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# A Review of Operating Experience Involving Passive Component Material Degrading in Periods of Extended and Long-Term Operation

CODAP Topical Report


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## CODAP Topical Report


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NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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# Foreword

The Nuclear Energy Agency (NEA) Component Operational Experience, Degradation and Ageing Programme (CODAP) is the continuation of the 2002-2011 OECD/NEA Pipe Failure Data Exchange Project (OPDE) and the 2006-2010 OECD/NEA Stress Corrosion Cracking and Cable Ageing Project (SCAP). In December 2020, 13 countries and economies (Canada, Czechia, Finland, France, Germany, Japan, Korea, the Netherlands, the Slovak Republic, Spain, Switzerland, Chinese Taipei and the United States) entered into an agreement for CODAP's fourth term (2021-2023).

The project prepares topical reports to communicate insights and results from systematic reviews of operating experience (OPEX) data on metallic passive component material degradation and failure. The topical reports provide summaries of trends and patterns in metallic material performance in different operating environments and for different time periods.

This report provides technical insights regarding material degradation mechanisms that have produced failures during periods of extended operation and long-term operation. It was approved by the NEA Committee on the Safety of Nuclear Installations (CSNI) on 8 December 2021 (see document NEA/SEN/SIN(2021)2/REV, not publicly accessible) and prepared for publication by the NEA.

# Acknowledgements

This report is the product of work carried out over a three-year period since June 2019 and of six meetings of the CODAP Management Board, chaired by Dr Jovica Riznic (Canadian Nuclear Safety Commission). The list of members of the Management Board can be found in the table below. This report could not have been produced without the valuable contributions of the many experts who helped to collect and assemble the information contained herein.

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# List of abbreviations and acronyms

| | |
|---|---|
| **AERM** | Ageing effects requiring management |
| **AHC** | Access hole cover |
| **AMP** | Ageing management programme |
| **ANVS** | Autoriteit Nucleaire Veiligheid en Stralingsbescherming (Authority for Nuclear Safety and Radiation Protection, Netherlands) |
| **AOT** | Allowed outage time |
| **APAL** | Advanced PTS analysis for LTO (Euratom) |
| **AR** | Action request |
| **ART** | Adjusted reference temperature |
| **ASME** | American Society of Mechanical Engineers |
| **ASN** | Autorité de Sûreté Nucléaire (Nuclear Safety Authority, France) |
| **BAC** | Boric acid corrosion |
| **BFB** | Baffle-former bolt |
| **BKW** | Bernische Kraftwerke AG (Switzerland) |
| **BMI** | Bottom mounted instrument nozzle |
| **BNI** | Basic nuclear installation |
| **BRUTE** | Barsebäck RPV material used for true evaluation of embrittlement |
| **BWR** | Boiling water reactor |
| **BWRVIP** | BWR vessel integrity programme |
| **CAPS** | CSNI activity proposal sheet |
| **CASS** | Cast austenitic stainless steel |
| **CBB** | Core barrel bolt |
| **CC** | Component cooling |
| **CFPR** | Carbon fibre-reinforced polymer repair |
| **CFR** | Code of federal regulations |
| **CHUG** | CHECWORKS® users group |
| **CNS** | Centre for Nuclear Safety (Finland) |
| **CNSC** | Canadian Nuclear Safety Commission |
| **CR** | Condition report |
| **CRDM** | Control rod drive mechanism |

| | |
|---|---|
| **CRIEPI** | Central Research Institute of Electric Power Industry (Japan) |
| **CTH** | Chalmers University of Technology (Sweden) |
| **CUF** | Cumulative usage factor |
| **CVC** | Chemical and volume control |
| **CW** | Cold worked |
| **DETEC** | Federal Department of the Environment, Transport, Energy and Communications (Switzerland) |
| **DHC** | Delayed hydride cracking |
| **DMW** | Dissimilar metal weld |
| **DPA** | Displacement per atom |
| **EAF** | Environmentally assisted fatigue |
| **EBO V2** | Bohunice Nuclear Power Plant units 3 and 4 (Slovak Republic) |
| **ECSCC** | External chloride induced SCC |
| **EDG** | Emergency diesel generator |
| **EFPY** | Effective full-power year |
| **EMDA** | Expanded material degradation assessment |
| **EMO** | Mochovce Nuclear Power Plant units 1 and 2 (Slovak Republic) |
| **ENS** | Event notification system |
| **ENSI** | Federal Nuclear Safety Inspectorate (Switzerland) |
| **ENVDEG** | Environmental degradation conference |
| **EOL** | End of licence |
| **EOLE** | End of licence, extended |
| **EPU** | Extended power uprate |
| **ESW** | Essential service water |
| **ETC** | Embrittlement trend curve |
| **FAC** | Flow-accelerated corrosion |
| **FAD** | Flow-assisted degradation |
| **FAMOS** | Fatigue monitoring system |
| **FAVOR** | Fracture analysis of vessels – Oak Ridge |
| **FEM** | Finite element method |
| **FFS** | Fitness for service |
| **FIV** | Flow induced vibration |
| **FSWOL** | Full structural weld overlay |
| **FW** | Feedwater |
| **GALL** | Generic Aging Lessons Learned (NRC) |
| **GE** | General Electric |

| | |
|---|---|
| **GSKL** | Gruppe der schweizerischen Kernkraftwerksleiter (Group of Swiss Nuclear Power Plant Managers, Switzerland) |
| **HCF** | High-cycle fatigue |
| **HDPE** | High-density polyethylene |
| **HPCI** | High pressure coolant injection |
| **HVAC** | Heating ventilation, and air conditioning |
| **HWC** | Hydrogen water chemistry |
| **HWR** | Heavy water reactor |
| **IA** | Instrument air |
| **IAD** | Irradiation-assisted degradation |
| **IAEA** | International Atomic Energy Agency |
| **ICES** | INPO consolidated events database |
| **ID** | Inside diameter |
| **IGALL** | International Generic Ageing Lessons Learned (IAEA) |
| **IGSCC** | Intergranular SCC |
| **IHSI** | Induction heat stress improvement |
| **INCEFA** | Increasing Safety in Nuclear Power Plants by Covering gaps in (EU) Environmental Fatigue Assessment |
| **INPO** | Institute of Nuclear Power Operations (United States) |
| **IPA** | Integrated plant assessment |
| **IR** | Incident report |
| **IRP** | International review panel |
| **IRS** | Incident reporting system |
| **IRSN** | Institut de Radioprotection et de Sûreté Nucléaire (Institute of Radiological protection and Nuclear Safety, France) |
| **JIE** | Jet impingement erosion |
| **JNO** | Japanese Nuclear Operators |
| **KATAM** | Katalogs der Alterungsmechanismen (Catalogue of Ageing Mechanisms, Switzerland) |
| **KEG** | Kernenergiegesetz (Nuclear Energy Act, Switzerland) |
| **KHNP** | Korea Hydro and Nuclear Power Co. |
| **KKB** | Nuclear Power Plant Beznau (Switzerland) |
| **KKG** | Nuclear Power Plant Gösgen (Switzerland) |
| **KKL** | Nuclear Power Plant Leibstadt (Switzerland) |
| **KTH** | Royal Institute of Technology (Sweden) |
| **KWU** | Kraftwerk Union AG (Germany) |
| **LBB** | Leak-before-break |
| **LCF** | Low-cycle fatigue |

| | |
|---|---|
| **LDIE** | Liquid droplet impingement erosion |
| **LLC** | Low-leakage core |
| **LR** | Licence renewal |
| **LTO** | Long-term operation |
| **LT-OAM** | Long-term overall ageing management |
| **LTOP** | Low temperature overpressurisation |
| **LWR** | Light water reactor |
| **LWRS** | Light water reactor sustainability programme |
| **MA** | Mill-anneal |
| **MB** | Management Board |
| **MC** | Master curve |
| **MEACTOS** | Mitigating environmentally assisted cracking through optimisation of surface condition. |
| **MHI** | Mitsubishi Heavy Industries |
| **MIC** | Microbiologically influenced corrosion |
| **MRP** | Materials reliability programme |
| **MSR** | Moisture separator reheater |
| **NCD** | Non-conformance disposition |
| **NFPA** | National Fire Protection Association |
| **NMCA** | Noble metal chemical addition |
| **NRA** | Nuclear Regulation Authority of Japan |
| **NRC** | Nuclear Regulatory Commission (United States) |
| **NPS** | Nominal pipe size |
| **NSSC** | Nuclear Safety and Security Commission (Korea) |
| **NSSS** | Nuclear steam supply system |
| **NUREG** | US Nuclear Regulatory Commission technical report designation |
| **NWC** | Normal water chemistry |
| **OD** | Outside diameter |
| **OL** | Operating licence |
| **OLNC** | On-line noble chem |
| **OPEX** | Operating experience |
| **PARTRIDGE** | Probabilistic analysis as a regulatory tool for risk-informed decision guidance |
| **PAUT** | Phased-array ultrasonic testing |
| **PCSG** | Pipe crack study group |
| **PEO** | Period of extended operation |
| **PIP** | Problem investigation process |

| | |
|---|---|
| **PIRT** | Phenomena identification and ranking technique |
| **PISA** | Pressure vessel integrity and safety analysis |
| **PLiM** | Plant life management |
| **PMDA** | Proactive material degradation assessment |
| **PSCR** | Primary systems corrosion research |
| **PSI** | Paul Scherrer Institute (Switzerland) |
| **PSR** | Periodic safety review |
| **PTS** | Pressurised thermal shock |
| **PWR** | Pressurised water reactor |
| **PWROG** | PWR owners' group |
| **PWSCC** | Primary water SCC |
| **RHR** | Residual heat removal |
| **PZR** | Pressuriser |
| **RHWG** | Reactor Harmonization Working Group (WENRA) |
| **RVH** | Reactor vessel head |
| **RVI** | Reactor vessel internals |
| **RVID** | Reactor vessel integrity database |
| **RVLIS** | Reactor vessel level indication system |
| **RWCU** | Reactor water cleanup |
| **SAFIR** | Safety of Nuclear Power Plants (Finnish National Research Programme) |
| **SBP** | Small bore piping |
| **SCC** | Stress corrosion cracking |
| **SEFW** | Super emergency feedwater system |
| **SG** | Steam generator |
| **SICC** | Strain-induced corrosion cracking |
| **SIL** | Service information letter |
| **SME** | Subject matter expert |
| **SMILE** | Studsvik material integrity life extension project |
| **SRE** | Selected representative event |
| **SRP** | Standard review plan |
| **SS** | Stainless steel |
| **SVTI** | Swiss Association for Technical Inspections |
| **SW** | Service water |
| **SWR** | Siedewasserreaktor (BWR) |
| **T/C** | Thermocouple |
| **TAE** | Thermal ageing embrittlement |

| | |
|---|---|
| **TASC** | Thermal stratification, cycling and striping |
| **TF** | Thermal fatigue |
| **TGSCC** | Transgranular SCC |
| **TLAA** | Time-limited ageing analysis |
| **TLR** | Technical letter report |
| **TPC** | Taiwan Power Company |
| **TPR** | Topical peer review |
| **TSO** | Technical support organisation |
| **TT** | Thermal treatment |
| **UCL** | Uniaxial constant load |
| **ÚJD** | Úrad jadrového dozoru (Nuclear Regulatory Authority of the Czech Republic) |
| **UT** | Ultrasonic testing |
| **VHP** | Vessel head penetration |
| **VISA** | Vessel integrity simulation analysis |
| **WENRA** | Western European Nuclear Regulators Association |
| **WGIAGE** | Working Group on Integrity and Ageing of Components and Structures (NEA) |
| **WGRISK** | Working Group on Risk Assessment (NEA) |
| **WPS** | Warm pre-stress effect |
| **WWER** | Water-cooled, water-moderated energy reactor |
| **XFEM** | Extended finite element method |
| **xLPR** | Extremely low probability of rupture |
| **ZIRP** | Zorita Internals Research Project |

# Executive summary

The Nuclear Energy Agency (NEA) Component Operational Experience, Degradation and Ageing Programme (CODAP) is the continuation of the OECD/NEA Pipe Failure Data Exchange Project (OPDE – 2002 to 2011) and the NEA Stress Corrosion Cracking and Cable Ageing Project (SCAP – 2006 to 2010). In December 2017, 13 countries and economies (Canada, Czechia, Finland, France, Germany, Japan, Korea, the Netherlands, the Slovak Republic, Spain, Switzerland, Chinese Taipei and the United States) entered into an agreement for a third term for CODAP (2018-2020).

The CODAP project prepares topical reports to communicate insights and results from systematic reviews of operating experience (OPEX) data on metallic passive component material degradation and failure. The data collection for this report ended in 2020, and only the validated data were used. The topical reports provide summaries of trends and patterns in metallic material performance in different operating environments and for different time periods.

During the April 2019 Working Group on Integrity and Ageing of Components and Structures (WGIAGE) meeting, it was decided to pursue an evaluation of material degradation OPEX during periods of extended operation (PEO) and long-term operation (LTO) of nuclear power plants. The objectives were to: a) document insights from evaluations of OPEX with passive metallic components during PEO/LTO; and b) based on the results of these evaluations, develop recommendations for possible enhancements to the CODAP data collection process. The results, conclusions and recommendations that are documented in this report were obtained using the following process:

- **STEP 1**. In 2015, the WGIAGE, operating under the Committee on the Safety of Nuclear Installations (CSNI), published the results of a "Questionnaire on Long-term Operation of Commercial Nuclear Power Plants" (NEA/CSNI/R(2015)13). In preparing this report, a comparison was made between the content of the CODAP event database and the results of the WGIAGE questionnaire. This step was taken in order to identify possible gaps in the scope and content of CODAP relative to the WGIAGE qualitative ranking of degradation mechanisms considered to be important with respect to PEO/LTO.
- **STEP 2**. The results of two expert panels on potential material degradation issues during PEO/LTO were reviewed. The subject expert panels were convened by the Office of Nuclear Regulatory Research of the US Nuclear Regulatory Commission (NRC). The Proactive Material Degradation Assessment Expert Panel (PMDA, 2004 to 2006) identified the material degradation scenarios that could affect plant systems for up to 40 years of operation, and the Expanded Material Degradation Assessment Expert Panel (EMDA, 2012 to 2014) extended the analysis time frame from 40 to 80 years. In preparing this report, a comparison was made between the CODAP and the results of the two expert panels. The PMDA and EMDA expert panel members were drawn from the United States and international materials science community. The PMDA performed a comprehensive evaluation of potential ageing related degradation modes for reactor internals, as well as primary, secondary and some tertiary piping systems. The PMDA output has supported the NRC staff evaluations of licensees' ageing management programmes and the prioritisation of research needs. The EMDA broadened the analytical time frame to 80 years to encompass a potential second and third 20-year licence renewal. The EMDA output has supported the NRC staff evaluations of licence renewal applications and the prioritisation of research needs.

- **STEP 3**. In 2019, the NRC organised an international workshop on age-related degradation of reactor vessels and internals. During this workshop the participating organisations were invited to summarise the country/economy-specific state of knowledge, ongoing research and associated operating experience. The technical information included in the workshop presentations and the summary report was used to complement the country/economy-specific overviews of PEO/LTO material degradation issues in this report.
- **STEP 4**. The material degradation issues as recorded in the CODAP event database were reviewed. This review included selected reactor vessel internals and safety-related and non-safety-related piping system components.
- **STEP 5**. A synthesis of the PEO/LTO material degradation insights based on the results of Steps 1 through 4 was formulated. Because of the different plant licensing regimes among the 13 CODAP project member countries and economies, this report considers OPEX from plants that have been in operation for more than 25 years.

## Conclusions

The technical and programmatic processes that have been implemented to address stress corrosion cracking (SCC) problems appear to be effective. In comparison with the SCC OPEX during the first 25 years of plant operation, a significantly lower SCC incident rate is noted for plants in PEO/LTO. A possible exception is the recent WWER-specific SCC OPEX on medium-diameter safety Class 2 pressure boundary components.

Since the late 1980s, major piping pressure boundary failures attributed to flow-accelerated corrosion (FAC) have been relatively rare events. This is attributed to steps taken to implement effective secondary-side chemistry control (in PWRs), proactive piping replacements using FAC-resistant materials, and implementation of plant-specific FAC monitoring programmes. It is noteworthy that the WWER-specific FAC experience appears to be different from that of PWR plants. There is also a “gap” between the CODAP event database content with respect to FAC in WWER plants and the ranking assigned to this material degradation mechanism in the WGIAGE questionnaire.

Corrosion in safety-related raw water-cooling system piping and fire water system piping poses unique ageing management challenges. Applications of non-metallic materials, advanced metallic materials and composite repair technology are receiving increased attention in mitigating corrosion effects on piping pressure boundary integrity.

## CODAP work in the fourth term (2021-2023) and beyond

This report looks into observed age-dependent material degradation of piping systems and reactor internals. While it is unclear whether the identified trends and patterns are representative across the countries and economies participating in the CODAP project, validating these trends would be technically challenging and time-consuming. Recommendations for potential future work include:

- Selecting a single “system-material-environment” grouping and, for a limited and recent time period (e.g. 2015-2020), to determine:
  - the applicable country/economy-specific ageing management programme requirements, regulations and operability determination procedures;
  - the country/economy-specific OPEX;
  - the country/economy-specific regulatory inspection programmes and the processes for documenting the findings from site inspections.
- Evaluating the WWER-specific FAC OPEX to determine if and how it differs from the corresponding PWR-specific OPEX.

- The CODAP event database contains a significant population of fire protection water system and service water system pipe failures in US plants. The project should consider determining if this "bias" is attributed to unique US piping system designs, material selections, operating environments and/or material ageing management programmes, or whether it is more of a reflection of the data exchange process.

In this study, the evaluation of possible age-dependent material degradation was done in a high-level manner using simple "visual tests" of graphical plots. It is a first step in a more rigorous analysis of the OPEX. The work done in producing this Topical Report would benefit from a benchmarking exercise involving multiple TSOs under an "umbrella project", e.g. by developing a CSNI Activity Proposal Sheet (CAPS) entry with joint CODAP/WGIAGE/WGRISK oversight and guidance.

## CODAP data collection process

Since the 1960s the exchange and analysis of nuclear power plant operating experience has been a point of focus for nuclear regulatory authorities, technical support organisations and the nuclear industry at large. The processes and systems for collecting and disseminating operating experience data continue to evolve. For the year 2021 and beyond the CODAP Management Board (MB) is considering the following activities to potentially enhance the data collection process:

- Each project member to identify the current national routines for recording and submitting information on material degradation issues, including OPEX access.
- Each project member to report on the current national regulatory and industry routines/practices and requirements for performing operability determinations or fitness for service assessments. In particular, the question of how such evaluations provide relevant OPEX data for submission to CODAP should be elaborated.
- The CODAP Management Board has devised the term "selected representative event" (SRE) as a means for simplifying data entry. However, it has promoted an ad hoc approach as opposed to systematic approach to the exchange of reactor internals OPEX. It is therefore recommended that the list of BWR, CANDU, PWR and WWER reactor internal parts of interest be more firmly defined and that a plan be developed for how to conduct the future reactor internals OPEX exchange.

# Chapter 1. Introduction

Since 2002, the NEA has operated an event database project that collects information on passive metallic component degradation and failure. The scope of the database includes primary system piping components, reactor pressure vessel internals (“reactor components”), main process and standby safety system piping, and support system piping (e.g. ASME III Class 1, 2 and 3, or equivalent) components, as well as non-safety piping components whose degradation or failure can have significant operational impact. The CODAP project prepares topical reports to communicate insights and results from analyses of operating experience data. This report evaluates material degradation operating experience during periods of extended operation (PEO) and long-term operation (LTO).

## 1.1 Objectives and scope

The objectives of this report are twofold. First, it aims to document insights from evaluations of operating experience (OPEX) with passive metallic components in commercial nuclear power plants during PEO/LTO. Second, based on the results of these evaluations, it aims to develop recommendations for possible enhancements to the CODAP data collection process.

The technical questions that are addressed in this report relate to the feasibility of determining trends and patterns in possible age-dependent occurrence rates of passive component failures. Also addressed are the national differences in materials ageing management practices. The scope of the OPEX evaluations are as follows:

- safety-related and non-safety-related piping systems;
- fire water system piping[1]; and
- selected non-piping passive components, e.g. reactor internals.

## 1.2 Background and related work

Research on nuclear power plant materials’ ageing degradation has been carried out for more than five decades, references [1]–[39]. CODAP prepares topical reports to communicate insights and results from systematic reviews of operating experience with metallic passive component material degradation. Since 2010 the following topical reports have been published:

- “Technical Basis for Commendable Practices on Ageing Management: Stress Corrosion Cracking Mechanisms” (NEA/CSNI/R(2010)5) “ [21];
- “Flow Accelerated Corrosion (FAC) of Carbon Steel and Low Alloy Steel Piping in Commercial Nuclear Power Plants” (NEA/CSNI/R(2014)6) [30];

1. This category was selected in order to develop an international perspective on the ageing management of fire water systems. As an example, the US fire water system ageing management requirements are defined in Section XI.M27 of NUREG-2191, Vol. 2, Generic Ageing Lessons Learned for Subsequent License Renewal (GALL-SLR),” www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr2191.

- “Operating Experience Insights into Pipe Failures in Electro-Hydraulic Control (EHC) and Instrument Air (IA) Systems” (NEA/CSNI/R(2015)6) [31];
- “Operating Experience Insights into Pressure Boundary Component Reliability and Integrity Management (RIM)” (NEA/CSNI/R(2017)3) [33];
- “Operating Experience Insights into Below Ground Piping at Nuclear Power Plants” (NEA/CSNI/R(2018)2) [34];
- “Basic Principles of Collecting and Evaluating Operating Experience Data on Metallic Passive Components” (NEA/CSNI/R(2018)12) [35]; and
- “A Review of the Post-1998 Experience with Thermal Fatigue in Heavy Water and Light Water Reactor Piping Components” (NEA/CSNI/R(2019)13) [38].

During its April 2019 Working Group Meeting, the CODAP Management Board (MB) decided to pursue an assessment of material degradation OPEX during extended or long-term operation of commercial nuclear power plants.

## 1.3 Technical approach

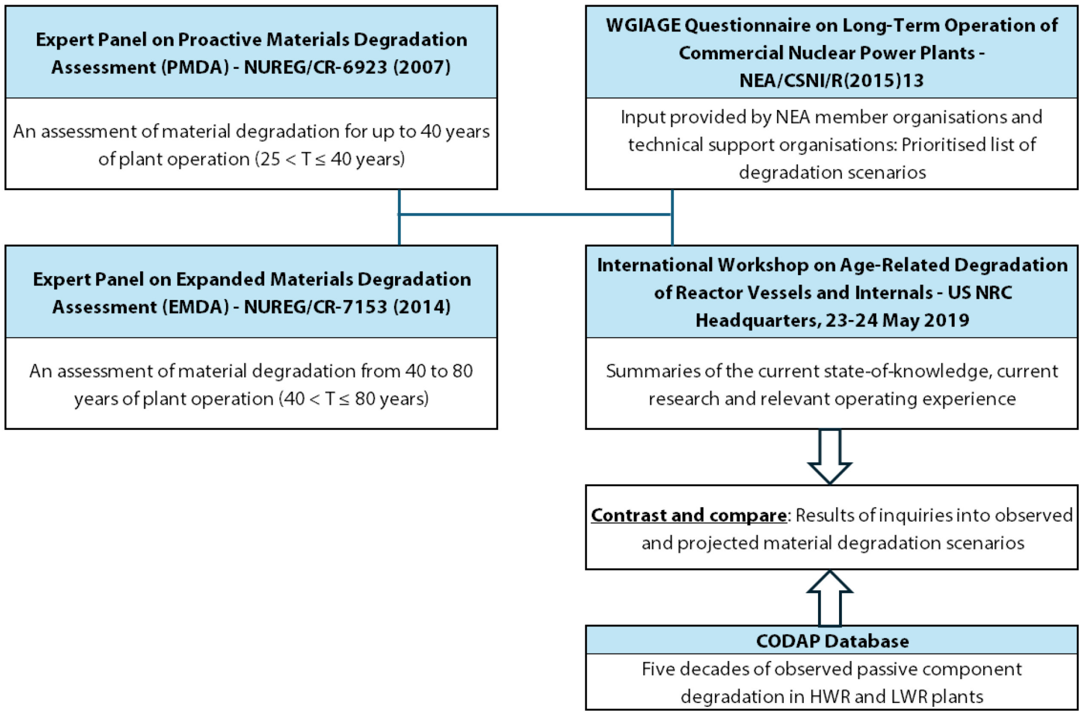
The results, conclusions and future activities that are documented in this report were obtained using a five-step approach (Figure 1-1):

- **Step 1**. The results from a questionnaire on LTO (see document NEA/CSNI/R(2015)13 [32]) served as a basis for determining to what extent the CODAP event database can be used to assess LTO trends in material performance. Prepared by the WGIAGE, this document includes a qualitative ranking of material performance issues of concern, Table 1-1. In preparing this report a comparison was made between the content of the CODAP event database and the results of the WGIAGE questionnaire.
- **Step 2**. A review of the results of two expert panels convened by the US Nuclear Regulatory Commission on material degradation issues during LTO. The proactive material degradation assessment (PMDA; 2004 to 2006) [17] identified the different material degradation scenarios that could affect plant systems for up to 40 years of operation, and the expanded material degradation assessment (EMDA; 2012 to 2014) [27][28][29] extended the analysis time frame from 40 years to 80 years. In preparing this report a comparison was made between the CODAP event database and the results of the PMDA and EMDA expert panels.
- **Step 3**. In 2019, the US Nuclear Regulatory Commission organised an international workshop on age-related degradation of reactor vessels and internals during which the participating organisations summarised their country/economy-specific state of knowledge, ongoing research and operating experience [39]. The technical information included in the workshop presentations and the workshop summary report was used to augment the country/economy-specific overviews of PEO/LTO material degradation issues in this report.
- **Step 4**. Material degradation issues as recorded in the event database were reviewed. This review included consideration of reactor vessel internals and safety-related (i.e. Safety Class 1 through 3) and non-safety-related piping system components. The evaluation of possible age-dependent material degradation was done in a high-level manner using simple “visual tests” of graphical plots.
- **Step 5**. A synthesis of the PEO/LTO material degradation insights was formulated based on the results of Steps 1 through 4.

### Table 1-1: Results of WGIAGE questionnaire on PEO/LTO [32]

| Country | Degradation mechanisms considered important to PEO/LTO | Ranking |
|---|---|---|
| Canada | CANDU Pressure Tube – fracture toughness reduction, delayed hydride cracking and irradiation induced creep | High |
| | CANDU Outlet Feeders in the Primary Coolant Circuit – FAC | High |
| | SG Tubes and External Surfaces – Alloy 800 SCC, Wastage | High |
| | Secondary Coolant Circuit – FAC | High |
| | Valves, pumps and other components – various degradation mechanisms | High |
| | CANDU Calandria vessel and its internals – various degradation mechanisms | Medium |
| | Bolting – IASCC/SCC | Low |
| Czechia | Ageing of reactor pressure vessels and reactor internals | High |
| | FAC of high energy pipelines | High |
| | Environmentally assisted fatigue of primary circuit metals | Medium |
| Finland | RPV radiation embrittlement, particularly in combination with hydrogen flaking and high nickel content | No ranking provided |
| | Thermal ageing and SCC in RPV safe ends and other dissimilar welds | |
| | Thermals fatigue caused by stratification and thermal cycling in branch connections (e.g. mixing tees) | |
| | Pitting and crevice corrosion near deposits, metallic contact points and stagnant conditions | |
| | Boric acid corrosion in bolted closures of the primary circuit pressure boundary | |
| | SCC and IASCC in RPV internals | |
| France | Irradiation embrittlement of reactor pressure vessel (RPV) | High |
| | Irradiation embrittlement of reactor vessel internals | |
| | Stress corrosion cracking of nickel alloys | |
| | Fatigue considering the environmental effects | |
| | Thermal ageing – e.g. RCS Hot Leg Steam Generator (S/G) inlet CASS elbows | |
| | External/internal corrosion of buried piping | |
| Japan | Neutron irradiation embrittlement of reactor pressure vessel (RPV) and core internals | High |
| | Irradiation assisted stress corrosion cracking (IASCC) | High |
| | Stress corrosion cracking mechanisms other than IASCC) | High |
| | Thermal ageing embrittlement of cast austenitic stainless steel (CASS) components | Medium |
| | Pipe wall thinning due to flow assisted degradation | Medium |
| The Netherlands | Thermal stratification leading to thermal fatigue in piping | Medium |
| Slovak Republic | Internal/external corrosion of buried emergency service water piping | High |
| Switzerland | RPV embrittlement | High |
| United States | Ageing of reactor vessel internals | High |
| | Thermal and neutron embrittlement of cast austenitic stainless steels (CASS) | Medium |
| | Metal fatigue of components considering environmental effects | Medium |
| | RPV embrittlement | Medium |

**Figure 1-1: The process for developing insights into long-term material degradation**




At the time when the US NRC PMDA expert panel was convened, the average age of the US commercial light water reactor (LWR) population was about 25 years (26.5 years for the BWR plant population and 23.6 years for the PWR plant population). The objective of the PMDA expert panel was to identify the potential key material degradation scenarios for the next 15 years of operation, up to the end of the original 40-year operating licence. At the time when the US NRC EMDA expert panel was convened, the average age of the US LWR population was about 33 years. The objective of the EMDA expert panel was to identify the potential key material degradation scenarios for extended periods of operation, i.e. 40+20 years (the licence renewal domain) and 60+20 years (the subsequent licence renewal domain).

Against this background, the review of the operating experience in the CODAP event database was limited to events that occurred at plants 25 years old or older. The review responded to the following questions:

1. What was the passive component operating experience from 2004 to 2014 (the start date of the PMDA expert panel and the conclusion of the EMDA expert panel)?
2. Are the apparent and underlying causes of material degradation during PEO/LTO similar to what was experienced during early plant life? If not, what are the main differences?
3. Is it feasible to validate the WGIAGE questionnaire results [32] against the CODAP event database? In other words, to what extent are any of the material degradation issues that were identified in the questionnaire also acknowledged and tracked in the CODAP project?
4. Based on the results of the operating experience evaluations that are documented in this report, are there any gaps in the lists of degradation scenarios that were identified by the PMDA, EMDA or WGIAGE?
5. What, if any, procedural changes should be contemplated to ensure that the CODAP project actively continues to track the long-term material performance for years to come?

There are well-known technical challenges in evaluating OPEX data. The quality of the database insights that are obtained through systematic evaluations is strongly correlated with the completeness of an OPEX database. From the point of view of PEO/LTO and possible trends in the rate of material degradation, the "latency" in populating an OPEX database with failure events presents the data analysts with another technical challenge. Since this report is concerned with operating experience through the end of the calendar year 2019, and as further explained in the next paragraph, there could potentially be latent gaps in the material degradation knowledge base.

Reference [35] introduced the term "inherent latency" of structured data collection efforts. Inherent latency refers to the amount of time required for OPEX to travel from the "originator" (i.e. nuclear power plant) to the data analyst, and this time could be considerable. The referenced report made the contention that it takes about five years to ensure OPEX data completeness. That is, five or more years could elapse before achieving high confidence in data completeness for the previous five years of OPEX collection and evaluation. In other words, around the years 2024 to 2025, and possibly later, the data mining for the previous five-year interval (i.e. 2015-2019) would be expected to approach "saturation" (meaning high confidence in the completeness of a database). Could the potential for "cliff-edge-effects" (i.e. small changes in the OPEX data resulting in large results variation) influence the outcome of an analysis due to database infrastructure factors? It depends on the maturity of inspection programmes and the state of knowledge concerning certain degradation mechanisms, as well as on the "effectiveness" of the CODAP project in capturing possible new material degradation insights. Considerations about the use of up-to-date failure data is intrinsically assumed to be factored into any OPEX data analysis task.

## 1.4 Nomenclature

There are two regulatory approaches to extending the operating time beyond an original licensing period. One approach is through a periodic safety review (PSR) and another approach is through licence renewal (LR). PSRs are performed to: (i) determine whether the plant complies with its licensing basis; (ii) identify the extent to which the original licensing basis remains valid, in part, by determining the extent to which the plant meets current safety standards and practices; (iii) provide a basis for implementing appropriate safety improvements, corrective actions, or process improvements; and (iv) provide confidence that the plant can continue to be operated safely.

The term "period of extended operation" (PEO) is used where an original operating licence may be renewed for a predetermined period only. Approval of a PEO is preceded by a periodic safety review and once the end of a PEO is reached the reactor is permanently shut down and decommissioned. The term "long-term operation" (LTO) is used where an original operating licence is extended through a formal licence renewal process. The United States has implemented a LR process that proceeds along two tracks: one for review of safety issues and another for environmental issues. The operating licence may be renewed multiple times in accordance with specific regulatory requirements.

The PEO/LTO definitions of the CODAP project member countries and economies are summarised in Table 1.2. As of mid-2021 the population of nuclear power plants in PEO/LTO (≥ 40 years of operation) in the CODAP project member countries and economies consisted of 81 reactor units (see Figure 1.2).

## 1.5 Report structure

This report consists of six chapters and three appendices. A summary of the PEO/LTO operating experience is given in Chapter 2. Chapter 3 gives examples of the country/economy-specific material degradation OPEX during PEO/LTO. Chapter 4 documents the results of the PEO/LTO operating experience data evaluations. A report summary with conclusions and future activities is presented in Chapter 5. A list of references is given in Chapter 6.

Annex A is a glossary of technical terms. Annex B is a summary of the US reporting requirements and routines for passive component degradation and failure.

**Table 1-2: Commercial nuclear power plant operating licence bases in CODAP project member countries/economies**[2]

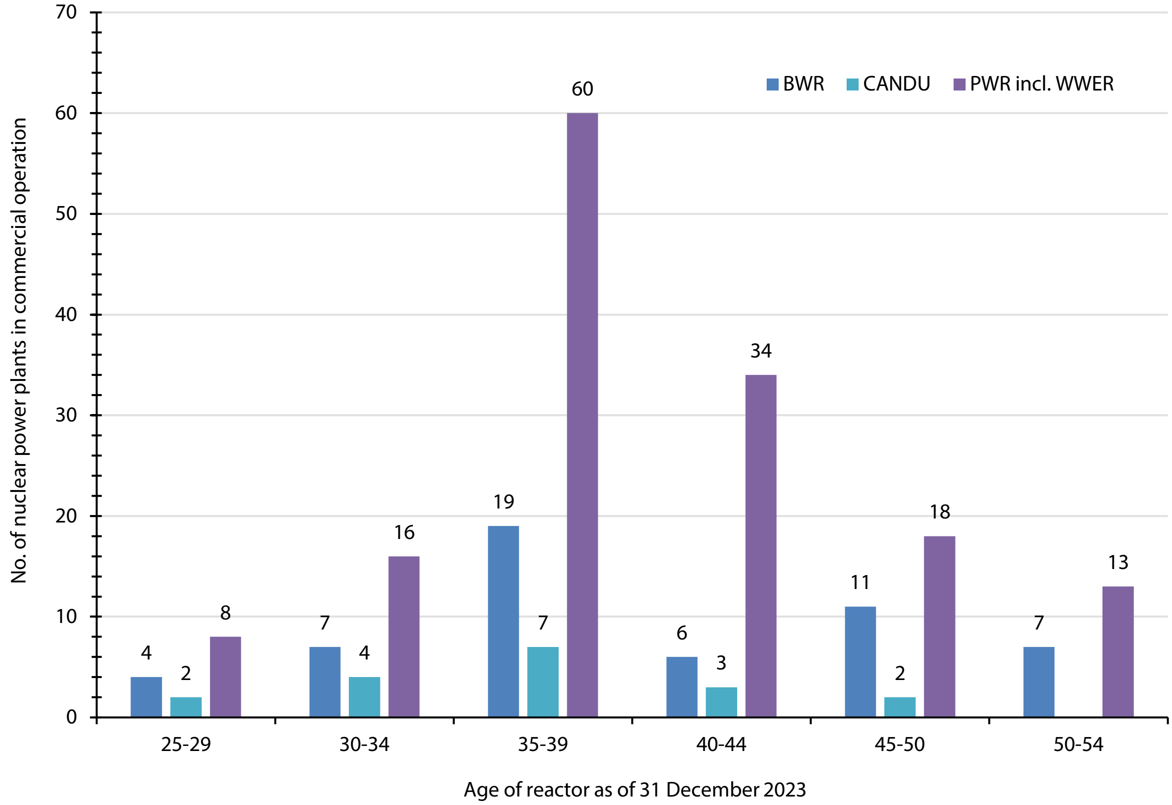
| Country/ economy | Basic principles |
|---|---|
| Canada | Canadian CANDU plants were originally designed assuming a nominal 30-year operating life. The plants are licensed to operate for periods of a maximum of ten years followed by periodic safety reviews (PSRs). Extended operation beyond 30 years is possible if accepted by the Canadian Nuclear Safety Commission (CNSC) based upon the results of a PSR. Long-term operation (for example for up to an additional 30 years of operation) typically requires major refurbishment activities which includes the replacement of the original fuel channels and feeder piping and, in some cases, steam generators. |
| Czechia | Initial design life is 30 years for main components, such as steam generators, reactor coolant pumps and pressurisers. The design life of the reactor pressure vessel is 40 years. The operating licence is unlimited, but the national regulator requires a safety review of the nuclear power plants every ten years and can authorise continued operation of a nuclear power plant for another ten years. Improvements required as a result of the PSR process, including updates to the safety standards, of operating experience feedback and of lessons learnt, must be implemented. |
| Finland | Periodic safety reviews are performed every ten years. The Loviisa Units 1 and 2 (WWER-type reactors) were initially designed for a 30-year operating life. The units were subsequently relicensed for an additional 20 years. The current operating licence ends in 2027 and 2030, respectively. Olkiluoto-1/2 (ASEA-Atom BWRs) were designed for a 40-year operating life. They have been relicensed to operate until 2038. The key issues in the licence renewal are ageing management, organisational issues, deterministic and probabilistic safety analyses, environmental fatigue, status of the safety improvements as well as matters relating to the environment, nuclear waste and nuclear fuel. |
| France | The initial design life is 40 years. As the plant ages, the safety authority conducts a periodic safety review (PSR) every ten years to ensure that a plant continues to be in conformity with the original safety standards and with any additional requirements. Improvements required as a result of the PSR process, of updates to the safety standards, of operating feedback and of lessons learnt, must be implemented. |
| Germany | As a legal basis for the operation of the German nuclear installations, the Atomic Energy Act (Atomgesetz, AtG) was amended in 2002 with the aim to phase out the use of nuclear energy for the commercial generation of electricity in a controlled and structured manner. The Act laid down the electricity production rights for each nuclear installation. With the amendment of the AtG on 6 August 2011, continued operation of eight nuclear installations for electricity generation (power operation) was terminated, whereas additional dates for the latest possible termination of power operation were fixed for the remaining nine nuclear installations. The last operating units to be permanently shut down at the end 2022 are Emsland, Isar-2 and Neckarwestheim-2. LTO is no longer an issue in Germany. |

2. For a comprehensive review, see NEA (2019), *Legal Frameworks for Long-Term Operation of Nuclear Power Plants*, OECD Publishing, Paris, www.oecd-nea.org/jcms/pl_1515. Of particular relevance are Chapter 2 ("Review of Approaches to Long-Term Operation") and Chapter 3 ("Country Reports").

