precipitation is water-clear. In the open air, these precipitates turn white through the formation of silica gel.

Acids

When attacked by acids, which may also be present in the ground and in natural waters, constituents of the cement stone will decompose and be converted into water-soluble compounds. The degree of attack, assessed according to the removal per unit area and time unit, depends on the strength and concentration of the acid as well as on the transport conditions of the attacking and dissolved substances. For example, the removal by the action of aggressive waters increases with the flow rate. Increased temperatures also increase the level of attack.

The decomposition of the cement stone upon acid attack is associated with a drastic drop in surface strength and is recognised, for example, when the surface is knocked off. When the broken cement stone is leached out by water, the concrete aggregates also become visible. The concrete surface assumes an exposed-aggregate-concrete appearance, so that the damage will be detected in the course of visual checking. Another visible sign of acid attack is the formation of salt crystals on the concrete surface, which may happen due to evaporation in the case of non-permanent exposure to acidic media. Typical of damage caused by carbonated liquids are calc-sinter plumes in the form of white deposits on the concrete surface, which will also be detected in the course of visual checking.

<u>Biological attack</u>

Biological attack means influences from the growth of mosses, lichens, and roots. Growths on concrete surfaces can increase existing damage (spalling, cracks) and in particular increase the risk of frost damage. Furthermore, there is the possibility that metabolic products (amino acids) may soften the concrete surface.

Damage caused or increased by plant growth is manifested by defects in the concrete surface (spalling, cracks, decomposition of the cement stone) and will be detected in the course of visual inspections.

<u>Radioactive radiation</u>

High levels of radiation exposure over a long period of time may degrade the mechanical properties of concrete /HIL 76/. This applies primarily to the effect of neutron and γ radiation; by contrast, α and β radiation have only a small penetration depth and consequently have only a slight influence on the strength properties of concrete. The following therefore refers to the possible deterioration of the mechanical properties of concrete due to neutron and γ radiation.

The damaging effect of neutron and γ radiation is based on the one hand on changes (disturbances) of the lattice structure of crystalline aggregates, which are associated with a strong volume increase. As a result, at correspondingly high radiation intensity, a gradual destruction of the concrete structure can take place. This degradation mechanism is countered by a suitable choice of radiation-resistant additives, backed up by corresponding tests.

Neutron radiation with a fluence of less than $1 \cdot 10^{19}$ n/cm^2 and γ radiation with an energy dose of less than approx. $2 \cdot 10^{19}$ Gy are harmless according to /HIL 76/ with regard to changes in the mechanical properties of the concrete. The values occurring during normal operation are lower by several orders of magnitude.

On the other hand, the effect of neutron and γ radiation is always associated with an increase in the temperature of the concrete component concerned, which is superimposed by operational temperature loads that may possibly occur (e.g. in a reactor). This alone can also have a considerable impact on the mechanical properties of concrete at a correspondingly high temperature level. Thus, a permanent temperature load of more than 100 °C leads to a significant reduction in the compressive strength and, in particular, in the tensile strength and the elasticity modulus. Particularly unfavourable impacts in this context can come from alternating thermal loads, possibly in conjunction with restraint stresses due to a limitation of thermal expansion. This degradation mechanism is taken into account in the design of the relevant structural part if it is relevant due to the occurring temperatures or temperature gradients.

The assessment of a possible degradation of the strength properties of concrete by radioactive radiation can be made in exposed areas by comparing the occurring (calculated or measured) radiation intensity or the absorbed dose rate with corresponding reference values from experiments under comparable conditions. The radiation resistance of concretes is relevant only in the area of the support skirt of the biological shield. The concrete compositions were chosen with regard to radiation resistance on the basis of many years of experience in older reactors or on the basis of the preliminary drafts of the “Radiation protection concrete” technical rules /BET 17/. These principles have not been substantially changed to date.

Corrosion of the reinforcement steel

A necessary prerequisite for corrosion is that the oxidic top layer (passive layer) on the surface of the reinforcing steels, which forms in the highly alkaline environment of the pore water during concrete hardening, is destroyed by carbonation or by chloride attack. The subsequent corrosion process is an electrochemical process that can be easily separated into two sub-processes:

1. After depassivation of the steel surface, positively charged iron ions are released from the reinforcement steel (anodic partial reaction).
2. The free electrons combine with oxygen and water to form hydroxide ions (cathodic partial reaction). After a few intermediate steps, the iron and hydroxide ions form rust, in principle iron oxide Fe_2O_3.

As becomes clear from the simplified description, the destruction of the oxide top layer (depassivation) alone does not necessarily lead to corrosion of the reinforcement. Additionally required are sufficient moisture as well as the supply of oxygen. For example, reinforcement in dry concrete cannot corrode because the conditions for the electrolytic process (sufficient moisture) are missing. With water-saturated concrete, too, corrosion is not expected due to insufficient oxygen supply, even if the passive layer of the reinforcement is destroyed. The strongest corrosion is expected when the concrete is subject to frequent changes of moisture intrusion and dehydration.

As regards the corrosion of reinforcement steel, the influence of cracks with common crack widths is relatively low compared to other parameters, such as concrete cover and w/c value, as has been shown in exposure experiments with reinforced concrete components having permanently open cracks under corrosive ambient conditions (see /DAF 86/). This applies both to cracks across the reinforcement (localised corrosion) with crack widths of up to 0.4 mm and to the slightly less favourable cracks parallel to the reinforcement (wider spreading of corrosion along the rod axis, risk of concrete cover spalling, possible failure of end anchorages of the reinforcement) with crack widths of up to approx. 0.3 mm.