## Table 1.2: Commercial nuclear power plant operating licence bases in CODAP project member countries/economies *(cont'd)*

| Country/ economy | Basic principles |
|---|---|
| Japan | Under the new regulatory requirements of 8 July 2013 based on the revised Reactor Regulation Act, the operating life of nuclear power reactors is set at 40 years, in principle. This may be extended one time only for another 20 years if approval is obtained before the expiration of the initial 40 years. With the introduction of this system, a 27 February 2013 NRA Commission Meeting decided that before any extension could be agreed, the current status of the relevant plant would be assessed to determine potential further anticipated deterioration during the proposed extension period. A maintenance policy for this extension period must be included in an application, with particular attention paid to preventive measures against ageing. Regarding commercial power reactors in operation for 30 years or more, a material degradation assessment and a long-term maintenance policy covering every ten years are already required, based on the Reactor Regulation Act and related legislation. |
| Korea | The operator of nuclear installations is required to perform a periodic safety review every ten years from the date of operating licence issuance and submit the results to the Nuclear Safety and Security Commission (NSSC). In case an operator wants to operate a nuclear installation beyond the design life, two additional items such as lifetime evaluation for major components and radiological environmental impact assessment are to be incorporated into the periodic safety review. |
| The Netherlands | The operating licence for the Borssele Nuclear Power Plant was issued in 1973 and it did not contain a predetermined expiration date. This means that as long as the requirements (as stated in the regulations and the operating licence) are fulfilled, the plant is allowed to operate. In 2006 the Dutch government signed an agreement (the "Covenant") with the owners of the Borssele Nuclear Power Plant, which allows for operation until the end of 2033, at the latest. In the meantime the Covenant conditions should be met, in addition to the requirements of the licence. The aforementioned end-date of operation is also a requirement in article 15a of the Dutch Nuclear Energy Law. |
| Slovak Republic | Based on the 2013 revision of the Atomic Act the operating licence is not limited in time, but the licence holder must comply with the provisions of law, i.e. every ten years demonstrate by a periodic safety review (PSR) the readiness of the facility for further operation. The regulatory authority ÚJD SR can complement the operating licence with specific conditions. |
| Spain | The Spanish nuclear power plants are subject to a system of operating permit renewal for a given period. Likewise, every ten years the plants carry out a PSR, updating the situation of the continuous safety assessment programmes that are performed systematically and analysing the applicability of the changes occurring in the standards over the ten-year period (Condition Application Standards, NAC). Facilities requesting long-term operation must include in their PSR an integrated ageing assessment and management plan, among other additional requirements. |
| Switzerland | In the aftermath of the 2011 Fukushima Daiichi accident, the Swiss government decided to phase out nuclear energy. Existing plants will continue to operate as long as they are considered safe by ENSI and fulfil all legal and regulatory requirements in this respect. Assessments of LTO have been performed for two Swiss nuclear power plants which have been in commercial operation for over 40 years. A detailed examination demonstrated that the conditions for taking out of service a nuclear power plant are not yet and will not be reached by Beznau Nuclear Power Plant and Mühleberg Nuclear Power Plant within the next ten years. Nevertheless, it is mandatory to continue with the scheduled ageing management, maintenance and backfitting activities. Mühleberg Nuclear Power Plant was permanently shut down at the end of 2019 after 47 years of operation. |
| Chinese Taipei | The nuclear power plants in Chinese Taipei have an initial design life of 40 years. According to the regulations, a life-extension application must be filed 5 to 15 years before the expiration of the operating licence (OL). However, in response to the Fukushima Daiichi accident, the government's national energy policy was revised in November 2011 with a provision of no life extension beyond 40 years for all nuclear units. As a result of this policy, Chinshan-1 was permanently shut down in December 2018 and Chinshan-2 was permanently shut down in July 2019. |
| United States | The Atomic Energy Act authorises the Nuclear Regulatory Commission to issue licences for commercial power reactors to operate for up to 40 years. These licences can be renewed for an additional 20 years at a time. The period after the initial licensing term is known as the period of extended operation. Economic and antitrust considerations, not limitations of nuclear technology, determined the original 40-year term for reactor licences. As of the end of 2020, 49 reactor units were in an extended period of operation, operating beyond 40 years. |

**Figure 1.2: Number of commercial reactor units in long-term operation in CODAP project member countries/economies**



# Chapter 2. Temporal changes in material degradation

This chapter presents high-level summaries of the material degradation mechanisms that have been observed to act on metallic piping systems as well as certain non-piping passive components. These degradation mechanisms are for the most part well understood and in some cases they have produced failures with significant operational impacts. Using the results from root cause evaluations and materials research, detailed ageing management processes have been implemented to prevent recurring failures or to mitigate the effects of active degradation mechanisms. Systematic OPEX data evaluations provide insights into the effectiveness of degradation mechanism mitigation processes and monitoring programmes.

## 2.1 The CODAP-OPEX knowledge base

Extensive material performance information exists, and it covers field experience obtained from more than five decades of commercial nuclear power plant operation. Organised by type of material, degradation mechanism and time period, the field experience is summarised in Tables 2-1 through 2-3 and Figure 2-1. For non-piping passive components, corresponding to about 10% of the total event population in CODAP, Figure 2-2 summarises the field experience by component type and the time period in which a failure was observed. Only selected representative events are included in CODAP. The four predominant types of metallic materials in current use are:

- Carbon steels are used extensively in primary-[1] and secondary-side piping systems. The OPEX is extensive. For piping in a raw water operating environment there is an upwards failure trend indicative of ageing management challenges.
- Low-alloy steels are used extensively in balance-of-plant systems and as a replacement for carbon steels in wet steam operating environments. Hundreds of failures have been reported for low-alloy steels.
- Nickel-base alloys are used extensively in primary systems of light water reactors in connections between ferritic components and austenitic piping. The OPEX consists of well over 500 failures. Since the 1990s a downward trend is noted.
- Stainless steels are used extensively in primary- and secondary-side piping systems. The OPEX is extensive and with a clearly discernible and well understood downward failure trend since the 1990s. There are multiple grades of stainless steels. Austenitic stainless steels are the most common.

The degradation mechanisms acting on the above listed material types are well understood and managed through different types of reliability and integrity management (RIM) programmes; reference [33]. For process piping systems the OPEX is organised by degradation mechanisms; Figure 2-3. Summarised in Figures 2-4 through 2-9 are “selected OPEX excerpts” from CODAP. These summaries do not account for the plant population that produced the failure data in a given time period. A formalised assessment of ageing trends is beyond the scope of this Topical Report.

1. As an example, carbon steels are used for the CANDU primary heat transport system (HTS) feeder piping.

**Table 2-1: Effect of ageing management on material performance in SCC- and fatigue-susceptible environments**

| | Degradation mechanism (DM) | Material(s) | Operating environment(s) | Steps taken to manage/ mitigate degradation (Post-1980s) | Ageing management effect on OPEX (plant age > 25 years) |
|---|---|---|---|---|---|
| Stress corrosion cracking | ECSCC | Stainless steel (SS) – multiple grades | Multiple | Improved fabrication practices, cleanliness, pipe surface decontamination | Inconclusive (i.e. not possible to make clear distinction between the "early-life OPEX" vs. "PEO/LTO OPEX") |
| | TGSCC | | | | |
| | IGSCC | Unstabilised austenitic SS | BWR primary water (HWC, NMCA) | Improved water chemistry control, application of stress improvement process (e.g. MSIP, peening), use of SCC-resistant material, application of FSWOL, minimise number of welds) | Significant reduction noted in SCC incident rates |
| | | Stabilised austenitic SS | | | |
| | | Alloy 82/182 | | | |
| | PWSCC | Alloy 600/82/182 | High-temperature PWR primary water | Use of SCC-resistant material (e.g. Alloy 690/52/152), application of stress improvement process, application of FSWOL | |
| Fatigue | HCF (small-diameter lines) | Multiple | Multiple | Elimination of socket welds, fatigue monitoring, improved piping configuration, installation of vibration dampers | Inconclusive |
| | LCF | Multiple | Multiple | Improved piping configuration (e.g. added supports), enhanced welding technology | Inconclusive |
| | Thermal fatigue | Multiple | Hot/cold fluid mixing | Fatigue monitoring, improved NDE technology, improved system operating procedures, piping system re-configuration | "Somewhat" inconclusive – to be further investigated, for more details see for example Reference [38] |
| | Corrosion fatigue/ environmental degradation | Multiple | High-temperature/ high-pressure primary/ secondary side system environment | Reconfiguration of piping (e.g. PWR feedwater nozzle configuration). Fatigue monitoring, improved plant operating procedures | Inconclusive (only limited OPEX exists, most of which are from the 1970s to early 1980s). Relationship between field experience and experimental and theoretical work yet to be determined |

**Table 2-2: Effect of ageing management on material performance in FAD-susceptible and corrosive environments**

| | Degradation mechanism (DM) | Material(s) | Operating environment(s) | Steps taken to manage/ mitigate degradation | Ageing management effect on OPEX (plant age > 25 years) |
|---|---|---|---|---|---|
| Flow assisted degradation (FAD) | Erosion-Cavitation | Multiple | Multiple | Piping system reconfiguration, improved system operating procedures | Inconclusive – a function of the effectiveness of knowledge transfer |
| Flow assisted degradation (FAD) | Erosion-Corrosion (E/C) | CS | Multiple | Use of E/C-resistant materials | Inconclusive |
| Flow assisted degradation (FAD) | Flow-assisted corrosion | CS, LAS | High-energy piping, single or two-phase flow conditions | Use of FAC-resistant material, implementation of FAC monitoring programme, improved secondary-side water chemistry | Significant reduction noted for FAC-induced leaks and ruptures |
| Flow assisted degradation (FAD) | Liquid droplet impingement erosion | LAS, SS | High-energy piping, single or two-phase flow conditions | Reconfiguration of piping, improved operating procedures/operational strategies (e.g. baseload vs. load following) | Inconclusive; the relationships between extended power uprate and LDIE-susceptibility to be determined |
| Corrosion | Corrosion – General | Al-Bronze, CS, SS (e.g. 300-Series) | Raw water (e.g. brackish, lake, pond, river, sea) | Chemical treatment of water, use of corrosion resistant material incl. non-metallic material, reconfiguration of piping, application of composite repair technology[1], installation of cathodic protection system, etc. | Inconclusive – however, high-alloy austenitic stainless steels (e.g. AL-6XN®, 254-SMO® and 654-SMO®) and HDPE materials appear to perform well in corrosive environments |
| Corrosion | Crevice corrosion | | | | |
| Corrosion | De-alloying – Selective leaching | | | | |
| Corrosion | Galvanic corrosion | | | | |
| Corrosion | Microbial corrosion | | | | |

1. For more information, see for example https://adams.nrc.gov/wba (Accession No. ML20014E476; "30 to 96-inch Pipelines Upgrade Case Study: Navigating Safety Related Pipeline Upgrades with CFRP" & ML20014E506, "Carbon Fiber Reinforced Plastic (CFRP) Repair – Future Applications and R&D Gaps").

**Table 2-3: Degradation mechanisms by the time period in which a failure was discovered**

| Degradation mechanism | | OPEX summary by time period (No. of failures) | | | |
|---|---|---|---|---|---|
| | | 1971-2005 | 2006-10 | 2011-15 | 2016-20 |
| Stress corrosion cracking | ECSCC – BWR – Piping | 83 | 9 | 3 | 2 |
| | ECSCC – PWR – Piping | 98 | 43 | 17 | 7 |
| | IGSCC – BWR – Piping | 1 380 | 43 | 13 | 18 |
| | IGSCC – PWR – Piping | 119 | 8 | 22 | 11 |
| | PWSCC – PWR – Piping | 92 | 47 | 6 | 14 |
| | PWSCC – PWR – Vessel head penetration | 119 | 10 | 29 | 13 |
| | TGSCC – BWR – Piping | 130 | 14 | 0 | 0 |
| | TGSCC – PWR – Piping | 34 | 4 | 7 | 2 |
| Fatigue | HCF – BWR – Socket weld | 144 | 9 | 12 | 9 |
| | HCF – PWR – Socket weld | 457 | 40 | 25 | 25 |
| | HCF – All plant types and all piping systems | 1 550 | 150 | 168 | 146 |
| | LCF – All plant types and all piping system | 161 | 30 | 42 | 32 |
| | TF – BWR | 73 | 5 | 5 | 3 |
| | TF – PWR | 117 | 6 | 15 | 20 |
| Corrosion | Corrosion – FP system | 144 | 64 | 91 | 50 |
| | Corrosion – BWR – SW system | 152 | 91 | 135 | 141 |
| | Corrosion – PWR – SW System | 683 | 178 | 292 | 280 |
| | MIC – All plant types and all systems | 426 | 169 | 371 | 415 |
| Flow-assisted degradation | FAC – BWR – All systems | 422 | 69 | 36 | 20 |
| | FAC – PWR – All systems | 1 305 | 159 | 67 | 29 |
| | Erosion-Cavitation – All plant types and all systems | 124 | 41 | 33 | 39 |
| | Erosion-Corrosion - All plant types and all systems | 314 | 86 | 60 | 60 |

**Figure 2-1: The piping operating experience by material type and time period**

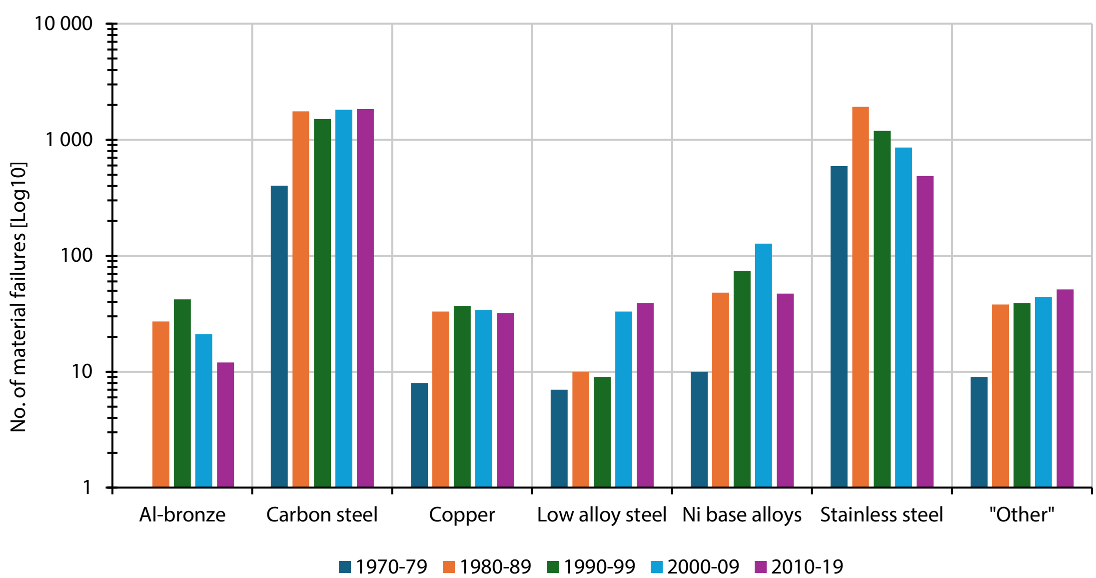


**Figure 2-2: The non-piping passive component OPEX by material type and time period**

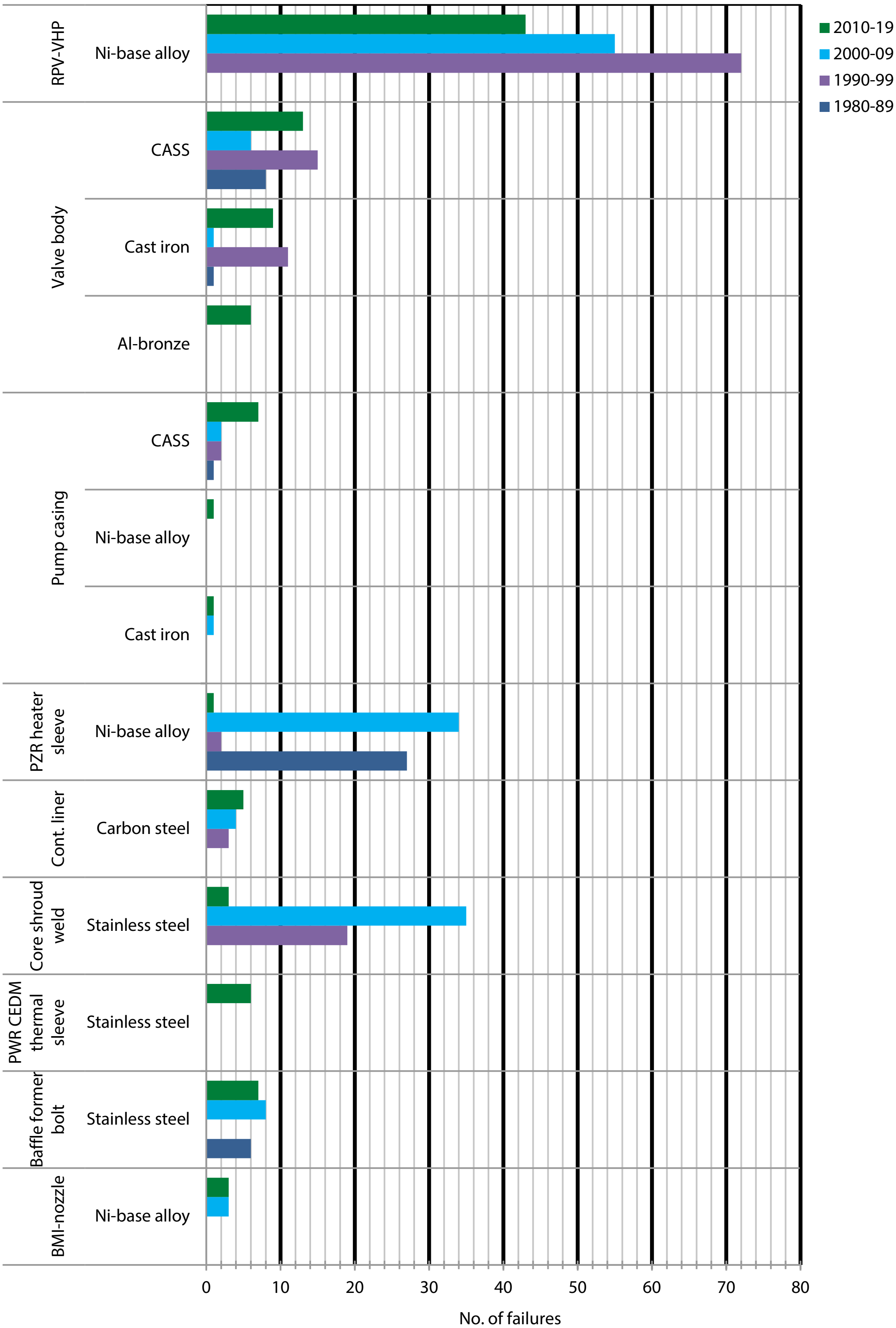


**Figure 2-3: Piping material degradation mechanisms observed during five decades of commercial nuclear power plant operation experience**

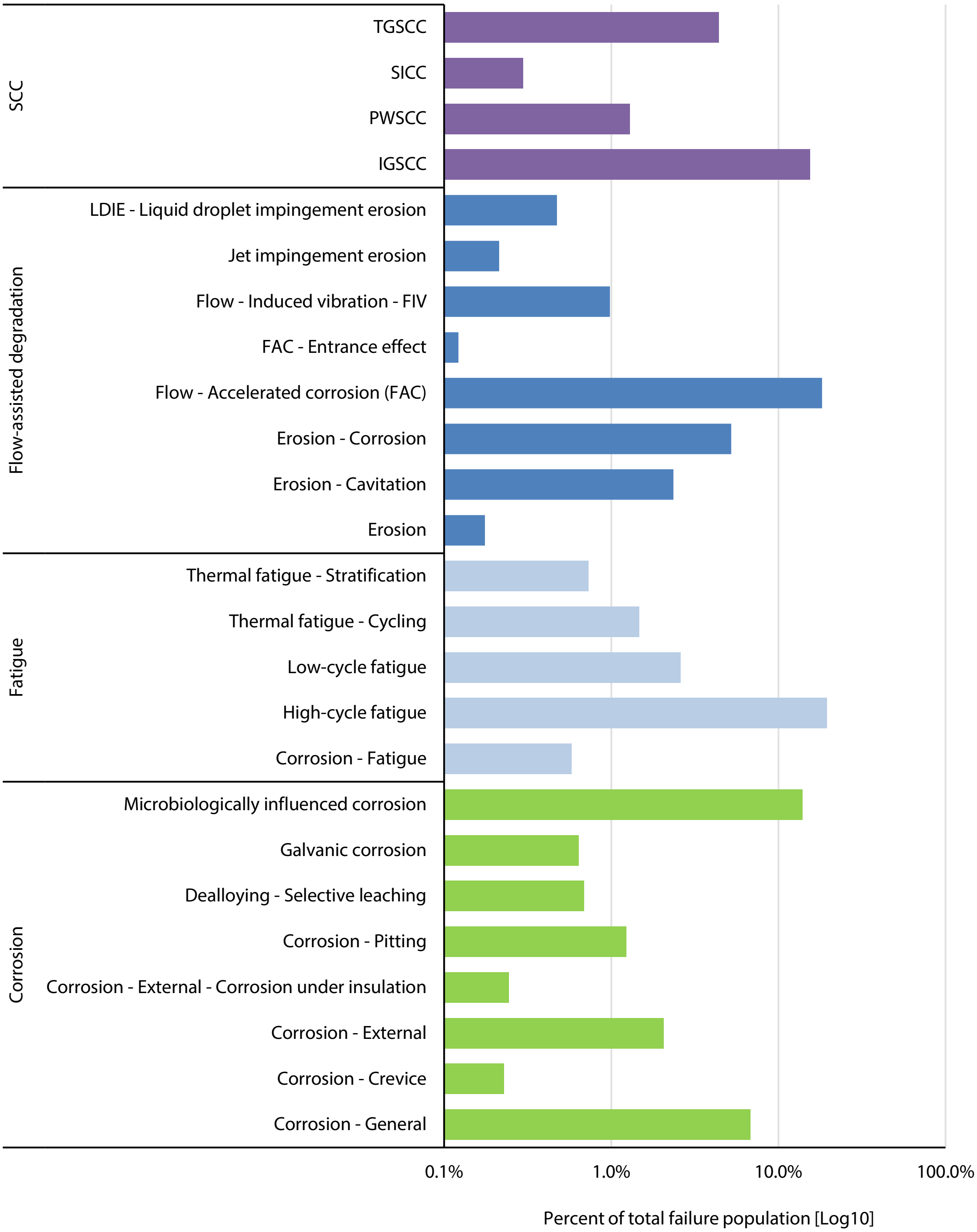


**Figure 2-4: Piping OPEX by system group and age of plant at time of failure**

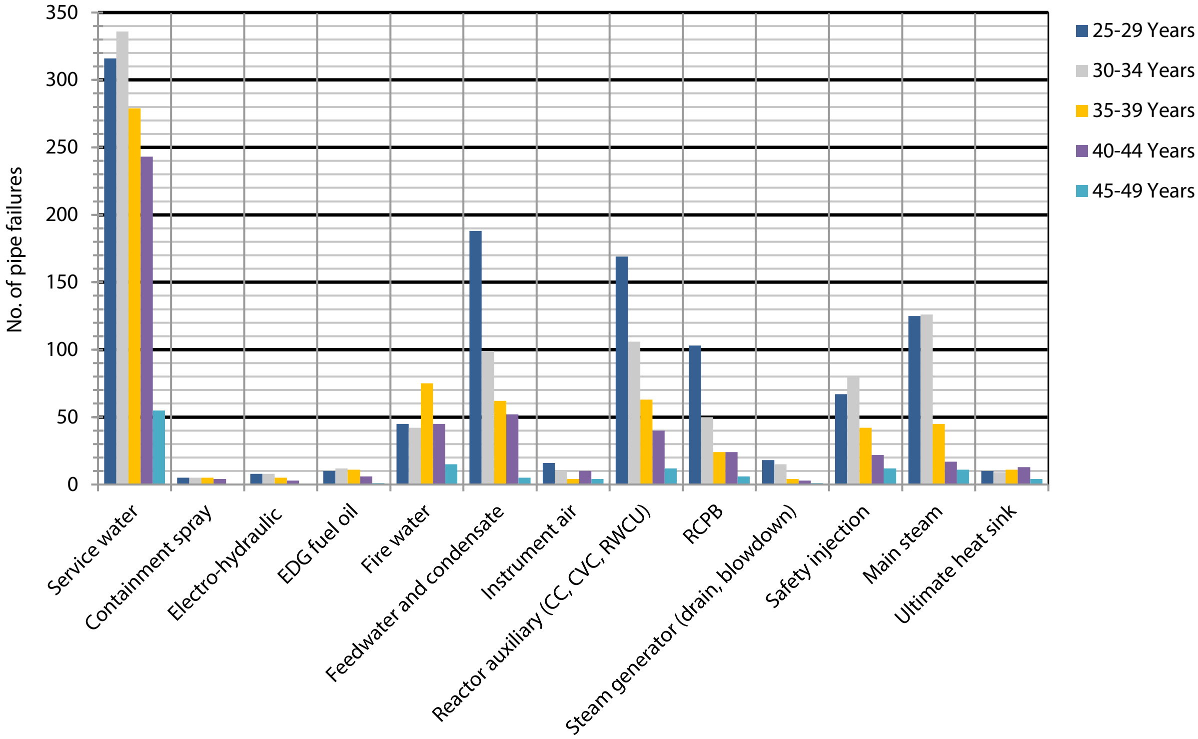



**Figure 2-5: Piping OPEX by system group and for 0 ≤ T < 25 reactor operating years**

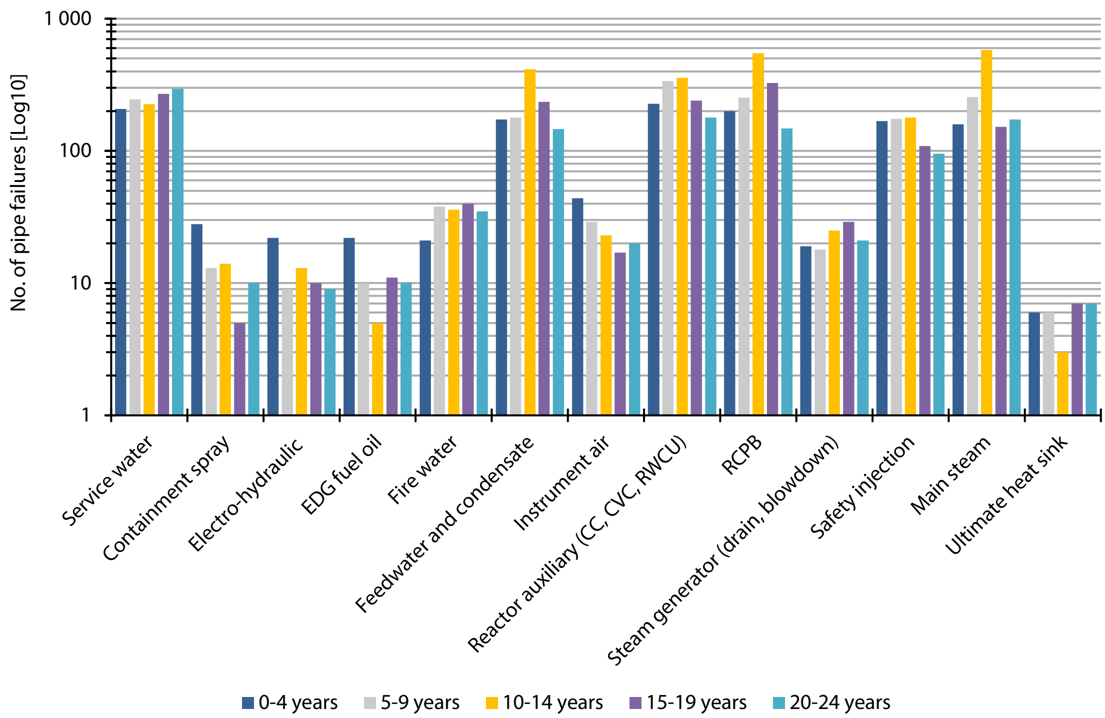


**Figure 2-6: Piping OPEX by safety class and plant age**

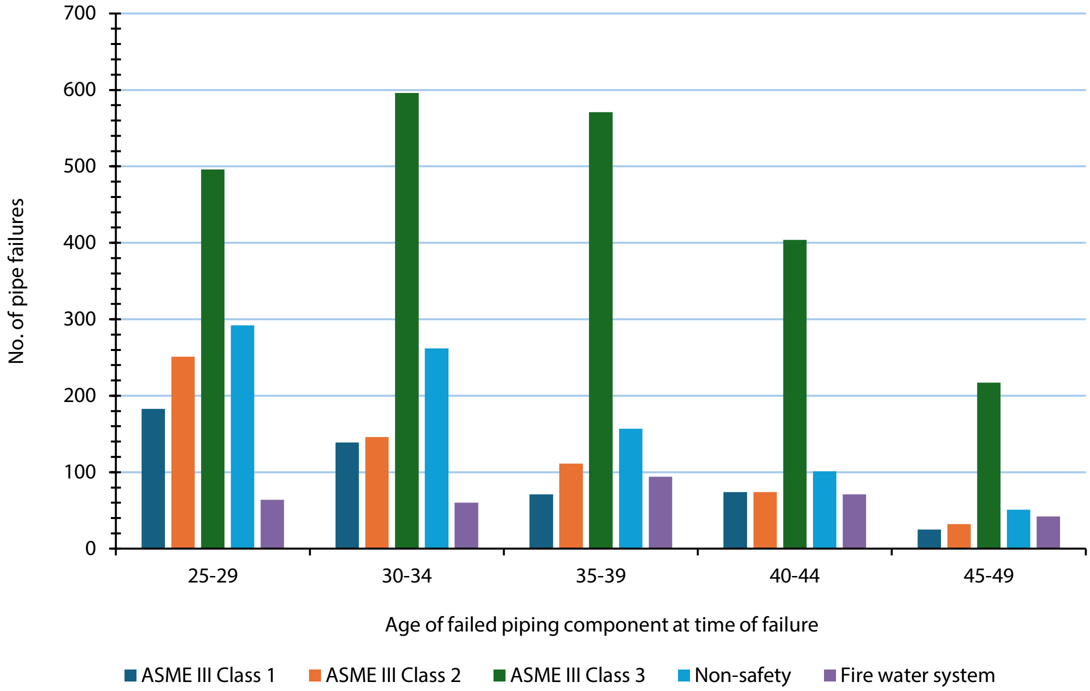



**Figure 2-7: The PWSCC operating experience**

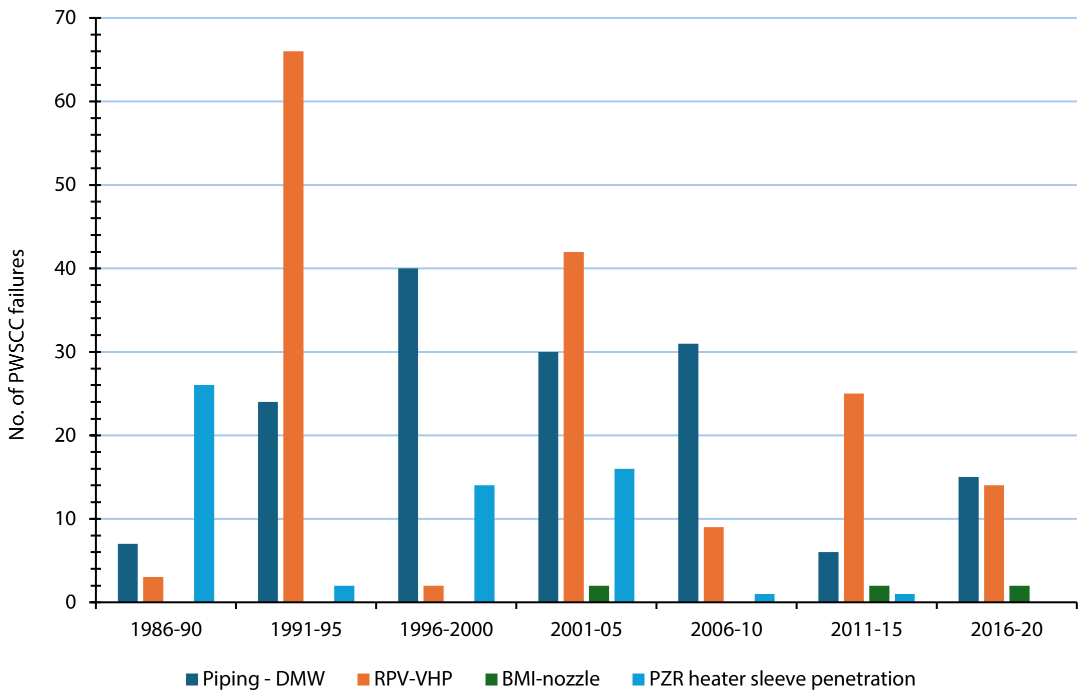


**Figure 2-8: Reactor internals OPEX summary by plant type, component type and time period**

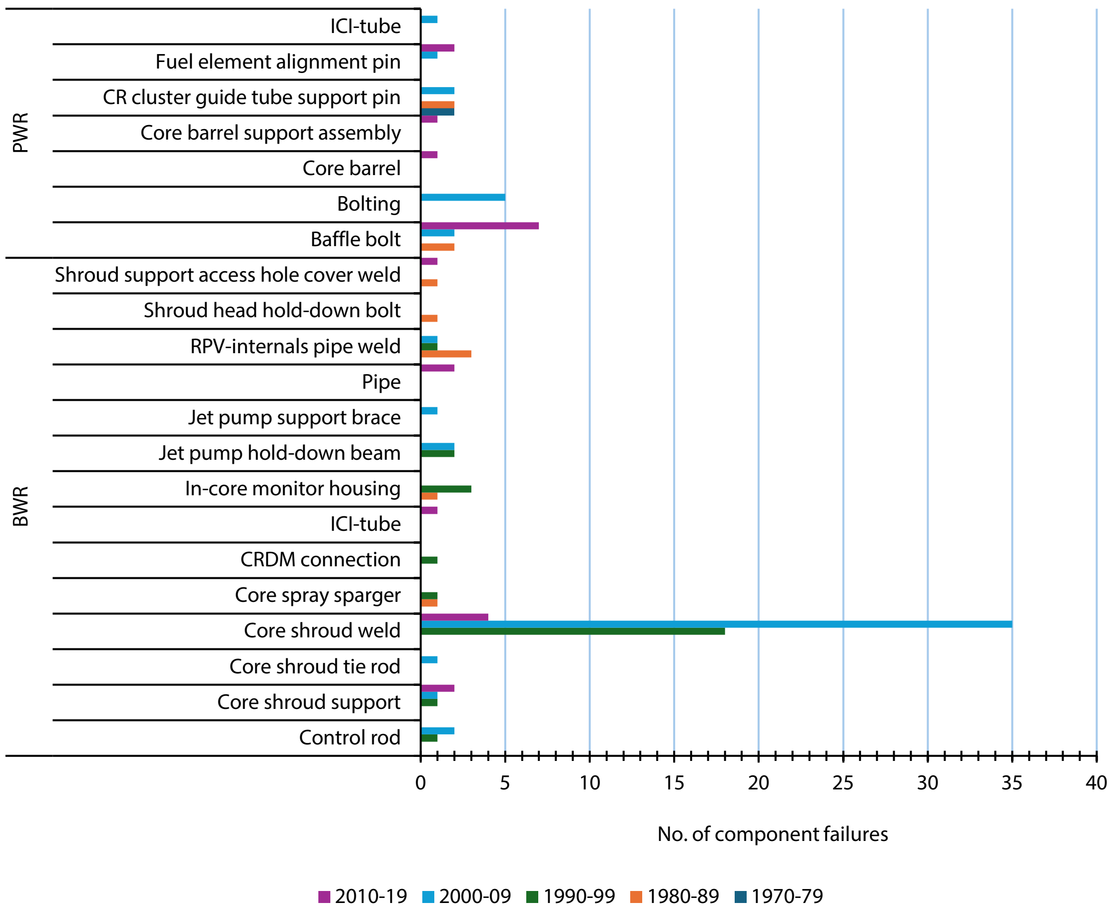
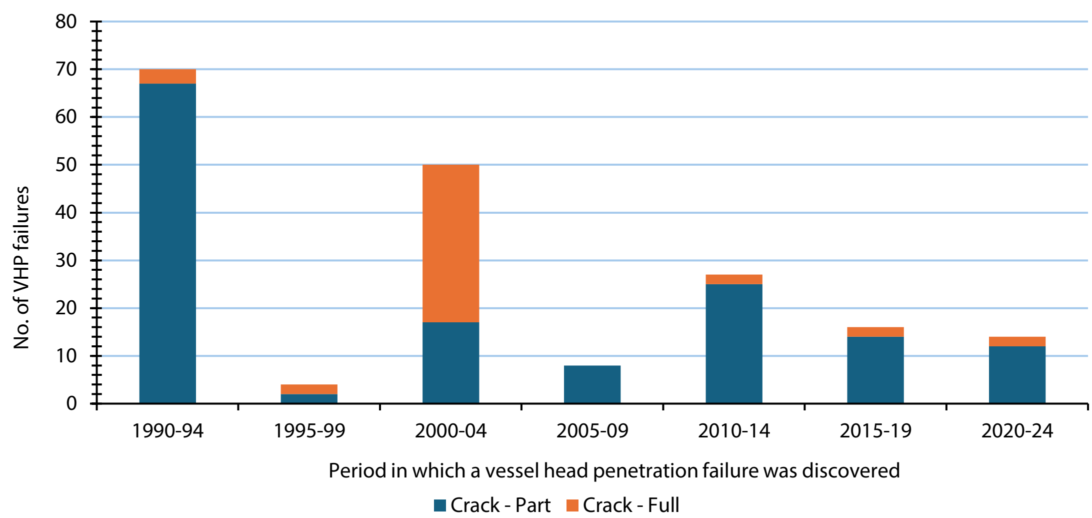



**Figure 2-9: PWSCC in PWR reactor pressure vessel head penetrations (VHP)**



## 2.2 Degradation mechanisms acting on process piping

This section includes tabular and graphical summaries of the piping degradation and failure OPEX. Represented in CODAP are multiple nuclear steam supply (NSSS) designs as well as multiple design generations; Figure 2-10. Indicated in Figure 2-11 are the respective CODAP member country/economy contributions to the OPEX database.

About 70% of the event population originates in the United States. This is not only due to the relative number of reactors in the United States, but also fundamental differences in event reporting requirements to the regulatory authorities among the member countries/economies. The disproportionality is also attributed to the origin of the event database, the data collection process as it was originally implemented, national regulations, and the ability to exchange restricted data on material performance and failures.

The piping OPEX data are organised by the time-period in which a failure was observed, as well as by the degradation mechanism that caused or contributed to a failure. The following summaries are provided:

- Table 2-4. Summary of the piping operating experience by time period in which a failure was observed, safety class and outside pipe diameter (OD) in mm. The fire water system OPEX in this chart comes from Finnish [11][37] and US nuclear power plants.
- Figure 2-12. Small-diameter piping OPEX. The predominant failure mechanisms are crack initiation and propagation via fretting or high-cycle fatigue (HCF). In this summary the term "small-diameter" refers to piping with outside diameter ø ≤ 60.3 mm (≤ NPS2). The chart includes butt welded and socket welded piping components.
- Figure 2-13. This chart shows the small-diameter socket weld OPEX sub-population (987 failures).
- Figure 2-14. Medium-diameter piping OPEX. In this summary the term "medium-diameter" refers to piping with outside diameter in the range of 60.3 mm < ø ≤ 273.1 mm (2 < NPS ≤ 10).
- Figure 2-15. Large-diameter piping OPEX ø > 273.1 mm (> NPS10).
- Figure 2-16. Summary of the OPEX involving the four different fatigue mechanisms: 1) corrosion fatigue, 2) high-cycle fatigue, 3) low-cycle fatigue, and 4) thermal fatigue.
- Figure 2-17. Summary of the OPEX involving stress corrosion cracking (SCC).
- Figure 2-18. Summary of the WWER-440/213 OPEX as recorded in CODAP. OPEX has been recorded for Bohunice-3/4, Dukovany-1/2/3/4, Loviisa-1/2, Mochovce-1/2 and Temelín-1/2.
- Figure 2-19. Summary of the OPEX involving flow-assisted degradation; 1) erosion-cavitation, 2) erosion-corrosion, 3) flow-accelerated corrosion (FAC), and 4) liquid droplet impingement erosion (LDIE).
- Figure 2-20. Summary of the OPEX involving corrosion failures, mainly in service water (SW) and fire protection (FP) water systems.
- Figure 2-21. Summary of the OPEX involving service water system pipe failures; safety-related and non-safety-related piping.
- Figure 2-22. Summary of the OPEX involving Fire Water system pipe failures. In this chart the term "Corrosion – Various mechanisms" includes "general corrosion, pitting, crevice corrosion, galvanic corrosion".