On the one hand, corrosion of reinforcement steel results in a reduction of the cross-section of the reinforcement; on the other hand, the transformation from iron to rust is associated with a strong increase in volume, which causes cracks and spalling in the concrete.

With the reduction of the steel cross-section as a result of corrosion, the yield strength and ultimate strength of the respective reinforcing rod decreases approximately linearly with the reduction in cross-section. Significantly greater is the influence on the deformation capacity and the fatigue strength, especially with thin reinforcement bars /DAF 86/). In the case of special load cases, for which the deformability of the reinforcement is exploited, this may result in a reduction of the maximum load-bearing capacity of the relevant reinforced concrete component.

The surface corrosion of reinforcement steel is associated with a strong increase in volume. This creates cracks, voids, and spalling in the concrete, all of which is detected by visual inspection and tapping of suspicious zones of the concrete surface. Chloride-induced corrosion generally occurs locally through deep pitting so that concrete does usually not spall over the reinforcement or, if it does, only at a more advanced stage. Obvious corrosion phenomena are also revealed by rust marks on the concrete surface.

Stress corrosion cracking of prestressing steel (only BWR)

Stress corrosion cracking means cracking and crack propagation at the simultaneous action of tensile stresses and a corrosive medium. For high-strength steels such as prestressing steel, stress corrosion cracking is induced by free, atomic hydrogen, which forms during corrosion processes on the steel surface. The damage sequence in the case of corrosion of the prestressed reinforcement is described e.g. in /DAF 96/ and /ISE 98a/.

The damaging effect of atomic hydrogen is based on the fact that it can be absorbed by the steel, accumulate on existing defects in the metal structure, and loosen the cohesion of the microstructures. In conjunction with an arising tensile stress, this can lead to cracking in the prestressing steel up to failure due to brittle fracture.

Hydrogen-induced stress corrosion cracking is not bound to a specific corrosion-inducing medium. In the case of “sensitive” prestressing steel, the influence of condensed water may already be sufficient. However, certain media (e.g. solutions containing chloride and sulphate) can multiply increase the risk of hydrogen-induced stress corrosion cracking. Traces of reactive poisons, such as sulphides and rhodanides, which inhibit the chemical recombination of atomic to molecular hydrogen and thus promote the absorption of atomic hydrogen in steel, are particularly critical. This effect is i.a. also specifically exploited for the estimation of the sensitivity of prestressing steels to stress corrosion cracking.

Regarding prestressed concrete components with post-tensioning, stress corrosion cracking is mainly due to the following causes (see also /ISE 95/, /REH 81/, /NÜR 90/, /REH 84/, /EIS 83/):

- improper handling of the prestressing steel during transport and storage
- longer residence times of the prestressing steel in the ungrouted
- incomplete compression of the duct
- concrete cover of to low thickness or quality (favouring corrosion attack)

Finally, the production, composition and heat treatment of the steel grade concerned have a significant influence on the occurrence of crack corrosion (see e.g. /RUS 92/). Generally speaking, given the microstructure of the steel alloy, the sensitivity to hydrogen-induced stress corrosion cracking increases with increasing strength. Tempered “old type” prestressing steels, which were produced until 1965, have been found to be susceptible to hydrogen-induced stress corrosion cracking. In the period between 1965 and 1978, an improvement in the degree of purity and uniformity of chemical composition in tempered prestressing steels was achieved, and production was changed in part to round material. Tempered prestressed steels from this production period are considered to be less susceptible to hydrogen-induced stress corrosion cracking. For production after 1978, no particular susceptibility to stress corrosion cracking has been known regarding tempered prestressing steels of the Sigma type, as the relevant tests carried out since 1978 in the context of quality control and numerous research results (e.g. /EIS 83/, /ISE 98b/) have shown.

Stress corrosion cracking of prestressing steel is not detectable in the course of visual inspection. There may be indirect evidence of prestressing steel corrosion or corrosion-induced failure of individual tendons from cracking of the concrete, as the failure of individual tendons leads to the shifting of forces to the other tendons and to the existing non-stressed reinforcement. However, such ductile “crack before break” behaviour usually requires the existence of several tendons or a sufficiently dimensioned minimum reinforcement.

The use of cold-drawn (instead of tempered) 1400/1600 tension wires precludes stress corrosion cracking on the material side. Further measures to prevent this ageing phenomenon on prestressing steel have been taken in the planning and construction phase and are listed in Chapter 6.1.4.

Loss of prestress of prestressing steel (only BWR)

Losses of prestress occur on the one hand directly when applying the initial tension and on the other hand in the course of the service life of the relevant prestressed concrete component. Losses of prestress are unavoidable and are taken into account in the dimensioning of the initial tension. The main causes of tension losses are:

- friction during prestressing,
- slippage in the anchorage,
- multi-strand initial tension,
- creep and shrinkage of the concrete,
- prestressing steel relaxation.

The prestress losses due to friction during prestressing arise due to the sometimes very high contact pressure of the tendons on the walls of the duct in conjunction with the change in length of the tendon in its longitudinal axis. The amount of friction losses also depends on the surface condition of the prestressing steel (smooth, profiled) and possible “clamping effects” on multi-wire prestressing steel bundles. Short-term overstressing and subsequent slackening can reduce friction losses.

In the case of friction, a distinction must be made between unwanted friction (unwanted deflection angle) due to the waviness of the tendon and friction in the planned curvatures (scheduled deflection angles) due to the selected tendon profile. In the context of ageing management, it is the unwanted frictional influences exceeding the calculated range that are of interest. The unintentional deflection angles to be assumed for the calculation are stated in the permits for the tensioning methods; however, the care taken in the construction of the building, the spacing of the supports and the stiffness of the ducts have a significant influence on their actual values. The care in the execution of construction with respect to the friction behaviour is directed in particular at the integrity of the ducts. Ducts that are dented, beginning to rust or leaky (into which concrete may flow) can significantly increase the calculated coefficients of friction. Sag between the tendon supports is almost inevitable. This must also be taken into account in the case of a planned, straight tendon guide. It is essential in this context that the distance and the rigidity of the support structure should be adapted to the conditions at the construction site and possibly be chosen to be lower than specified in the permit.

Slippage in the anchorage occurs when the prestressing force “settles” on the anchoring elements when the prestressing strands are anchored to the anchor plate. Many tensioning methods have the ability to fix this slip to a precise degree. Thus is done in by some tensioning methods in order to control the slackening path after overstressing by means of the existing slip or by correcting washers. Although slippage in anchorage is theoretically conceivable as a damage phenomenon at later times, it is practically impossible with modern tensioning methods. For slippage to occur, the anchoring elements have to fail (e.g. under the influence of signs of fatigue).

With several tendons or a series of tendons that are tightened one after the other, noticeable losses of prestress can also occur, depending on the component's softness. In the prestressing of shells, therefore, this interaction of the tendons with each other has to be computationally recorded for each prestressing section by a reserve provided by overtensioning for the first prestressed tendons. As damage phenomenon, this effect plays practically no role. However, cases are conceivable where a similar effect (e.g. in the case of the vertical prestressing of a shell) due to a higher or previously unconsidered load resulting from a change in utilisation leads to a shortening of the concrete and thus to a loss of tension similar to that of multi-strand prestressing.

The shortening of the concrete under the effect of a high permanent load (creep) as well as by volume changes in the course of dehydration (shrinkage) is the most important cause for the loss of prestress. Both effects are time- and temperature-dependent and are taken into account mathematically in the design. In addition to an increased loss of prestress, the redistribution of stresses to non-creeping components (e.g. liners, steel components, etc.) may occur as a damage phenomenon, so that the concrete is relieved and the non-creeping components receive higher loads. Such unintentional “prestressing” may possibly only show after several years of operation by the appearance of cracks in the concrete unless these effects have been taken into account mathematically in the design. A further damage phenomenon could be caused through changes in use if temperatures other than the design temperatures occur over time. At higher temperatures, an increase in creep and shrinkage deformations is to be expected.

Prestressing steel relaxation occurs when the extended length of prestressing steel is kept constant and the original stress decreases, i.e. when a relaxation occurs. This behaviour is caused by dislocations in the crystal lattice when the steel is stressed permanently. The amount of tension loss due to relaxation is essentially dependent on the magnitude of the initial stress and the temperature. In particular, the initial tension has a significant influence. Thus, the loss of prestress at an initial tensioning of 60% of the tensile strength is only 2% to 3%. With an initial stress of 80% of the tensile strength, however, prestress losses of 12% to 15% may occur. This fact is especially to be taken into account when e.g. by agreement in an individual case, a higher initial tension is to be achieved for a prestressing steel or for the prestressing method.

Tension losses as damage phenomena during the course of the ageing process (creep and shrinkage, prestressing steel relaxation) are practically only detectable from an increase of deformations and possibly from initial cracking. However, with prestressed shells, such as the BWR containment – especially if designed to withstand high levels of pressure that do not occur under normal operating conditions – no deformation or cracking will be apparent.

The ageing phenomenon “tension losses of the prestressing steel” was taken into consideration in the planning and construction of the BWR. This is described in the following chapters.

Voids in the grout (BWR only)

Voids can be caused by the fact that the grout does not reach all the prestressing steel sections in the duct or that liquid grout flows out of a prestressing steel section again. In a defective location, the prestressing steel can thus be completely free of mortar or enveloped by an incomplete layer of mortar. There is also the possibility that voids in the grout are filled with residual water from the grouting process.

Voids in the grout can cause corrosion in the prestressing steel or intensify any already existing corrosion. Selective tests /REH 84/ on the influence of voids in the grout on the corrosion of prestressing steel have revealed the following findings:

- A risk of corrosion due to voids mainly concerns mortar-free gaps in the grout or gaps filled with residual water from the pressing process. Even a thin coating of the prestressing steel with fine mortar in the area of a defect can be sufficient to prevent corrosion, as long as this coating is not carbonated.

- In partially grouted ducts, the risk of corrosion is higher than in completely non-compressed ducts because small anodic areas develop in the area of the gaps, where the corrosion rate greatly increases.
- In mortar-free voids, temperature fluctuations within the component may cause changes between condensation and dehydration on the prestressing steel surface. Such changes in humidity may cause corrosion, in particular with a corresponding oxygen supply. This is especially true if the defect in question lies in the area of open grouting- or venting tubes or in the area of cracks in the concrete as the access of oxygen from the air is favoured in such locations.
- If voids contain residual water from the grouting process, the chlorides and sulphates contained therein can also lead to corrosion of the prestressing steel if oxygen supply is sufficient.

In order to be able to exclude voids in the grout, special measures were taken during the planning and construction phase of the BWR, which are explained and assessed in the following chapters.

Experience feedback

As part of the evaluation of national and international experience, operators have assessed increased containment leak rates of French power plants. The increased leak rate was attributed to cracks in the vicinity of the equipment air lock and to the general increase in the permeability of the concrete due to the drop in prestress. This phenomenon was not seen as applying to German BWRs since in their case, the prestressed containment is built with an internal steel liner, which ensures its leaktightness. In addition, there are no cracks in the concrete and thus there is no evidence of decreasing prestress.