**Figure 2-10: The CODAP OPEX by NSSS design**

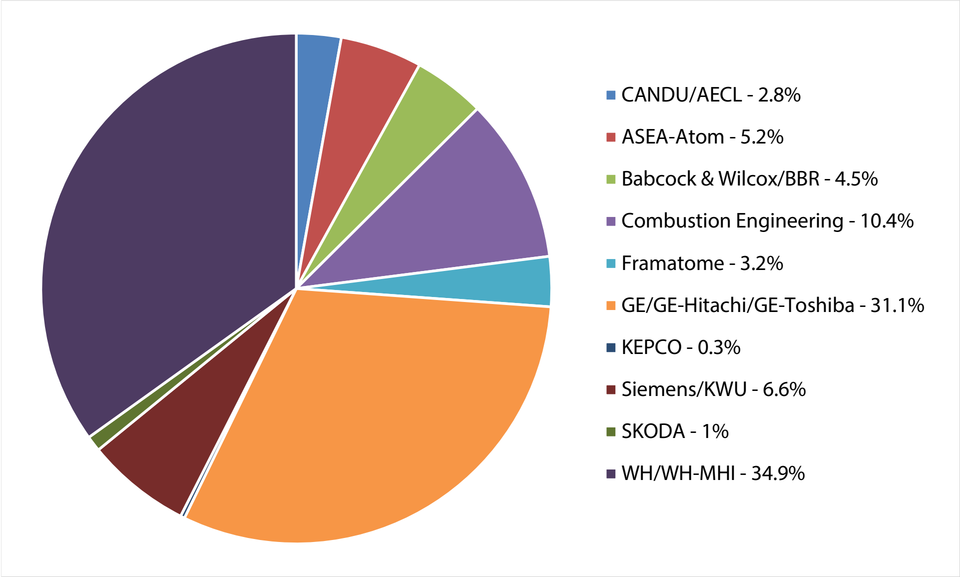
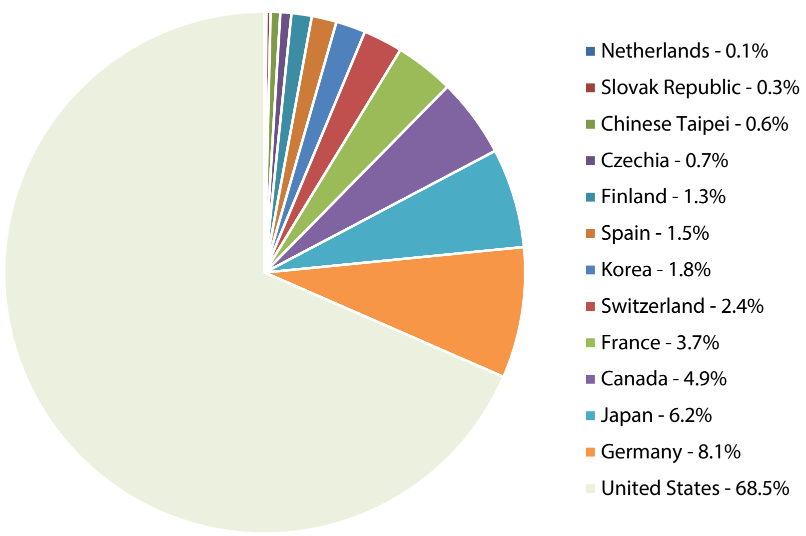



**Figure 2-11: The piping and non-piping passive component OPEX by CODAP member country/economy**



## Table 2-4: Piping operating experience by time period, safety class and outside diameter

| Safety class | Outside diameter [mm] | Number of failures per time period | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | 1970-74 | 1975-79 | 1980-84 | 1985-89 | 1990-94 | 1995-99 | 2000-04 | 2005-09 | 2010-14 | 2015-19 |
| 1 | ∅ ≤ 21.3 | 2 | 14 | 27 | 14 | 9 | 25 | 18 | 22 | 13 | 8 |
| | 21.3 < ∅ ≤ 60.3 | 18 | 44 | 54 | 77 | 60 | 78 | 67 | 64 | 47 | 60 |
| | 60.3 < ∅ ≤ 114.3 | 18 | 20 | 31 | 13 | 24 | 17 | 22 | 35 | 11 | 11 |
| | 114.3 < ∅ ≤ 273.1 | 2 | 13 | 91 | 40 | 101 | 47 | 9 | 7 | | 4 |
| | ∅ > 273.1 | | 19 | 348 | 223 | 55 | 151 | 123 | 65 | 17 | 10 |
| 2 | ∅ ≤ 21.3 | 2 | 10 | 24 | 13 | 21 | 20 | 15 | 29 | 6 | 4 |
| | 21.3 < ∅ ≤ 60.3 | 29 | 131 | 131 | 130 | 161 | 125 | 117 | 76 | 67 | 40 |
| | 60.3 < ∅ ≤ 114.3 | 6 | 33 | 31 | 143 | 27 | 23 | 16 | 17 | 27 | 10 |
| | 114.3 < ∅ ≤ 273.1 | 7 | 43 | 113 | 53 | 55 | 44 | 32 | 24 | 36 | 9 |
| | ∅ > 273.1 | 4 | 22 | 47 | 57 | 40 | 38 | 45 | 40 | 18 | 9 |
| 3 | ∅ ≤ 21.3 | 3 | 11 | 27 | 21 | 9 | 18 | 22 | 14 | 25 | 20 |
| | 21.3 < ∅ ≤ 60.3 | 16 | 77 | 105 | 82 | 97 | 157 | 143 | 149 | 200 | 116 |
| | 60.3 < ∅ ≤ 114.3 | 3 | 16 | 37 | 16 | 55 | 67 | 83 | 83 | 117 | 80 |
| | 114.3 < ∅ ≤ 273.1 | 5 | 44 | 14 | 20 | 78 | 77 | 79 | 108 | 124 | 124 |
| | ∅ > 273.1 | 2 | 9 | 19 | 83 | 89 | 75 | 127 | 124 | 173 | 158 |
| 4 – Balance of plant | ∅ ≤ 21.3 | | 9 | 14 | 17 | 11 | 11 | 9 | 10 | 6 | 2 |
| | 21.3 < ∅ ≤ 60.3 | 20 | 58 | 84 | 136 | 131 | 119 | 122 | 94 | 55 | 39 |
| | 60.3 < ∅ ≤ 114.3 | 9 | 14 | 19 | 42 | 39 | 33 | 48 | 66 | 32 | 19 |
| | 114.3 < ∅ ≤ 273.1 | 7 | 20 | 144 | 324 | 49 | 46 | 74 | 81 | 38 | 21 |
| | ∅ > 273.1 | 7 | 14 | 123 | 331 | 46 | 73 | 77 | 86 | 41 | 13 |
| Fire water | ∅ ≤ 21.3 | | | 1 | | 1 | | | | | |
| | 21.3 < ∅ ≤ 60.3 | | | 4 | 4 | 7 | 6 | 2 | 11 | 18 | 9 |
| | 60.3 < ∅ ≤ 114.3 | | | 5 | 1 | 4 | 10 | 7 | 8 | 23 | 7 |
| | 114.3 < ∅ ≤ 273.1 | | 1 | 5 | 8 | 9 | 20 | 17 | 31 | 42 | 27 |
| | ∅ > 273.1 | | 1 | 9 | 2 | 4 | 11 | 25 | 20 | 18 | 14 |
| Total no. of pipe failures: | | 160 | 623 | 1 507 | 1 850 | 1 182 | 1 291 | 1 299 | 1 264 | 1 154 | 814 |

**Figure 2-12: Small-diameter pipe failure OPEX by time period, safety class and outside diameter**

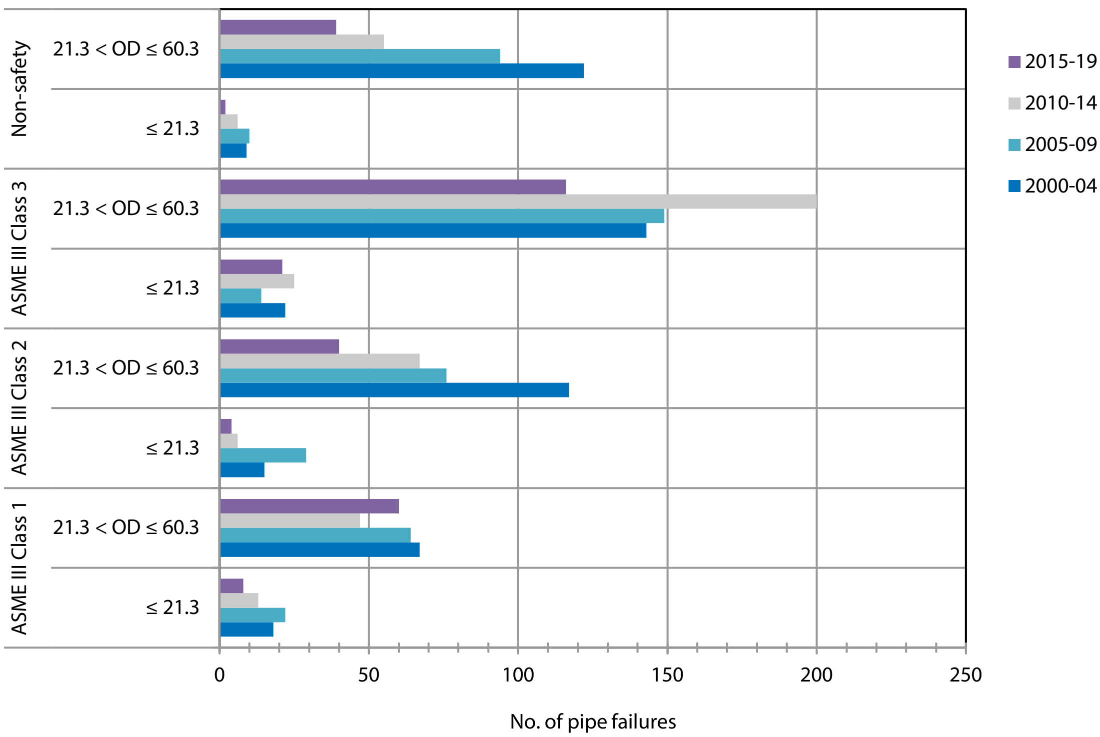



**Figure 2-13: Socket weld failure OPEX data by time period**

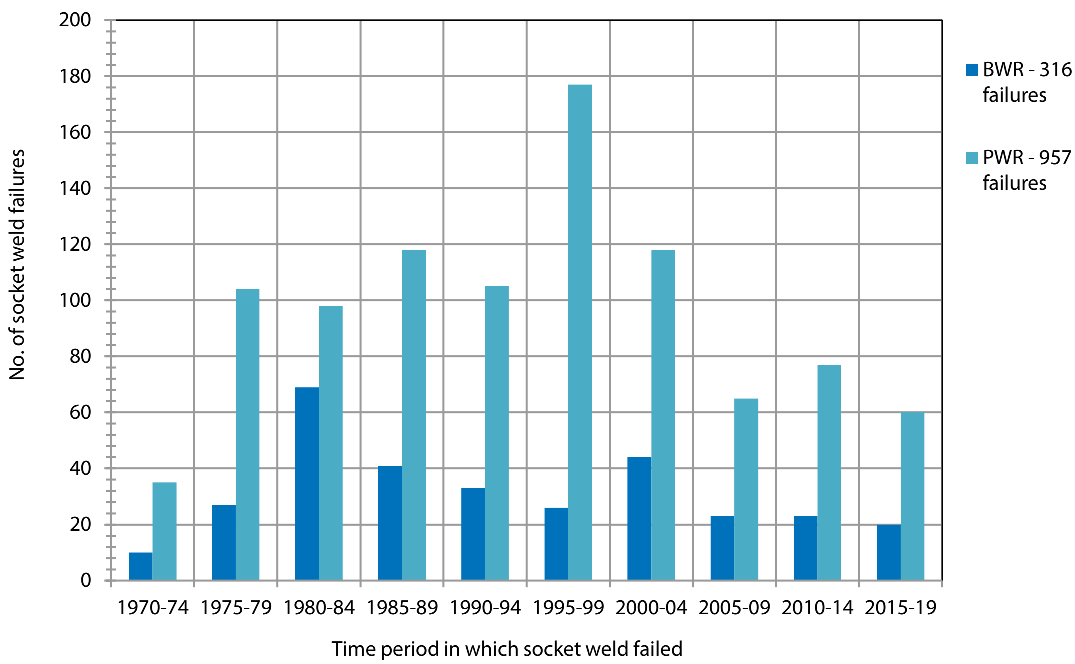


**Figure 2-14: Medium-diameter pipe operating experience by time period, safety class and outside diameter**

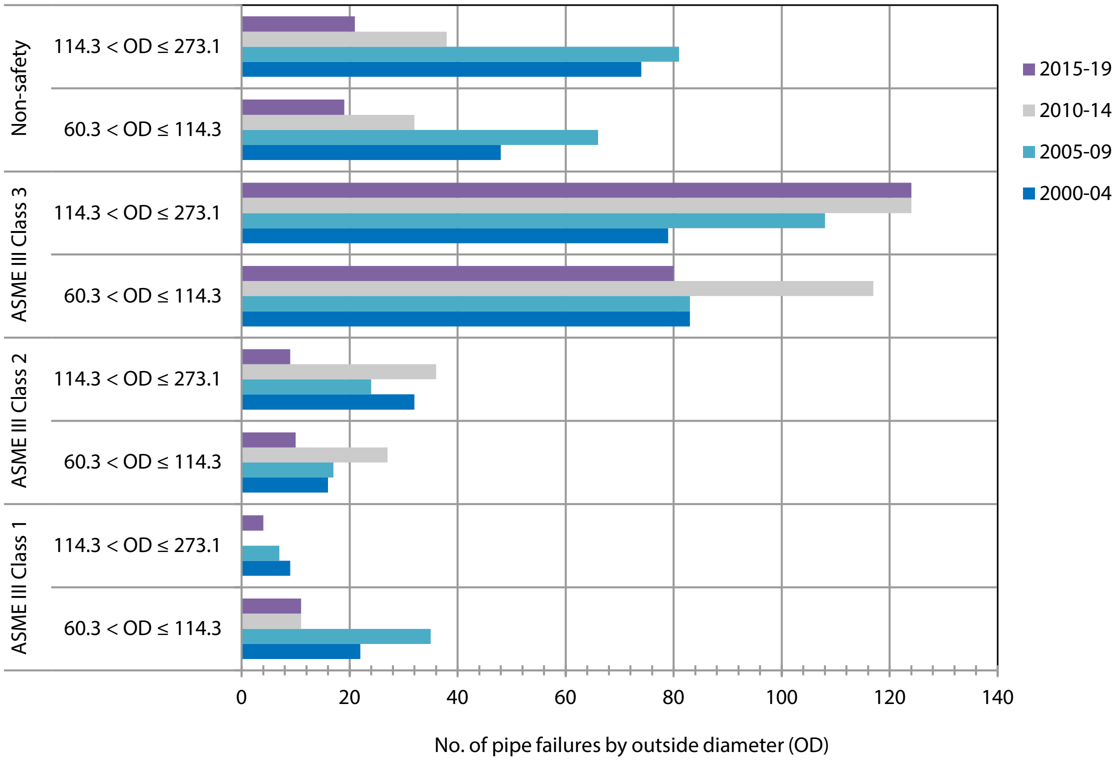



**Figure 2-15: Large-diameter (OD > 273.1 mm) pipe OPEX by time period and safety class**

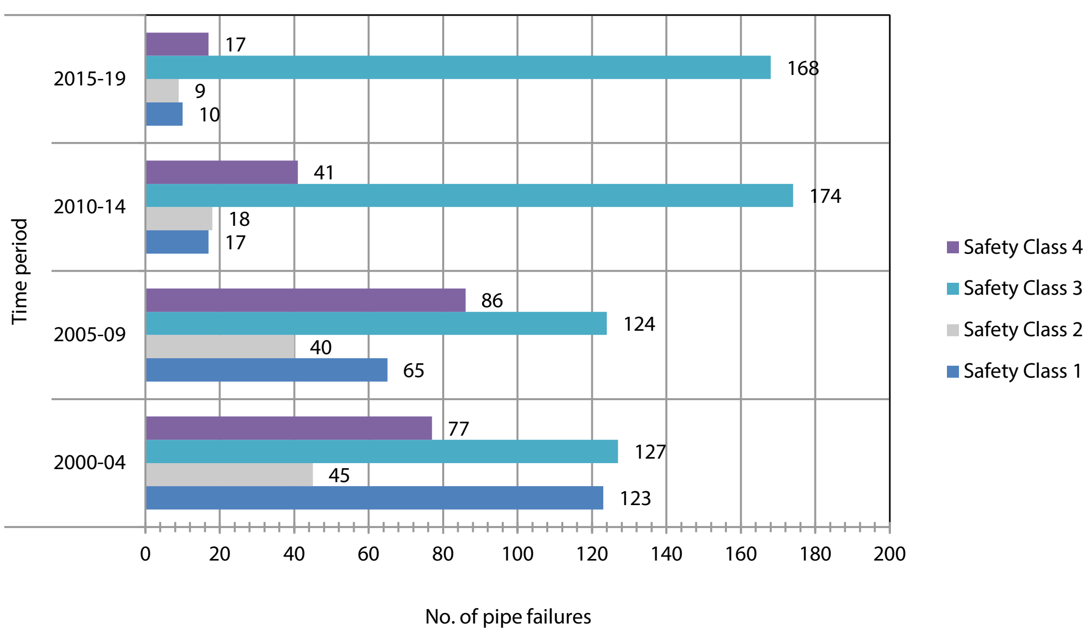


**Figure 2-16: Piping fatigue operating experience**

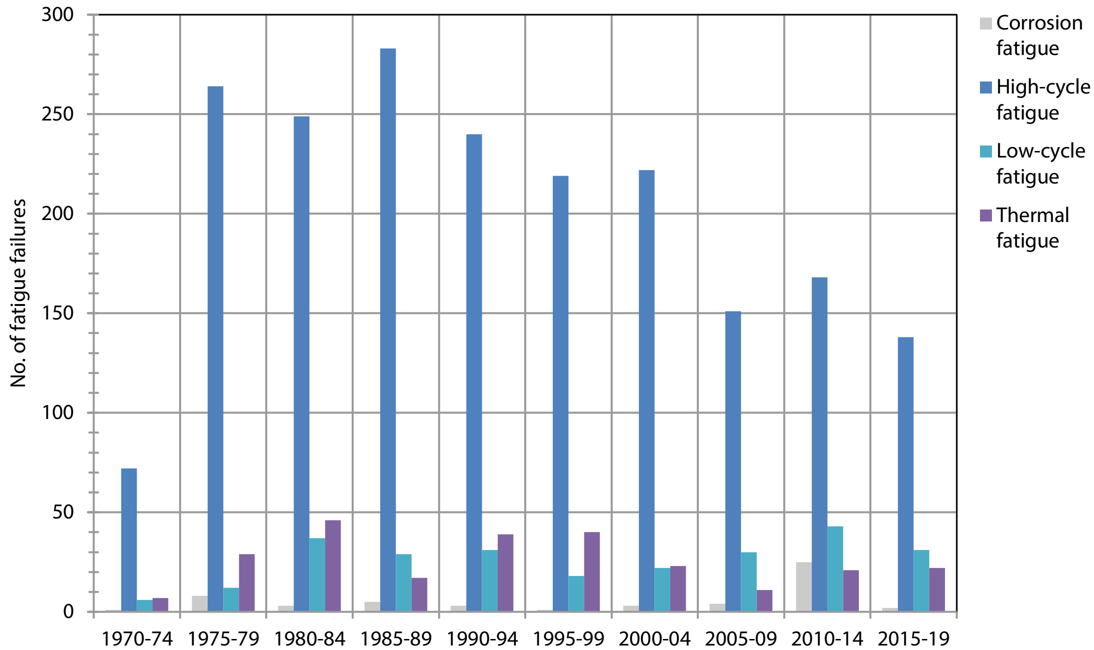



**Figure 2-17: Piping operating experience involving stress corrosion cracking**

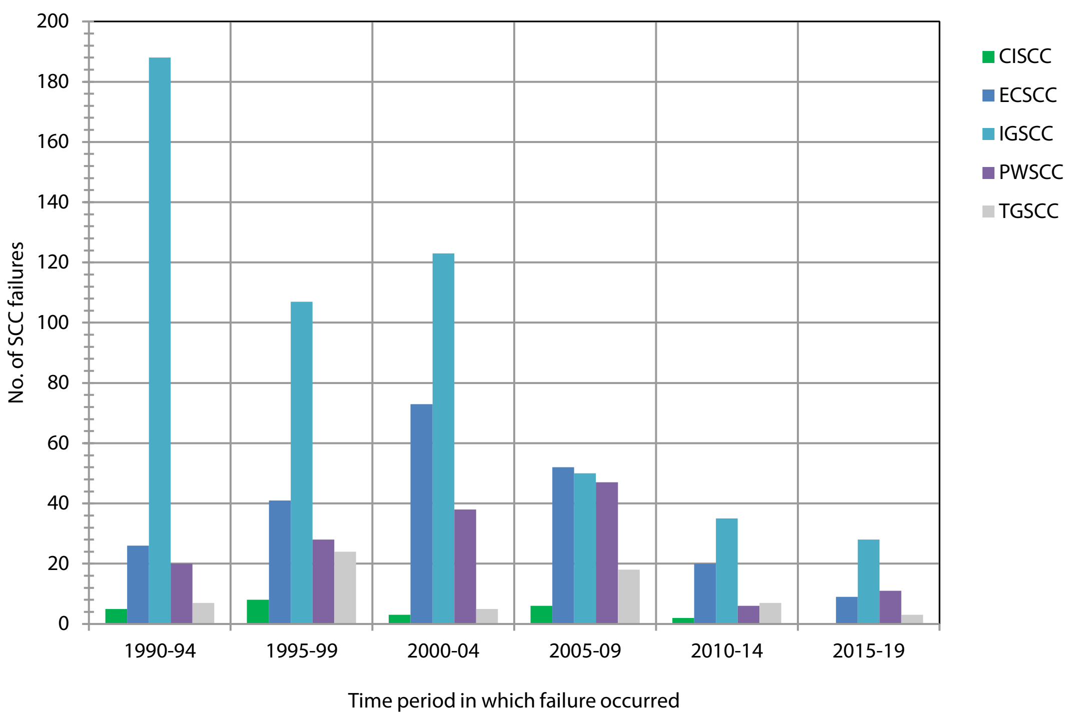


**Figure 2-18: The VVER-440/213 passive component OPEX in CODAP organised by safety class**

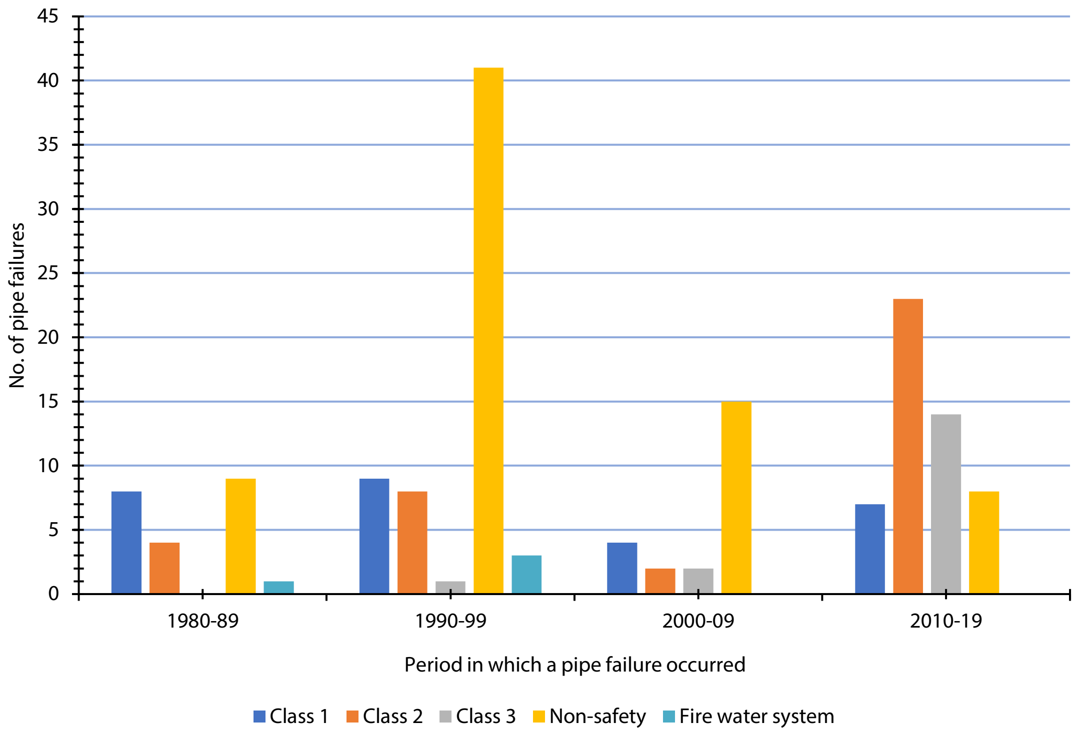



**Figure 2-19: Piping operating experience involving flow-assisted degradation mechanisms**

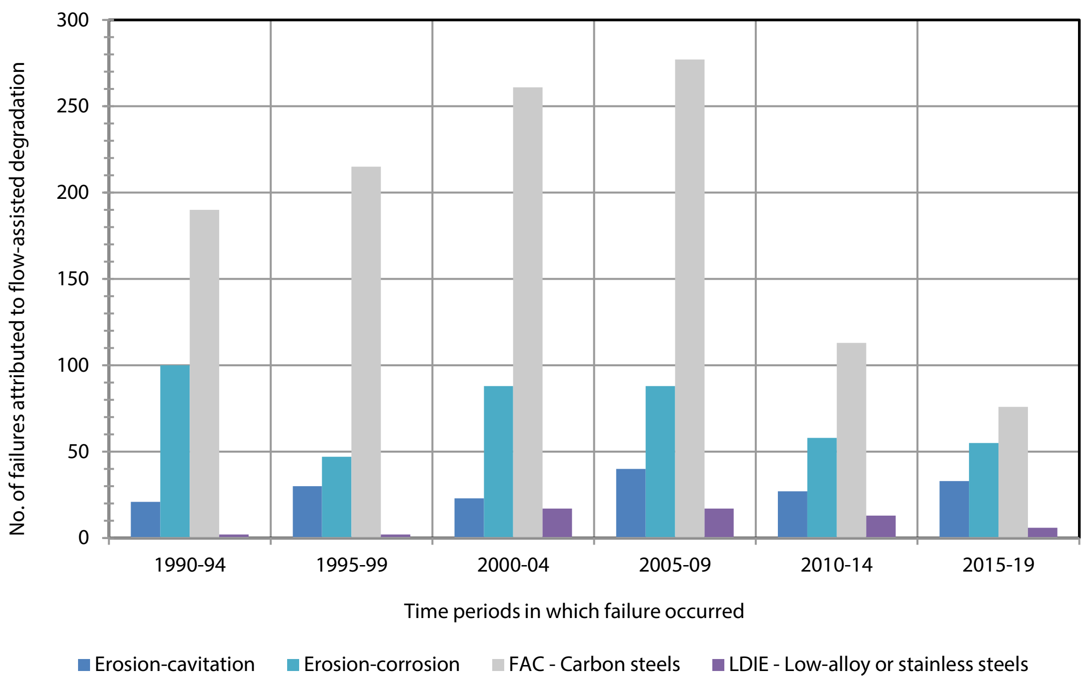


**Figure 2-20: Piping operating experience involving corrosion failures**

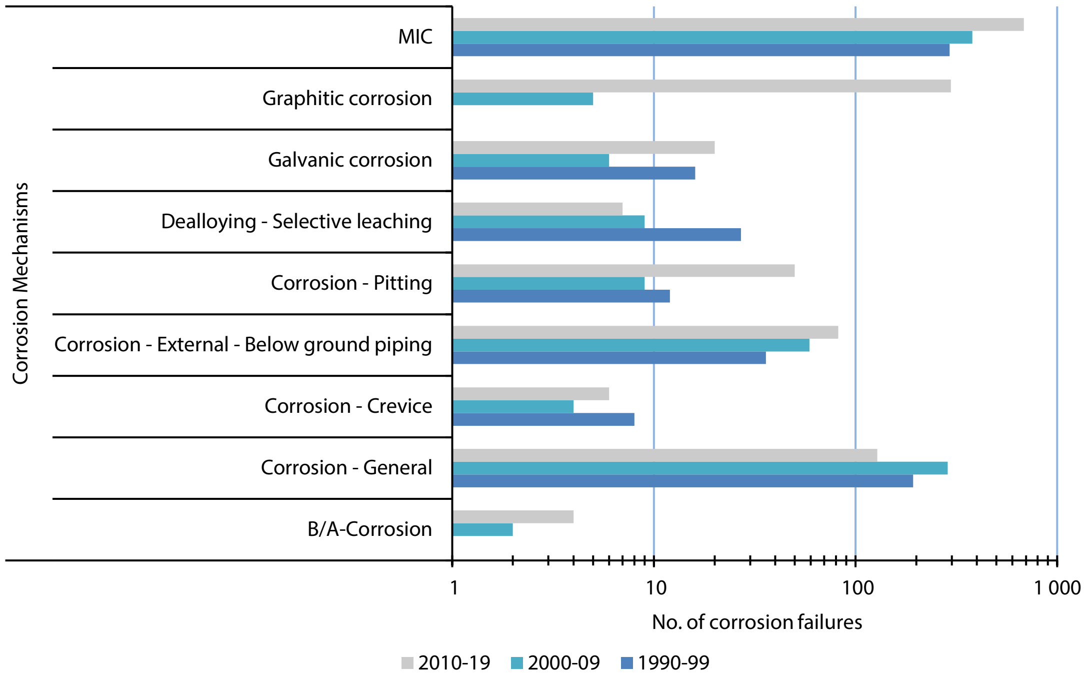



**Figure 2-21: Service water piping failure operating experience**

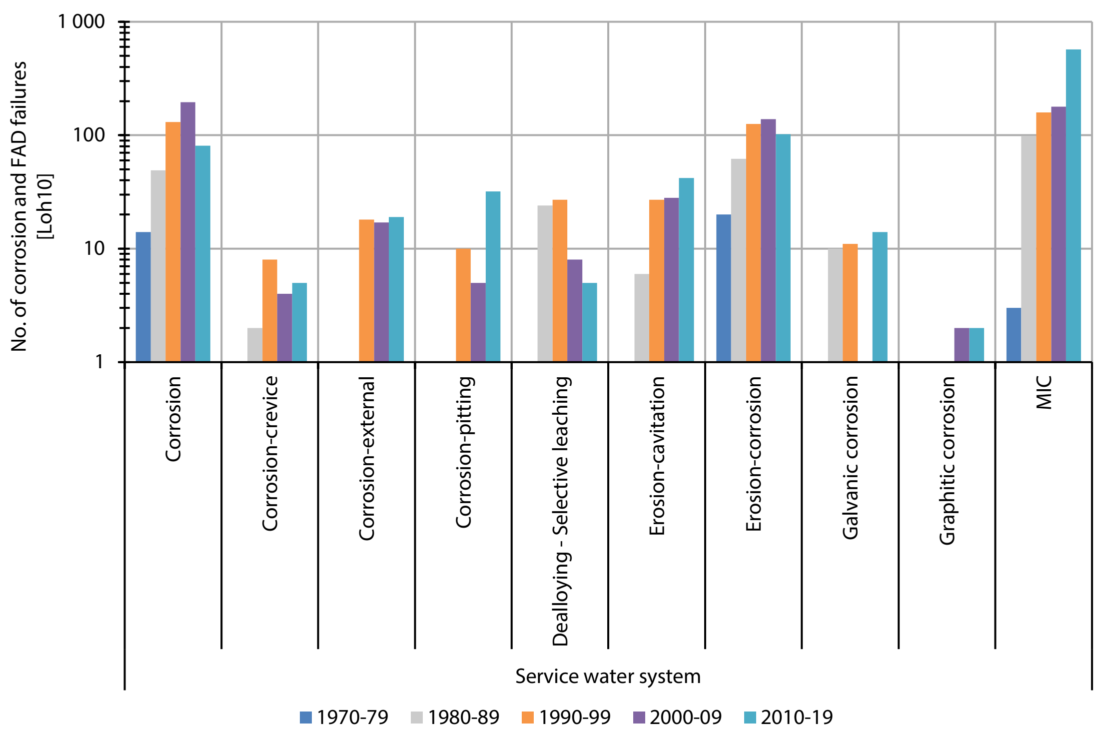
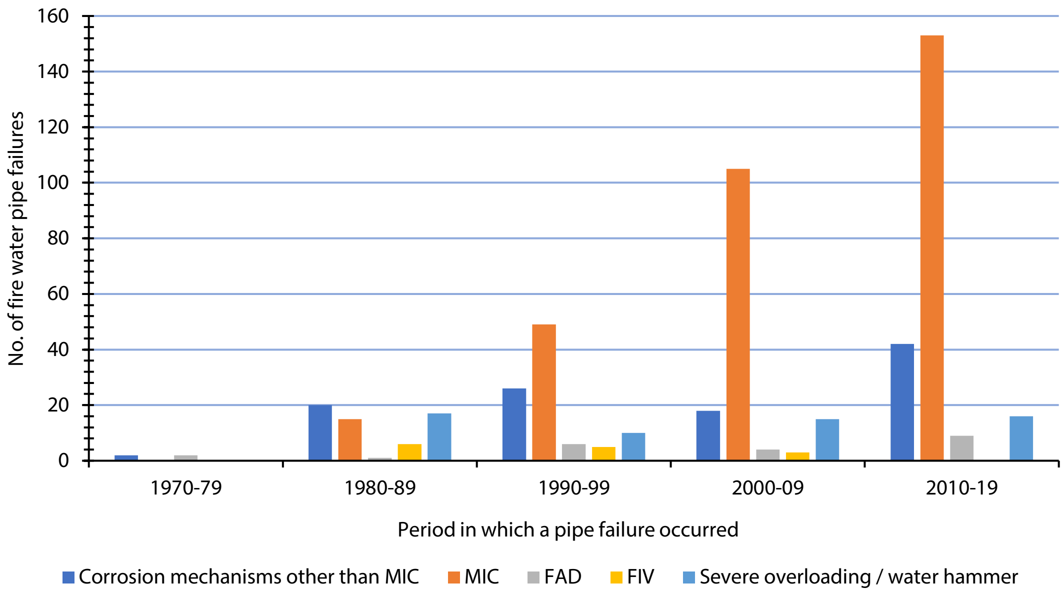


**Figure 2-22: Fire water piping operating experience**




## 2.3 Degradation mechanisms acting on cast austenitic stainless steel

Cast austenitic stainless steels (CASS) used in pump casings, valve bodies, pipe elbows, and other components in reactor coolant systems of HWRs and LWRs can suffer a loss in fracture toughness due to thermal ageing embrittlement (TAE) after many years of service at temperatures in the range of 280-320°C. Thermal ageing of cast stainless steels at these temperatures causes an increase in hardness and tensile strength and a decrease in ductility, impact strength and fracture toughness of the material. There are no failures (as in through-wall leaks) that are attributed to TAE, however.

The CODAP event database includes 41 failures involving CASS valve bodies. Included in this event population is the discovery of non-surface penetrating intergranular stress corrosion cracks in all eight DN600 CASS valves in the Reactor Recirculation System of Oskarshamn Unit 1 (ASEA-Atom BWR) during the 1995 refuelling outage [10]. At the time of this discovery the reactor had been in commercial operation for 23 years[2]. The CASS material was of Type EN 1.4552, which corresponds to ASTM A-351-CF8C. The longest measured IGSCC crack was 145 mm and the deepest crack was 14.5 mm. All flaws were removed using the boat sampling technique.

## 2.4 Degradation mechanisms acting on reactor vessel internals

Austenitic stainless steels (SS) are used extensively as structural alloys in the internal components of LWR pressure vessels because of their relatively high strength, ductility, corrosion resistance and fracture toughness. Intergranular stress corrosion cracking (IGSCC) in austenitic piping was a major issue for BWRs in the 1970s and 1980s. The susceptibility of reactor vessel internals (RVI) to IGSCC was also recognised. Shroud cracking in 1993-1994 confirmed that IGSCC of RVI is a significant issue for BWRs. The susceptibility of materials to IGSCC is enhanced by the effect of radiation. Neutron irradiation increases the IGSCC susceptibility of austenitic stainless steels by changing the material microstructure and microchemistry. This effect is referred to as irradiation-

2. Oskarshamn Unit 1 was permanently shut down on 18 June 2017.

assisted stress corrosion cracking (IASCC). IASCC was not considered in the design of LWR core internal components, but it has been considered in addressing plant ageing and licence renewal issues in the United States. Microstructural changes in austenitic SS due to neutron irradiation vary with the irradiation temperature, neutron fluence, flux and energy spectrum. Neutron irradiation can decrease the fracture toughness of austenitic SS significantly. Fracture toughness data on austenitic SSs irradiated in LWRs indicate:

- little or no loss of fracture toughness below ~0.5 dpa (displacement per atom);[3]
- a substantial and rapid decrease at exposures of 1-5 dpa

In BWRs, in-service failures have been seen in the core shroud, jet pump assembly, top guide and core plate. Failures of baffle-former bolts have been observed in PWRs. In the United States, BWR plant owners formed the BWRVIP (BWR Vessel and Internals Programme) in 1994, and subsequently a similar initiative was established by the US PWR plant owners. The two programmes issue annual inspection results summaries that are available on the US NRC website.

## 2.5 "Uncommon" and "unexpected" material degradation scenarios

In this section the terms "uncommon" and "unexpected" are used to characterise events that have received attention by regulators, industry and/or material scientists because of the operational impact, the modes of degradation and failure, and/or because of the potential effect on reliability and integrity management (RIM) practices. For illustrative purposes, seven events are described in further detail below. These events were selected for two reasons: (1) they occurred after more than 25 years of operation; and (2) they highlight possible ageing management programme challenges.

1. December 2001 reactor pressure vessel head cooling pipe rupture at a German BWR plant; CODAP Record #2747. At the time of the event the reactor had been in commercial operation for 25 years. On 14 December 2001, control room staff received indications of drywell leakage. These indications were initially interpreted as a flange leakage. Further evaluation of the control room indications resulted in the determination that the leakage was from a non-safety related system. It was later determined that the affected system was the RPV head cooling system (TC), a system only used during reactor shutdown. The apparent leakage was located on an isolable section of DN100 stainless steel TC piping. On 21 February 2002, an investigation team was chartered to evaluate the event. During the initial drywell inspection on 18 February 2002, it was discovered that a large section (length of two to three metres) of DN100 stainless steel piping had ruptured. On the order of 25 piping fragments could be located at the time. The initial hypothesis regarding the cause of rupture pointed to deflagration (hydrogen explosion) due to a local, high concentration of hydrogen in the pipe. Possible issues relating to operating practices were not considered in the root cause analysis.
2. In December 2014 multiple pinhole size leaks were discovered in stainless steel service water supply piping to emergency diesel generator coolers at a Finnish PWR plant; CODAP Record #5102. After about 35 years, the original service water piping had been replaced with corrosion resistant material during the 2014 refuelling and maintenance outage. The replacement material was a high-alloy austenitic stainless steel especially developed for corrosive operating environments; stainless steel of Type EN 1.4547 (UNS S31 254). The leak locations were at or adjacent to butt welds. Except for periods of emergency diesel generator operation and surveillance testing, the raw water in the affected piping is stagnant.
3. During the March 2018 refuelling outage at a US PWR, crack indications were observed on the core shroud and the outside diameter of the core support barrel during MRP-227-A inspections [24] that were conducted during the outage; CODAP Record #5079. At the time

3. 1 dpa (displacement per atom) is equated to a fast fluence of 6.7 x $10^{20}$ n/cm$^2$.

of the event, the reactor had been in commercial operation for 42 years. Cracks were found on the OD surface of the core barrel in the beltline elevation using the enhanced visual testing (EVT-1) method. The Owner informed industry and the NRC during the outage about this first-of-a-kind EVT-1 inspection finding. Additional EVT-1 and UT inspections were performed to further characterise the indications. One crack-like indication found in base-metal adjacent to the middle-girth weld, 45 crack-like indications found in base-metal adjacent to the middle-axial weld. None of these indications were through-wall, however.