7.1.2.b Research reactors

The concrete structure of the FRM II is evaluated for cracking (crack length and depth) and by way of integral visual inspection. Settlement measurements are used to monitor the position of the buildings (see Chapter 7.1.3.b).

Until now, there have been no findings on concrete structures with any safety-related significance at the FRM II. Further measures are not required as the FRM II does not have a containment against a pressure build-up caused by cooling system leaks.

7.1.3 Monitoring, testing, sampling and inspection activities for concrete structures

7.1.3.a Nuclear power plants

The monitoring or control of the ageing phenomena regarding concrete structures already starts in the planning and manufacturing phase. This is described in detail in Chapter 7.1.4. It is particularly true for the prestressed structure of the BWR containment, which does not require monitoring for the planned operation of the nuclear power plant with regard to the relevant ageing phenomena concerning prestressing steel.

For the monitoring of the concrete structures of all nuclear power plants (PWR and BWR), monitoring instructions for the structures/substructures in terms of KTA 1403 have been drawn up and described

- the walk-down intervals,
- the required qualification of the personnel doing the walk-downs,

- the necessary tools and aids,
- the corresponding inventory documents (drawings or other documents),
- the possible ageing-related material changes,
- limits from when on such changes should be classified as harmful, and
- notes on how to describe, localise and document possible findings.

As part of the inspections of the structures (PWR and BWR) as required by the monitoring instructions, coordinated monitoring records are prepared. These document among other things

- the name of the person doing the walk-down, time and place
- the system and/or its components inspected,
- location and type of any findings, including photo documentation if required, and
- further action.

The entries made in the inspection logs during the walk-down are summarised in the summary protocols. The logs are kept directly in the database of the ageing management system.

In addition, statements on ageing-related activities, measures and monitoring results are written down, and insights from external sources are also added. Furthermore, relevant findings from any fault/defect reports are to be included. In addition, reports from other nuclear installations may give rise to inspections. The cause and the results are included in the report. Thus, all findings from regularly conducted inspections are combined, as far as they are of importance for the physical ageing of building materials.

As shown in Chapter 7.1.2, all relevant ageing phenomena of concrete structures are first manifested by cracking. The general condition of the concrete surfaces with regard to cracks, spalling, and damage is visually checked during regular inspections.

The crack patterns, taking into account the crack width and length on the component surface, are determined with sufficient accuracy using a crack width comparison scale. The documentation of the position and orientation in the component is done systematically.

Ageing monitoring of the concrete structures of pressurised and boiling water reactors relies essentially on the visual checks for cracking. Based on the recorded characteristics, an assessment by the expert personnel doing the walk-downs is already carried out on-site as part of ageing management. The regular inspections of the relevant concrete structure surfaces showed no signs of damage-relevant cracking. Thus, the decisive characteristic of ageing phenomena for concrete structures has been taken into account and a deterioration of the building fabric need not be expected in the coming years according to the state of the art in science and technology.

7.1.3.b Research reactors

At the FRM II, the following key actions are taken to monitor the structure as part of ageing management:

- ISI “Settlement measurements”, interval annually, since 2004 with participation of authorised expert only every two years
- ISI “Visual inspection in selected areas (decontamination coating)”, interval annually with participation of authorised expert
- plant walkdown “Buildings and other civil engineering structures” (annually)
- plant walkdown “Containment: structural leak tightness” (annually)

- plant walkdown “Containment: engineered systems” (annually)
- plant inspections as part of the daily rounds

Explicitly, cracks in the concrete are analysed in the ISIs. Cracks longer than 0.3 mm have not been observed to this date. Hence no action has been required. Neither have any actions been required as a result of any of the other above-mentioned activities.

7.1.4 Preventive and remedial actions for concrete structures

7.1.4.a Nuclear power plants

Structural provisions were taken in the planning phase of the nuclear power plants already to control possible ageing degradation mechanisms. Based on a variety of authorised experts' opinions, findings were already implemented in the planning phase that today are state-of-the-art in science and technology. This manifests itself in the fact that additional requirements for the corresponding safety certification beyond the standards that were valid at the time had to be observed (e.g. temperature-dependence of the creep and relaxation coefficients, creep behaviour of mass concrete, consideration of redistribution, etc.). The assumptions used in the planning and construction phase as well as their implementation also correspond to the currently valid DIN 25449 /DIN 16/. For the prestressed concrete containment of the BWR, the following measures were taken into account in the planning phase, which also correspond to the current version of DIN 25459 /DIN 17/:

- The prestressed concrete containment is partially prestressed. The prestressed reinforcement forms only part of the entire reinforcement. Local peak stresses are to a considerable extent covered by common reinforcement. Due to the relatively high degree of common reinforcement, the partial pre-stressing, results in a more ductile construction, smaller crack widths, and greater robustness even in the unlikely event of a single tendon failing.
- The containment is designed reversibly for all load cases. Reversibility means that the containment remains globally within the elastic range for all design load cases. Plasticising is only allowed in fault zones (openings, transition areas between wall and base plate). This design leads to structures with little deformation and cracks.
- The influence of temperature on the creep behaviour of the concrete and the relaxation of the prestressing steel were determined according to procedures that are still valid today. These influences lead in particular to higher prestress losses, which were taken into account mathematically. This way it is ensured that a minimum prestressing level counteracts the corresponding stress level during the entire operating life.

In the construction phase in many structures, additional design measures beyond those that were the usual provisions at the time were implemented to ensure the required quality of execution, such as increasing the concrete cover by an additional allowance of tolerances, tighter support of the tendons, shorter ungrouted residence time of the tendons in the duct. The following additional measures were already implemented during the construction phase for the prestressed concrete structure of the BWR:

- Grouting of the ducts with simulation of the flow to determine the optimum consistency of the grout and the possible layers of the prestressing strands in the duct. These measures ensure a uniform and complete embedding of the tension wires in the grouting material so that voids in the grout can be excluded.
- After prestressing, the anchor heads were covered with a concrete cap. The concrete cap is monolithically connected with the buttresses via a connecting reinforcement. A particularly low-shrinkage concrete was used for the concrete cap to prevent the construction joint from opening through shrinkage. This way, the corrosion protection of the tension anchorage can best be ensured.

- The support distance of the ducts was reduced to < 1.50 m. Careful planning of the support ladders in sections at a distance of approx. 1.50 m ensures that the assumed characteristic values for the friction or the deflection angles are not exceeded.
- During assembly of the common reinforcement, the ducts were protected against damage by temporary measures. Thus, for example, protective plates were arranged in front of the ducts so that no duct would be pierced during the installation of long reinforcing bars
- The use of cold-drawn (rather than tempered) 1400/1600 tension wires means that there is no stress-corrosion susceptibility on the material side, as has been experienced with “old-style” tension wires.
- The installation of the tendons was done in sections immediately before prestressing and grouting in order to reduce the residence time in the open casing.
- To reduce the effects of creep and thus the loss of prestress force, special concrete aggregates were used which show relatively good creep behaviour at high temperatures.

In 2002, one-off additional strain measurements were carried out on the prestressed concrete containment of a BWR. These investigations, together with the mathematical determination of the deformation sensitivity of the containment to assumed initial tension losses, have confirmed the robust design. Other insights additional to what visual inspections for possible cracks that are associated with tendon failure are able to reveal could not be gained.

During operation, regular walkdowns and inspections by trained personnel ensure that changes and damage to building structures are identified and, if necessary, suitable maintenance and repair measures are initiated.

Protection against ageing phenomena due to external weather conditions, as listed in Chapter 6.1.2, is not required for internal structures. Exterior components are provided with coatings.

All structures are subject to a regular visual inspection in accordance with KTA 1403 and VGB Standard /VGB 13b/, so that ageing phenomena can be recorded in a timely manner as described above.

The secondary concrete shell of the BWR is protected from the weather by a facade construction and thermal insulation, so that regular checking can be dispensed with here.

The above-mentioned measures in connection with the planning and execution of the prestressing of the BWR containment also ensure, according to current knowledge, that no ageing degradation need be expected for the duration of planned operation.

So far, there has not been any need for repairs due to ageing phenomena, neither of the reinforced concrete shell of the PWR nor of the containment of the BWR, which is protected against external weather conditions.

7.1.4.b Research reactors

So far, only safety-irrelevant findings have occurred at the FRM II. Therefore, no further preventive or remedial actions have been required.

7.2 Licensee's experience of the application of AMPs for concrete structures

7.2.a Nuclear power plants

The visual inspection is a fundamental element for detecting ageing processes. The results are systematically and comprehensibly documented in accordance with KTA 1403 and the VGB standard "Implementation of Ageing Management in Civil Engineering According to KTA 1403" /VGB 13b/. The effectiveness of the ageing management programme is determined on the basis of the annual structural status reports that are part of the structure condition reports to be prepared every ten years.

It is in particular due to the actual realisation of the concrete structures, as described above, with the corresponding provisions regarding durability that ageing phenomena, such as corrosion of the reinforcement and crack formation, develop at an extremely slow pace. Through systematic ageing management, the first signs of changes are detected in good time, and necessary maintenance and repair measures are initiated if required. The evaluation of experience feedback and the consideration of the state of the art in science and technology take place according to internally defined processes as well as across all plant operators via the VGB, also taking international developments into account.

During the inspections that were carried out, no relevant abnormalities were found in concrete containments made either of reinforced or of prestressed concrete. The design rules, which also ensure the durability of the structures, had already matured at the time of construction. Even then, the state of the art in science and technology behind these design rules essentially corresponded to the current level of knowledge and was already taken into account conservatively in their preparation.

In the course of the regular visual inspections according to KTA 1403, cracks patterns were occasionally detected. In most cases, following an expert assessment of these crack patterns, new coatings were applied merely for visual reasons and only occasionally as preventive maintenance.

Neither the results of the German nuclear power plant operators' ageing monitoring programmes nor the evaluation of national and international experience with concrete structures or the consideration of the state in science and technology regarding the ageing of concrete components suggest that there are any indications of an imminent undue impairment of functional characteristics of the concrete structures.

7.2.b Research reactors

At FRM II, there have been no significant findings in terms of the inspection targets. Therefore no further measures were required.

7.3 Regulator's assessment and conclusions on ageing management of concrete structures

7.3.a Nuclear power plants

Assessment of the ageing management processes for the concrete structures

The ageing management regarding the concrete structures of the reinforced-concrete shield building (PWR) or the containment and outer shell (BWR) – hereinafter referred to as "concrete containment structure" – is correctly described in Chapter 7. The safety features of reinforced or prestressed concrete containment (BWR) and reinforced-concrete shield building (PWR) are accurately presented. The details of the structure of concrete containments in German nuclear power plants correspond to the state in which they are realised.

The degradation mechanisms acting on concrete containment structures are described in detail in Chapter 7.1.2 and accurately reflect the mechanisms considered in the ageing management of German nuclear power plants.

In addition to the possible damage phenomena that may occur in reinforced-concrete components, the degradation mechanisms are listed that are specific to the reinforced-concrete shield building of the PWR and the containment of the BWR Product Line 72, such as stress corrosion cracking, loss of prestress, and voids in the grout. These are correctly described in Chapter 7.1.2 and accurately reflect the mechanisms considered in the ageing management of German nuclear power plants.