4. In January 2019 through-wall leaks were discovered in the stainless steel large-diameter turbine cross-under piping at a Spanish PWR plant; CODAP Record #5061. At the time of the event, the reactor had been in commercial operation for 30 years. During a refuelling and maintenance outage in 1988 and in an effort to mitigate material degradation through flow-accelerated corrosion the original carbon steel piping was replaced with new piping made of stainless steel material (Type EN 1.4541). The turbine cross-under piping is the large-diameter piping (e.g. 900-1 500 mm diameter) that carries wet steam from the high-pressure turbine to the moisture separator reheater (MSR) and dry(-er) steam from the MSR to the low-pressure turbine.

5. March 2019 brittle fracture of a DN25 pipe coupling in the reactor vessel level indication system (RVLIS) at a US BWR. At the time of the event, the reactor had been in commercial operation for 43 years. On 28 March 2019, while operating at 100% reactor power, the narrow range reactor water level instrument (1-C32-LI-R606B) failed high; CODAP Record #5083. It is an instrument tap off the steam space of the reactor vessel. Drywell pressure and drywell floor drain leakage increased. Investigations inside the containment determined that a DN25 coupling on line 1-B21-774 located on the steam side of a reactor level condensing chamber experienced a 360° circumferential separation at the approximate centre of the coupling [40]. This failure resulted in a primary system leak rate of approximately 0.5 kg/s. The post-event metallurgical report determined that the coupling showed no evidence of localised plastic deformation. The coupling experienced hydrogen embrittlement and did not exhibit a leak-before-break failure mechanism. The coupling was made of a shape memory alloy material composed primarily of Nickel-Titanium-Iron (Tinel material). Examination of the failed coupling was conducted at a metallurgical laboratory. Microhardness testing, visual microscopy, and scanning electron microscopy were used to characterise the failed material. The examinations confirmed that the failure was caused by hydrogen embrittlement. This was supported by the transgranular cleavage on the fracture surface, high hardness values in the region exposed to the process fluid, and a hydrogen rich environment, which are all consistent with hydrogen embrittlement. The root cause of this event was that the selection of Tinel was inappropriate for long-term application in a high temperature process that contains elevated levels of hydrogen.

# Chapter 3. Country/economy-specific overviews of material degradation state-of-knowledge, research and operating experience

This chapter summarises selected country/economy-specific material degradation issues with respect to plant ageing management. For countries/economies with a fixed operating time limit (i.e. 40 years) the OPEX review addresses material degradation issues that have been observed at plants that have been in commercial operation for 25 years or longer. For countries/economies without a predetermined operating licence time limit, the OPEX review addresses material degradation issues that have been observed since a plant has been in commercial operation for 25 years or more, as well as degradation issues during the extended period of operation beyond 40 years.

## 3.1 Introduction

The country/economy-specific information on material degradation issues during PEO/LTO addresses three areas: 1) overview of the current state-of-knowledge, 2) ongoing research, and 3) operating experience during LTO and solutions. The specific topics that are being addressed include but are not limited to:

- **State-of-knowledge**. Degradation issues and ageing management.
  - Description of the degradation mechanisms being managed during the period of extended operations.
  - Description of the regulatory framework used to manage degradation/ageing during the PEO/LTO. Specifically, the programme or guidance that forms the basis of the respective ageing management programme.
  - Description of the most significant technical issues related to the PEO/LTO.
    - Plans, if any, to update the regulatory guidance to address any of these ageing management issues.
  - Summaries of, or references to, any operating experience, research programmes or ageing management programmes related to irradiation-induced void swelling or creep.
  - With respect to reactor internals and, if applicable, description of the use of embrittlement trend curves (ETC)[1]:
    - Which trend curve is used and when was it implemented in the country/economy's regulations?
    - Are the ETCs used for predictive purposes or reliance placed on surveillance results and with ETCs used for interpolating between surveillance results?

1. RPV & reactor internals material embrittlement trend curves (ETCs) are used to estimate the magnitude of neutron irradiation embrittlement as a function of both exposure (fluence, flux, temperature) and material composition variables.

- **Ongoing research**
  - Summarise the planned and ongoing research addressing degradation issues being managed during the PEO/LTO.
- **Operating experience during the PEO/LTO and solutions**
  - Discussions on select operating experience from plant operation during the PEO/LTO.

## 3.2 Canada

In Canada, there are several material degradation issues that must be managed for safe LTO of CANDU nuclear power plants. Many of these issues arose towards the end of the original 30-year operating life of the CANDU reactors. Some examples are discussed below.

### 3.2.1 *De-alloying of Monel steam generator tubes*

The Pickering Nuclear Generating Station is comprised of six 515 MWe CANDU reactors, each with 12 steam generators using tubes made of Monel 400 SB-163/N04 400 material with nominal outside diameter (OD) of 12.60 mm and 1.25 mm nominal wall thickness. The seamless tubes were supplied in the annealed condition. These are recirculating steam generators with an integral preheater above the cold-leg tube sheet, and the steam generators have eight carbon-steel (A245B) lattice-bar type of tube-to-bundle supports.

During the 2016 Pickering Unit 4 steam generator inspection outage a new outside surface, volumetric degradation mechanism (subsequently characterised as de-alloying degradation) was detected by CTR-1 and X-Probe electromagnetic probes above the top preheater baffle plate P19 in the cold-leg side of two steam generators. Based upon eddy current sludge mapping, these indications were located within the magnetite sludge pile on top of the upper preheater baffle plate; the height of the sludge pile extends for approximately 15 cm and it appears to be unchanged since 2001. The de-alloying features were localised and characterised as volumetric wall loss, but a few had depths of about 80% of the wall thickness, indicating they had initiated and grown relatively rapidly since the degradation had not been detected in previous inspection outages. The following activities were undertaken by the operator to characterise the degradation and develop a management strategy [42]:

- inspection scope expansion;
- examination of removed tubes;
- burst testing of removed tubes and fabricated specimens with simulated flaws;
- development of a flaw model and acceptance standards, that include:
  - growth rate predictions;
  - tube plugging limits.

### 3.2.2 *Steam generator tube support plate degradation*

The Bruce Nuclear Generating Station is comprised of eight CANDU units providing approximately 6 300 MWe in total. Each unit has eight steam generators (SG) and four preheaters (PH) configured into two circulating loops with four SGs and two PHs per loop. Unit 8 operates original recirculating steam generators containing 4 200 tubes each and external preheaters containing 2 820 inverted U-tubes each. The tubing material for both steam generators and preheaters is mill-annealed Inconel 600 alloy with design dimensions of 12.95 mm (0.510 inch) outside diameter (OD) and 1.12 mm (0.044 inch) wall thickness.

Three of the eight steam generators at Unit 8 have been experiencing significant degradation of the carbon steel tube support plates due to flow accelerated corrosion, which appears to be attributable to the low chromium content of the material [43]. Replacement of the steam generators is planned during the major component replacement outage planned to commence in

2030. Until that time, the degradation of the supports is monitored through inspection activities and there have been changes to feedwater chemistry control to reduce the rate of degradation.

### 3.2.3 *Material degradation related to steam generator lay-up*

In the mid-2000s, the alloy 600 steam generator tubes at the Bruce Nuclear Generating Station were subject to outside diameter intergranular attack/stress corrosion cracking at the hot-leg top of tube sheet region [44]. The degradation mechanism was attributed to aggressive reduced sulphur species in the sludge pile arising from exposure to oxygen when the steam generators were fully drained for maintenance activities. In 2007, the operator implemented controls to reduce oxygen ingress during maintenance activities which has arrested the degradation. These controls included only fully draining the steam generators for maintenance when necessary and limiting the time which the steam generators are fully drained. These controls have been adopted across the Canadian CANDU nuclear power plant fleet.

### 3.2.4 *Buried piping*

CANDU nuclear power plants have sections of buried piping on several safety significant systems, including emergency core cooling systems, service water systems and fire protection systems. Materials of construction included carbon steel, reinforced concrete pipe and cast iron. The exterior of the piping was usually coated to protect against soil-side corrosion and often sacrificial anode cathodic protection was often used. Many of these systems contained raw service water. As the plants age the external coatings have been subject to deterioration leading and anodes have been consumed leading to the potential for increased soil side corrosion. Furthermore, the interior of the piping may have contained raw service water and been exposed to microbiologically influenced corrosion (MIC). The Canadian utilities have implemented buried piping programmes to assess the current condition of safety significant buried piping to confirm the condition of piping, repair protective coatings, perform selective replacement of piping (sometimes using non-metallic materials) and improve cathodic protection to support LTO; see reference [34] for additional information.

### 3.2.5 *Calandria and pressure tubes*

In the CANDU®, RBMK and Fugen-ATR[2] reactors, the pressure tubes (PTs) serve the same purpose as the pressure vessel in light water reactors. These tubes are long zirconium alloy tubes that pass through the centre of the reactor core and therefore are subjected to neutron irradiation, high temperatures and pressures, and corrosion from the heat transport fluid. For these tubes to remain fit for service from an economic perspective, they have to maintain their integrity for at least 30 years at a 90% capacity factor.

The pressure tubes used in a CANDU reactor are made from zirconium alloyed with niobium; (Zr-2.5%Nb. During service the pressure tubes operate at temperatures between about 250°C and 310°C and with variable coolant pressures up to 11 MPa, corresponding to hoop stresses of up to 130 MPa. The pressure tubes are subjected to a maximum flux of fast neutrons from the fuel of about 4E+17 $n{\cdot}m^2\ s^{-1}$. These operating conditions can lead to degradation of the pressure tube material with respect to dimensional changes because of irradiation creep and growth, deterioration in mechanical properties due to irradiation embitterment, initiation and growth of new flaws like fretting damage due to debris. Thermal creep, irradiation creep and irradiation growth change the shape of pressure tubes during service. The tubes elongate axially, expand in diameter and sag. The rate of deformation is linearly related to fast neutron flux.

The products of the chemical reaction between the pressure tube and the PHTS heavy water are zirconium oxide and deuterium. The loss of metal from corrosion is structurally insignificant but some of the deuterium is absorbed by the tubes. Deuterium also enters the tubes via the

2. Fugen-ATR was a Japanese 165 MWe Prototype Heavy Water Reactor. It began commercial operation in March 1979 and it was permanently shut down on 29 March 2003.

stainless steel end fittings at each end of the channels. It may also enter from the outside surface of the tube if there is deuterium present in the annulus between the pressure tube and Calandria tube and the oxide on the pressure tube loses its effectiveness as an adequate barrier. The increase in deuterium concentration increases the susceptibility to delayed hydride cracking (DHC) [46] and may decrease the fracture toughness. Pressure tube cracking and fracture have been observed since the early 1970s.[3]

An example of pressure tube fracture is an event at Pickering-A Unit 2. On 1 August 1983 a pressure tube failed by developing a two-metre-long crack near the coolant outlet of the tube. While at full power, a sudden loss of heavy water from the PHT system occurred. The loss rate was about 17 kg/s. The failure was caused by the contact of the pressure tube and the Calandria tube for a relatively long time (two to five years) due to installation error of the outlet garter spring. The outside surface of pressure tube was cooled where it contacted the Calandria tube. The deuterium built up in the operating pressure tube, then migrated to the cool areas and formed zirconium hydride blisters. A crack was formed through four of the zirconium hydride blisters.

Canadian licensees periodically inspect pressure tubes in accordance with clause 12 of CSA standard N285.4 (Periodic Inspection of CANDU Nuclear Power Plant Components). Among several volumetric and dimensional inspection requirements, clause 12 of CSA N285.4 requires the determination of the pressure tube to Calandria tube gap and stipulates that the pressure tubes are considered acceptable for operation only when “no pressure tube to Calandria tube contact” is predicted to exist at the end of the next periodic inspection interval. When the screening criteria of CSA N285.4 are not satisfied, N285.8 (“Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors”) provides additional requirements to disposition. Additionally, N285.8 requires a risk assessment for the entire core performed either deterministically or probabilistically.

## 3.3 Czechia

In Czechia, there are four nuclear units of type WWER-440/V213 in operation at the Dukovany site. The four Dukovany reactor units were commissioned in 1985-1987. Commissioned in 2000 and 2002, the two reactor units at Temelín are of type WWER-1 000/320. There are no fixed operating time limits in Czechia.

The definitions and procedures concerning requirements for ageing management are found in SÚJB Safety Guide BN-JB-2.1 “Ageing Management for Nuclear Installations.” Addressed in this guide are the following activities:

1. Scoping and screening process of the SSCs that should be subject to ageing assessment;
2. Understanding dominant ageing effects/mechanisms of SSCs selected within the screening process and finding or developing effective and usable methods for monitoring and mitigating their ageing effects; and
3. Ageing degradation management of specific SSC by implementing effective measures in the field of ISI, maintenance and management of facility operation.

### *3.3.1 The Czech AMP approach*

The overall ageing management programme (AMP) for the Dukovany and Temelín nuclear power plants is consistent with the relevant IAEA Safety Standards (e.g. Safety Series Guide SSG-48 “Ageing Management and Development of a Programme for Long-Term Operation of Nuclear Power Plants” [36]) and the WENRA Safety Reference Levels. The quality assurance process of the ageing management programme is defined in document ČEZ_PG_0001 “Operational Ageing Management Programme for NPPs.”

3. Limited to selected “representative events” the current version of the CODAP event database includes three events that involve DHC-induced degradation of pressure tube integrity.

In view of the fact that the Dukovany Nuclear Power Plant has already achieved its original lifetime limit set by design and that ČEZ, a. s., declared a strategic goal for its nuclear power plants to extend their life span by 20 to 30 years as a minimum, the LTO programme was implemented in accordance with international best practice. Therefore, ČEZ, a. s. took an active part in the IAEA programme called the safety aspects of long-term operation (SALTO) and has been involved in the IAEA International Generic Ageing Lessons Learned (IGALL) Programme [26]. In accordance with the requirements of Section 49(1)s of the Atomic Act and Sections 11 and 12 of Decree No. 21/2017, an operational ageing management programme was implemented by the nuclear power plant operator ČEZ, a.s.

### 3.3.2 *Systems, structures and components subject to AM*

The scope of the ageing management process is defined in Decree No. 21/2017 Coll. The following should be included in the selection of systems, structures and components subject to the ageing management process:

- To fulfil the requirements of Decree No. 21/2017 Coll., the methodology ČEZ_ME_0987 – Selection and assessment of equipment for AM and LTO, the following criteria for the selection of systems subject to the ageing management process are set out:
  - Equipment classified as safety class one, two, three according to Decree No. 329/2017 Coll.
  - Equipment with specific safety functions according to the methodology ČEZ_ME_0901; classification of systems, structures and components of nuclear power plants in terms of Nuclear Safety, i.e. equipment with criticality level 1 and level 2, determined according to the methodology ČEZ_ME_0608 – SSC categorisation in the production division.
  - Equipment classified as safety-relevant per PSA insights and as articulated in Decree No. 162/2017 Coll.
  - Equipment recommended from global good practice, operating experience and outcomes of common walkdowns.

### 3.3.3 *Component-specific AMPs*

To ensure the required service life of SSC, a graded approach is implemented to ageing management according to the standard ČEZ_ST_0072, "Requirements for Nuclear Power Plant Reliability Management." The graded approach is selected on the basis of the strategy defined for the care of SSC.

The list of equipment for which the ageing management is ensured by introducing the component specific AMPs or by preparing TLAA analyses, results from the strategies for care of SSCs and is given by a list of specific equipment (Category A), which is the outcome of the implementation of activities under working procedure ČEZ_PP_0425. Selected component-specific AMPs are itemised below:

- low-cycle fatigue – passive mechanical components;
- flow-accelerated corrosion – nuclear power plant secondary circuit piping;
- radiation damage to reactor pressure vessels;
- service water piping;
- welds and weld-HAZ susceptible to degradation;
- passive components of main circulation pumps (MCP);
- passive components of main isolation valves (MIV);
- pressuriser (PZR);
- steam generators (SG);
- safety class one piping;
- high-energy pipelines.

### 3.3.4 *A Czech perspective on WWER age-dependent material degradation*

A Czech perspective on "open material issues before extended operation of reactor vessel internals" is outlined in Reference [49]. The first widely reported IASCC event in the internals of WWER plants was the cracking of Loviisa-2 baffle bolts, which were discovered in September 2006 [50]. Several bolts were investigated as a result of NDE indications after 30 years of operation. The material of the bolts is cold worked ti-stabilised austenitic stainless steel (AISI Type 321 steel).

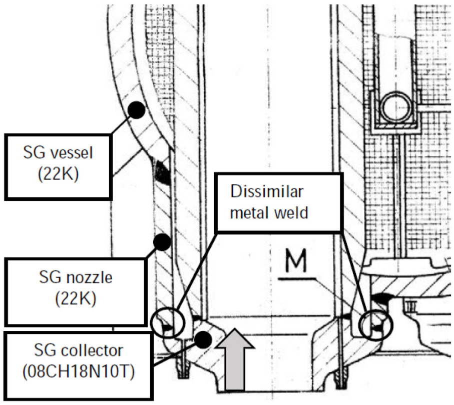
The issue of high dose irradiation is considered more significant for WWERs than BWRs because estimated doses can reach 120-150 dpa. It is known that material properties change in long-term irradiation and saturation is expected after 5-15 dpa. Experimental data of material irradiated at the thermal reactor conditions are available up to 80 dpa, and include data for mechanical properties, the ultimate tensile strength and the fracture toughness. It is not fully clear whether the material behaviour remains at the saturation level after more than 100 dpa.

### 3.3.5 *PEO/LTO-OPEX and solutions*

The WWER reactor of Type 440/213 has six primary loops. Each loop has a horizontal steam generator (SG). This reactor type was designed in the 1970s and four units are in operation in Czechia (Dukovany Units 1, 2, 3 and 4). The primary circuit is made of seamless titanium-stabilised austenitic stainless steel piping of grade 08Ch18N12T steel. The steam generator (SG) shell is made of carbon steel. There are dissimilar metal welds (DMWs) between the stainless steel piping and the steam generators. There is one DMW on the steam generator hot collector (inlet side) and one DMW on the steam generator cold collector (outlet side) for a total 12 DMWs per reactor unit.

After approximately 25 years of operation, NDE examinations performed at Dukovany Units 2 and 4 in 2012 and 2013, respectively, revealed circumferential stress corrosion cracks along the fusion line between the carbon steel and the first layer of the stainless steel weld butter; [51][52]. The degraded welds were completely removed followed by the installation of new welds with higher chromium content stainless steel weld material. The DMWs are located at the lowest point of respective SG in the area of closed pockets on the secondary side; Figure 3-1. Corrosion products from the secondary side are deposited on the lowest point of the SG and it creates a corrosive environment with higher concentrations of salts. This creates an environment that promotes electrochemical corrosion due to a difference in the electrochemical potential of the carbon steel and stainless steel. This in turn produces pitting which is a significant stress concentrator and promotes stress corrosion cracking.

**Figure 3-1: Steam generator collector DMW**




Source: Reproduced from a presentation prepared for the NEA WGIAGE Metal Subgroup Meeting, 11-13 April 2015; "Current Status of Weld Joints in CZ NPPs," by Lubomír Junek (IAM, Brno) and Zdeněk Čančura (ČEZ a.s.).

Multiple instances of SCC of DMWs in the super emergency feedwater system (SEFW) piping were identified at the Dukovany Nuclear Power Plant (and at the Bohunice Nuclear Power Plant in the Slovak Republic) in 2015. The affected DMWs were completely removed and re-welded. In response to continuing material degradation, a first-in-kind weld overlay (WOL) following the general guidelines of ASME Code Case N-740-2 was installed in March 2018 on a DMW joining the SEFW piping (OD = 89 mm) to the steam generators in Dukovany Unit 2 [53]; Figure 3-2. This was the first WOL installed in Czechia and it was accepted by the Czech regulator and two different inspection agencies following successful mock-up demonstrations, welding procedure qualification, a non-destructive examination demonstration and weld residual stress analyses.

**Figure 3-2: Dukovany-2 first-in-kind SEFW weld overlay**

## 3.4 Finland

STUK Regulation on the Safety of a Nuclear Power Plant (STUK Y/1/2018)[4] stipulates that the design, construction, operation, condition monitoring and maintenance of a nuclear power plant shall provide for the ageing of systems, structures and components (SSCs) important to safety in order to ensure that they meet the design-basis requirements with necessary safety margins throughout the service life of the facility. Furthermore, systematic procedures shall be in place for preventing such ageing of SSCs which may deteriorate their operability, and for the early detection of the need for their repair, modification and replacement. Safety requirements and applicability of new technology shall be periodically assessed in order to ensure that the technology applied is up to date, and the availability of the spare parts and the system support shall be monitored.

### 3.4.1 *Regulatory AM framework*

Issued in February 2019, Regulatory Guide YVL A.8 “Ageing Management of a Nuclear Facility”[5] imposes requirements on licensees related to the management of physical ageing as well as obsolescence of SSCs and presents the regulatory oversight relevant to the licensees’ duties. The Guide applies to all nuclear power plant life cycle phases and all SSCs important to nuclear and radiation safety. Regulatory requirements set in the Guide aim at ensuring both short and long-term operability and technological conformance of SSCs, whether in service or stand-by.

4. STUK (2018), “Radiation and Nuclear Safety Authority Regulation on the Safety of a Nuclear Power Plant”, www.stuklex.fi/en/maarays/stuk-y-1-2018.
5. STUK (2019), “Ageing Management of a Nuclear Facility”, 15.2.2019, www.stuklex.fi/en/ohje/YVLA-8.

Annex A of YVL A.8 includes a list of "typical ageing mechanisms." The key documents for the regulator's review are a conceptual plan for ageing management and an ageing management programme along with construction and operating licence applications, respectively, and ageing management follow-up reports issued annually by the licensees.

In 2018 the two Finnish licensees TVO and Fortum issued their updated ageing management programmes according to Guide YVL A.8. The role of comprehensive ageing management programmes will be more emphasised as the operation of the nuclear power plant units is extended beyond the original design lifetime. The design basis operability of SSCs has to be maintained even though their degradation rate may be hard to anticipate. A generic lesson learnt in Finland is that the closer nuclear power plants get to the end of their design lifetime, the more challenging it is for the licensees to start large and expensive investments to modernise or modify the nuclear power plants.

### 3.4.2 *Active research programmes*

In accordance with Chapter 7a of the Finnish Nuclear Energy Act, the objective of the SAFIR2018 National Nuclear Power Plant Safety Research Programme is to ensure that, as new matters related to the safe use of nuclear power plants arise, the regulatory authority possesses sufficient technical expertise and other competence required for rapidly determining the significance of the matters. With respect to the structural safety and materials area the aim of the research is to increase knowledge supporting the long-term and reliable use of nuclear power plants, particularly with respect to the integrity of barriers and material issues that affect safety. A summary of selected research results of the SAFIR2018 project [54] is given below.

#### *3.4.2.1 Analysis of fatigue and other cumulative ageing effects to extend lifetime (FOUND)*

The project FOUND (2015-2016)[6] was a cross-disciplinary assessment of ageing mechanisms for safe management and extension of operational plant lifetime. It developed deterministic, probabilistic and risk-informed approaches in computational and experimental analyses. The focus areas were:

- remaining lifetime and long-term operation of components having defects;
- susceptibility of BWR RPV internals to degradation mechanisms;
- fatigue usage of primary circuit, with emphasis on environmental effects and transferability;
- fatigue and crack growth caused by thermal loads, with emphasis on modelling;
- elaborations on certain aspects of RI-ISI methodology[7];
- nuclear power plant piping analysis methods;
- residual stresses in BWR nuclear power plants.

#### *3.4.2.2 Long-term operation aspects of structural integrity (LOST)*

The main objective of LOST is to develop methods and tools to assess the structural integrity of primary circuit components, considering particularly the long-term operation aspects. LOST is divided into two work packages, where the first work package is carried out in SAFIR2018 and the second work package in SAFIR2022, in the AMOS project. The first package focuses on RPV

6. VTT (2019), "FOUND Analysis of Fatigue and Other Cumulative Ageing to Extend Lifetime", http://safir2018.vtt.fi/finalseminar/22032019/3_2_FOUND%20final%20seminar.pdf.
7. Examples of work products include VTT-R-04644-17 (Computation of Conditional Large Early Release Probabilities of Piping Components) and VTT-R-03666-16 (Computation of Consequences of Piping Component Failures in PRA Software); these reports are available from https://cris.vtt.fi/en/searchAll.

safety; 1) fast fracture in the upper shelf region and 2) pre-investigations for BREDA (Barsebäck Research and Development Arena). The second package focuses on dissimilar metal welds; 1) residual stresses 2) materials characterisation and 3) numerical simulations.

Fast fracture testing was performed in a temperature above the range covered by the 5% master curve, known as the upper shelf region. Specimens were cooled from 300°C to the room temperature at a cooling rate of 2°C/s (rate determined from centre location of the specimen). The effect of quick cooling on tearing resistances of the material, the J-R curves, were examined in reference [55]. It was concluded that for the first time, a testing method was used for the assessment of cooling rate effect on fracture toughness in upper shelf region – extreme fast fracture was not observed during testing of the fast cooling transient. However, conclusions were based on the obtained load-displacement data and further analysis is required. SAFIR2022 will continue this work package in the AMOS project. Remarks were also made that the developed method should be applied for an actual RPV-material in the future.

Pre-investigations for BREDA retained testing of fracture toughness with 4 mm thick miniature C(T) specimens for characterisation of the $T_0$-temperature of respective materials (miniature specimens are essential in LTO-perspective since there is a limited amount of nuclear power plant materials available in surveillance programmes for monitoring of ageing of the RPV). The use of miniature C(T) specimens is validated by comparing the initiation locations of the brittle fracture to standard sized specimens' initiation locations. The initiation location with respect to the mid-plane of miniature C(T) specimens has the same distribution as conventional, standard sized specimens, and side-grooves did not show a significant effect on the obtained $T_0$; these results validate the use of miniature C(T) specimens. Large data sets, regarding obtained $T_0$ with miniature and thicker C(T) specimens for various materials, show that the values are in the same range; see reference [56].

However, based on the previous work reported in open literature, the following was concluded: In almost all the previous investigations, the materials characterisation included chemical, hardness, macrostructure and fracture toughness analysis in the through-thickness direction. These analyses are necessary to explain the variation of the material properties in thickness direction. Taking this into account, analyses were not comprehensive enough to draw definitive conclusions.

#### *3.4.2.3 Mitigation of cracking through advanced water chemistry (MOCCA)*

Corrosion problems in the PWR secondary circuit are mostly related to deposition of magnetite particles in the steam generator (SG) inside surfaces and the enrichment of impurities into crevices under these deposits. About 70% of tube failures are caused by stress corrosion cracking (SCC), which is a localised phenomenon enabled by certain conditions, e.g. susceptible material, sufficiently high stress levels and localised aggressive environments. MOCCA studies the effect of alternative water chemistry on:

- the feed water line corrosion rate;
- the tendency of magnetite to deposit into the SG;
- the lead (Pb) concentrations, which may interact adversely within the deposits.

A previously used method to minimise the SG feed water line corrosion is to use an oxygen scavenger, hydrazine ($N_2H_4$). However, there is a distinct possibility that, because of the health and environmental risks related to the use of hydrazine, the EU will in the future pass a directive forbidding its use. Therefore, different measures to mitigate the above-mentioned detrimental effects of deposits, e.g. variation of secondary side circulation pH, searching good oxygen scavengers (as replacement to hydrazine) and variation of surface (electrical) charge of particles, were also studied in MOCCA; see references [57][58].

It was concluded that the two possible candidates to replace hydrazine as an oxygen scavenger were found to be carbohydrazine and iso-ascorbic acid. However, their efficiency as oxygen scavengers was about 50% of that of hydrazine. The effect of different pH of water chemistry regimes was found to mitigate the deposits into SG's when the pH was more alkaline. The effect of surface charge of particles was suggested to play an insignificant role in magnetite deposition process.

It has been a previous concern that lead may promote SCC in carbon steel made SG's. In this work, MOCCA, it was shown that the addition of 100 ppm of lead was in fact beneficial to carbon steel under both acidic and alkaline crevice conditions; see reference [59]. Without the addition of lead, it was found that the carbon steel was more susceptible to SCC under acidic conditions than under alkaline crevice conditions.

#### *3.4.2.4 Thermal ageing and EAC research for plant life management (THELMA)*

The project THELMA, thermal ageing and EAC (environmental assisted cracking) research for plant life management, deals with nuclear materials behaviour in light water reactor environments. The development of thermal ageing of stainless steels, both welds in wrought material and cast material, have been investigated in THELMA – in other words, the focus of the study was the ageing of austenitic primary circuit materials. The application of an electrochemical measurement method, double loop electrochemical potentiokinetic reactivation (DL-EPR), was applied to determine material ageing in the study as well. One of the objectives of THELMA was to increase the knowledge on material behaviour in nuclear power plant environments, which is needed to assist STUK, licensees and other entities in questions concerning plant life management, material degradation assessment and failure analysis. The following topics were studied:

- thermal ageing of cast stainless steel;
- thermal ageing and short-range ordering (SRO) of alloy 690;
- RPV weld brittle fracture initiation;
- characterisation of irradiated stainless steel materials.

Measurements were conducted on hot leg (HL) and cold leg (CL) materials of the SG, when the stainless cast materials were examined with DL-EPR; see reference [60]. Results showed the complexity of ageing and that the DL-EPR can be used for the evaluation of the degree of thermal ageing in materials that have aged at higher than operational temperatures, so it may not be so useful for plant-aged materials. It was concluded that there is a need for a method capable of determining the degree of thermal ageing from plant components without the need to dismantle the component. It has been noted that a method called transient granting (TG) spectroscopy has been developed at MIT, United States, that has given promising preliminary results that are in line with the DL-EPR's observation of a stronger "ageing response" in the CL material compared to the HL material.

In the case of alloy 690, there is a clear influence of SRO on the hardness. This was verified in seven different material conditions, i.e. ageing temperatures and post-treatment states (e.g. cold-worked, solution annealed). The combination of intergranular (IG) carbide precipitation and growth in combination with SRO is deemed detrimental to PWSCC resistance, and it would deserve further investigations on the matter with more commercial alloy 690 materials; see reference [61].

#### *3.4.2.5 NDE of primary circuit components (WANDA)*

The ageing of the passive components of nuclear power plants is monitored through in-service inspections (ISI), which are heavily related to condition monitoring by NDE methods. The WANDA project focuses on three important aspects of proactive ageing management: early detection of deterioration, monitoring of deterioration and application of prognostics for the estimation of remaining component life. NDE is one of the recommended tools for the early detection of deterioration of nuclear power plant materials.

WANDA was divided into two work packages. The first package focused on early detection and inspection reliability. This included research on plausible ways to detect material deterioration and efficient ways to evaluate the probability of detection (POD) curves for ISI. In work package two, the focus was on concrete infrastructure. The main objective of work package two was to bring concrete NDE research on a par with metal NDE.

The new approach to determining the POD-curves was introduced in WANDA with promising results; physical flaws were developed with eFlaw technology using only limited amounts of real physical flaws. With eFlaw, scanned UT-data from a single sample was digitally altered and extended to an unlimited number of examples of data for POD determination. With the ability to create flawed data files on demand, it was possible to gather hit/miss data from various inspectors and to generate POD-curves on these data – even with the very limited number of actual physical flaws, inspectors were able to find flaws and generate a plausible POD-curve; see reference [62].

Concrete NDE research was done by building a mock-up wall, loosely representing the cross-section design of the containment wall of Olkiluoto-2 containing steel reinforcements, steel liner and tendon ducts. During construction of the mock-up wall, numerous common types of simulated defects were embedded within the wall and the slab to determine how the current state-of-the-art NDE techniques are able to determine various forms of degradation in nuclear power plant concrete structures. It was also concluded that the mock-up wall enables future research on concrete NDE and opens the possibility of training future experts; see reference [63].

#### *3.4.2.6 RPV integrity research*

As part of a Nordic nuclear safety research programme [64], the Technical Research Centre of Finland (VTT), Chalmers University of Technology, and the Royal Institute of Technology (KTH) have continued earlier efforts regarding extraction, mechanical and microstructural testing, and analysis of materials from a retired reactor pressure vessel (RPV). The objective of the study is to increase the current knowledge base on the correctness of the existing surveillance programmes, as well as the influence of long-term thermal ageing of the materials. During 2016, a baseline study was performed to prepare the basis for a test programme to analyse the as-aged material properties of the RPV from the decommissioned Barsebäck Unit 2.[8]

Included in the SAFIR2022 (2019-2022) research plan [65] is the BRUTE Project, “Barsebäck RPV Material Used for True Evaluation of Embrittlement”, which is intended to pioneer the new infrastructure of the VTT Centre for Nuclear Safety (CNS) hot cells, and to determine the properties of RPV material after thermal ageing and neutron irradiation. All methods used are verified and commissioned in the new CNS hot cell environment.

Mechanical tests (e.g. impact, fracture toughness and tensile tests) will be performed to determine the thermal and neutron embrittlement in high-Ni reactor pressure vessel weld metal, typical for Nordic RPVs. The microstructure is to be characterised to obtain an improved understanding of factors affecting brittle fracture and embrittlement. The results will be used to compare the results from surveillance programmes, used for assessment of the embrittlement using specimens manufactured at the time of the RPV manufacturing with results from material from the component itself. The data will be used to update the existing prediction curve(s) for embrittlement and improve the understanding of both neutron and thermal embrittlement.

8. The Swedish Barsebäck 2 BWR began commercial operation in 1977 and was permanently shut down in 2005. Following this, the BREDA project (Barsebäck Research & Development Arena) was initiated in 2016 with the purpose of assessing the ageing of RPV materials. In the BREDA project, cylindrical trepans with a diameter of 200 mm were extracted from beltline welds subjected to a neutron irradiation from the beltline as well as from welds from the RPV head. This extraction was completed during 2018. The BRUTE project was launched in 2018 as part of the SAFIR2018 programme, and an application for its continuation in the SAFIR2022 programme has been submitted.

### 3.4.3 *PEO/LTO-OPEX and solutions*

TVO has two power units in Finland: The OL1 and OL2 BWR units are of type AA-IV BWR-2500. They were designed by Asea-Atom (AA) of Sweden. OL1 reached the 40-year design age in 2018 and OL2 will do so in 2020. Both units were granted a licence to operate until 2038. During licence renewal of OL1/OL2, the owner/operator re-calculated thermal transients of the processes. It was concluded that no changes are required to be implemented in the processes. However, there are known locations where high cycle thermal fatigue has induced cracks. These are:

- FW/RHR (312/321-system) mixing tees where cracks up to a few mm depth occurred after 25 years of operation; see Topical Report 6 [38] for more detailed information. The cyclic thermal load occurs during hot standby when cold water is injected to keep the reactor water level in the allowable range. These tees were recently replaced. Process changes are under investigation to mitigate these thermal cycles by changing the circulation of cold feedwater from FW/RHR (312/321) mixing tees to go through systems RWCU/RHR (331/321).
- FW-distributors (213.1-system), which distributes the feedwater evenly to the reactor: On the short-term thermal shields were added to protect thermally loaded piping. On the long-term changes have been made to the spargers and these cycles no longer occur.
- Core-spray pipelines (213.3-system). These pipelines were thermally loaded due to the above-mentioned FW-distributor problem. The cracks have arrested, but recently some of the existing cracks opened due to a test where cold water was injected into these pipes.

Thermal fatigue in the above-mentioned locations is monitored with ISI. Assessment of the risk for thermal fatigue can also be managed by analysis. In addition to deterministic fatigue evaluation, TVO has developed a probabilistic risk assessment method (PAMPRI) to evaluate the risk for piping or component damage as well as the risk for core damage. With this approach all significant damage mechanisms are considered. Although the programme itself is ready, validation and implementation of PAMPRI is still ongoing.

## 3.5 France

The nuclear power plants currently in operation in France comprise a total of 56 pressurised water reactors (PWRs). The two oldest units, Fessenheim Unit 1 and Unit 2, were permanently closed in 2020. One unique feature of the French nuclear power plant fleet is its standardisation, with many technically similar reactors spread over 19 nuclear sites. Each site includes two to six PWRs. The nuclear reactor fleet is divided into four NSSS designs: CP0 Series (900 MWe), CPY Series (900 MWe), P4 and P’4 Series (1 300 MWe) and N4 Series (1 450 MWe). There are four reactors in the N4 Series; two units at Chooz and two units at Civaux.

### 3.5.1 *Regulatory AM framework*

According to the French regulatory framework, ageing management must be demonstrated, relying on OPEX feedback, the maintenance provisions, and the possibility of either repairing or replacing the components. Other factors must also be considered, in particular physical phenomena which can modify the characteristics of the equipment, depending on their function or their conditions of use. Consideration must therefore be given to the deterioration of replaceable items and the lifetime of non-replaceable items. The periodic safety review is an opportunity for an in-depth examination of the effects of ageing on the equipment. The regulatory requirements concerning ageing management are summarised as follows:

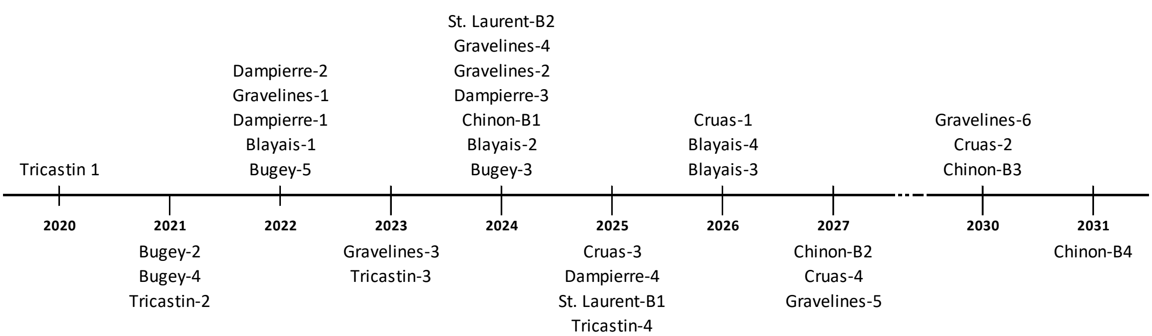
- The periodic review process contained in articles L. 593-18 and L. 593-19 of the Environment Code. The Environment Code defines the conditions for a periodic safety review of the nuclear facilities and stipulates the role of each of the stakeholders: including a) the licensee, which is responsible for the safety of its facilities, b) ASN, which is responsible for ensuring that the licensee implements the means needed to ensure a high level of safety, and c) IRSN, which is ASN’s technical support organisation.

- Article R. 557-14-2 of the Environment Code for nuclear pressure equipment (reactor vessels in particular).
- The provisions of the order of 30 December 2015 relative to nuclear pressure equipment.
- Article 2.5.1 of the order of 7 February 2012 setting the general rules for basic nuclear installations (BNIs), requiring that "design, construction, testing, inspection and maintenance provisions ensure that this qualification is maintained for as long as necessary".
- Provisions of the order of 10 November 1999 relative to the monitoring of operation of the main primary system and the main secondary systems of nuclear pressurised water reactors.
- ASN positions.
- The ageing management provisions of the "PWR Design Guide" produced jointly by the ASN and IRSN and published on 18 July 2017.

### 3.5.2 *Fourth periodic safety review of 900 MWe reactors*

In 2018, assessments were initiated for the periodic safety review of 900 MWe series reactors associated with the fourth ten-yearly inspections (VD4 900); see Figure 3-3. These reactors were commissioned between 1979 and 1988. The IRSN assessed the measures implemented by EDF to successfully manage the ageing of these reactors and presented its report to the advisory committee for reactors (GPR) and the advisory committee for nuclear pressure equipment (GPESPN). It also continued to assess design-basis accident studies, Level 1 and 2 PSA and studies of internal and external hazards as well as severe accident studies, with a view to presenting its conclusions to the GPR in 2019 at the four dedicated meetings.

**Figure 3-3: The fourth PSR schedule for the French 900 MWe reactor units**




Note: This timeline indicates the year in which a PSR is expected to be completed.

In addition, the in-service performance of 900 MWe reactor vessels between the two ten-yearly inspections was subject to a first review, which was presented to the GPESPN and was concluded in 2019 by a second assessment. This generally entails assessing the relevance and demonstrative nature of the studies submitted by EDF, evaluating the adequacy of the changes proposed by EDF with regard to the objectives of the VD4 900 safety review (in particular working towards the safety objectives defined for third-generation reactors), and ensuring that the reactors can continue to be operated safely over the coming decade.

### 3.5.3 *Insights from the PSR 2018-2020*

The approach to controlling ageing implemented by EDF is based on a process of examining structures, systems or components and the way in which their integrity or functionality may be affected by an ageing mechanism. This process accounts for the operating and maintenance provisions in force, as well as repair or replacement considerations.

The IRSN considers that this process is satisfactory with a view to extending the operating life of the 900 MWe reactors beyond 40 years, even if the processing of experience feedback and the anticipation of decisions to be taken will have to be improved (the generalisation to all reactors of lessons from a particular reactor on corrosion, fouling of piping, etc.).

EDF's maintenance strategy aims to decide on the actions to be taken with a view to controlling the ageing or obsolescence of systems, structures and components. These actions may relate to preventive maintenance (routine) and exceptional maintenance or advance studies of repair or replacement solutions, integrating research and development actions where appropriate. The IRSN considers that EDF must improve its preventive maintenance programmes to better guarantee the compliance of its installations. This requires better anticipation, a strengthening of the integration of safety issues into risk analyses, an improvement in the quality of execution, controls and requalification, as well as the speed of detection of deviations. The commitments made by EDF concerning exceptional maintenance are satisfactory, with, for example, the replacement of the cylinder heads of the back-up generator sets by the end of 2025. Finally, concerning obsolescence, the EDF process is likely to guarantee a satisfactory and lasting treatment.

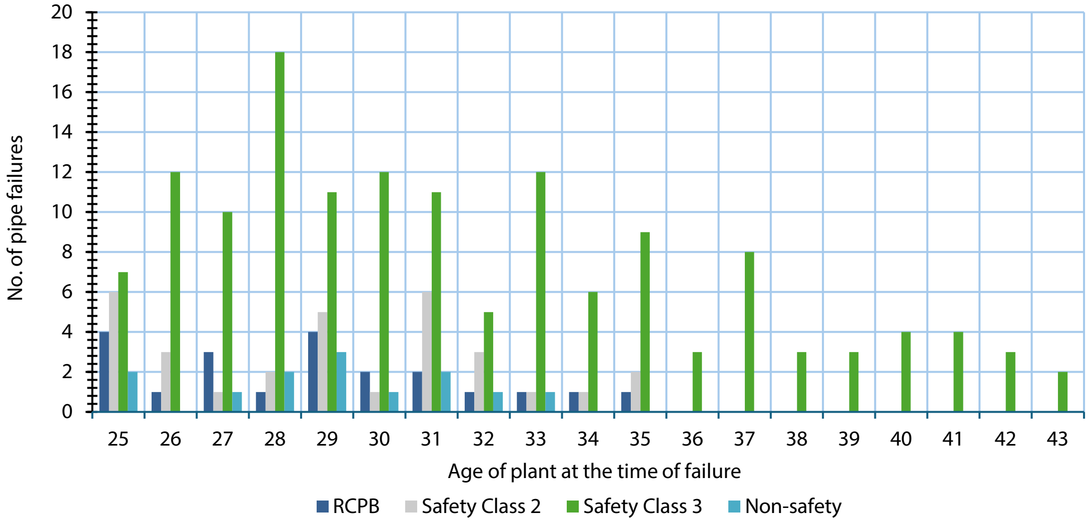
As part of the extension of the operating life of the reactors, a specific approach concerns the replaceable equipment that would remain in place beyond the life selected during their initial qualification. EDF has defined a progressive qualification programme which concerns some electrical and mechanical equipment, which is satisfactory given the commitments made. The complete demonstration of this progressive qualification will be acquired once the results of the expertise and tests of materials to be taken on-site have been obtained.

Some structures and equipment, if they are not replaceable, are the subject of a suitability check for continued operation. In this regard, based on the files examined, the IRSN considers that the serviceability of the reactor pressure vessel (RPV) of reactors No. 1 at Tricastin and No. 2 at Bugey, the first reactors concerned by the RP4-900 review, has been demonstrated for ten years of operation beyond the fourth ten-yearly outages. For the other reactors, additional information is still necessary to reach conclusions. Likewise, additional controls and studies are planned for the primary and secondary circuits. The elements presented by EDF for the civil engineering structures, the internal structures of the RPV, the primary pump units (hydraulic part), the general electrical installation cables, the instrumentation and control system and the electrical feedthroughs of the enclosure wall are generally satisfactory, with some supplements provided by EDF. For the reactors at the Bugey Nuclear Power Plant, the IRSN considers it necessary for EDF to implement a programme to check the condition of the shell screws of the vessel internals.

## 3.6 Germany

In Germany, a fixed shutdown date for each plant is mandated by law, with the end of 2022 being the final date for the plants to be shut down. Considering the final shutdown dates, no operating plant in Germany will exceed the design lifetime of 40 years. Indeed, none of the plants which were already shut down in the past have exceeded the 40-year design lifetime. The longest operating time in the past and the longest estimated operating time for respective plant currently in operation is about 37 years. The German piping OPEX for plants that have been operated for ≥ 25 years is summarised in Figure 3-4. Included in this OPEX are failures discovered during periods beyond final shutdown and during decommissioning.

**Figure 3-4: The German piping OPEX after 25 years of operation**




One of the most challenging issues for LTO is the neutron embrittlement of the RPV beltline, especially in PWRs. In the design of the RPV of German 1 300 MW PWRs, the water gap between the core barrel and RPV wall was increased compared to other Western PWR RPVs to reduce the neutron fluence at the RPV wall. Together with the early implementation of the low-leakage core strategy, the end-of-life fluence in German PWR RPV beltlines is calculated to be around 3·1 018 n/cm$^2$ (E > 1 MeV) and hence around one order of magnitude lower than for other Western PWR RPVs. The ductile-to-brittle transition temperature of German PWR RPVs is still around 0°C or below. From a technical point of view, neutron embrittlement would not be an issue, not even for LTO.

## 3.7 Japan

Nuclear power plant operating periods are limited to 60 years in Japan. Most of the Japanese nuclear power plants experienced, or continue to experience, prolonged shutdown periods after the Great Japan Earthquake of 2011. These prolonged shutdown periods have significantly reduced the remaining licensed operating times for the nuclear power plants. As a result, the 11 Japanese nuclear operators (JNOs) have completed technical assessments of ageing-related degradation for nuclear power plant infrastructure originating from the long-term shutdown period. In 2018, the Electric Power Research Institute (EPRI) published the results of an independent review of the JNO technical assessments; see reference [66]. The independent review focused on: 1) low-cycle fatigue of Class 1 components, 2) neutron irradiation embrittlement, 3) irradiation assisted stress corrosion cracking, and 4) thermal ageing of cast austenitic stainless steels (CASS). The following conclusions were reached:

1. **Low-cycle fatigue (LCF)**. Since no transient events are expected to occur during long-term shutdown, long-term plant shutdowns will not cause significant effects on LCF.
2. **Neutron irradiation embrittlement**. No embrittlement effects from neutron fluence are expected because of the absence of a fission reaction in the reactor core during long-term shutdown. Furthermore, during the long-term shutdown, no pressurised thermal transients, at either high pressure, low temperature or both, are possible. Thus, no pressurised thermal shock (PTS) events that challenge the integrity of the RPV under brittle conditions, nor any events that could induce a ductile failure via an RPV rupture, are predicted to occur. It is therefore concluded that there are no consequential ageing effects or damage that occurs to the RPV during the long-term shutdown.

3. **Irradiation assisted stress corrosion cracking**. Neither SCC nor IASCC are relevant for reactor components exposed to low temperatures below 60°C (140°F) during a long-term shutdown. Also, during shutdowns, while the reactor is not critical, there is no neutron irradiation acceleration of SCC (i.e. IASCC) due to the absence of neutron fluence.
4. **Thermal ageing of CASS**. Based on the existing body of research, for CASS plant components where temperature exposure is below 250°C (482°F), no reduction in fracture toughness is expected. The ageing effect of thermal embrittlement at lower temperatures is not considered relevant and long-term ageing effects below 250°C (482°F) do not need to be considered for CASS components. Therefore, there is no ageing effect associated with thermal ageing of CASS materials during a long-term shutdown.

### 3.7.1 *Active research programmes*

The Division of Research for Reactor System Safety, in the Secretariat of the Nuclear Regulation Authority (NRA) will conduct a safety research project using materials from decommissioned plants to confirm the adequacy (conservatism) of past results of safety research programmes applied in the confirmation of the adequacy of the ageing management technical evaluation and the review of the extension of operational period [67]. As of March 2020, the S/NRA/R has been conferring with utilities on the following items to start the research project since their co-operation is essential [68]:

- candidate units, candidate materials and available time frame, among other things;
- the possibility of disclosing the research information that utilities have already acquired to avoid duplication of resources;
- legal issues such as transportation and the status of material like waste or research material.

Tables 3-1 and 3-2 show the candidate metal materials for this project. From 2020 to 2024 the plan of this research project for metal ageing has included testing or test preparation to evaluate RPV (reactor pressure vessel) embrittlement, fracture toughness of BWR core internals, the effectiveness of preventive maintenance of BWR core internals and the integrity of thermally aged CASS (cast austenitic stainless steel) [69].

**Table 3-1: Candidate materials for first stage of R&D programme**

| Materials | Structures, systems and components |
|---|---|
| **1st stage of R&D Programme (2020 – TBD)** | |
| **Low alloy steel (BWR)** | RPV |
| **CASS (BWR)** | Primary loop recirculation (PLR) pump casing |
| **Stainless steels** | PLR piping where induction heat stress improvement (IHSI) was carried out |
| | Core shroud (BWR) |

**Table 3-2: Candidate materials for second stage of R&D programme**

| Materials | Structures, systems and components |
|---|---|
| **2nd stage of R&D Programme (TBD – 2024)** | |
| **Low alloy steel (PWR)** | RPV |
| **CASS (PWR)** | Main coolant piping |
| **Ni-based alloy** | Instrumentation tube where peening was carried out |
| **Stainless steels** | Baffle-Former bolt, core barrel |

### 3.7.2 *PEO/LTO-OPEX and solutions*

As one example of recent OPEX, on 16 February 2017, during the 17th inspection period of Shimane Unit 2 (BWR/5), a crack was found in the weld joint (Alloy 82) of one of the access hole covers (AHCs) while performing a visual inspection of the RPV internals using an underwater camera; CODAP Record #5168; see reference [70]. The inspection was part of the ageing management technical evaluation for the 30th year of operation of the reactor unit.

Both access hole covers (including a part of the shroud support plate) were cut and removed for detailed investigation. As a result of the investigation, cracks were found in the weld joint of both access hole covers. Moreover, the cause of the cracks was determined to be stress corrosion cracking due to the overlap of hardening in the vicinity of the weld joint of the access hole cove; water quality deterioration in the gap between the shroud support plate and the access hole cover and tensile residual stress in the vicinity of the weld joint of the access hole cover Upon investigation of the shroud support plate side in the reactor, no cracks remained, and the extent of crack propagation was confirmed. Therefore, an integrity evaluation of the core support structure (shroud support) was performed, and it was confirmed that the core support function was not affected. The corrective action involved replacements with new access hole covers that do not have a welded structure (bolt fastening structure).

## 3.8 Korea

Korea has 24 nuclear power plants in commercial operation: 21 PWR units and 3 CANDU-6 units. An additional four units are under construction. The oldest reactor unit, Kori-2 has been in operation since 1983. A total of nine (9) reactor units have been in operation for more than 25 years. Periodic safety reviews (PSRs) are conducted to address the cumulative effects of plant ageing, operating experience and the evolution of science and technology. In particular, the assessment and management of plant ageing is one of the major areas of work. It includes identification of the systems, structures and components (SSCs) for ageing management, assessment of ageing effects and planning of the ageing management implementation programme.

### 3.8.1 *Ageing management technology reports*

Prepared by Korea Hydro and Nuclear Power Co. (KHNP), two AM technology reports were accepted by the Nuclear Safety and Security Commission (NSSC) in 2019. The specific technology report under Article 100 paragraph 1 of the Enforcement Decree of the Nuclear Safety Act shall refer to reports containing the following information: (1) methodologies and computing codes related to technical information concerning the selection of a site for a nuclear reactor facility, design, manufacture, construction, pre-service test, trial operation, operation and disassembly; (2) information concerning safety that can be applied repeatedly for an identical purpose; and (3) information that provides the basis for preparing documents to be attached to the application for a permit related to a nuclear reactor facility. Examples of ageing management guidelines include:

- **Integrated Guidelines for Management of Alloy 600 Locations (Review Period: 31 August 2015 until 30 June 2019).** Cracks were continuously observed in the component with Alloy 600 materials; an example is the boric acid leakage in the reactor upper head of the Hanbit Unit 3 in November 2012 (CODAP Event Record #4584). The NSSC required an integrated ageing management plan for alloy 600 material of KHNP. Accordingly, KHNP established the comprehensive management plan for the integrity of alloy 600 material. KHNP applied for specific technology report approval in August 2015 and received the permission in 2019. This report includes a non-destructive testing programme for each device using alloy 600 material, defect repair maintenance, preventive maintenance and water quality management to prevent PWSCC.

- **Methodology for Fatigue Monitoring of ASME Class 1 Components in LWR Environments (Review Period: 8 September 2017 until 31 October 2019)**. Although the conservative fatigue life design is applied to safety class 1 equipment, it is difficult to predict the actual fatigue life of equipment of a plant in LTO, because transients are not considered. In Korea, a fatigue monitoring system is applied for the quantitative fatigue life management of Safety Class 1 equipment as part of ageing management of the PSR. In July 2013 KHNP applied for a specific technical subject report to establish and operate the Nuclear Power Plant Fatigue Monitoring System (NuFMS) and was approved; see reference [71]. However, recent studies have shown that the fatigue life of equipment tends to decrease in the reactor coolant environment of LWRs compared to an air environment. Thus, the needs for fatigue evaluation of Safety Class 1 equipment have been raised considering the environmental impact of the reactor coolant. In this regard, US NRC describes the use of Regulatory Guide 1.207 Rev.01[9] and NUREG/CR-6909 [72], to reflect the Environmental Fatigue Correction Factor in the air environment fatigue assessment. In Korea, to reflect the environmental fatigue monitoring methods in the existing fatigue monitoring system (NuFMS), this specific technology report was issued and approved.

### 3.8.2 *Statutes and requirements for periodic safety review of long-term operation*

Periodic safety reviews (PSRs) are performed by the power plant operator (Korea Hydro and Nuclear Power Co., KHNP) to assess the integrity and safety of operating nuclear power plants in accordance with the Nuclear Safety Act. The PSRs are submitted to the regulatory body for evaluation. The Korean nuclear regulatory body consists of the NSSC and its technical support organisation, the Korea Institute of Nuclear Safety (KINS). The highest law related to PSR is the Nuclear Safety Act[10]:

- Nuclear Safety Act – Article 23 (Periodic Safety Review)
  - The operator of the nuclear power plant reactor shall periodically review the safety of the nuclear power reactor and related facilities, and report the results thereof to the Commission under the conditions as prescribed by the Presidential Decree. (Intermediate omission)
  - The Commission may, when the results of the periodic safety review referred to in Paragraph (1) and the safety measures taken based on such results are deemed insufficient, order the operator of the relevant nuclear power reactor to correct or supplement such insufficiencies.
  - Matters concerning the method and contents of review, etc. in Paragraph (1) shall be prescribed by the Presidential Decree.

The details (timing, contents, etc.) of the PSR are described in the Enforcement Decree of the Nuclear Safety Act (Presidential Decree), Enforcement Regulation of the Nuclear Safety Act (Ordinance of the Prime Minister). Each operator of a nuclear power reactor shall comprehensively review the safety of the reactor facilities every ten years from the date they have obtained an operating licence. The requirements related to ageing and degradation management are as below: Article 37[11] of Presidential Decree, Paragraphs 1, 2 and 7.

- Actual status of safety-related structures, systems, and devices;
- Ageing (referring to physical or chemical processes that will result in the degradation of systems, structures and equipment in a nuclear power plant over time and with use) management plans.

9. NRC (2018), "Guidelines for Evaluating the Effects of Light Water Reactor Water Environments in Fatigue Analyses of Metal Components", RG 1.207, www.nrc.gov/docs/ML1631/ML16315A130.pdf.
10. Nuclear Safety Act, 대한민국 영문법령 (https://klri.re.kr).
11. Enforcement Decree of the Nuclear Safety Act (Presidential Decree), Article 37 "Details of Periodic Safety Review", 대한민국 영문법령 (https://klri.re.kr).

The details above are described in Article 20 of the Ordinance of the Prime Minister, Paragraphs (1) 2 and 7. Based on the law of PSR, the KINS provided the Safety Review Guidelines for Periodic Safety Review.

The contents of ageing management are located in Chapter 7 of this guideline. The targets of ageing management are the concrete containment structure; the nuclear safety related concrete structures; and nuclear facility components. To review the ageing mechanism and phenomenon of each facility, these documents are referenced: KEPIC MNZ App. W (corresponding to or ASME BPVC Section III, Appendix W), IAEA SRS-82 (IGALL), NUREG-1801(GALL).

The evaluation target components of a PWR are the:

- reactor vessel;
- reactor internals;
- control element drive mechanism(CEDM);
- steam generators;
- pressurisers;
- reactor coolant piping;
- reactor coolant pumps;
- supports and pipe whip restraints;
- valve system;
- pump system;
- pressure vessels;
- heat exchangers;
- HVAC components;
- diesel generator system;
- fire protection system;
- secondary side water system; and
- turbine system.

### *3.8.3* ***Ongoing R&D related to ageing management at KINS***

The project called Regulatory Technologies for Ageing Management of Crack Defect in Operating Nuclear Power Plants, which was started in 2018, is in development. The objective of the research is to develop regulatory technologies for ageing detection, integrity assessment and repair methods of major nuclear components in long-term operation. The scope of this project is as follows:

- Development of regulatory techniques for crack evaluation and ageing management:
  - Residual weld stress to dissimilar metal weld parts considering the phase transformation;
  - Ultrasonic crack signal analysis techniques based on artificial intelligence;
  - Non-linear ultrasonic technique for cast austenitic stainless steel (CASS) thermal ageing.
- Development of regulatory technologies for component integrity assessment considering environmental factors:
  - Reactor vessel safety assessment considering long-term irradiation;
  - Damage evaluation of reactor vessel internals;
  - Development of regulatory tools for probabilistic analysis.
- Development of regulatory technology for weld integrity and repair weld safety in reactor components.

## 3.9 The Netherlands

The Netherlands has a single nuclear power plant in operation, the two loop Siemens PWR of Borssele. This power plant went into operation in 1973 and was granted a LTO licence in 2013 for a lifetime extension of 20 calendar years, bringing the lifetime from 40 to 60 years. Currently, there is no plan to extend the lifetime of the power plant further than 2033. The Dutch Nuclear Energy Act states in article 15a that the power plant of Borssele will operate until 2033 with no possibility of licence renewal.

In the LTO licence, the subject of ageing is broadly addressed. The licence contains some specific requirements for the licensee, the most relevant of which are listed below:

- To submit to the regulatory authority an implementation plan regarding the extension and improvement of the existing ageing management programme with extra ageing management measures specifically thought for the period between 40 and 60 years of the lifetime of the reactor. The implementation plan should be based on the comprehensive Ageing Management Review for passive SSC important to safety, the PSR assessment of Safety Factors 10 (Organisation) and 12 (Human Factor), the outcome of the SALTO PR in 2012 and the regulatory review.
- To enhance the existing ISI-programme with specific items to be performed in the first years of LTO.
- To assure that active components important for safety are adequately addressed in the preventive maintenance or monitoring programme of the plant.
- To confirm the safety margin of the RPV against brittle fracture with additional surveillance capsules.
- To confirm the safety margins against fatigue failure for specific components and perform yearly monitoring of fatigue loading with a dedicated monitoring system.
- To carry out yearly monitoring and determination of the residual lifetime of electric equipment for design base accidents to preserve the qualification of this equipment.

### 3.9.1 *Operating experience during LTO*

*RPV Irradiation Surveillance Programme (SOP)*

According to the conditions in the LTO licence, the licensee must estimate the safety margin to brittle fracture of the RPV by means of an additional irradiation surveillance programme, with the goal of evaluating the transition temperature to brittle behaviour for 60 years of operation. To estimate the changes in material properties of the RPV, several samples of the ferritic steel of the vessel material were irradiated in the reactor and retrieved after the desired equivalent lifetime for destructive testing. Specimens from the base materials of the RPV beltline as well as from the weld material of the circumferential core weld of the RPV were tested. The harvesting and testing of the specimens was carried out following the guidelines of the German standard KTA 3203, issue 11/17. The second specimen set, irradiated until an equivalent lifetime of at least 55 EFPYs, was retrieved in 2018 and the results of the destructive testing are currently under review by the regulatory authority. So far, results are encouraging: Based on the neutron flux average over cycles with full low leakage core loading and low leakage with MOX fuel assemblies, the maximum fluence value at the most irradiated position of the RPV after 55 EFPYs is determined to be $3.20 \times 10^{19}$ $cm^{-2}$. An evaluation of uncertainties shows that even in the most conservative case the fluence will still be smaller than the RPV design fluence of $3.5 \times 10^{19}$ $cm^{-2}$. To determine the brittle to ductile reference temperatures both Charpy V- and three-point bending specimens (Master Curve) were used. With both concepts (RTNDT and RTT0) high safety margins have been proven for 60 years of operation.

*Degradation of baffle-to-former bolts*

Baffle-to-former bolts (BFBs) are part of the reactor vessel internals of a pressurised water reactor. These bolts connect the vertical baffle plates that frame the core region to the horizontal formers, thus forming a connection between the baffles and the core barrel. While all these bolts are mostly addressed as baffle bolts in the international community, a distinction is generally made between baffle-to-former bolts (BFBs) and core barrel-to-former bolts (CBBs) in Siemens/ KWU plants. The region between the barrel and the baffle plates is cooled by the primary water, flowing through holes in the formers. In the Dutch power plant of Borssele (Siemens/KWU design) the formers, and thus the bolts, are cooled with an upward flow. The plant is equipped with seven formers and two sizes of baffle bolts: M16 bolts in the first and the seventh former and M12 bolts in all the others.

The baffle-to-former bolts of the Borssele Nuclear Power Plant have a history of degradation due to SCC. The bolts where originally made of nickel alloy X-750. In response to the operational experience of other S/KWU reactors in the 80s, the bolts were replaced in 1988 with new bolts made of austenitic stainless steel 1.4571, which is comparable to 316Ti steel and should have less sensitivity to stress corrosion cracking. The lowest row of bolts (in the seventh former) was not replaced. In 2006 degradation of the replaced bolts was observed via visual inspection. These findings led to the replacement of 39 bolts in the same year. The design and material of the new bolts has not been changed.

It has been confirmed that Inter-Granular SCC was the leading mechanism in the degradation of the S/KWU bolts. Indeed, sensitisation at the grain boundary, even though reduced by using titanium, is possible. In general, this corrosion mechanism is enhanced by the local presence of impurities, such as chloride and sulphate, and of dissolved oxygen. As such, dissolved oxygen is hardly present in the primary water. However, residual amounts of oxygen could be introduced in the primary circuit, during refuelling for instance. Local conditions (low flow rate, accumulation of dissolved oxygen) will favour radiolysis and then create an oxidising environment in the shaft, promoting IGSCC.

At KCB, Oxygen is kept at a low level during refuelling and gassed out when restarted, avoiding then the use of hydrazine. Currently the baffle–to-former bolts are visually inspected every three years. There are several damaged bolts and they are mostly concentrated on the highest positions (above the core). The next visual inspection is planned during the refuelling in 2020.

The two-loop PWR of Borssele is one of the earlier reactors built by Siemens/KWU. The design as built is unique, limiting thus the comparison with other nuclear power plants. The design of the core structure is, however, based on the Westinghouse design and is also used in the other nuclear power plants of the Siemens/KWU reactor fleet. With the "Atomausstieg" in Germany, the number of reactors with identical baffle-to former bolts is reduced.

The Dutch nuclear regulatory authority is setting up a supervision strategy on this specific topic to challenge the licensee regarding the chosen inspection methodology and its frequency.

## 3.10 Slovak Republic

In the Slovak Republic, there are currently four nuclear units of type WWER 440/V213 in operation, two units at the Bohunice site (EBO V2) and two units at the Mochovce site (EMO). Two additional WWER 440/V213 units at the Mochovce site are under construction (EMO V2). The EBO V2 units were commissioned in 1984 and 1985, respectively. The two EMO units were commissioned in 1998 and 1999, respectively.

### *3.10.1 Regulatory AM framework*

A legal framework for ageing management is established in regulatory Decree No. 33/2012 on Periodic Safety Review. Referring to AM, the regulatory authority issued safety guide BNS I.9.2/2019 "Ageing Management of Nuclear Power Plants, Requirements", which specifies and supplements the requirements of Decree No. 33/2012. This guide is based on IAEA and WENRA recommendations and provides guideline on development, establishment and implementation of ageing management programmes for safety-related SSCs.

Based on the regulator Decision No. 1012/2013, the licensee is obliged to submit on a yearly basis the following information referring to the lifetime of main components:

- Remaining service life of the reactor pressure vessel and main components, including critical temperature of the brittle fracture of the RPV.
- Assessment of the critical brittle temperature of the RPV based on surveillance specimens programme.
- Results of in-service inspection programmes.

The effectiveness of the ageing management process is evaluated every ten years within the Periodic Safety Review (PSR). The licensee is required to implement an action plan for corrective actions resulting from the PSR.

### 3.10.2 *AM approach*

For implementation of ageing management, an operating organisation issued the methodological guide "Ageing Management of SSC in NPPs". This guide – an overall ageing management programme – is valid for all nuclear units in the Slovak Republic (both in operation and under construction). The guide is consistent with the relevant IAEA Safety Standards (SSG-48 "Ageing Management and Development of a Programme for Long-term Operation of Nuclear Power Plants") and WENRA SRL and defines roles and responsibilities which are necessary for proper implementation of ageing management.

For specific SSCs, the individual ageing management programmes are developed. These ageing management programmes are also valid for all nuclear power plant units in the Slovak Republic. Individual ageing management programmes are developed with consideration of a 60-year operation time frame.

### 3.10.3 *Systems, structures and components subject to AM*

According to national requirements, all safety important SSCs in safety class one to four are subject to ageing management. In addition, the operator can enlarge the scope of SSC subject to AM, e.g. taking into account the economical viewpoint. For each nuclear power plant, lists of systems, structures and components for ageing management are developed.

### 3.10.4 *Component-specific AMPs*

For specific SSCS, the operating organisation developed and implemented individual ageing management programmes. A list of these component-specific amps is provided below:

- reactor pressure vessel;
- steam generators;
- main circulation pumps;
- main isolating valves;
- primary piping;
- pressurisers;
- secondary piping;
- essential service water piping;
- main condensers;

- reactor internals;
- electrical cables;
- reactor building;
- diesel generator station;
- forced draft cooling towers;
- building of chemical treatment water of nuclear power plant EBO V2;
- auxiliary building;
- turbine hall and basement of turbo-generator.

The specific type of AMP which is not component-oriented is the amp for monitoring corrosion of SSCS.

### 3.10.5 *PEO/LTO-OPEX and solutions*

A current technical issue being addressed for the WWER 440/V213 is the modification of dissimilar metal welds (DMWs) of steam generators in the Bohunice Nuclear Power Plant. DMWs are specific joints between the stainless-steel piping and the carbon steel steam generator body. A general problem associated with DMWs after 30 years of operation is the difficulty of making repairs and their relatively short service life. Modification is focused on diameters from 18 to 90 mm, and it includes removal of the original DMWs and replacement with new weldments with dissimilar weld metal. The required quality of weldments with DMW will be achieved by using a unique welding procedure (production in workshop by machining devices). In addition, protection of the weld root from the secondary medium will be achieved by application of a protective nickel layer. The above-mentioned factors have ensured the required reliability of new DMWs during long-term operation. The modifications are planned from 2021 to 2025.

Also, a modification of DMW of feed water pipeline with diameters of 270 mm inside the steam generator is being prepared. In this case, the original DMW will be removed by the flange joint according to the solution applied and verified on steam generators of WWER 440/V213 units in other European countries, i.e. Czechia and Hungary.

## 3.11 Spain

Since the 1990s, considerations of ageing management have led to a series of recurrent activities at the Spanish plants that have arisen from the operation of the plant itself and to respond to regulatory requirements. These requirements were initially established based on the specific conditions set out in the operating licence and later, from 2009, in CSN Safety Instruction IS-22[12], which includes the requirements for ageing management and long-term operation.

The main objective of the first ageing management activities applied by the Spanish plants was to preserve the option of extending operation beyond the foreseen design lifetime, with similar approaches to those implemented at US pilot plants for the application of 10 CFR 54 ("Requirements for Renewal of Operating Licensees for Nuclear Power Plants").[13] The requirements established in IS-22 apply to all plant operating conditions, from initial start-up to definitive shutdown, for which reason common and specific requirements are defined for each of the two plant lifetime phases, i.e. design lifetime of the facility (40 years) and long-term operation (LTO).

12. CSN Instruction IS-22, Revision 1, of July 1st 2009, on safety requirements for the management of ageing and long-term operation of nuclear power plants, https://piramidenormativa.sne.es/Repositorio/CSN/is-22i.pdf.
13. Available on the NRC website at www.nrc.gov/reading-rm/doc-collections/cfr/part054.

As shown below, the Spanish fleet of operating nuclear power plants consists of seven units at five sites. None of the operating units has reached LTO, but all of them have been operating more than 30 years and some are close to LTO:

- Trillo-I (PWR-KWU 3-LP), first criticality in May 1988;
- Vandellós II (PWH 3-LP), first criticality in November 1987;
- Cofrentes (GE BWR/6), first criticality in August 1984;
- Ascó (WH 3-LP), Unit I first criticality in June 1983 and Unit II in September 1985;
- Almaraz (WH 3-LP), Unit I first criticality in April 1981 and Unit II in September 1983.

Two Spanish nuclear power plants have reached the definitive shutdown phase: Santa María de Garoña Nuclear Power Plant (GE BWR/3 Mark I, was permanently shut down on 2 August 2017) and José Cabrera-1 (Zorita Nuclear Power Plant, PWR 1-Loop-Westinghouse was permanently shut down on 30 April 2006).

### 3.11.1 *Zorita reactor internals research project (ZIRP)*

The ZIRP project was an R&D project in which the CSN participated together with entities from other countries (EPRI Materials Reliability Program (MRP), US NRC, SSM (Swedish Radiation Safety Authority), Tractebel, and AXPO [operator of Beznau Nuclear Power Plant]). The project has assessed the state of stainless steel Type 304 corresponding to certain reactor internals of the Zorita Nuclear Power Plant (1968-2006, 26 EFPY) to characterise the effects of neutron irradiation on the mechanical and microscopic properties of stainless steel, irradiated under service conditions, in order to increase awareness of the effects of irradiation towards the end of the life of the nuclear power plants (40 and 60 EFPY, and beyond). Therefore, material samples were extracted from the baffle plates inside the Zorita RPV, from which a series of specimens were manufactured and have been subjected to the different characterisation tests. The different specimens that have been obtained from the extracted material had irradiation doses ranging from about 10 dpa to almost 50 dpa.

After an initial engineering study carried out in Spain (Tecnatom developed the vessel temperature analysis and Gas Natural Fenosa performed neutron creep analysis), the material that was extracted from the Zorita vessel was transported to the laboratories of Studsvik (Sweden) where mechanical tests were carried out, while microstructural tests were carried out at MHI laboratories in Japan (samples were sent from Studsvik to MHI). Itemised below are the material tests performed to date:

- Mechanical testing:
  - Tensile testing:
    - Tensile testing of 10, 30 and 50 dpa material at room temperature and 320°C;
    - Significant radiation hardening observed at all doses.
  - Fracture toughness:
    - Fracture toughness testing of 9, 24 and 35 dpa material at 320°C in air and 9 dpa material also in PWR environment at 320°C;
    - Low fracture toughness values observed;
    - PWR environment showed slight reduction in fracture toughness.
  - Crack initiation:
    - Crack initiation testing of UCL and O-ring specimens of ~40-45 dpa material to 9 000 hours;
    - No failures of O-ring specimens;

    - Failure of UCL specimens ranging from 60% to 100% YSirr with failure time increasing with decreasing stress.
  - Crack growth rate testing:
    - Crack growth rate testing of 9, 24, and 41 dpa material at three temperatures and 2 K levels in a PWR environment;
    - Low crack growth rates with no apparent K dependency;
    - Rapid crack growth rates after decrease in test temperature or operational upset (chemistry or load).
- Microstructural testing:
  - Void swelling:
    - Transmission electron microscopy (TEM) examination of nine areas including highest temperature and highest dose regions;
    - Maximum average swelling value was 0.08% at the location of the highest temperature.
  - Microstructure:
    - Grain size typical of Type 304 stainless steel;
    - Gamma prime phase and Frank loop sizes and densities quantified;
    - Precipitates analysed.
  - Hardness:
    - Vickers hardness testing (1.0 kg and 0.1 kg) of three areas;
    - Results typical of irradiation-hardened Type 304 material.
  - Gas content:
    - Hydrogen (thermal conductivity) and helium (mass spec) gas analyses of specimens from three locations;
    - Microstructural bubbles.

The ZIRP project was scheduled to be completed at the end of 2016, but there were some delays in the Studsvik testing and some of them were finished during 2018. EPRI issued the final report at the end of 2019, with which the project is fully concluded (Materials Reliability Programme: Zorita Internals Research Project [MRP-440] Testing of Highly-Irradiated Baffle Plate Material 3002016015 Final Report, October 2019)[14].

The project has been a pioneer in its field since, at the time of its initiation, the properties of this steel in real irradiated state (not in experimental reactor) were not available, or almost not available worldwide, at doses up to approximately 50 dpa. The project lasted almost 15 years, within which the characterisation tests represented only a relatively small part. The large number of years required to perform the project was due to, among other factors, the complexities associated with the participation of different nations, the need to plan all the steps properly in advance due to irradiated material handling, and, especially, the difficulties associated with the experiments themselves.

In relation to the results obtained, the tests constitute an important milestone in the knowledge of the properties of the irradiated material (of type 304 stainless steel) and are of great interest, especially for those countries that plan to operate their nuclear power plants in LTO (beyond the original design life).

14. The report is available at no cost to funding project members only.

### 3.11.2 ESW inspection plan

All Spanish nuclear power plants have implemented dedicated inspection plans (IPs) to monitor and control the degradation phenomena affecting the essential service water (ESW) piping, by means of an ageing management programme (AMP) based on the NUREG-1801 AMP.XI-M20 "Open-cycle Cooling Water System." Each Spanish nuclear power plant has ad-hoc IP based on the defects found due to each degradation mechanism, which guarantees the functionality, operability and structural integrity of the ESW system.

As a result of material degradation due to corrosion, MIC, galvanic corrosion, or corrosion under deposits, all the Spanish nuclear power plants have currently established IPs using UT or PAUT (Phased Array UT) to monitor pipe thickness, both in areas already inspected and in new areas, to detect the growth of the indications and/or the appearance of new indications. To control and mitigate the loss of material, the IPs consider the previous OPEX and identifies the most susceptible locations, including both preventive and corrective actions, such as:

- Selection and prioritisation of lines, following the EPRI Technical Report 1010059 "Service Water Piping Guideline" [73].
- Improvement of the monitoring and control of degradation mechanisms. This includes inspection equipment, inspection criteria and inspection processes. Also included are contingency measures in case of active leakage, flaw evaluation strategies, repair strategies and validation of new ASME Code Cases.
- Monitoring of the evolution of pipe wall loss and consideration of using less susceptible material system operational changes to reduce the progression of pipe wall loss, using longer circulation times through piping, and changes to valving operations.

Based on the minimum thickness established, the IP defines ranges of remaining thickness and establishes actions to be taken in relation to the thickness measured, as well as sample extensions, evaluations (ASME Code Case N-513-3/4, "Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 3 Piping") or substitutions, temporary repairs and inspections in replaced, recharged or repaired sections. Laboratory analyses are carried out to determine the cause of the defect when applicable.

There is also continuous monitoring of the corrosion rate by means of corrosion probes, spools, or corrosion coupons, and where it is possible, chemical and biological treatment of the system and operational practices have been optimised. In some cases, the inner lining of the sections containing stagnant water will be carried out as well as inspections after these mitigation measures are considered.

### 3.11.3 Vibrational fatigue AMP

In response to a socket weld failure that occurred in 2018 (CODAP Record #5026), some licensees have developed an augmented non-destructive examination programme for small-diameter piping susceptible to high-cycle fatigue. The programme is to be implemented prior to the LTO. The new inspections will be included in the AMP based on GALL AMP-XI.M35[15] "One-time Inspection of ASME Code Class 1 Small-Bore Piping." All Spanish nuclear power plants are developing a validated and qualified procedure to inspect socket welds. Additionally, a specific AMP for vibrational fatigue is being developed in some Spanish plants, which will include the following inputs:

- The most susceptible locations within the reactor coolant system (RCS), chemical and volume control system (CVCS) and emergency core cooling system (ECCS).
- Inspection of socket welds in ASME III Code Class 1, 2 and 3 pipelines of diameter < DN100.

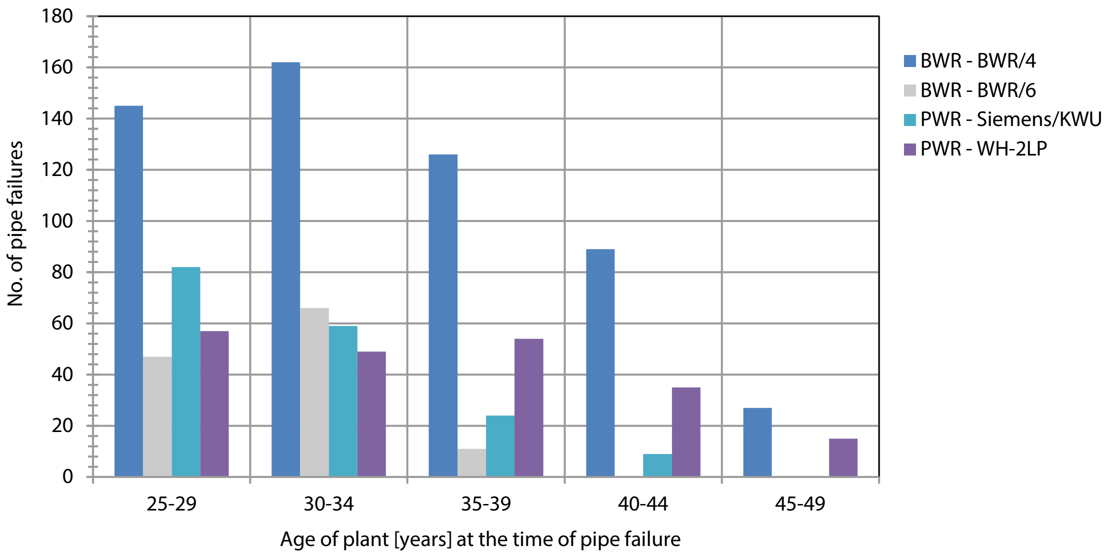
15. NRC (2010), "Generic Aging Lessons Learned (GALL) Report", NUREG-1801, Rev. 2, US Nuclear Regulatory Commission.