The cause and effect of the relevant degradation mechanisms are presented in appropriate detail.

The inspection methods used in structural inspections and the relevant damage indicators are accurately presented. Based on the requirements of KTA 1403, the inspection procedures used in German nuclear power plants are presented with regard to the boundary conditions of the tests to be carried out and the test criteria, with different plant-specific methods being applied.

Some elements, such as visual inspections and settlement measurements, have been performed and documented at the plants since commissioning.

In Chapter 7.1.4, preventive measures taken during the construction of the power plants are mentioned, e.g. the steel liner on the inside and the decontamination coating on the outside of the BWR containment, which are suitable for the prevention or minimisation of damage to the concrete containment structures. Also listed are repair measures to restore a bearing capacity and durability in the case of any degradation.

The monitoring measures and ISIs used to monitor the ageing of the containment of the BWR are listed in full in the report. Some elements, such as visual inspections, have been performed and documented at the plants since commissioning.

When evaluating the annual status reports, the results of which are included in the structure condition reports that have to be prepared in accordance with KTA 1403 and updated at least every ten years, it can be seen that if the currently practiced procedure regarding the ageing management of German plants is retained, the future durability of the concrete containment structures will continue to be guaranteed.

Experience with the application of ageing management programmes for concrete containment structures

The ageing management regarding the concrete containment structures of the German plants is based on the specifications of KTA 1403 for structural facilities. The concrete containment structures are assigned to the Ageing Management Group B1.

In addition to general information on the ageing management procedure, the basic reports contain descriptions of the possible degradation mechanisms in the context of civil engineering and a list of the buildings to be considered.

The monitoring of the ageing of structures is generally based on visual inspections and settlement measurements to identify degradation mechanisms such as cracking in concrete or coatings, concrete spalling, corrosion on the reinforcement, mechanical damage due to wear, or weather-related ageing phenomena.

Monitoring of the BWR containment is also based on visual inspections to identify degradation mechanisms such as cracking in concrete or coatings and concrete spalling. Being an internal component within the reactor building, the containment is not exposed to the weather. Carbonation or the intrusion of chlorides is highly unlikely due to the steel liner and the decontamination coating. In view of the slowly developing degradation mechanisms on structural facilities, which usually become visible on the surfaces of the partial buildings (e.g. cracks, spalling), visual inspections for the assessment of the durability of the containment are sufficient from the point of view of the nuclear regulatory authority. This assessment was confirmed for the KRB C plant by special tests (strain measurements on the inner prestressed concrete containment and non-destructive tests on the tendons – applying the remanent magnetism method – for a partial area of the containment).

In order to ensure that effective monitoring takes place, the German licence holders have drawn up rules (e.g. work instructions) for this purpose. In plant-specific procedures, the boundary conditions for structural inspections are laid down, concerning for example the areas to be inspected, inspection intervals, required aids, the manner of execution of the inspection, or the recording of abnormalities.

The structural inspections are supplemented by regular rounds by shift personnel, which are carried out by specially instructed staff. In addition, special inspections can take place.

If relevant, the recorded abnormalities are summarised in an annual technical status report prepared for the purpose of ageing management, together with a description of the main work carried out. Repair measures are initiated promptly.

There have been no inadmissible findings in the past for the BWR containment classified in group B1.

External events (e.g. VGB Notifications, events at domestic and foreign plants, GRS information notices) are checked by the plant operators for applicability. This also includes an assessment of the relevance of events for the ageing management programme. As part of the annual reporting of the licence holders to the respective supervisory authorities, accounts are given of the activities carried out in the respective year under review and of any abnormalities detected and their assessment. The assessment of the applicability of the events at other plants and the measures that may have been initiated are also part of the annual reporting.

In addition to the status reports, structure condition reports are prepared for the civil structures in accordance with the specifications of KTA 1403. The structure condition reports are intended to demonstrate that all safety-relevant structural facilities have been fully inspected and assessed within a 10-year interval and that all structures meet the requirements with regard to load-bearing capacity, serviceability, durability and, if required, leaktightness.

In past inspections, findings were recorded in some plants that were mainly due to weather-related ageing phenomena. The findings so far have been of minor importance. Those that were observed were biological deposits, damage to coatings, corrosion of the reinforcement, and spalling of concrete and coatings.

The plant-specific assessments showed that following the execution out suitable repair measures, the findings had no effect on the durability of the concrete containments. The repair measures were carried out promptly.

The findings yielded no indications of any damage to the concrete containments that would have been insufficiently recorded in the ageing management.

Effectiveness assessment of the existing ageing management programme

At the German plants, processes, events and measures that are relevant in connection with ageing management are documented by means of suitable computer software (e.g. integrated operation management system). This ensures that relevant processes that develop during plant operation are fully and comprehensively taken into account. A relevance assessment for ageing management takes place at least once a year.

The measures implemented at German nuclear power plants in the context of ageing management are altogether suitable for ensuring that the quality of the concrete containment meets the specified requirements. The detected degradation mechanisms that are relevant in connection with the ageing management of the concrete containment are controlled.

The annual evaluation of the results of the ageing management programme for the German plants confirms the effectiveness of the ageing monitoring programmes for the concrete containments.

Main strengths

Ageing management at the German plants takes place with consideration of the established processes in the power plants. The measures in the context of these processes (e.g. maintenance measures) are steered by the integrated operation management system and are comprehensively and systematically evaluated component- and mechanism-specifically with regard to their relevance for ageing management.