- Proximity assessment of rotating/pulsating equipment to susceptible lines where transmission of vibrations can favour high-cycle fatigue (HCF) cracking.
- Prioritisation in the selection of non-insulated lines in case of fatigue cracking in order to facilitate repair without affecting the plant operability.

## 3.12 Switzerland

Three different reactor types are in commercial operation in Switzerland: the GE-BWR/6, the KWU/Siemens 3-loop PWR, and the Westinghouse 2-loop PWR. A fourth reactor type, a BWR/4 at KKW Mühleberg, was permanently shut down at the end of 2019 and is undergoing de-commissioning. Summarised in Figure 3-5 is the international piping OPEX applicable to the four NSSS-types. In 1991 the Swiss regulatory body implemented requirements for plant ageing management. Subsequently the GSKL (Group of Swiss Nuclear Power Plant Operators) initiated the development of plant-specific ageing management programmes.

**Figure 3-5: Piping OPEX as a function of plant age and NSSS-design**




### 3.12.1 *State of knowledge regarding degradation mechanisms during LTO*

The degradation mechanisms considered to be of most relevance to LTO include fatigue mechanisms, irradiation embrittlement, stress corrosion cracking (SCC) and flow-accelerated corrosion (FAC). Other mechanisms like corrosion, wear, erosion and thermal ageing embrittlement are also taken into account.

For the monitoring and assessment of fatigue, three of the four Swiss nuclear power plants are using automated fatigue monitoring systems: FatiguePro[16] at Leibstadt (KKL) since 2007,

16. IAEA (n.d.), Advanced Approach For Fatigue Monitoring, IAEA-CN-194-040, https://inis.iaea.org/collection/NCLCollectionStore/_Public/43/070/43070840.pdf?r=1&r=1

WESTEMSTM [17] at Beznau-1/2 (KKB) since 2002 and FAMOSi [18] at Gösgen (KKG) since 2014. At Mühleberg (KKM), fatigue relevant data were recorded followed by a manual data evaluation. The fatigue evaluations combine stress-based and cycle-based approaches. The stress-based approach is done using measurements of temperature-time histories to identify transients. Prior to the installation of fatigue monitoring systems, a simpler and more conservative cycle-based approach was used. This is referred to as baseline fatigue and is estimated from the number of transients based on design specifications.

Environmental effects on fatigue must be considered according to ENSI-B01 [74]. The results of the fatigue monitoring in the form of current fatigue usage factors and the corresponding levels extrapolated to 60 years of operation are submitted to ENSI in an annual report. Currently, the results of the fatigue monitoring indicate that the long-term operation of the Swiss nuclear power plant is not subject to any limitations because of fatigue. Potential future challenges are preventive actions to reduce the probability of small-diameter piping fatigue damage caused by vibrations and, possibly, operating plants in a load-following mode[19].

Another important issue related to LTO is irradiation embrittlement of reactor pressure vessel materials. The ageing management for embrittlement includes the assessment of neutron fluence, which is reduced by use of low leakage core (LLC) configuration, and the testing of surveillance specimens. The determination of Adjusted Reference Temperature (ART) is based on the requirements of ENSI-B01. This guideline references three ART determination methods. Method I is based on US NRC Regulatory Guide 1.99 Rev. 2[20], and Methods IIA and IIB are based on the master curve (MC) approach according to ASTM E 1921[21]. Method IIA is the application of MC approach for irradiated material, while method IIB combines the MC approach for unirradiated material with the "Charpy-shift model."

The embrittlement has to meet the so called "DETEC[22] Criteria" (ART < 93°C in ¼ wall depth, Charpy energy ≥ 68 J). For the PWR units pressurised thermal shock (PTS) analyses are performed by deterministic fracture mechanics studies. The current PTS-Analyses are guided by the KTA 3203 methodology.[23] For the normal operation, the licensees must specify pressure-temperature limiting curves. The P-T-limiting curves are reviewed and, if necessary, updated periodically.

Stress corrosion cracking (SCC) remains an important issue with respect to LTO. Components affected by SCC in the past included the core shroud in Mühleberg, the baffle bolts in Gösgen as well as in Beznau Units 1 and 2, the RPV head penetration welds in different plants, the dissimilar metal welds at the pressuriser in Gösgen and the N5 RPV core spray nozzle in Leibstadt. Some of these components were substituted by less susceptible materials. One mitigation action, especially for BWRs, is the application of hydrogen water chemistry combined with noble metal injection. All relevant areas that are susceptible to SCC are part of the periodic inspection programme.

17. NRC (2012), "WCAP-17577", Revision 0 (Topical Report on ASME Section III Piping Fatigue Analysis Utilising the WESTEMSTM Computer Code), US Nuclear Regulatory Commission, www.nrc.gov/docs/ML1206/ML12061A377.pdf.
18. NDT net (2012), "Structural health monitoring of power plant components based on a local temperature measurement concept", www.ndt.net/article/ndtnet/2013/65_Rudolph_Rev1.pdf.
19. IAEA (2012), *Non-baseload Operation in Nuclear Power Plants: Load Following and Frequency Control Modes of Flexible Operation*, Nuclear Energy Series No. NP-T-3.23, www.iaea.org/publications/11104/non-baseload-operation-in-nuclear-power-plants-load-following-and-frequency-control-modes-of-flexible-operation (Section 5.2 of the IAEA report addresses "Phenomenological Aspects of Flexible Operation.")
20. NRC (1988), "Radiation Embrittlement of Reactor Vessel Materials", RG 1.99, www.nrc.gov/docs/ML0037/ML003740284.pdf.
21. Standard Test Method for Determination of Reference Temperature for Ferritic Steels in the Transition Range.
22. Federal Department of the Environment, Transport, Energy and Communications.
23. KTA (2017), "Surveillance of the Irradiation Behaviour of Reactor Pressure Vessel Materials of LWR Facilities", Nuclear Safety Standards Commission (KTA), KTA-3203 (2017/11); Salzgitter, Germany, www.kta-gs.de/e/standards/3200/3203_engl_2017_11.pdf.

Consideration of flow-assisted corrosion (FAC) is integrated into the periodic inspection programmes. The wall thickness of potential affected areas of these piping is measured by ultrasonic testing.

### 3.12.2 *Regulatory AM framework*

The Swiss ageing management framework is based on the Nuclear Energy Act[24], Nuclear Energy Ordinance[25], DETEC Ordinance "Suspension of a NPP Operating Licence", and the IAEA Specific Safety Guide SSG-48 [26]. Results of other programmes such as IGALL [36] and national and international OPEX are also considered. The operating licence for the Swiss nuclear power plants is not limited to a fixed period, but there are conditions for suspending plant operation if certain requirements are not met. The periodic safety review, when reaching 40 years of operation, must include an LTO safety evaluation based on licensees' safety cases. The regulation requires continued backfitting measures to improve nuclear safety or to reduce nuclear risks. The required TLAAs include the following issues:

- RPV Integrity (irradiation surveillance programme, neutron fluence assessment, PTS analyses) based mainly on ENSI-B01, US NRC Regulatory Guide 1.99 Rev. 2 and KTA 3203[26].
- Fatigue analyses for pressure vessels and piping, mechanical and thermo-mechanical based on ENSI-B01, 10 CFR 50.54[27], NUREG/CR-6909 [72] and NUREG-1800 [75].
- Leak-before-break analyses for pressure vessels and piping based on NUREG 0800 SRP 3.6.3[28], NUREG-1061 [76], Reg. Guide 1.45[29] and KTA 3206[30].

Other TLAAs are safety analyses for the nuclear steam supply system, corrosion monitoring and corrosion assessment of the steel containment, and structural analyses of the concrete parts of the containment.

With the ratification of the Nuclear Energy Act (KEG) and the Nuclear Energy Ordinance (NEO) in 2005, more detailed conditions for AMP were established at the legislative level. The KEG and the NEO are accompanied by the ordinance for safety-classified vessels and piping in nuclear power plants, and the DETEC Ordinance on "Suspension of a NPP Operating Licence"[31], which became effective in 2008.

The latter requires that a nuclear power plant be shut down if the integrity of the physical barriers is compromised by ageing mechanisms. The definition of non-acceptance criteria is, for the RPV embrittlement, an adjusted reference temperature of ≥ 93°C or an upper shell Charpy energy < 68 J, for degradation of primary piping or of the primary containment a wall thickness below the original design thickness, and for primary piping the violation of allowable fracture mechanics-based limits by cracking. For LTO the licensee has to demonstrate compliance with all these criteria for ageing over the expected operation time by TLAA. The Swiss Regulator ENSI has issued different regulatory guidelines related to LTO. In particular, these are the regulatory guideline on Ageing Management ENSI-B01 [74] in 2011, the regulatory guideline on Periodic Reporting ENSI-B02 [77] in 2008, and the regulatory guideline on Periodic Safety Review (PSR) ENSI-A03 in 2014 [78].

24. Kernenergiegesetz (KEG) 732.1, 2003, www.fedlex.admin.ch/filestore/fedlex.data.admin.ch/eli/cc/2004/723/20230901/de/pdf-a/fedlex-data-admin-ch-eli-cc-2004-723-20230901-de-pdf-a-1.pdf.
25. Swiss Nuclear Energy Ordinance, 2004, www.fedlex.admin.ch/eli/cc/2005/68/en.
26. KTA (2017), "Surveillance of the Irradiation Behaviour of Reactor Pressure Vessel Materials of LWR Facilities", KTA 3203 (2017/11), Nuclear Safety Standards Commission (KTA), Salzgitter, Germany.
27. US NRC Regulations, Title 10, Code of Federal Regulations, Section 50.54, Conditions of Licenses.
28. NRC (2007), "Leak-Before-Break Evaluation Procedures", www.nrc.gov/docs/ML0636/ML063600396.pdf.
29. NRC (2008), "R.G. 1.45: Guidance on Monitoring and Responding to Reactor Coolant System Leakage", Rev. 1.
30. KTA (2014), "Break Preclusion Verifications for Pressure-Retaining Components in Nuclear Power Plants", KTA 3206 (2014/11), Nuclear Safety Standards Commission (KTA), Salzgitter, Germany.
31. See www.bfe.admin.ch/bfe/en/home/supply/nuclear-energy.html.

ENSI-B01 defines the specific requirements for AMP. The scope of the SSCs to be covered in the AMP includes all safety classified structures and electrical components and all safety class 1 to 3 vessels, piping and related mechanical components (e.g. pumps, valves, supports and attachments). It defines the methods used for the verification of the embrittlement resistance of the RPV (mandatory time-limited ageing analyses, or TLAA, Annex 5) and the scope of the components to be covered in the fatigue monitoring and assessment programme (mandatory TLAA, Annex 6). Further it requires basic AMP documents to be prepared by the licensees. These are generic industry guidelines (by GSKL, the Swiss Nuclear Power Plant Owners Group), the KATAM (catalogue of ageing mechanisms) and the so-called fact sheets ("Steckbriefe"), which summarise the basic data, tables of ageing mechanisms relevant for the individual components, the component history, and the periodical updates to be done by the licensee. The KATAM is a proprietary document prepared by the GSKL. The KATAM is a comprehensive compendium of potential materials ageing mechanisms for mechanical equipment. Originally issued in 1991, the KATAM, which is in German only, is updated periodically.

ENSI-B02 defines the requirements for annual reporting. This includes an overview of changes performed in the AM-documents, the results of the fatigue surveillance programme, the results of the review of the operating experience and the follow-up of the current state of science and technology, a description of the consequences of these results on the AMP and the evaluation of the efficiency of the AMP based on failure statistics and maintenance indications. The guideline ENSI-A03 defines additional requirements on reporting within the PSR. This includes a review of the design and qualification of the in-scope SSCs, an assessment of technological obsolescence and the review of existing time limited ageing analyses with respect to LTO.

Based on the results of the Swiss participation in the ENSREG topical peer review process[32] [33] [79] and the IAEA Specific Safety Guide SSG-48 [16] for AMP, ENSI will review the regulatory guidelines ENSI-B01 and ENSI-B02.

### 3.12.3 *Significant technical issues related to PEO/LTO*

From a Swiss regulatory perspective, the most relevant technical issues concern embrittlement for the older reactors, especially the PWRs, fatigue including vibrational fatigue, SCC and FAC. A key challenge is the implementation of updated regulatory requirements (like increased hazards or safety margins). The AMP addresses all currently relevant issues and will be periodically reviewed based on the state-of-the-art of technology and on OPEX. The guidelines ENSI-B01 and ENSI-B02 will be reviewed to determine the need for updates consistent with IAEA SSG-48 and the results of the ENSREG topical peer review 2017.

Currently, irradiation induced void swelling or creep is not an issue and there is no ongoing research programme. The phenomenon is mentioned in the KATAM, but not addressed separately. ENSI intends to monitor the state-of-the-art knowledge on the issue and the international OPEX. If necessary, it will be included into the AMP.

With respect to reactor internals, the applied embrittlement trend curve (ETC) is the one defined by US NRC Regulatory Guide 1.99 Revision 2. It was implemented in the Swiss regulatory framework in 2008 through the DETEC Ordinance "Suspension of a NPP Operating Licence" and in 2011 through ENSI-B01. The ETC of RG 1.99 was used prior to 2008, but it was not a mandatory requirement. If surveillance data are available (which is the case for all Swiss nuclear power plants), ETC is used only for interpolating/extrapolating these results.

32. EU Topical Peer Review 2017 on "Ageing Management of Nuclear Power Plants", www.ensreg.eu/news/eu-topical-peer-review-2017-ageing-management-nuclear-power-plants.
33. The plant-specific Topical Peer Review reports are available at www.ensi.ch/en/documents/document-category/gutachten-ensi.

### *3.12.4 **Active research programmes***

The ongoing research projects funded by the Swiss regulator are the projects LEAD and PROACTIV, performed by the Paul Scherrer Institut (PSI). The NORA III project (Noble Metal Deposition Behavior in Boiling Water Reactors) was completed in 2018. Furthermore, ENSI as well as other Swiss organisations participate in the PIONIC and the IAEA IGALL projects.

The LEAD Project (LTO concerns environmental assisted degradation), scheduled to run from 2018 to 2020, is a continuation of the SAFE and SAFE II projects[34]. The research topics are SCC initiation in austenitic ni-alloys and stainless steel, environmental effects on fracture and on fatigue and investigations of synergies and superposition of ageing mechanisms. The project has connections to other research activities like the EU Horizon 2020 projects MEACTOS[35] and INCEFA+[36] and the ICG-EAC Round Robin on SCC initiation[37]. Results from the forerunner projects significantly affected the technical content of Revision 2 of BWRVIP-233 [80].

The PROACTIV project, scheduled to run from 2019 to 2021, is a follow-up of the PROBAB project and its predecessors PISA-I and II[38]. It investigates "Probabilistic Integrity Assessments for Critical RPV Locations and Piping" under consideration of active degradation mechanisms, experimental validation of XFEM with respect to application to crack growth simulations and fracture toughness evaluation of RPV-steels by use of small specimen. The project includes a participation in PARTRIDGE [81]. A Swiss participation in three OECD NEA Benchmark Projectson leak-before-break, extended finite element method (XFEM) and leak rate calculations are included in the PROACTIV project as well.[39]

The recently completed NORA project (2010 to 2019)[40] investigated the platinum deposition behaviour in BWRs and its possible impact on structural materials and fuel element performance [82][83]. It led to a deeper understanding of the mechanisms of SCC mitigation by noble metal application. One important result of direct practical use is a further optimisation of EPRI's OLNC procedure.

Project ZINC, a continuation of the NORA project, was initiated in 2019. It will investigate water chemistry related issues with respect to the interactions between zinc and noble metal injection.

ENSI has expressed interest in supporting the international SMILE project[41]. Furthermore, PSI and ENSI have expressed interest in participating in two proposed EURATOM/EU projects; APAL (Advanced PTS Analysis for LTO) and FRACTESUS (Fracture mechanics testing of irradiated RPV steels by means of sub-sized specimens). APAL is planned to develop advanced probabilistic PTS assessment methods, improved quantification of safety margins for LTO and the development of best-practice guidance for advanced PTS analysis with respect to LTO. FRACTESUS has the goal to join European and international efforts to incorporate the foundation of small specimen fracture toughness validation into codes and standards to allow the application and acceptance by various national regulatory bodies.

34. PSI (n.d.), "Safe LTO in the context of environmental effects on fracture, fatigue & EAC", www.psi.ch/sites/default/files/import/lnm/SafeEN/SAFE.pdf.
35. CORDIS (n.d.), "Mitigating Environmentally Assisted Cracking Through Optimisation of Surface Condition", https://cordis.europa.eu/project/id/755151.
36. CORDIS (n.d.), "Increasing Safety in NPPs by Covering gaps in Environmental Fatigue Assessment", https://cordis.europa.eu/project/id/662320.
37. LWRS (2016), "SCC Initiation in Alloy 600 and Alloy 690", https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/SSC_Initiation_in_Alloy_600_and_Alloy_690.pdf.
38 ENSI (2012), PISA, Pressure Vessel Integrity and Safety Analysis, www.ensi.ch/en/2012/03/09/pisa.
39. Additional details can be found in ENSI "Erfahrungs- und Forschungsbericht 2018" pp 30-32 (German language only).
40. PSI (n.d.), NORA (Noble Metal Deposition Behavior in Boiling Water Reactors): NanoTech for an Efficient Mitigation of Stress Corrosion Cracking in BWRs – A Decade of Research on NobleChem, www.psi.ch/en/lnm/nora.
41. Refer to Table 3-1 of this report.

### 3.12.5 *PEO/LTO-OPEX and solutions*

Three reactors (KKB1 & 2, KKM) entered into LTO during the period from 2009 to 2012. A fourth reactor (KKG) entered LTO at the end of 2019. For the fifth reactor (KKL), LTO will begin in 2024. For KKM (Mühleberg), LTO was approved by the regulator in 2012 for continued operation until the end of calendar year 2019. For KKB-1&2 (Beznau), a first term LTO was approved for the period 2008 until 2018 (Unit 1) and 2010 until 2020 (Unit 2). The documents for the second 10-year LTO period were submitted to the regulator in 2018. The KKG project for the first LTO period was submitted to the ENSI by the end of 2018. Summarised below is a Swiss perspective on material degradation during LTO.

**Irradiation embrittlement**: Four reactors have concluded the surveillance programmes for radiation embrittlement. It is continued only in KKL. KKL has tested two sets of surveillance specimen. The third capsule was to be extracted in 2020 and is expected to be tested in 2021. The two tested sets cover more than 24 years of operation. The third capsule will cover more than 36 years of operation. The embrittlement in KKL is very low. The adjusted reference temperature (ART) established using the RG 1.99 method is 9°C and the minimum upper shelf Charpy energy is at least 119 J. Additional observations of relevance include:

- KKM has tested three sets of specimens. The embrittlement for the base material and the core weld is low. Leading in KKM is the weld V2 with an ART of 62°C outside the core region because of the elevated copper content. Due to the permanent shutdown in December 2019, one remaining surveillance capsule will not be tested. The untested surveillance specimen will be used for future research activities.
- In KKG, all irradiation specimens have been removed and tested. Because of the high lead factor of around 3 in the PWR reactor manufactured by KWU, a theoretical operation time of more than 100 years would be covered. The predicted ARTs for 60 years for the leading base material II are lower than 33°C established by the conservative RG 1.99 method.
- In KKB-1 all six and in KKB-2 five of six surveillance capsules have been removed and tested. The coverage for both units exceed the expected operation time of 60 years. The highest values of embrittlement are found in the base material of ring C in Unit 1. Applying the method IIA according to regulatory guideline ENSI-B01, an extrapolated RTRef of 74°C for 60 years of operation was calculated in a depth of 1/4 of the wall thickness. With the more conservative method IIB, the value of $RT_{Ref}$ is 83°C. The UT indications in the PRV base material of KKB Unit 1 discovered in 2015, identified as $Al_2O_3$ inclusions from manufacturing, do not influence the reference temperature.[42] With many investigations within a safety case it was shown that these inclusions do not have a negative effect on fracture toughness.

It can be concluded that all Swiss reactors fulfil the requirements of the DETEC Ordinance (the 93°C shut down criteria). The existing results show that the long-term operation of the Swiss nuclear power plants is not subject to any limitations because of RPV irradiation embrittlement. PTS analyses showed for all three Swiss PWR that the allowable $RT_{PTS}$ (with consideration of warm pre-stress effect, WPS) is higher than the 93°C DETEC shut down criteria.

**Baffle-former bolt degradation at Beznau-1/2 and Gösgen**: Based on negative international operating experiences and problems during performing ultrasonic testing due to the geometry of the bolt head, the licensee decided to replace preventive numerous baffle-former bolts at Beznau units 1&2 in 2009 and 2010. The replacements were done based on a calculation of an optimised pattern. In all, 192 bolts in Unit 2 and 195 bolts in Unit 1 were replaced. The

42. ENSI (2018), ENSI Review of the Axpo Power AG Safety Case for the Reactor Pressure Vessel of the Beznau NPP Unit 1, www.ensi.ch/wp-content/uploads/sites/5/2018/03/14H2573-Rev.-1-1.pdf.

replacement bolts have an optimised notch contour to reduce stresses and are made of steel AISI 316 CW. Additional observations of relevance include:

- At KKM four actions were taken to address the issue: 1) periodic UT and visual testing; 2) numerical simulations; 3) optimisation of water chemistry conditions; and 4) a mechanical modification.
- In 1996, as a precautionary safety action against unexpected fast crack growth, pull rods and radial stabilisers were installed at four positions. After 2014, based on new knowledge from numerical analyses, it was planned to replace the four pull rods by a new design with six pull rods. After the decision for the final shutdown in 2019 this modification was not realised.
- In response to a regulatory request, an intensive periodical inspection programme for the cracks was initiated. In 1993 qualified systematic (1 or 2-year inspection interval) VT and UT (using a NSSS vendor inspection system, inspection from inner diameter) were started. In 1996 the visual inspection was expanded to also include the newly installed pull rods. In 2011 a new UT-system was qualified and introduced. This system enhanced the coverage, and it allowed the measurement of crack depth. The inspection was done from outer diameter surface.[43]
- In 2014 for the first time transversal cracks or so called "off-axis flaws" were detected. The last inspection of the core shroud, including off-axis cracks, was done during the outage in 2018. It confirmed that for the circumferential flaws, no further crack propagation had occurred since previous inspections in 2015 and 2016. For the off-axis cracks no further crack propagation was detected. The actual time period of crack initiation is not fully clear.
- A third action to guarantee the structural integrity of the core shroud was to carry out numerical simulations. In 1994 this action was started with a simplified fracture mechanics-based model to provide screening criteria. With respect to the LTO the simplified model was replaced in 2011 by a three-dimensional finite element model for fracture mechanics analyses. The cracks were conservatively modelled as through-wall cracks with their measured crack length. The leading load cases were combinations of seismic loads from the most adverse direction in combination with one failed tie rod and operational loads.
- In 2013, after the UT was able to size the crack depth, the cracks were realised as circumferential cracks with a depth of 60% wall thickness. This kind of modelling has the advantage that all crack lengths are covered if there is no growth in depth direction above 60%. The model was upgraded in 2014 to include the off-axis flaws. As additional load, case acoustic loads were added. In connection with new seismic loads, an improved model of the reactor building was created. This model integrates the major components of the nuclear steam generation system into the building model.
- In 1992/93 the conductivity of the reactor water was optimised to around 0.1 mS/cm. In 2000 the hydrogen water chemistry was complemented by platinum injection and in 2005 improved based on the Online Noble Chem method (ONLC™) developed by General Electric.[44] While the simple hydrogen water chemistry didn't show significant effects on the crack propagation, the usage of ONLC™ correlates with significant lower crack growth rates. As a result of continued research efforts (projects NORA I to III) a further optimisation of the platinum-feed-in strategy was achieved [82]. No further crack propagation was detected since 2009.

43. ENSI (n.d.), Fissures in the KKM Core Shroud: ENSI's Previous Assessment is Confirmed, www.ensi.ch/en/fissures-in-the-kkm-core-shroud-ensis-previous-assessment-is-confirmed.
44. Hettiarachchi (n.d.), NobleChem™ for Commercial Power Plant Application, https://inis.iaea.org/collection/NCLCollectionStore/_Public/34/062/34062664.pdf.

**Overview of SCC of a BWR RPV feedwater nozzle (CODAP Event ID 4 500)**: SCC of a dissimilar metal weld of an RPV feedwater nozzle was identified during an in-service inspection performed in 2012 per procedure SVTI-NE-14[45]. A full structural weld overlay was applied using the corrosion-resistant Inconel 52M. The qualification and implementation took place according to the requirements of design code ASME III, Subsection NB as well as ASME Code Case N-740-2[46] and guideline ENSI-G11 [84]. As part of the evaluation of the indications on the feedwater nozzle, a revised maintenance and ageing management concept for the pipe connection welds of all RPV nozzles was prepared and submitted to ENSI. This essentially comprises shortened inspection intervals and special remedial actions (MSIP, or Mechanical Stress Improvement Process). Currently, an application for approval of the technical specification for the MSIP is being prepared based on ASME III, Subsection NB, ASME Code Case N-770-1[47] and guideline ENSI-G11.

## 3.13 Chinese Taipei

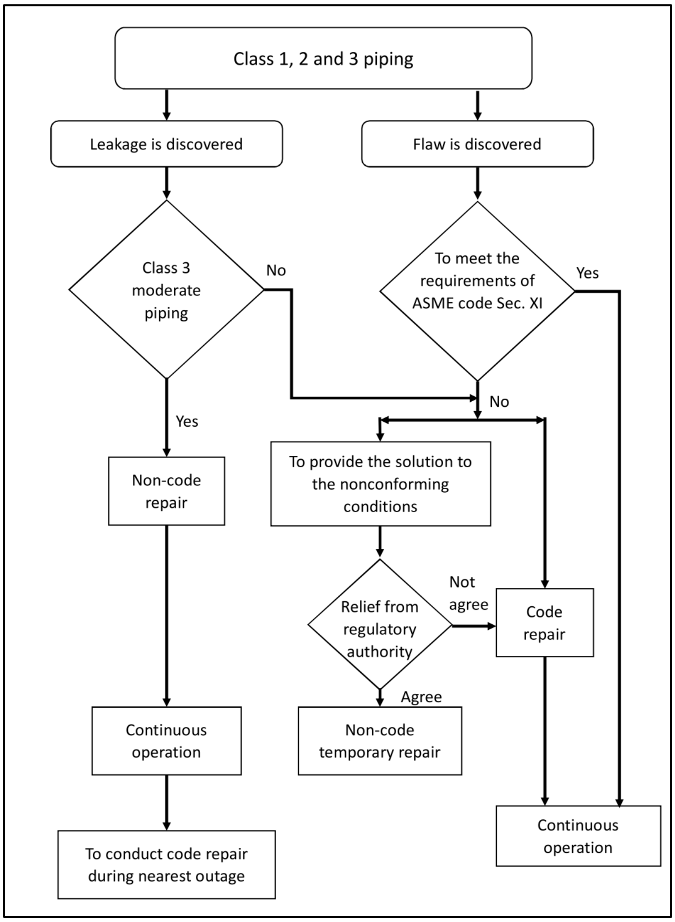
The Taiwanese experience with non-code repairs of moderate-energy piping is used as an example of how a frequently occurring ageing degradation mechanism is managed. Based on the US NRC Generic Letters 90-05[48] and 91-18[49], the implementation of non-code repairs to stop leakage without a plant shut down for repairs has been established. Reference [47] summarises the regulatory experience from the time of programme implementation in 1993 up to the year 2000. This section summarises the post-2000 regulatory experience.

### *3.13.1 Overview*

Regulations and plant technical specifications (TS) require that the structural integrity of safety-related piping be maintained according to Section XI of the ASME boiler and pressure vessel code. The information concerning pipe flaw behaviours must be recorded in non-conformance disposition (NCD) reports and submitted to the regulatory authority for the evaluation of the structural integrity and the corrective actions taken. To implement the code repairs, usually the plant is required to shut down. However, when leakage is discovered in Code Class 3 moderate-energy piping during operation, the guidelines for non-code repairs provided by generic letter 90-05, "Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping," and US NRC Inspection Manual Chapter (IMC) 0326, "Operability Determinations and Functionality Assessments for Conditions Adverse to Quality or Safety" can be applied without the plant being shut down. Under the request of the regulatory authority, the licensee (Taiwan Power Company, TPC) has established the procedures of the code and non-code repairs for nuclear safety-related piping as shown in Figure 3-6.

45. SVTI (2005), Festlegung NE-14 Wiederholungsprüfungen von nuklear abnahmepflichtigen mechanischen Komponenten der Sicherheitsklassen 1 bis 4, Revision 6, www.svti.ch/fileadmin/SVTI/NUK/NE-14.pdf.
46. ASME Code Case N-740-2, "Dissimilar Metal Weld Overlay for Repair or Mitigation of Class 1, 2, and 3 Items."
47. ASME Code Case N-770-1, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material with or Without Application of Listed Mitigation Activities", 2009.
48. NRC (1990), Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping www.nrc.gov/reading-rm/doc-collections/gen-comm/gen-letters/1990/gl90005.html.
49. NRC (1991), Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability, www.nrc.gov/reading-rm/doc-collections/gen-comm/gen-letters/1991/gl91018.html.

**Figure 3-6: The code and non-code repair procedure**




### 3.13.2 *Non-code repair of code class 3 moderate-energy piping*

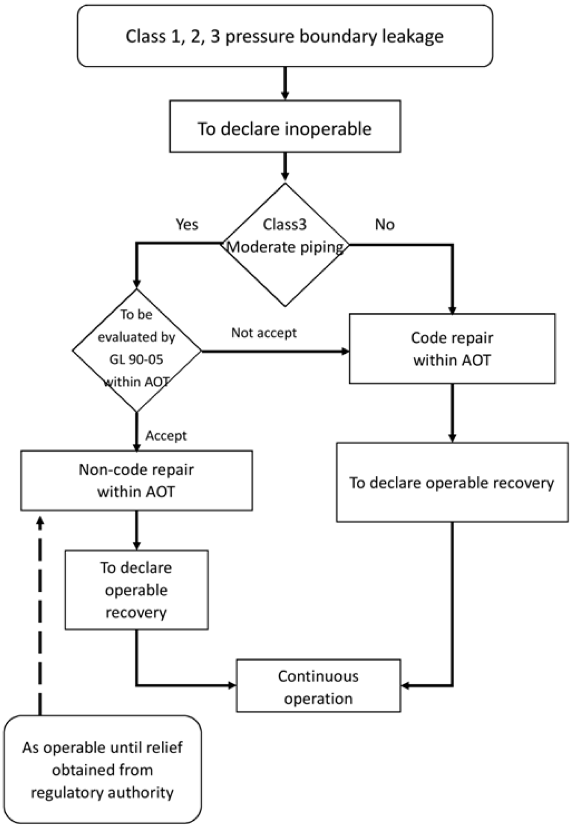
If a leak is discovered in a code class one, two or three component while conducting in-service inspections, maintenance activities, or during plant operation, the licensee should declare the "component" inoperable (this not necessarily results in the relevant "system" being inoperable) and the component should be repaired or replaced according to the requirements of corrective measures in ASME Section XI IWA-5 250. The repair/replacement work should be completed within the specific allowed outage times (AOTs) in the TS. If not, the limiting condition for operation (LCO) must be entered upon the discovery of pressure boundary leakage. A non-code repair of code class three piping is applicable until the next scheduled outage exceeding 30 days, but no later than the next scheduled refuelling outage.

Temporary non-code repairs of code class three piping in high energy systems, that is, the maximum operating temperature exceeds 200°F or the maximum operating pressure exceeds 275 psig, must have load-bearing capability like that provided by engineered weld overlays or engineered mechanical clamps. Generic letter 90-05 does not allow requests for high energy code class three piping based on repairs such as encapsulation of leaking pipes in cans using liquid sealants, clamps with rubber gasketing, or non-engineered weld overlays (patches). For temporary non-code repairs of code class three piping in moderate energy systems, that is, other than high energy systems, TPC may consider non-welded repairs. Furthermore, the structural integrity of the temporary non-code repair of code class three piping should be assessed periodically.

For code class three moderate-energy piping, a flaw evaluation must be provided to demonstrate the structural integrity of the through-wall cracked piping and then implement a non-code repair without plant shut down and prior written relief from the regulatory authority. If the result of the evaluation meets the acceptance criteria in generic letter 90-05, the component can be considered as operable until the next outage schedule. The licensee should then submit the evaluation report to the regulatory authority for review and approval in one month to obtain the written relief of this non-code repair. However, a code repair must be performed during the next up-coming refuelling outage.

At the request of the Atomic Energy Council, in 1993 the TPC established the process of the non-code repair for the moderate-energy class three piping. This process is illustrated in Figure 3-7. Most of the piping that required non-code repairs for leakage was in the seawater piping system with internal epoxy or rubber liner. The fracture of the inner liners due to ageing/degradation resulted in direct attack of the inner surface of piping by the seawater. A crack due to corrosion could penetrate the wall thickness and cause leakage.

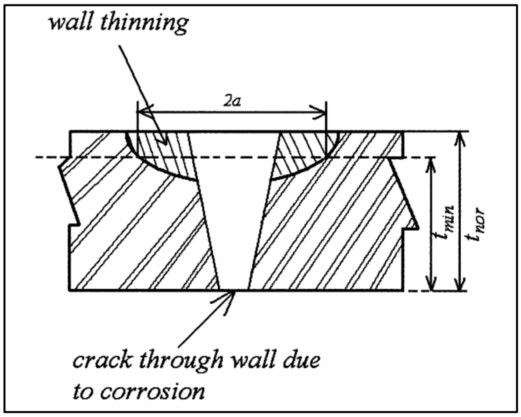
**Figure 3-7: Procedure for assessing leakage of safety-related piping**




### 3.13.3 *Flaw evaluation*

When a leakage in code class three moderate-energy pipe components is identified, the flaw geometry must be characterised by volumetric inspection. The flaw geometry has to be considered to account for measurement uncertainties and limitations. The structural integrity of the flawed piping is assessed by the fracture mechanics approach provided by GL 90-05. The "LBB.NRC Analysis Method for Circumferentially Through-Wall Cracked Pipes Under Axial Plus Bending Loads" [48] is used to evaluate the through-wall flaw independent of the orientation with respect to the piping, as shown in Figure 3-8.

**Figure 3-8: Geometry of corrosion-induced pipe flaw**




Based on linear elastic fracture mechanics and assuming a pipe thickness of "$t_{min}$", the stress intensity factor "K" resulting from the flaw under the applied load is:

$$K = 1.4 \times S \times F\sqrt{\pi a} \tag{3.1}$$

$$F = 1 + Ac^{1.5} + Bc^{2.5} + Cc^{3.5} \tag{3.2}$$

$$c = a/(\pi R) \tag{3.3}$$

R = mean pipe radius

$$A = -3.26543 + 1.52784r - 0.072698r^2 + 0.0016011r^3 \tag{3.4}$$

$$B = 11.36322 - 3.91412r + 0.18619r^2 - 0.004099r^3 \tag{3.4}$$

$$C = -3.18609 + 3.84763r - 0.18304r^2 + 0.00403r^3 \tag{3.5}$$

$$r = \frac{R}{t_{min}} \tag{3.6}$$

Where

$t_{min}$ = the code-required minimum wall thickness

2a = the length of the through-wall flaw

If the length 2a exceeds either 75 mm or 15% of the length of the pipe circumference, the flaw is not acceptable by this approach. The stress S at the flawed location should be determined from the combination of deadweight, pressure, thermal expansion and safe-shutdown earthquake. A safety factor of 1.4 should be applied to the stress. The value of "r" should be within 4 and 16. If it is outside this range, "LBB.NRC" assumes either 4 or 16 as appropriate. For the stability of the crack, the value of the calculated stress intensity factor K should be less than 35 or 135 ksi $\sqrt{\text{in}}$ corresponding to ferritic steel or austenitic stainless steel, respectively.

### 3.13.4 *Types of non-code repairs*

The considerations for temporary repairs should be to avoid further degradation and impractical periodic inspections. The typical acceptable temporary repairs suggested by GL 90-05 are clamps with rubber gaskets and encapsulation of leaking pipes in cans using liquid sealants. The patch welding to the cracked area of the pipe is not acceptable to prevent crack propagation near the original flaw regions due to thermal stresses during welding process.

### 3.13.5 *Monitoring and augmented inspection*

If the flaw still exists in the cracked piping component after the temporary repair, the crack growth must be monitored during the operation and the procedures given below must be followed continuously:

1. Increase the frequencies of the walkdown inspection for the qualitative assessment of the leakage through the temporary repair.
2. Apply volumetric examination of the defect area at the specific period. At least five of the most susceptible locations shall be examined to evaluate the global degradation of the affected region.

If either of the actions mentioned above does not ensure the effectiveness of the temporary repair, a code repair is required to maintain the structural integrity of the flawed pipe component.

### 3.13.6 *Implementation of non-code repairs*

The pipe components in the nuclear service coding water (NSCW) system leaked several times as listed in Table 3-2. The pinholes induced by the throttle flow near the valve caused the inner rubber fracture and the corrosion of the pipe wall in CCW heat exchanger as shown in Figure 3-9. To prevent these drawbacks, the stainless liners were installed, and 316 stainless pipe components were selected to resist the corrosion. The orifice plates were also embedded in the pipe components to reduce the throat flow. For the period 2007 to 2019, Figure 3-10 shows the number of non-Code repairs at Chinshan-1/2 (GE BWR-4), Kuosheng-1/2 (GE BWR-6) and Maanshan-1/2 (WH-3LP).

**Table 3-3: Examples of leak locations in the NCSW piping system**

| Leakage location | Root cause | Treatments |
|---|---|---|
| The elbow near the downstream of CCW-A discharge valve | Throttle flow | Code repair<br>To install stainless liner |
| The elbow near the downstream of CCW-A discharge valve | Throttle flow | Mechanical clamp<br>To replace 316L new elbow at the next outage |
| The elbow near the downstream in CCW-B heat exchanger inlet valve | Throttle flow of the valve and the flow direction changed to impact the rubber liner | Mechanical clamp<br>To replace new elbow at the next outage<br>To install the stainless liner |
| The straight pipe near the downstream of chiller inlet valve | Throttle flow of the valve and the flow direction changed to impact the rubber liner | To replace the rubber and code repair |
| The straight pipe near the downstream of chiller discharge valve | Discharge throttle flow | Mechanical clamp<br>To replace new pipe at the next outage |
| The straight pipe near the downstream of NSCW discharge valve | Discharge throttle flow | Mechanical clamp<br>To replace new pipe at the next outage |
| The elbow near the flange in the downstream of CCW-B heat exchanger discharge valve | The rubber liner near the flange fracture due to the compression | To replace new elbow |
| The upstream of vent valve of the side of CCW-B inlet | The rubber fracture | To replace new pipe |

**Figure 3-9: NCSW leak locations adjacent to CCW heat exchanger**

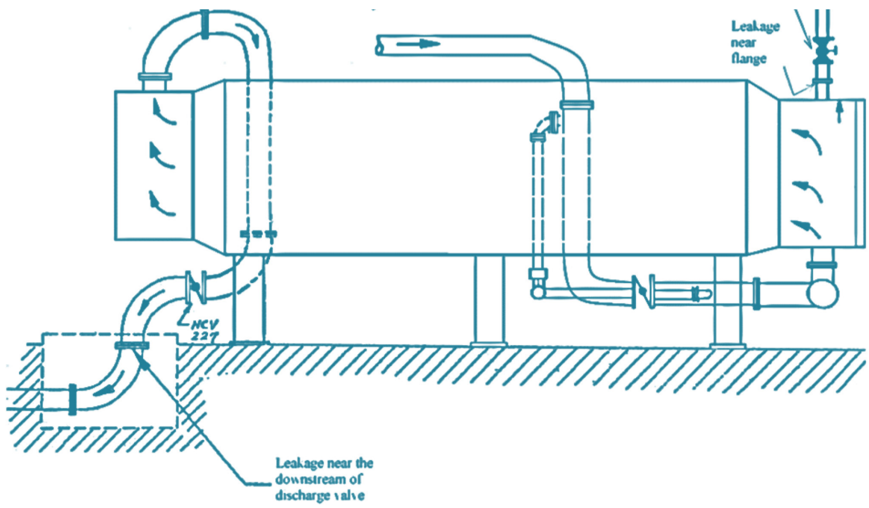
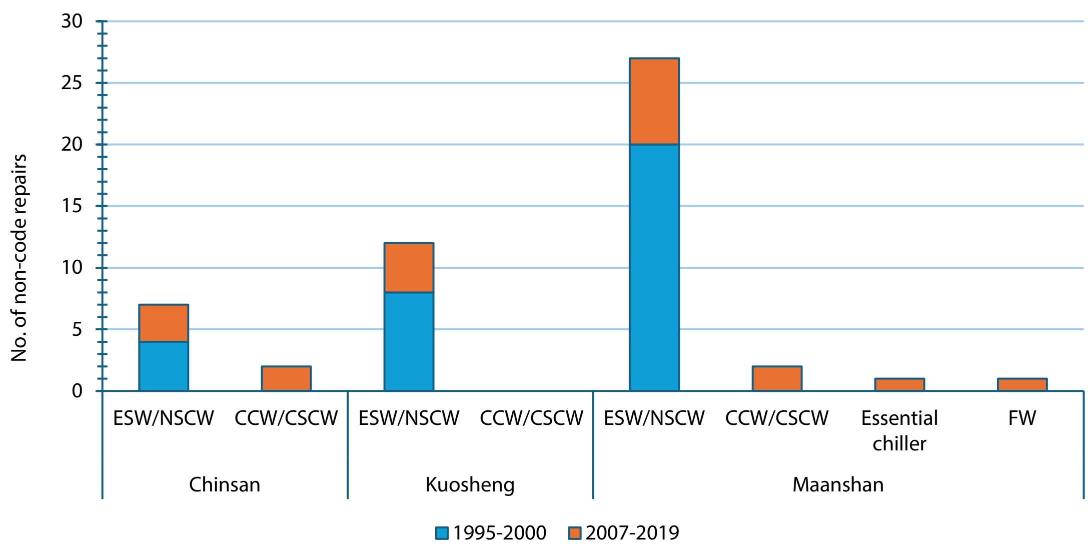



**Figure 3-10: Non-code repairs at Taiwanese nuclear power plants in 1993-2000 and 2007-2019**




### 3.13.7 *Conclusions*

Surveys of the non-conformance documents indicated that piping liner failure causing corrosion of the piping is the most common ageing degradation mechanism. Due to the successful applications of fracture mechanics analysis and temporary non-code repairs, the structural integrity of the safety-related class three moderate pipe components with cracks is confirmed to be able to prevent the leakage without the shutdown of the reactors. The non-code repaired piping components are to be replaced in the next scheduled outage. Complete procedures including the inspection programme, flaw evaluation, repair and quality control have been established in Chinese Taipei to enhance the safety of long-term nuclear operation.

## 3.14 United States

There are 93 (as of the end of 2021) nuclear power reactors licensed to operate in the United States (62 PWRs and 31 BWRs). The average age of these reactors is about 39 years old. As of the end of 2021, 94 reactors have received renewed licences (9 units have subsequently shut down), and at the time of writing this report, the NRC has approved the subsequent licence renewal applications for six units. Ageing management is a major focus area of licence renewal and subsequent licence renewal (to authorise plant operation for 80 years). The US NRC and the US nuclear industry are actively managing and monitoring the effects of ageing on nuclear plant systems, structures and components through the US regulatory framework and via voluntary industry initiatives and programmes. The NRC issued NUREG-1800 and NUREG-1801 as companion documents for reviewing licence renewal applications, and NUREG-2192 and NUREG-2191 as companion documents for reviewing subsequent licence renewal applications. These documents are discussed below. A significant document issued by industry, NEI 03-08, "Guideline for the Management of Materials Issues,"[50] outlines the policies and practices the industry commits to follow for managing materials ageing issues through voluntary initiatives. Several industry issue programmes are responsible for implementing the NEI 03-08 guidance. The issue programmes include, for example, the BWR vessel and internals project (BWRVIP), the BWR owners' group, the materials reliability programme (MRP), the steam generator management programme (SGMP) and the PWR owners group.

### *3.14.1 Building the state of knowledge regarding the management of material degradation during LTO*

Long-term operation of the US nuclear reactor fleet is being addressed by stakeholders that include both industry and the NRC through various initiatives and programmes in preparation for licensee renewal and subsequent licensee renewal. In 2006, the NRC published NUREG/CR-6923, "Expert Panel Report on Proactive Materials Degradation Assessment," or the PMDA, to address operations up to 40 years [17]. To address 80 years of operation, the NRC published NUREG/CR-7153, "Expanded Materials Degradation Assessment (EMDA)," in 2014 [27]. The PMDA and the EMDA identified materials and components where future degradation may occur in specific light water reactor systems, and gaps in knowledge. In 2019, the NRC organised an International Workshop on Age-Related Degradation of Reactor Vessels and Internals. The workshop participants addressed the state of knowledge, operating experience and research activities related to reactor pressure vessel embrittlement at high fluences, degradation of reactor vessel internals for operating periods up to 80 years and degradation related to LTO of other safety-significant pressure boundary components [39].

To assist with regulatory decision-making regarding ageing-related materials degradation, the NRC conducts targeted confirmatory research on age-related degradation to ensure the effectiveness of existing ageing management programmes and inform the development of revised guidance. The NRC research programme is summarised in NUREG-1925 and is updated every two years [88].

The US Department of Energy (DOE) Office of Nuclear Energy's (NE) light water reactor sustainability (LWRS) programme conducts research to, in part, extend the operating life of US nuclear power plants. The LWRS programme is focused on (1) developing knowledge to understand the ageing of materials, structures and components; (2) applying that knowledge to support long-term operations; and (3) researching new technologies that enhance performance, economics and safety. The LWRS programme addresses the following technical areas: plant modernisation, flexible plant operation and generation, risk-informed systems analysis, materials research and physical security. The DOE publishes an annual report providing an overview of the LWRS programme and its accomplishments [89]. More information on the LWRS programme, including reports published on each of the technical areas, can be found on the DOE NE website[51].

50. Nuclear Energy Institute (2010), "Guidelines for the Management of Materials Issues", NEI 03-08 [Rev 2], www.nrc.gov/docs/ML1010/ML101050337.pdf.
51. DOE NE website: www.energy.gov/ne/office-nuclear-energy.

Industry, through EPRI, developed the materials degradation matrix (MDM) [90] to identify potential degradation mechanisms, knowledge of degradation mechanisms (how well they are understood), and strategic long-term operation issues. EPRI also developed and maintains Issue Management Tables (IMTs) [91][92] to assess the consequences of failure, identify gaps in guidance, and prioritise and direct R&D efforts related to materials degradation. The MDM and IMTs are used by industry to manage materials degradation/ageing issues through the various issues programmes mentioned above.

The NRC, DOE and EPRI developed and maintain a research roadmap that includes research they are conducting related to reactor systems materials. The purpose of the roadmap is to help co-ordinate research activities among the various US stakeholders to provide focus and reduce unnecessary duplication. The roadmap is based on information from various sources which include EPRI's MDM and IMTs.

Based on the activities of the US stakeholders described above, several ageing mechanisms are being addressed during the licence renewal and subsequent licensee renewal periods. The following are examples: reactor pressure vessel material radiation embrittlement, stress corrosion cracking (including primary water stress corrosion cracking and irradiation assisted stress corrosion cracking), thermal ageing, fatigue (high-cycle, low-cycle and thermal), selective leaching, flow accelerated corrosion and microbiologically influenced corrosion. The NRC has developed a regulatory framework to manage these, and other, ageing mechanisms, that is described in the next section.

### *3.14.2* ***Regulatory framework***

The Atomic Energy Act and Nuclear Regulatory Commission (NRC) regulations limit commercial power reactor licences to 40 years but permit the operating licences to be renewed. The original 40-year term was selected based on economic and antitrust considerations rather than technical limitations; however, many of the technical safety evaluations were subsequently based on a 40-year operating period. The decision to seek licence renewal rests entirely with the nuclear power plant owners and typically is based on the plant's economic situation and whether it can continue to meet NRC requirements.

The provisions of Title 10, Code of Federal Regulation (CFR) Part 54[52] allow an operating licence to be renewed for 20 years and with no limit on the number of times a licence can be subsequently renewed, provided it is justified, and that the effects of age-related degradation will be adequately managed during the period of extended operation. The earliest that a licensee can submit a licence renewal application is 20 years before the expiration of its current licence. Therefore, a licensee is eligible to apply for an initial licence renewal once it has operated for 20 years. A licensee is eligible to apply for a subsequent licence renewal (SLR) once it enters the initial period of extended operation (i.e. the 20-year period beyond its initial 40-year licence period). In 2000, the NRC issued the first renewed licences for the Calvert Cliffs Nuclear Power Plant and the Oconee Nuclear Station. As of the end of 2021, 94 reactors had received renewed licences (nine units have subsequently shut down). In 2019, 2020 and 2021, the NRC approved the SLR applications for Turkey Point Units 3 and 4, Peach Bottom Units 2 and 3 and Surry Units 1 and 2, respectively. The NRC had received four other SLR applications (North Anna Units 1 and 2, Oconee Units 1, 2 and 3, Point Beach Units 1 and 2, and St. Lucie Units 1 and 2) at the time this report was drafted.

**Licence renewal (40 to 60 years)**: "Licence renewal" (LR) is the process used in the United States for a nuclear power plant to request renewal of the plant's operating licence for an additional 20 years of operation via a licence renewal application (LRA). The NRC issued NUREG-1800, "Standard Review Plan for Review of Licence Renewal Applications for Nuclear Power Plants," (i.e. SRP-LR) to provide NRC staff with guidance on how to review licensee applications to extend operating licences from 40 years to 60 years. The SRP-LR covers scoping and screening methods for identifying systems, structures, and components subject to ageing management review and time-limited ageing analyses. NUREG-1800 [95] references NUREG-1801,

52. Available on the NRC website at: www.nrc.gov/reading-rm/doc-collections/cfr/part054.

"Generic Aging Lessons Learned (GALL) Report" [96], which evaluates existing programmes to determine if they are adequate for managing identified ageing effects during the period of extended operations or require changes. The GALL Report describes acceptable ageing management programmes (AMPs) for the structures and components within the scope of the SRP-LR. Applicants can reference the GALL Report in their applications and treat it as an approved Topical Report. As stated in the GALL Report, "however, if an applicant takes credit for a programme in the GALL, it is incumbent on the applicant to ensure that the conditions and operating experience at the plant are bounded by the conditions and operating experience for which the GALL Report programme was evaluated." The initial GALL report and SRP-LR were issued in 2001 and were revised in 2005 and 2010.

The guidance in the GALL report and the SRP-LR have been modified using the Interim Staff Guidance process (ISG-LR), as described at www.nrc.gov/reading-rm/doc-collections/isg/license-renewal.html.

**Subsequent licence renewal (60 to 80 years)**: The term "subsequent licence renewal" (SLR) refers to the second (or subsequent) renewal of a licence that was previously renewed. For example, SLR may approve continued operation for the period from 60 to 80 years. SLR also refers to the process for applicants to apply for the staff to review and evaluate. The term "period of extended operation" or PEO is used in 10 CFR 54.3 in the definition of "integrated plant assessment" (IPA). This term describes plant operation beyond the initial 40-year licence term; for example, plant operation from 40 to 60 years under a renewed licence. The period from 60 to 80 years is referred to as the "subsequent period of extended operation." A framework like the licence renewal process was established for reviewing applications for subsequent licence renewal. NUREG-2192, "Standard Review Plan for Review of Subsequent Licence Renewal Applications for Nuclear Power Plants" [97], was issued in 2017, and the companion reference NUREG-2191, "Generic Aging Lessons Learned for Subsequent Licence Renewal (GALL-SLR) Report" [98], was published previously in 2015. Like the GALL Report, the GALL-SLR Report contains AMPs that were reviewed specifically for SLR and modified as necessary.

The GALL-SLR report and the SRP-SLR have been modified using the ISG-SLR process, as described at www.nrc.gov/reading-rm/doc-collections/isg/license-renewal.html.

### 3.14.3 *Materials research in support of LTO*

The development of the GALL report (NUREG-1801), the technical bases for the SRP-LR (NUREG-1800), involved periodic operating experience reviews (domestic and international), and the consideration of comments from stakeholders in the process (e.g. licensees and the Nuclear Energy Institute). As a result, revisions subsequent to NUREG-1801, Rev. 0 incorporated changes that made it more efficient and responsive to current operating experience. The GALL Report is a "living" document meant to capture lessons learnt from licence renewal reviews and operating experience, as is the GALL-SLR Report. The NRC seeks to continuously improve its understanding of materials ageing mechanisms and their effects on plant systems, structures and components and to incorporate the information into relevant regulatory documents via the ISG-LR and ISG-SLR process referenced in the previous paragraph.

The Office of Nuclear Regulatory Research (RES) provides data, techniques and methods to the US Nuclear Regulatory Commission's (NRC's) regulatory offices to support their reviews of material performance-related licensing submittals and safety issues. The confirmatory research on materials performance focuses on both the development of methodologies needed to support regulatory actions and the work supporting the technical bases for codes and standards development. A common theme in this work is a proactive approach to the management of ageing degradation. As industry's interest in licence renewal for operation beyond 60 years increases, the staff has begun to assemble technical information on ageing phenomena that can affect materials in nuclear power plants and to develop technical guidance for the staff's review of subsequent licence renewal (SLR) applications. Outlined below are the key research areas.

- **Steam generator tube integrity**: Research is underway to develop a technical basis for steam generator tube integrity to support regulatory decisions and code applications.

The goal is to ensure appropriate inspection intervals while still maintaining public safety. To provide this basis, research is focused on two areas: (1) inspection reliability and in-service inspection technology; and (2) the evaluation and experimental validation of tube integrity prediction modelling of known degradation modes.

- **Reactor pressure vessel (RPV) integrity and internals**: Safe plant operation relies on maintaining the structural integrity of the RPV during routine operations and postulated accident scenarios. Two key capabilities underpin the assessment of RPV structural integrity: (1) the ability to predict the behaviour of cracked components under loading; and (2) the ability to predict the effects of irradiation embrittlement on the fracture toughness of RPV steels. Ongoing NRC efforts are aimed at understanding the adequacy of existing approaches and predictive procedures, as well as surveillance requirements to address the integrity of the RPV under long-term operation. For RPV internals, ongoing research is focused on (1) irradiation-assisted degradation (IAD) of RPV internals to assess the structural and functional integrity of pressurised-water reactor internal components; and (2) performing confirmatory analysis of the performance of RPV internal materials during extended operation up to 80 years. Research is being conducted on harvested ex-plant materials as well as on representative materials irradiated in test reactors.
- **Piping degradation**: To better understand how the reactor is influenced by the phenomenon of primary water stress corrosion cracking (PWSCC), the NRC is conducting confirmatory testing on both crack initiation and crack growth rate for susceptible materials. Additional research is focused on the development of analysis methods and computational tools, and the performance of experimental testing to assess the impact of PWSCC on the overall safety of piping systems of the reactor coolant pressure boundary. The NRC also is assessing impacts of PWSCC on the leak-before-break behaviour of piping systems.
- **Non-destructive evaluation (NDE)**: Current NDE research is focused on the areas of effectiveness and reliability of NDE methods, assessment of NDE modelling tools, and evaluation of human factors on NDE performance; [100][101][102][103]. The results of ongoing research are used to support regulatory decisions associated with NDE and in-service inspection (ISI) of safety-related systems. The research efforts also support review of relief requests and proposed changes to ASME Code requirements.
- **Probabilistic codes**: Through a co-operative agreement, the NRC's Office of Nuclear Regulatory Research and the Electric Power Research Institute (EPRI) developed the xLPR (Extremely Low Probability of Rupture) probabilistic code to evaluate leak-before-break analysis requirements for primary pressure piping systems per the US Nuclear Regulatory Commission (NRC) Standard Review Plan 3.6.3.

### 3.14.5 *Operating experience during the PEO*

A search in the CODAP database for US plants with OPEX during the PEO was conducted based on components with 40 or more years of service at the time of the event (e.g. leakage, flaw detection). The search resulted in 69 records. Figure 3-11 is the distribution of the events in CODAP versus component age and shows that most of the events occurred between 40 and 45 years of component service. The data in Figure 3-11 is not normalised by the population of plants based on age, which would be required to determine event frequencies.

Figure 3-12 shows the distribution of damage/degradation mechanisms for US plants during PEO. Figure 3-13 shows the specific systems involved in US plant OPEX during the PEO. Systems with four or more events recorded in CODAP include: chemical and volume control, condenser system, reactor coolant system (PWR), reactor pressure vessel and service water system. The data in Figures 3-12 and 3-13 are not normalised. Hence, using the data to develop event frequencies requires an understanding of how events are characterised in CODAP. In certain cases, for example, a single inspection of a component piece/part may result in multiple cracking events being entered in the database.

**Figure 3-11: Distribution of OPEX during the PEO for US plants vs. component age**

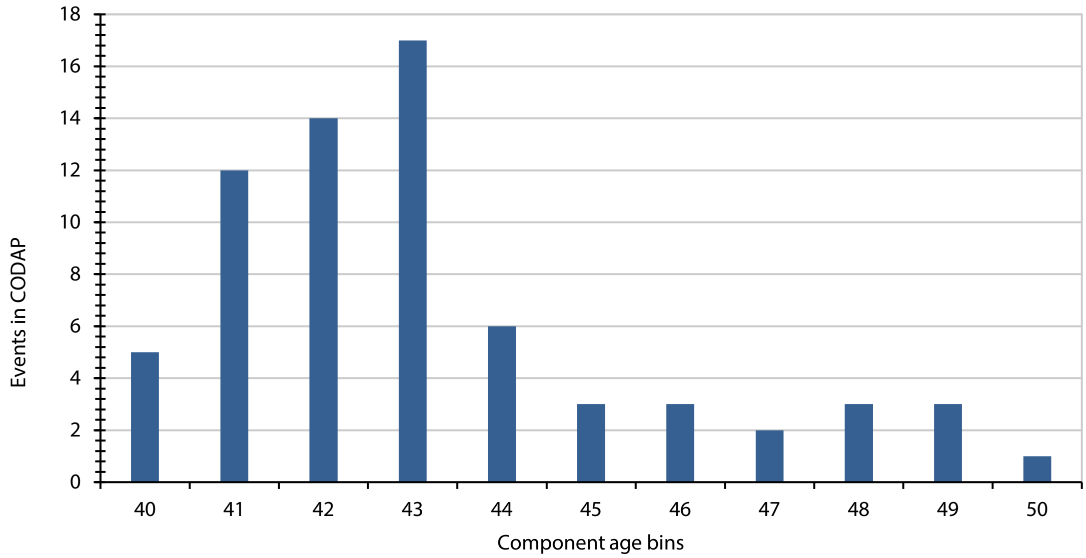



**Figure 3-12: Distribution of damage/degradation mechanisms for US plants during PEO**

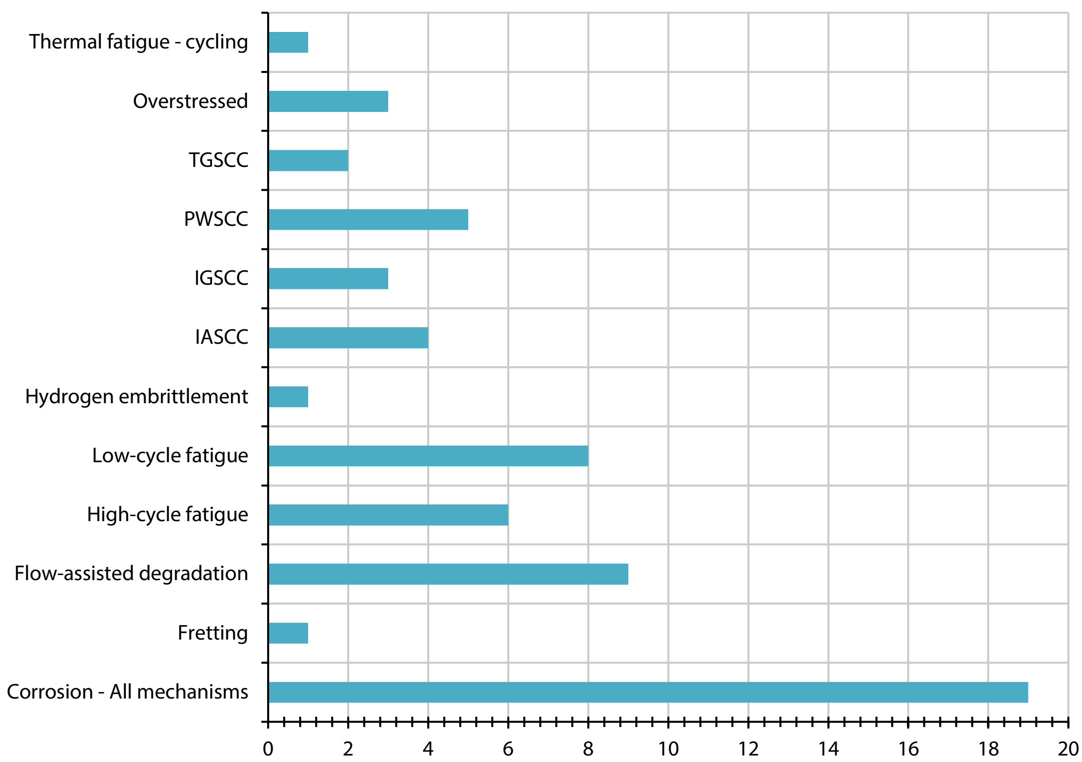
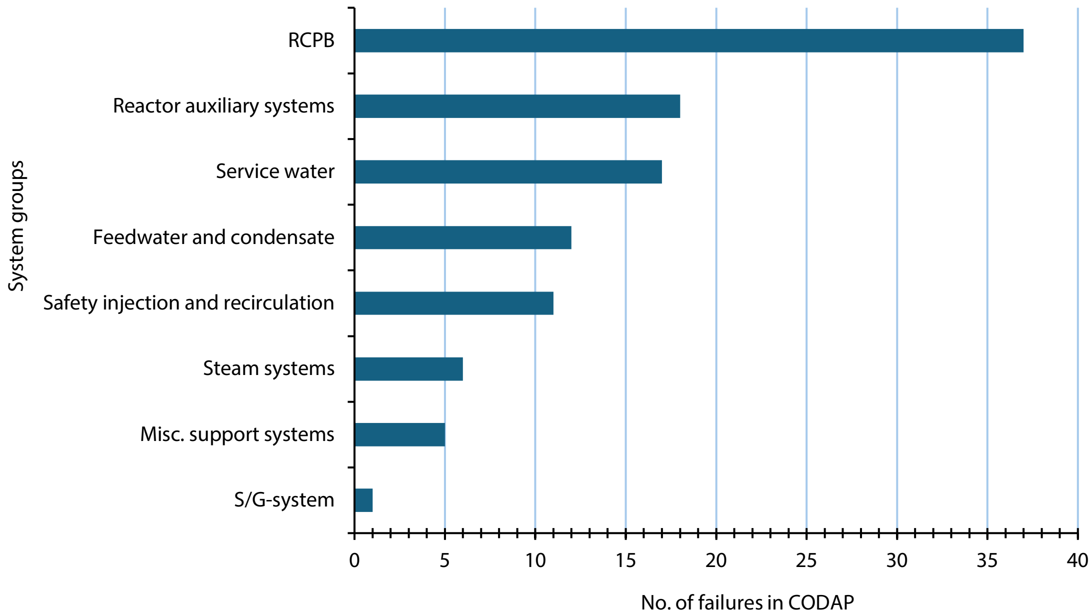


**Figure 3-13: Distribution of OPEX during the PEO across system groups**




**Baffle-former bolt cracking OPEX**: An example of RPV materials degradation being managed during the PEO is the SCC of baffle-former bolts in PWR plants. Through the EPRI materials reliability programme (MRP) and the BWRVIP, the NRC receives biennial summaries of the BWR and PWR reactor internals inspection results, respectively. These public domain summaries are available from the NRC website.[53] Below is an overview of the PWR baffle-former bolt OPEX. These reports identify the reactor internals inspected and generally include the date and frequency of inspection, the inspection method used, a summary of results including repair or replacement activities and flaw dimensions.

The core baffle is a portion of the reactor vessel internals and is located within the core barrel and functions to direct the coolant flow through the core and provide some lateral support to the fuel assemblies. Figure 3-13 shows a typical Westinghouse baffle-former assembly and the locations of various bolt types. Vertical baffle plates are bolted to the edges of horizontal former plates, which are attached to the inside surface of the core barrel. There are typically eight levels of former plates located at various elevations within the core barrel. The bolts that secure the baffle plates to the former plates are referred to as baffle-former bolts. To cool the baffle structure, some water flowing through the reactor vessel is directed between the core barrel and the baffle plates (through holes in the former plates) in either a downward direction (down-flow configuration), or an upward direction (up-flow configuration).

Degradation of baffle-former bolts was first noted in 1988 in the French Bugey Unit 2 reactor. The NRC communicated operating experience on baffle-former bolt degradation to US licensees in Information Notice (IN) 1998-11, “Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants,” dated 25 March 1998 [105]. Subsequent inspections in the US plants identified limited degradation at most. As a preventive measure, some of these plants also replaced some of their bolts, including those without indications, to an evaluated pattern to ensure structural integrity of the baffle-former assembly, using a different stainless-steel alloy and an enhanced geometry to reduce stress risers in the bolt geometry.

53. See https://adams.nrc.gov/wba; Accession No. ML19232A181 (2018 report for US BWRs) and ML18204A161 (2018 report for US PWRs).

The degradation of the baffle-former bolts is attributed to irradiation assisted stress corrosion cracking. Baffle-former bolts are subjected to significant stresses and irradiation over years of plant operation. PWRs with a down-flow configuration place additional stress on the baffle-former bolts because of the pressure differential across the vertical baffle plates. At this time, the most significant degradation of the baffle-former bolts has been observed in Westinghouse four-loop PWR reactors with the down-flow configuration and bolts made of Type 347 stainless steel. Seven reactors in the United States match this description.

In 2011, the EPRI MRP issued MRP-227-A, "Pressurised Water Reactor Internals Inspection and Evaluation Guidelines" [24], which includes an inspection of the baffle-former bolts during the time frame when bolt degradation is most likely to appear, as shown by operating experience. The NRC endorsed MRP-227-A in 2012. The guidelines of MRP-227-A provide for the development of an ageing management programme for PWR reactor vessel internals that meets the NRC requirements for issuance of a renewed operating licence.

From spring 2016 through spring 2017, four US nuclear power plants identified many Type 347 stainless steel baffle-former bolts with indications of degradation during ultrasonic inspections following MRP-227-A guidelines. From spring 2017 through spring 2019, several other US nuclear power plants also performed ultrasonic inspections following MRP-227-A guidelines but identified relatively few stainless steel baffle-former bolts with indications of degradation. In general, these plants replaced potentially degraded bolts and proactively replaced bolts that did not exhibit indications with Type 316 stainless steel bolts to improve the design.

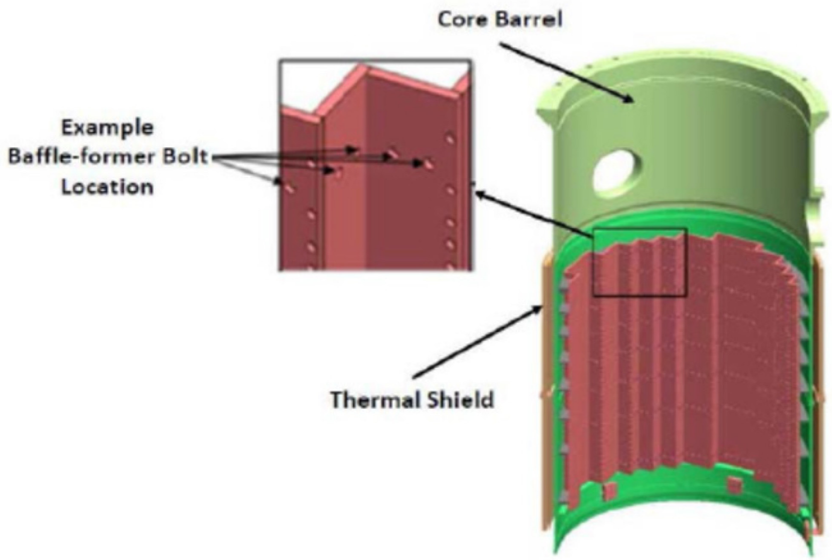
The NRC performed a risk-informed evaluation of the safety impact that the degradation of baffle-former bolts could present to operating reactors. The NRC concluded at that time that this issue did not require the immediate shutdown of any facilities. Between 2016 and 2017, EPRI issued interim guidance on baffle-former bolt inspections in Westinghouse-design PWRs. The NRC's assessment of this interim guidance is discussed in "Staff Assessment of EPRI MRP Interim Guidance on Baffle Former Bolts," dated 20 November 2017.[54] Analyses were performed to determine the material condition of the baffle-former bolts that were removed from these plants to provide additional insights on the degradation mechanism. The US nuclear industry formed a working group to consider potential changes to the MRP-227-A inspection regime and plans to incorporate the interim guidance in future revisions to MRP-227.

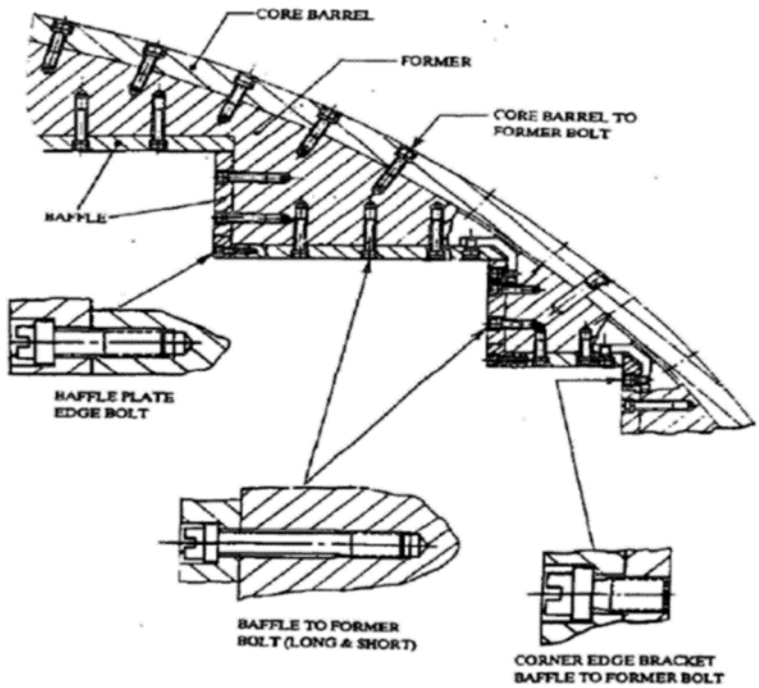
On 21-22 May 2019, the NRC met with representatives from the nuclear industry for an exchange of technical information on the industry's materials programmes.[55] During this public meeting, EPRI provided an update on baffle-former bolting inspection results. In all instances except one, the inspection results have been as, or better than, expected. The exception occurred in the spring of 2019 at a 4-loop down-flow unit. Visual inspections identified significant degradation of baffle-former bolts. Based on these results, the unit performed ultrasonic testing inspections of all original baffle-former bolts. The ultrasonic testing inspections identified a significant number of baffle-former bolts with indications of degradation and a significant clustering of degraded bolts. At the time of the public meeting, the utility was planning to replace approximately 272 baffle-former bolts and was considering other potential future actions in fall 2020.

54. NRC (2017), "Staff Assessment of EPRI MRP Interim Guidance on Baffle Former Bolts," www.nrc.gov/docs/ML1731/ML17310A861.pdf.

55. Amberge, K.J. (2019), "Update Baffle-Former-Bolting (BFB) Inspection Results in the United States", a presentation made at the EPRI-NRC technical information exchange meeting on 21-22 May 2019; https://adams.nrc.gov/wba/view (Accession No. ML19134A161).

**Figure 3-14: Illustrations of (a) a typical Westinghouse baffle-former assembly and (b) cross section of one octant of a typical Westinghouse baffle-former assembly showing locations of bolt types**



# Chapter 4. PEO/LTO operating experience data analysis

This chapter gives the results of the PEO/LTO operating experience data evaluations. The data evaluations were performed in three phases. First, through comparisons of the content of the CODAP event database against the observations and results of the WGIAGE survey [32], in Section 4.1. Next, a comparison was made of the CODAP event database against the findings of the two expert panels on material degradation [17][27], in Section 4.2. Mainly, these comparisons were done to identify possible gaps in the CODAP event database as well as to identify ways in which the project can improve its communications to the material science community. Finally, a simplified OPEX assessment was performed to identify the age-dependent trends and patterns of different combinations of systems, materials and degradation mechanisms; see Sections 4.3 through 4.7.

## 4.1 CODAP event database vs. WGIAGE conclusions

Specific to LWR operating environments, a synthesis of the WGIAGE conclusions [32] is found in Table 4-1. According to the survey, embrittlement of RPV material and RPV internals was recognised by four organisations as being of "high importance" with respect to regulation of PEO/LTO. With respect to RPV material, irradiation embrittlement is extensively monitored throughout the nuclear industry through surveillance programmes to measure changes in the properties of actual vessel materials due to the irradiation environment. The CODAP event database includes OPEX data on RPV head and bottom head penetrations. According to the current Terms of Reference for the project, CODAP does not address RPV minor fabrication defects such as the 2012 Doel-3 and 2015 Beznau-1 RPV base material flaw indications.

Three respondents to the WGIAGE questionnaire assigned a "high importance" to material degradation scenarios related to corrosion mechanisms such as "pitting and crevice corrosion near deposits, metallic contact points and stagnant conditions." These corrosion mechanisms are extensively addressed in the CODAP event database.

## 4.2 CODAP event database vs. PMDA/EMDA conclusions

In comparing the PMDA/EMDA conclusions and the CODAP event database content no fundamental gaps have been identified. CODAP is concerned with material degradation mechanisms that have produced failures. The key observations from this comparison are as follows:

- The long-term effectiveness of the various degradation mitigation strategies does warrant a continued and systematic OPEX data evaluation effort.
- The environmental effects on fatigue are a research area that continues to receive significant attention. The related OPEX is sparse, however. The continued OPEX data mining efforts would be of significant value in supporting the future research activities.
- Considerable R&D has been directed to furthering the TAE knowledge base, including extensive mechanical testing of CASS components that have been removed from service. The conjoint requirements of TAE are well understood. No TAE failures have yet been reported, however.

**Table 4-1: A synthesis of the results of the WGIAGE survey**

| Degradation mechanisms considered important to PEO/LTO of light water reactors (BWRs, PWRs and WWERs)[1] | Ranking | No. of respondents (of 8 total)[2] | How is this degradation mechanism addressed in CODAP? |
|---|---|---|---|
| RPV irradiation embrittlement, particularly in combination with hydrogen flaking and high nickel content | High | 5 | No service induced OPEX is available |
| | Medium | 1 | |
| Irradiation embrittlement of RPV internals | High | 5 | CODAP includes "selected representative events" |
| Pitting and crevice corrosion near deposits, metallic contact points and stagnant conditions. | High | 3 | The related OPEX is extensively covered in the database. NEA/CSNI/R(2018)2 [34] summarises the below ground corrosion OPEX |
| Internal/external corrosion of below ground piping | | | |
| Thermal and neutron embrittlement of cast austenitic stainless steels (CASS) | High | 2 | Only limited OPEX in the current version of the event database |
| | Medium | 2 | |
| Irradiation assisted SCC | High | 1 | CODAP includes "selected representative events" |
| SCC of Ni-base alloys | High | 1 | The related OPEX is extensively covered in the database. For additional details refer to NEA/CSNI/R(2010)5 [21]. |
| Thermal fatigue caused by stratification and thermal cycling in branch connections (e.g. mixing tees) | High | 1 | The related OPEX is extensively covered in the database. For additional details refer to NEA/CSNI/R(2019)13 [38]. |
| | Medium | 1 | |
| FAC of high energy pipelines | High | 1 (WWER) | For WWERs the database includes some "representative events" and potentially this OPEX does not reflect the "high" or "medium" priority in the WGIAGE Survey[3] |
| | Medium | 1 (WWER) | |
| Environmentally assisted fatigue of primary circuit metals | High | 1 | Only a few records in the database and limited to corrosion fatigue/SICC failures |
| | Medium | 2 | |
| Boric acid corrosion in bolted closures of the primary circuit pressure boundary | High | 1 | Yet, bolted connections are not part of the scope of the CODAP event database |

1. CANDU-specific degradation mechanisms are not addressed in this table.

2. Eight CODAP project member countries/economies responded to the survey.

3. The CODAP event database includes 13 CZ records covering the period 1985 to 2013 and 4 SK records covering the period 2000 to 2010.