Through their annual reports, the evaluation processes of the power plant operators are presented in a transparent and comprehensible manner. The documentation of the ageing management regarding the concrete containment is continuously updated by each plant operator.

Weaknesses identified

Ageing management at the German plants is carried out in accordance with KTA 1403 in terms of a continual improvement process (PDCA cycle) with an updated knowledge base.

The annual evaluation of the results of ageing management of the reinforced-concrete shield building (PWR) and the containment and outer shell of the reactor building (BWR) in the form of status reports as well as the presentation of the structural condition in the structure condition report confirm the effectiveness of the plant-specific ageing management system. There are no indications of any ageing degradation.

The procedures regarding ageing processes described in Chapter 6.1 “Concrete containment structures – Description of ageing management programmes for concrete structures” fully correspond to the actual ageing management practiced.

No weaknesses have been identified in the ageing management process for the ageing management of concrete structures.

7.3.b Research reactors

Within the framework of the operation as licensed for the FRM II, the competent nuclear licensing and supervisory authority checks all measures specified by the licence holder, applying the graded approach described in Chapter 2.7. After examination by the competent authority, the abovementioned measures described by the licence holder are considered to be suitable for the early detection of ageing effects.

8 Prestressed concrete reactor pressure vessels (AGR)

Reactors of the AGR type are not operated in Germany, thus there is no reporting on this topic.

9 Overall assessment and general conclusions

Based on the above presentation, the overall assessment on the ageing management of the nuclear installations under review can be summarised as follows:

1. In Germany, consideration of ageing effects of safety-related SSCs already began with the design/layout of the nuclear power plants still in operation today. By appropriate design and construction as well as the operation of the nuclear power plants, precautions against undue impairment from ageing effects known at that time have been taken, which were also laid down plant-specifically in the construction and operating licences.
2. Identification, documentation and consideration of ageing effects has been continuously expanded based on the progressing state of knowledge. For this purpose, various sources were used and, where necessary, appropriate measures have been implemented for the control of ageing phenomena, including the replacement of affected parts and various constructive improvements.
3. The requirements of safety standard KTA 1403, in which international practice is also fully taken into account, led to a further systematisation of ageing management. In addition, KTA 1403 created a standardised assessment basis.
4. The knowledge required for effective ageing management is summarised in a knowledge base and regularly updated so that the identification of safety-related degradation mechanisms is ensured and appropriate measures are derived.
5. Ageing management in German nuclear power plants is essentially characterised by a proactive approach. Extensive monitoring of the known causes and consequences of degradation mechanisms reliably ensures that the safety margins provided for in the design are maintained.
6. Where possible, conditions have been created in German nuclear power plants to prevent ageing of safety-related SSCs on the basis of extensive research and development work. So, for example, the occurrence of various corrosion mechanisms on pressure-retaining components could be prevented by optimising these components and the operating conditions.
7. In summary, it can be concluded that the ageing management practised in German nuclear power plants provides an effective instrument for the detection and monitoring of ageing-related phenomena. The measures are suitable for identifying and controlling ageing-related mechanisms and thus maintaining the condition of the SSCs meeting the requirements.
8. In German research reactors, ageing management takes place within the framework of maintenance by appropriately applying safety standard KTA 1403. Previous operating experience confirms the effectiveness of measures for the control of ageing mechanisms also for research reactors.
9. Ageing management in German nuclear power plants is reviewed by the nuclear supervisory authorities of the *Länder* and the effectiveness of ageing management is confirmed, i.e. the practised procedure ensures that for German nuclear power plants and research reactors the high level of safety during operation is maintained.

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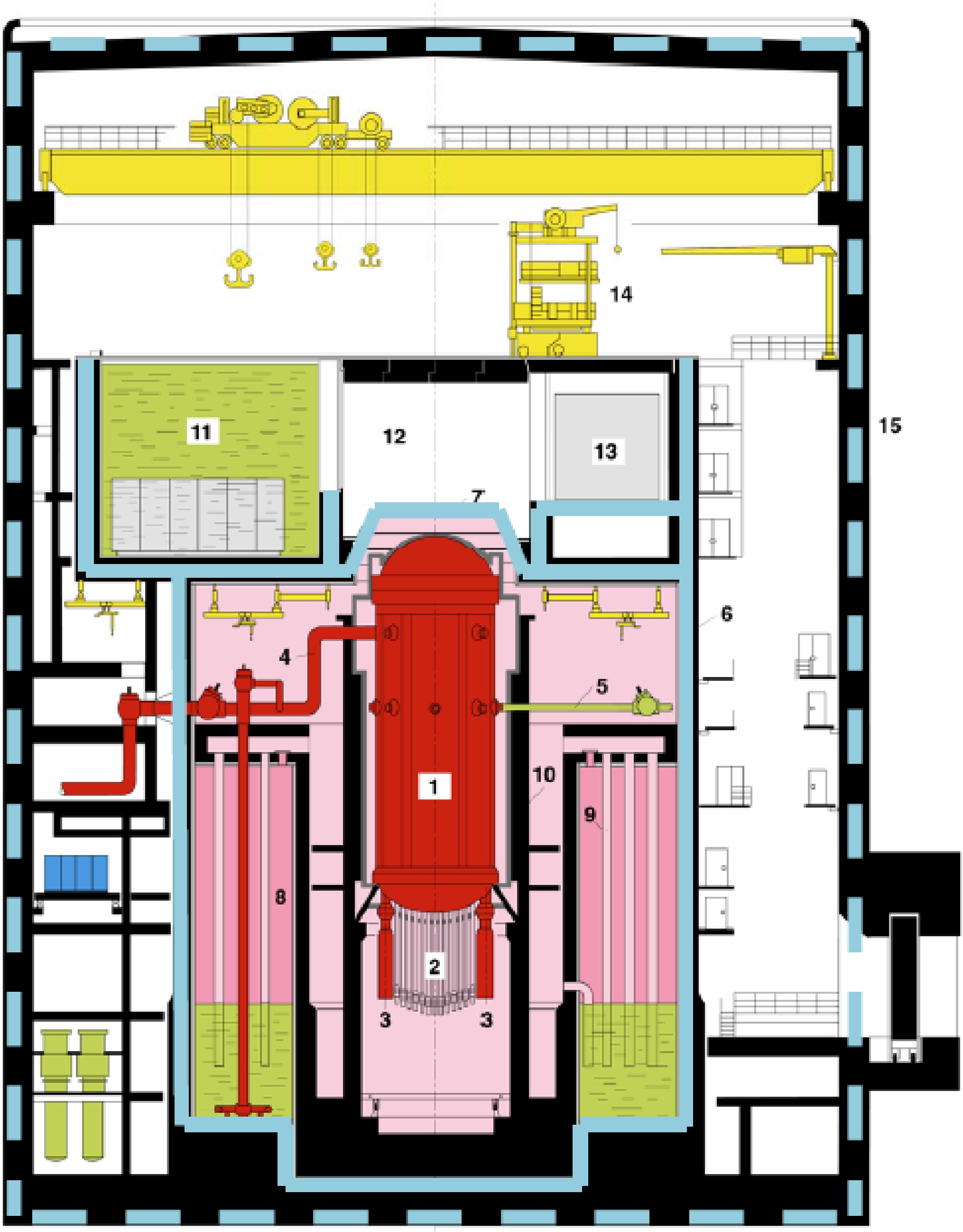
11 Annex 1: Schematic diagrams of BWR and PWR containments



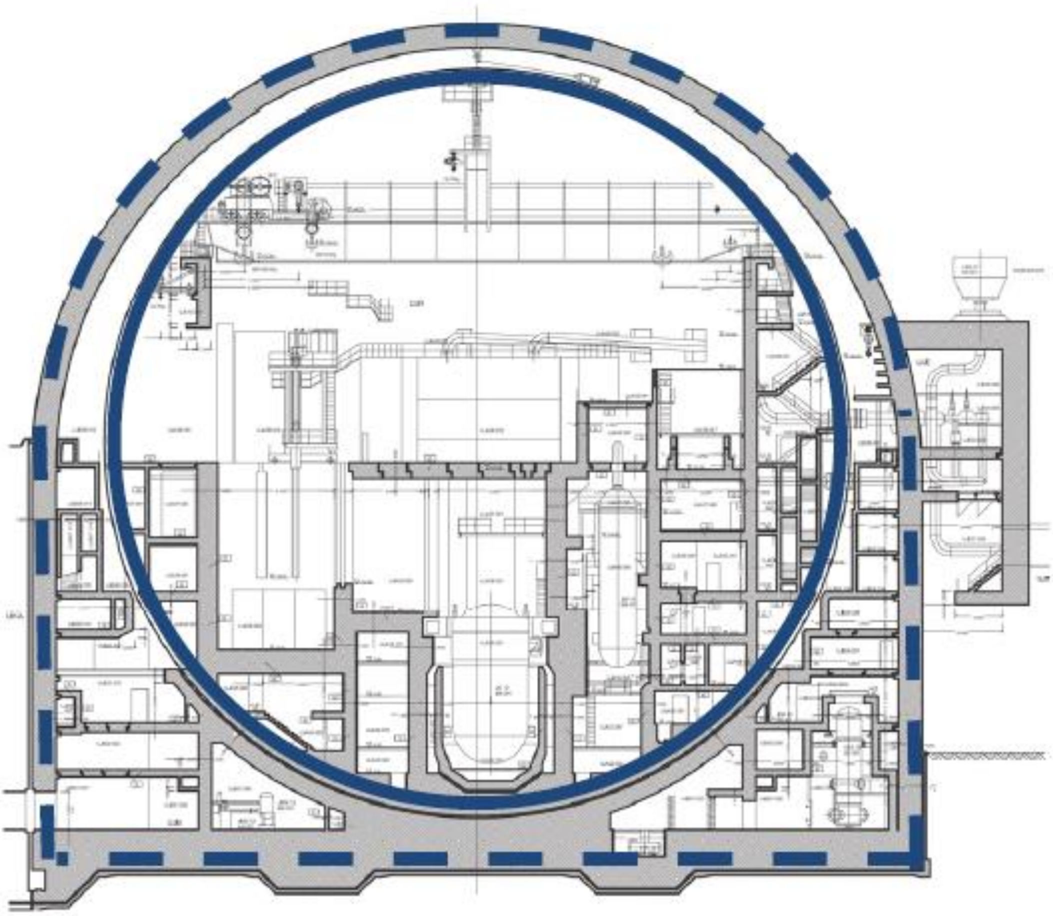
Containment of a boiling water reactor with pressure suppression system

1 Reactor pressure vessel
2 Control rod drives
3 Reactor coolant pumps
4 Main steam line
5 Feedwater line
6 Containment with steel liner
7 Loading cover
8 Pressure suppression pool
9 Pressure suppression pipes
10 Biological shield
11 Fuel pool
12 Reactor well
13 Steam separator storage pool
14 Refuelling machine
15 Reactor building

Primary containment with monolithically connected structures of the pools

Secondary containment

Figure 11-1 Containment boiling water reactor



Primary containment made of steel with a thickness of 38 mm

Secondary containment made of reinforced concrete

Figure 11-2 Containment pressurised water reactor