## Table 4-2: Comparison of PMDA/EMDA conclusions with CODAP event database insights

| BWR primary system components | | |
|---|---|---|
| **PMDA: 25 < T ≤ 40 Years** | **EMDA: 40 < LTO ≤ 80 years** | **CODAP OPEX data insights** |
| SCC of Alloy 82/182 weldments used in joints between ferritic and austenitic components (e.g. thermal sleeves, attachment pads) and particularly alloy 182 under NWC conditions. | SCC was identified as a high knowledge, moderate Susceptibility mode of degradation for alloy 182/82 weldments in BWR HWC environments. Reduced fracture resistance in alloy 182/82 welds at lower temperatures has been noted in laboratory testing although additional mechanistic understanding is needed. | Extensive OPEX coverage in the database. As an example, refer to Figure 4-1 for the BWR-specific OPEX. |
| SCC of Type 304/316 stainless steel heat affected zones (even under HWC conditions) at moderate fluences. | The following knowledge gaps were identified: 1) impact of irradiation on fracture toughness and SCC in both NWC and HWC environments, 2) SCC susceptibility at very long operation periods, particularly in NWC environments, and 3) Cumulative impact of fatigue on corrosion and component integrity, particularly weldolets, sockolets, and components in the upper core internals. | The CODAP database is structured so that material degradation of reactor internals can be tracked. So far, no systematic process has been implemented for the collection of failure data on reactor internals, however. |
| Fatigue of type 304/316 stainless steel in steam dryers under NWC and HWC conditions. This is probably design-specific, but as of now, is not quantified across the whole fleet. | Not explicitly addressed beyond noting that the steam dryer holds down brackets are alloy 182. | CODAP includes selected representative event reports on BWR steam dryers. So far, no systematic process has been implemented for the collection of failure data on reactor internals, however. |
| SCC cracking of alloy 600 components and heat affected zones throughout the reactor coolant system, particularly under NWC conditions. | SCC was identified as a high knowledge, high susceptibility mode of degradation for alloy 600 in BWR NWC environments by the expert panel. This is a known issue for these alloys in high potential environments. | Most, if not all, BWR plants have implemented HWC to mitigate SCC susceptibility; refer to the first line item of this table. |
| SCC of Alloy X-750 components throughout the reactor coolant system under both NWC and HWC conditions. | Extended service will result in increased time exposed to the high-temperature water environment and under stress. Further, for core internals, an increased fluence will be experienced due to longer service and power uprates. PIRT scoring for X-750 in the high Knowledge, high Susceptibility category due to problems identified in service. | The first Alloy X-750 failures (in-vessel bolting) were observed in Germany in the late 1970s. The most recent and a first-of-kind failure of X-750 were reported in the United States in February 2006. It involved a core shroud tie rod upper support. |
| Cast stainless steel components (e.g. CF8M brackets and guide rods and CF3 control rod guides), may experience a reduction in fracture resistance as a result of thermal ageing. For the most part, however, the BWR RCS components operate at lower temperatures than PWR RCS components and, therefore, possible problems would be expected to become evident in PWRs earlier than in BWRs. Also, the possibility of SCC occurring in cast stainless steel components should not be dismissed without further investigation of the possible synergistic effect of thermal ageing on this degradation mode. | The effects of long-term thermal ageing for extended operating periods may drive changes in mechanical or corrosion performance that are relatively unknown. | No thermal ageing failure data has been identified for inclusion in CODAP. The definition of TAE failure modes is an issue for the CODAP-MB to consider. |

**Table 4-2: Comparison of PMDA/EMDA conclusions with CODAP event database insights** *(cont'd)*

| BWR primary system components | | |
|---|---|---|
| **PMDA: 25 < T ≤ 40 Years** | **EMDA: 40 < LTO ≤ 80 years** | **CODAP OPEX data insights** |
| The long-term effects of HWC on SCC are not yet clear and it is also not clear that the low potentials needed for mitigation are in fact achieved at all the high-susceptibility locations. The role of hydrogen in the stress corrosion cracking process of austenitic stainless steels is critical but is expected to be less than in high strength nickel-base alloys. The effect of hydrogen on fatigue and its interaction with ripple loading and dynamic strain ageing was noted as a potential contributor to degradation under HWC conditions. | Continued "potential knowledge gap" regarding the SCC susceptibility at very long lifetimes. | This "degradation scenario" is being monitored through the review of OPEX. Relevant information has been reported at conferences such as EDF FAC conferences, Fontevraud1 through Fontevraud8 and ENVDEG1 through ENVDEG19 (Environmental Degradation of Materials in Nuclear Power Systems). To date, CODAP does not contain any explicit information, however. Noteworthy is the observation of the Swedish National Ageing Management Action Plan (SSM 2019:17) regarding HWC. The Swedish BWR plants with internal recirculation pumps (Forsmark-1/2/3 and Oskarshamn-3 abandoned HWC mitigation in the 1990s. According to the Owners "… it is almost impossible to create the mitigating environmental conditions in the bottom of the pressure vessel." www.stralsakerhetsmyndigheten.se/en/publications/reports/safety-at-nuclear-power-plants/2019/201917. |
| Accelerated environmental effects on fatigue of austenitic stainless steels and nickel alloys at low electrochemical corrosion potentials under HWC conditions. These effects have been demonstrated in the laboratory and therefore are anticipated to apply in the field. | Continued "potential knowledge gap" regarding the cumulative impact of fatigue on corrosion and component integrity at very long lifetimes. | No OPEX available as of yet. |
| Stress corrosion cracking of severely cold worked or mechanically strained stainless steels in BWR water, e.g., weld shrinkage strains in heat affected zones, cold bent elbows. | The EMDA Report includes extensive details on this subject. | Extensive OPEX included in CODAP. The role of cold work on the susceptibility of stress corrosion cracking (SCC) in metallic nuclear materials, especially austenitic stainless steels, and nickel-base alloys, has been recognised for the past several decades and the first operational events occurred during the early days of commercial nuclear energy production. |
| "Abusive" grinding or machining during weld preparation or surface finishing and their adverse effects on stress corrosion and fatigue crack initiation. | Not explicitly addressed. However, the EMDA Expert Panel refers to the adverse effects of improper weld preparation or surface finishing. | Extensive OPEX on this topic can be extracted from CODAP. See for example Reference [21]. |

## Table 4-3: Comparison of PMDA/EMDA conclusions with CODAP event database insights (Part 2)

| PWR primary system components | | |
|---|---|---|
| **PMDA: 25 < T ≤ 40 years** | **EMDA: 40 < LTO ≤ 80 years** | **CODAP OPEX data insights** |
| SCC of alloy 82/182 weldments throughout the primary system and especially in the highest temperature components such as the pressuriser. | The expert panel identified only one potential knowledge need for alloy 182/82 in PWR environments. Specifically, fracture resistance in alloy 182/82 welds at lower temperatures has been noted in laboratory testing although limited mechanistic understanding has been established. | Addressed extensively in CODAP event database and the Topical Report on SCC [21]. |
| SCC of alloy 600MA and 600TT steam generator tubes and alloy 600 forged components and, especially, cold worked or mechanically strained material such as steam generator tube expansion transitions, small radius U-bends, and the high-temperature components of the pressuriser. | SCC is a known issue for alloy 600 in all alloy forms. SCC has been observed in both primary and secondary-side applications for all forms of alloy 600 in service. The expert panel scored all 12 categories of SCC in the high knowledge, high susceptibility grouping. alloy 600 in the MA form was given the highest possible susceptibility score, consistent with operational experience. Most plants have or will be performing steam generator replacements and in the process are upgrading from Alloy 600 to Alloy 690. | The CODAP work scope considers alloy 800 and alloy 690 SG tube degradation. The recent (2018 and 2019) German OPEX involving degradation of alloy 800 tubes is included in CODAP. As of end of calendar year 2019, the CODAP event database includes two (2) data records on SG tube degradation. |
| Corrosion of low-alloy steel components in boric acid concentrate originating from primary water leaks, specifically in the annuli between the alloy 600 penetrations and the low-alloy steel reactor vessel or pressuriser. | Several trends for low alloy steels in PWR environments were identified via the PIRT process: 1) BAC of low alloy steel was identified as a high Knowledge, high Susceptibility mode of degradation, but only in the event of a leak of primary coolant. This is a well-known form of degradation, and 2) Crevice corrosion, pitting, and FAC of low alloy steel were identified as high Knowledge, moderate Susceptibility modes of degradation, but only in the event of a loss of water chemistry control. These are well-known forms of degradation. Stress corrosion cracking and fatigue are possible for these alloys, but unlikely in service. Changes in loading or increases in chemical conditions may drive increased susceptibility over a long operating period. | This is covered in the CODAP event database. |
| Irradiation-induced creep and SCC of austenitic stainless steels at more than 0.5 dpa, (e.g. baffle bolts, and other high strength fasteners) and swelling in internals components that reach higher temperatures and much higher doses. | Several knowledge gaps for 316 SS in PWR environments were identified via the PIRT process: 1) Effect of irradiation on fracture toughness, irradiation creep, swelling, and SCC, and 2) SCC susceptibility at very long lifetimes. Additionally, while scored in the high Knowledge category, the panellists noted that the cumulative effect of fatigue on corrosion and component integrity deserved additional examination due to possible changes and uncertainties in cyclic and flow-induced loading over extended service condition. | Extensive baffle former OPEX exists. Selected representative events are included in the CODAP event database. |

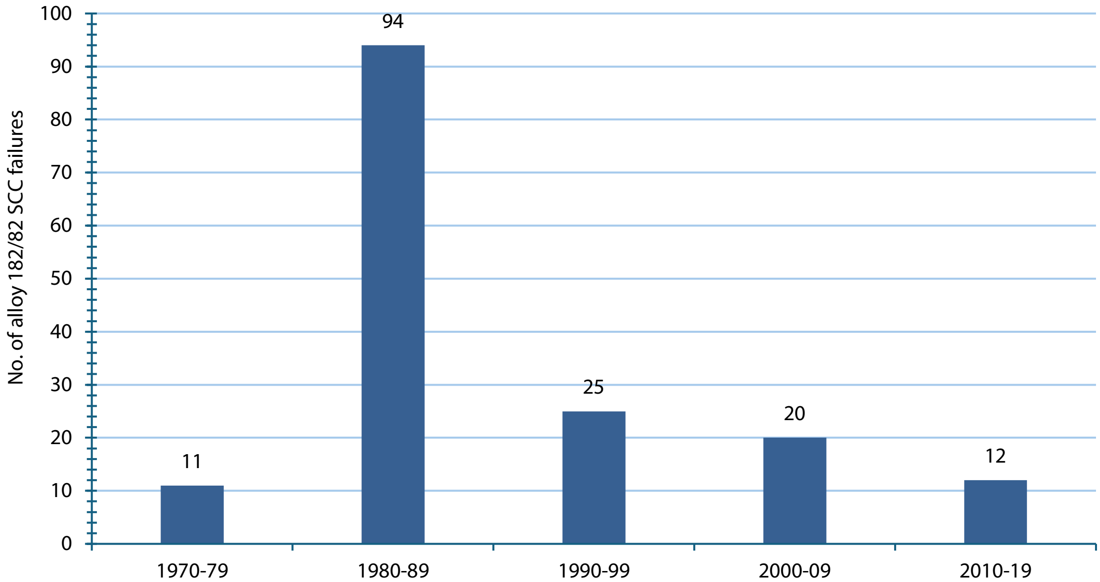
**Table 4-3: Comparison of PMDA/EMDA conclusions with CODAP event database insights (Part 2)** *(cont'd)*

| PWR primary system components | | |
|---|---|---|
| **PMDA: 25 < T ≤ 40 years** | **EMDA: 40 < LTO ≤ 80 years** | **CODAP OPEX data insights** |
| Fatigue due to unanticipated vibration and thermal fluctuations, e.g. in socket welds (which occur throughout the RCS), primary circuit deadlegs (e.g. abandoned pipe sections), and baffle bolts after irradiation-induced relaxation of pre-stress. | Fatigue and corrosion fatigue are known issues in reactor service. As service life increases, so does the total number of loading cycles experienced by a component. Further, power uprates may also increase flow-induced cyclic loading and impact component lifetime. The CUF must be evaluated for extended service. All environments for 316 SS were scored in the high knowledge, moderate susceptibility grouping. | Although not yet done, a correlation (if any) between power uprates and the fatigue incident rate can in principle be determined using the CODAP event database. This could be a topic for a future Topical Report. |
| Although alloy 690TT has exhibited good resistance to SCC compared with alloy 600 MA during more than 16 years of use in steam generator tubing, some panel members questioned its long-term resistance in other PWR primary water applications (e.g. pressure vessel penetrations). The potential for this degradation increases when considering effects of cold work and the use of the alternate welding alloys (alloys 52 and 152) which present welding challenges associated with, for example, lack of fusion and ductility dip cracking. | No knowledge gaps were identified for alloy 690 under subsequent operating periods in PWR environments following the PIRT scoring activity. The panellists did note that SCC, fatigue cracking, and pitting should be minimal for alloy 690 in service, although good water chemistry must be maintained. SCC is a known issue for alloy 690 in all its forms. SCC has not been observed to date in actual service, although it has been observed in some laboratory experiments. | Terms of reference of the 3rd project term clearly states that any "service-induced material failures involving alloy 690 or alloy 52/152 shall be considered for inclusion in the event database." According to NUREG-1841, "As of 31 December 2004, it [Alloy 690] was being used in about 43% of the operating PWRs in the US. Of the 577 070 alloy 690 TT tubes placed in service, only 333 tubes (0.06%) have been plugged after approximately 173 calendar years of operation. Most of these tubes (65%) were plugged prior to placing the steam generators in service. The dominant in-service degradation mode, responsible for about 24% of the plugged tubes, has been wear caused by a support structure or loose part. No corrosion or cracking had been detected as of the time this report was prepared." |
| SCC of severely mechanically strained stainless steels in PWR primary water, e.g., weld shrinkage strains in heat affected zones, pressuriser heater cladding and cold-bent piping elbows. | Not explicitly addressed in the PIRT scoring activity. | No OPEX has yet been identified. |
| Cyclic thermal loading in the primary circuit hot leg caused by the difference in temperature between primary coolant streams exiting the centre and periphery of low leakage cores which, despite turbulence, can persist up to the steam generator channel head. While studies have shown that this cyclic loading does not produce significant thermal fatigue issues, the possible effect of ripple loading on other degradation mechanisms (such as stress corrosion cracking) could be a concern. | Not explicitly addressed in the PIRT scoring activity. | No OPEX has yet been identified. |

**Table 4-3: Comparison of PMDA/EMDA conclusions with CODAP event database insights (Part 2)** *(cont'd)*

| PWR primary system components | | |
|---|---|---|
| **PMDA: 25 < T ≤ 40 years** | **EMDA: 40 < LTO ≤ 80 years** | **CODAP OPEX data insights** |
| Secondary side degradation (e.g. SCC) of steam generator tubes by potentially corrosive species not previously appreciated to be widespread in the secondary system and which may also affect "more resistant" tubing materials such as alloy 690. | Ferritic stainless steel grade 405 and 409 are also used in the secondary side of steam generators in PWRs. The expert panel rated these materials in the secondary environment for SCC and crevice corrosion. In both cases, the panellists noted the importance of specific water conditions, which will drive susceptibility for these alloys. | Refer to Section 3.2.3 of this report. |
| Potential degradation related to end of fuel cycle primary water chemistry, especially during cycle stretch-out, when the concentration of boron may be reduced below the minimum recommended value, thereby leading to possible localised corrosion due to excess LiOH, in, e.g. the boiling crevices in the pressuriser. | Not explicitly addressed in the PIRT scoring activity. | No OPEX has yet been identified. |
| Accelerated environmental effects on fatigue of austenitic stainless steels and nickel alloys at low electrochemical corrosion potentials, as demonstrated in the laboratory and therefore anticipated to apply in the field. | Not explicitly addressed in the PIRT scoring activity. | No OPEX has yet been identified. |
| Less-than-anticipated fatigue resistance of some components due to the current uncertainty in the characterisation of the cyclic loading conditions, particularly thermal loading, and the material response under those specific conditions. | Not explicitly addressed in the PIRT scoring activity. | No OPEX has yet been identified. |
| Reduced fracture resistance and increased susceptibility to stress corrosion cracking of cast stainless steels (and perhaps ferrite-containing stainless-steel cladding and welds) as a result of thermal ageing. | While CASS components have an excellent performance record, extended service will result in increased exposure time to the primary water environment and stress. The expert panel considered eight different categories of SCC for cast austenitic alloys in PWR environments. The panel scored the CASS and HAZ in primary water in the low Knowledge, moderate Susceptibility category. As above, this was driven by uncertainty in the effects of thermal ageing over an extended operating period on microstructure and mechanical performance. Other categories were ranked at lower susceptibility, although thermal ageing effects were considered by the panel as the key factor for this mode of degradation in all environments. One potential knowledge gap for CASSs was identified for both PWR and BWR environments using the PIRT data. Specifically, the effects of long-term thermal ageing for extended operating periods may drive changes in mechanical or corrosion performance that are relatively unknown. | The current version of the database includes 12 records involving CASS components, none of which has TAE as cause of failure. |
| "Abusive" grinding or machining during weld preparation or surface finishing and their adverse effects on stress corrosion and fatigue crack initiation. | Not explicitly addressed in the PIRT scoring activity. | This is covered in CODAP; extensive OPEX exists. Details can be found in the Topical Report on SCC [21]. |

**Figure 4-1: The BWR OPEX involving SCC of safety class 1 alloy 182/82 welds**




Note: In interpreting this OPEX summary it should be noted that by approximately 1990 most BWR operators had implemented (or were in the process of implementing) hydrogen water chemistry (HWC). Also, by this time, other types of IGSCC mitigation plans had been implemented, or were in the process of being implemented.

### 4.2.1 *Note on SCC-susceptibility of alloy X-750*

According to Reference [86]:

> "Alloy X-750 with high temperature solution annealing (1 093°C) and single stage (704°C) HTH ageing treatment is very resistant to SCC in high temperature primary water if any surface oxide layer due to heat treatment is removed by machining. However, this SCC resistance can be reduced by the presence of sulphide inclusion clusters in the matrix. The presence of certain surface layers built up during the last step of the manufacture such as the ageing treatment. If the final step is the ageing treatment, the nature of the atmosphere of the furnace has a direct influence on the behaviour of the material when stressed in high temperature primary water. Treatment under vacuum can induce SCC. Heat treatment under nitrogen or machining after heat treatment leads to the best resistance to SCC. The exposure of these surfaces to primary water at high temperature modifies the surface conditions. Stress corrosion cracking may be then triggered when tensile stresses are applied. This phenomenon can be accounted for by the formation of grain boundary damage in the sub-surface matrix during the ageing heat treatment or during the exposure to primary water depending on the composition of the surface layer. The composition of the surface film may also have a direct influence on the initiation phase for SCC. This effect of the surface layer on the SCC initiation explains the beneficial effect of polishing the control rod tube guide pins in the French PWRs and suggests that a better resistance may be obtained by oxidising the surface in a well-defined environment to create a surface layer enriched in Cr that protects the sub-surface matrix against internal oxidation."

In Germany, Alloy X-750 material degradation was first discovered in 1978 at KWB-A (Biblis A).[1] An ultrasonic examination revealed that several of the Inconel X-750 baffle-former bolts were defective. When the bolts were removed, some of them were found to be broken and others had cracks of different sizes. During 1980 through 1989, flawed baffle-former bolts were discovered in another four reactor units: KWB-B (Biblis Unit B), GKN-1 (Neckarwestheim-1), KKU (Unterweser) and KKS (Stade). These PWR units were commissioned from 1972 to 1978. Studies of the cracks showed that the cause of damage was IGSCC. The crack locations in the damaged baffle-former bolts were in the transition between the bolt head and the necked-down shaft (extending into the hexagonal socket), within the necked-down shaft and in the transition between the shaft and the thread and in the thread itself. All the baffle-former bolts were examined by ultrasonic testing. Damage to Inconel X-750 baffle-former bolts was subsequently found in other German plants as well. No defects were found in bolts made from austenitic stainless steel.

In the German PWR plants the baffle-former bolts made of Inconel X-750 (a total of over 5 100 bolts) were replaced with bolts made of material 1.4571 (corresponding to Type 316Ti, cold worked material). In the newer plants constructed after the early detection of this kind of cracking, bolts made from the alternative material were used. However, in 2005 some indications in the austenitic baffle-former bolts were detected at several Siemens/KWU-type PWR plants [105].

Reference [86] documents the French operating experience with Inconel X-750 bolts. After 11.6 EFPYs, 5-of-80 core barrel bolts were reported broken in 1984 during in-vessel visual inspections at Chooz-A. These bolts were replaced with the same material, but heat treated for better SCC resistance. Metallurgical examinations attributed the failures to intergranular SCC.

### 4.2.2 Note on PWR CRDM thermal sleeve degradation

Wear on CRDM thermal sleeves was initially noted in 2007 at a Westinghouse designed plant, while performing vessel head penetration J-groove weld examinations. Since this discovery, similar wear has been noted at other Framatome and Westinghouse designed PWR units; most recently in August 2019. The wear indications on the thermal sleeves were in the area where the thermal sleeve exits the CRDM head adapter tube. The wear is attributed to the thermal sleeve contacting the inside diameter of the CRDM head adapter tube due to flow-induced motion of the thermal sleeve.

### 4.2.3 Note on raw water-cooling system piping OPEX

The safety-related raw water-cooling piping operates in a corrosive environment. The piping system designs vary extensively. The system takes cooling water from a freshwater lake, reservoir, river or ocean. Depending on the specific system design, multiple important safety functions may be cooled directly by raw water ("open-loop" system), or indirectly by a closed-loop component cooling water system.

The safety significance of raw water-cooling piping integrity is evaluated within the scope of the internal flooding (IF) probabilistic safety analysis (PSA) element of plant-specific PSA. Insights from the review of IF-PSA studies indicate that for some plants a break of a raw water-cooling pipe is a major contributor to core damage frequency (CDF). Therefore, the quantitative assessment of raw water-cooling pipe break frequency in some cases is an important aspect of IF-PSA.

1. Rubel, H., J. Tautz and G. Micheel (1987), "Inconel~X-750 als Werkstoff für Kerneinbauten und Kernbauteile – Bisherige Erfahrungen und Austauschaktionen," *Proc. 13. MPA-Seminar*, Stuttgart, Germany, 8-9 October 1987, https://inis.iaea.org/search/search.aspx?orig_q=RN:20059527.

## 4.3 Basic principles of PEO/LTO-OPEX data analysis

The opportunity to identify and evaluate the potential effects of ageing on passive metallic component integrity was made possible by two important factors. The first is the commitment of the NEA and its members countries to support technical activities in the areas of structural integrity and ageing management (e.g. WGIAGE[2]) and operating experience data exchange (e.g. CODAP). The second is the availability of insights from the application of analysis methods and techniques to explore operating experience data in a structured manner. Thus, identifying trends in operating experience is made possible by the continuous and systematic application of relevant data analysis methods combined with a well maintained OPEX database.

There are multiple technical approaches to the analysis of age-dependent component failure rates [106]. In this report only simple "visual tests" of graphical plots are performed to obtain high-level insights into age-dependent material performance. It is a first step in a more rigorous analysis of the OPEX data. The objectives of ageing factor analysis account for:

- Ageing plant fleet. Numerous nuclear plants have entered into PEO/LTO. Ageing factor assessments measure the effectiveness of an existing ageing management programme with respect to long-term material performance. Specifically, a detailed identification and assessment of temporal trends would generate insights into the effectiveness of material ageing management and to determine whether event recurrence patterns exist.
- Renewal processes. Piping systems are routinely replaced-in-kind or upgraded, for example, by replacing original material with material known or assumed to be resistant to degradation. Also, there is plant-to-plant variability in piping system design, which impacts the material degradation propensity.
- New operating experience data. The operating experience with metallic passive components is continuously being updated. It needs to be determined if the data collection process is sufficient to support quantitative ageing factor assessments.
- Enhancements in reliability and integrity management (RIM). Plant life extension initiatives together with applications of non-destructive examination (NDE) techniques and inspection qualification processes continue to evolve. Embedded in the operating experience data are effects of NDE and changes in the reporting of pipe failures. It needs to be determined if the OPEX data reflects improvements in RIM programme implementation or unanticipated changes in material performance.

As seen in Figures 4-2 and 4-3, the occurrence rate of pipe failure shows distinct negative and positive trends over the nuclear power plant lifetime. In Figure 4-2 the pipe failure rate was calculated as the total number of observed pipe failures in an arbitrarily selected 3-year interval over the total number of plants in that interval that could produce a failure. The OPEX that is represented in Figure 4-2 covers all systems and all safety classes. Listed below are some observations:

- An inverted "bathtub curve" may be discerned from this data processing step. However, further analysis of the OPEX is required to reach conclusions about ageing-dependency. Included in Figure 4-4 are BWR- and PWR-specific insights about degradation mechanisms and their implications on plant ageing management. For example, the BWR plants experienced an increasing pipe failure rate during the first ten years of operation. This increase was attributed to IGSCC. Subsequent regulatory requirements and industry initiatives to address IGSCC have had significant mitigating effects. The IGSCC causal factors are well understood and controlled.
- An uptick and increase in the calculated pipe failure rate is observed for plants that have operated for >35 and >41 years, respectively. This increase is mainly attributed to corrosion mechanisms acting on raw water piping systems, e.g. circulating water system, service water system and fire water system piping. Examples of corrosion mechanisms include crevice corrosion, microbiologically influence corrosion and pitting corrosion.

2. See the dedicated WGIAGE page on the NEA website: www.oecd-nea.org/wgiage.

Age-dependent pipe failure rate estimation can be performed according to different analysis strategies to obtain piping reliability parameters as a function of the age of an affected piping component at the time of failure observation, or as a function of the temporal changes in the piping operating experience. The term “temporal change” means that descriptive failure statistics (mean, upper/lower bound, etc.) change over time. The temporal trends may be due to ageing, such as change in the physical properties of piping material (e.g. thinning or cracking), but may also be due to changing reporting routines and the data collection processes.

In the analysis of piping reliability the term “cohort effect” is sometimes used to describe variations in observed structural integrity factors (e.g. onset of crack initiation and subsequent crack growth) as a function of operating time, plant age, plant design generation and degradation mitigation implementation strategies (e.g. full structural weld overlay, peening, induction heating stress improvement). The commercial nuclear power plant designs have evolved, and those reactor units designed in the 1960s and commissioned in the early 1970s exhibited quite different service experience histories than reactors designed and commissioned in later time periods.

As one example, cohort effects relating to intergranular stress corrosion cracking (IGSCC) of boiling water reactor (BWR) primary system unirradiated stainless steel piping are particularly strong. Those BWR units commissioned in the 1960s and 1970s exhibited a very high IGSCC incidence rate, which was attributed to a lack of recognition of the relationships between the operating environments (e.g. water chemistry), material characteristics (e.g. carbon content) and stress conditions. The worldwide service experience with IGSCC in Code Class 1 piping is summarised in Figures 4-4 and 4-5 (a total of 1 126 IGSCC failure records). The service experience is organised by “IGSCC Class” where each class corresponds to a uniquely defined event population:

- IGSCC-1 (Gen I) corresponds to the first BWR design generation (essentially demonstration plants). These units were brought online in the 1960s and except for one unit, they have all been decommissioned.
- IGSCC-2 (Gen II “Early”) corresponds to the second BWR design generation; plants commercialised in early 1970s. Several units have entered into an extended period of operation, beyond 40 years, and a few of units are undergoing decommissioning.
- IGSCC-3 (Gen II “Midi”) corresponds to the third BWR design generation. Multiple IGSCC-mitigation projects were implemented in the latter part of the 1980s.
- IGSCC-4 corresponds to two specific BWR NSSS design generations (SWR69 and SWR72, respectively) that utilised stabilised austenitic stainless steel materials. This class is limited to German plants for which IGSCC-mitigation consisted of a re-design of the primary piping systems. The steps taken have significantly reduced the IGSCC susceptibility.
- IGSCC-5 (Gen II “Late”) corresponds to BWR designs for which IGSCC-mitigation consideration was an integral part of the original piping design (e.g. use of nuclear grade austenitic stainless steel) and plant operating practices (e.g. enhanced primary water chemistry control).

Not only do Figures 4-4 and 4-5 include five different sets of OPEX data; the charts show how the IGSCC incidence rates have changed over time. Correlating the five event populations with corresponding exposure terms (plant population that generated the failures, operating time and weld populations) provides a basis for establishing “a priori” failure rate distributions for each of the IGSCC categories.

**Figure 4-2: Pipe failure rate as a function of the age of the piping at the time of failure of safety- and non-safety-related piping**

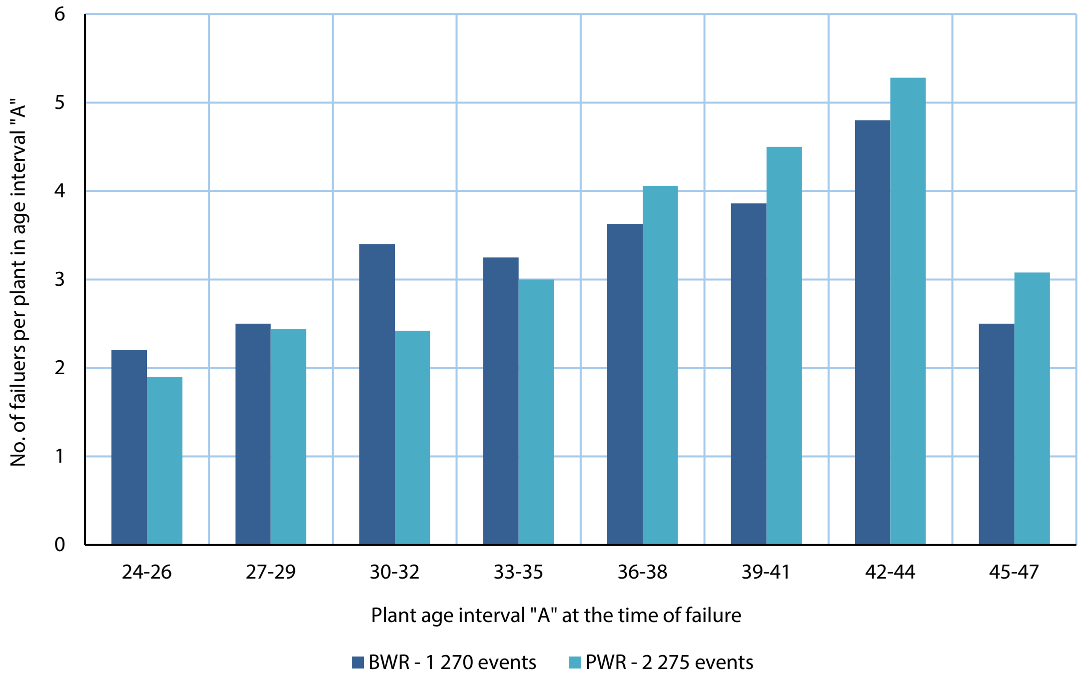



**Figure 4-3: The small-diameter (≤ DN50) pipe failure OPEX**

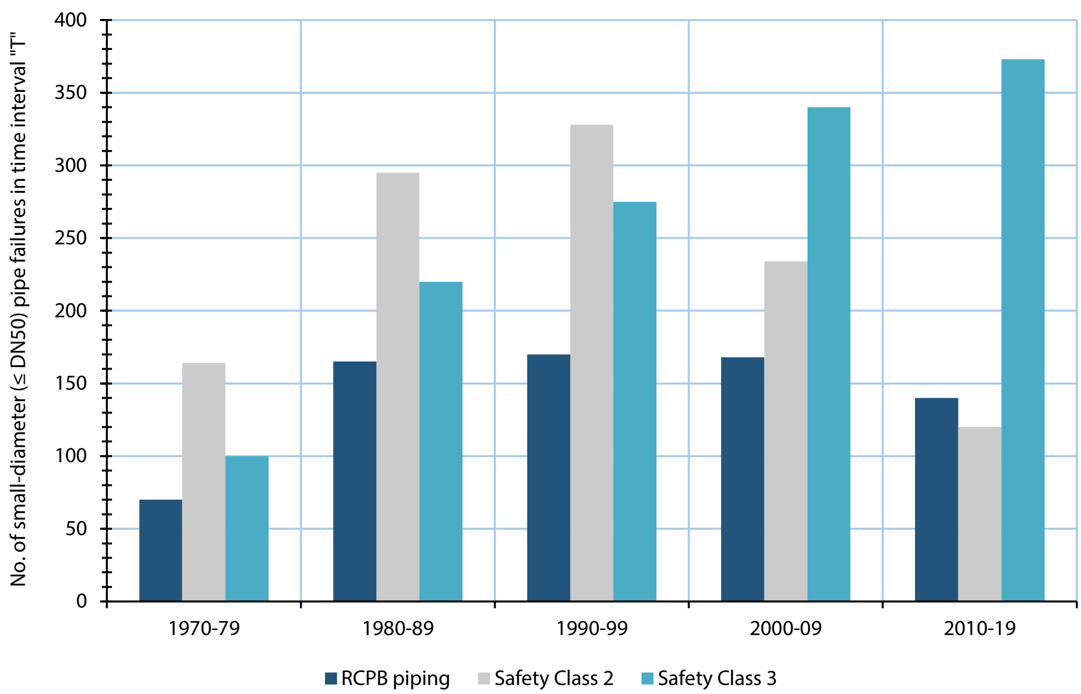


**Figure 4-4: The BWR-specific IGSCC operating experience data**

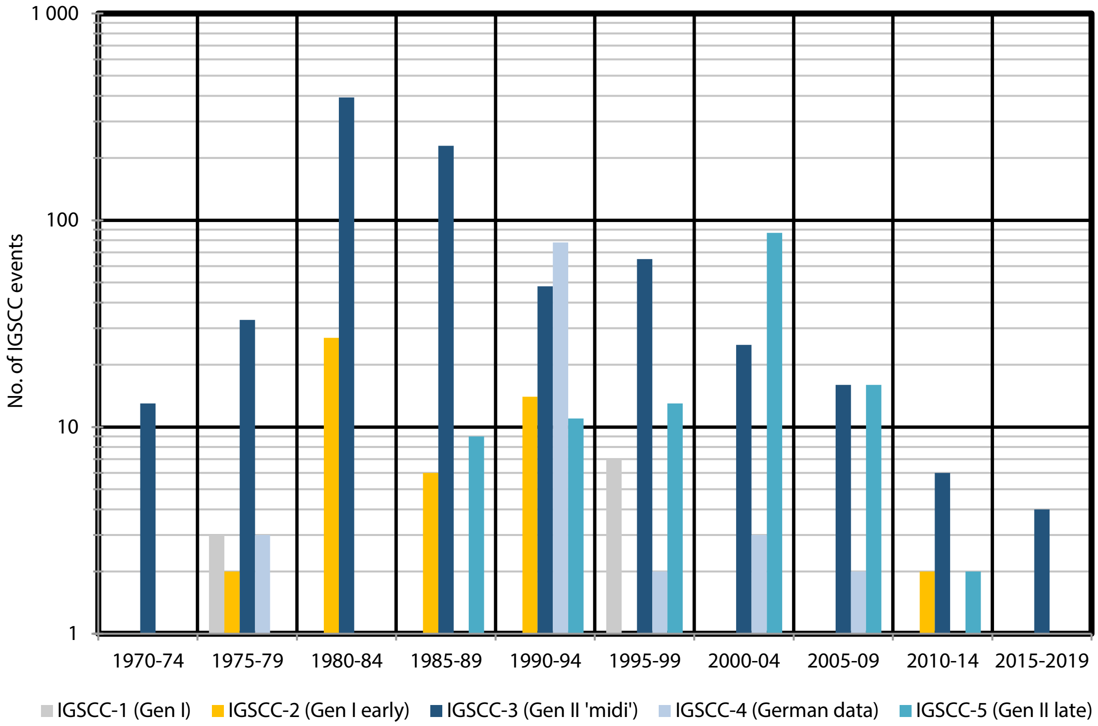
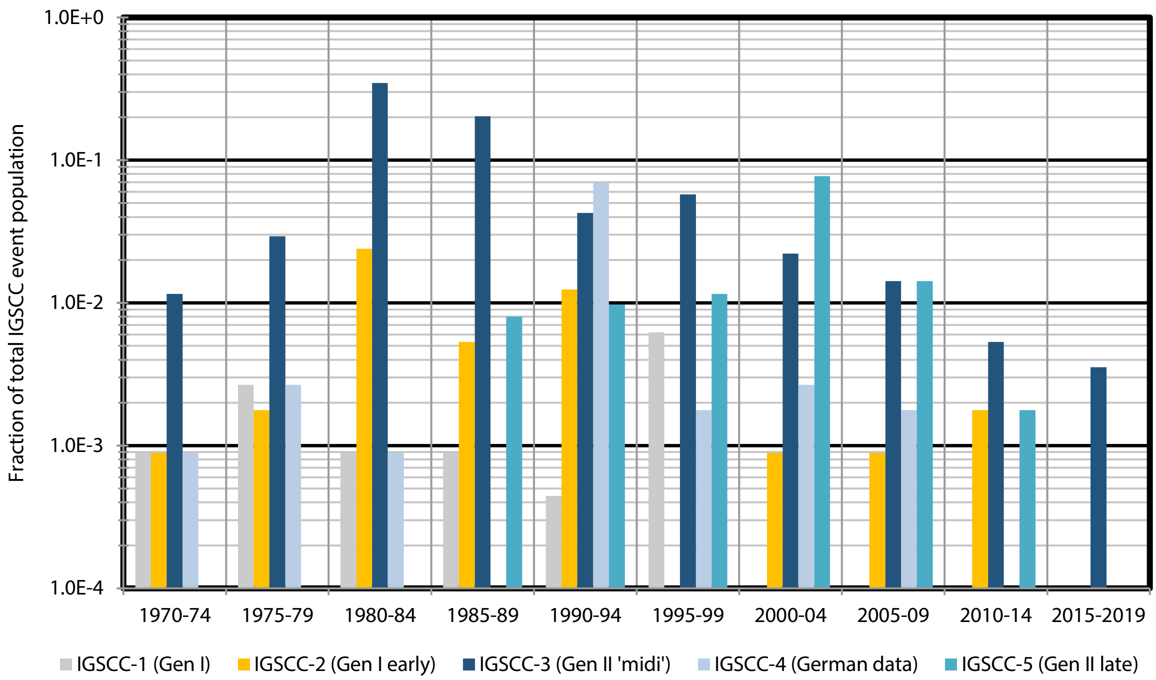



**Figure 4-5: The IGSCC "incident rates" by time period and BWR design generation**

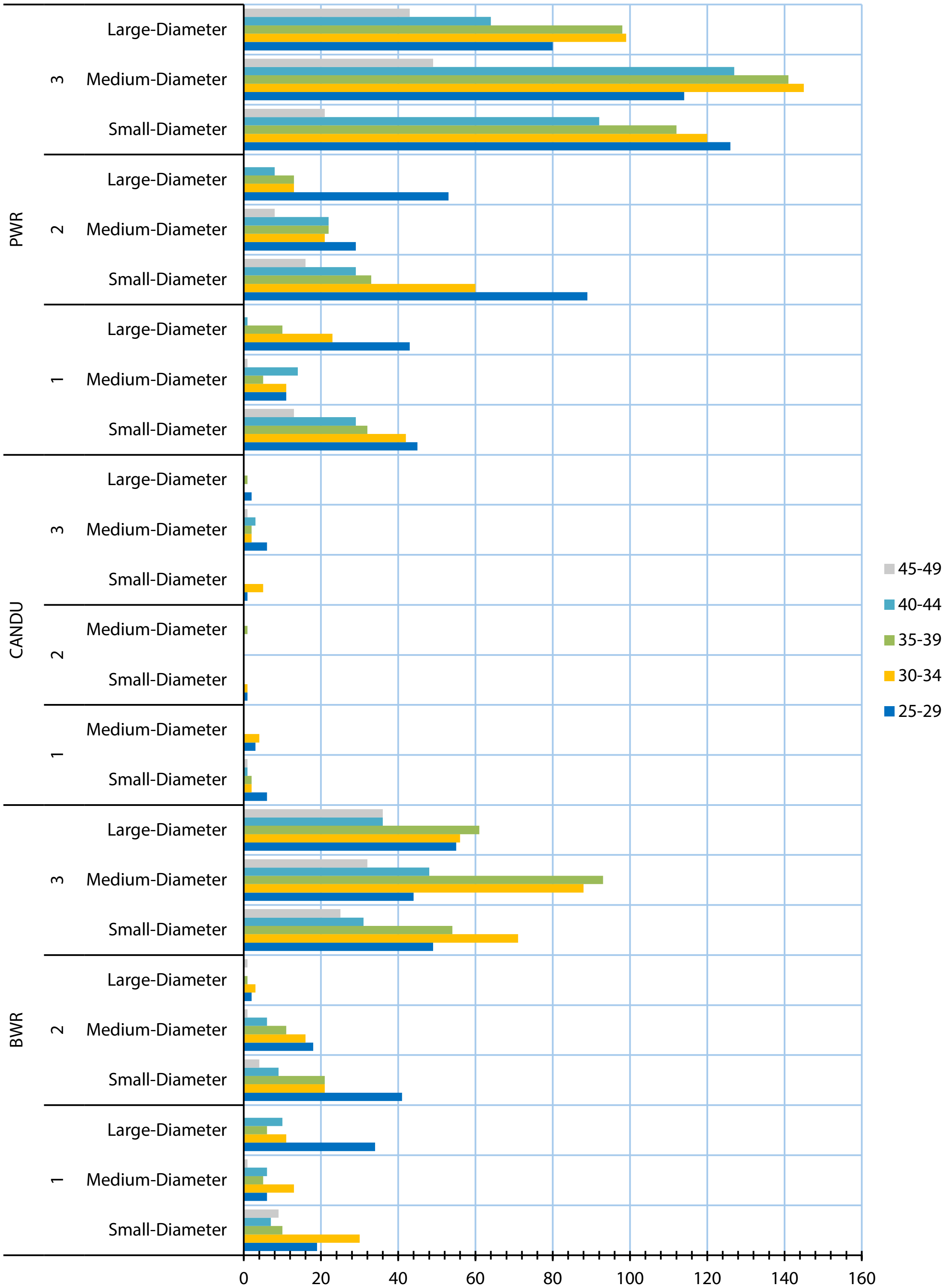


## 4.4 PEO/LTO-OPEX data analysis insights – safety-related piping

An ever-present data analysis issue is concerned with the completeness of the OPEX data, the statistical significance of an event population, and the applicability of the service experience across different plant design generations. Figures 4-6 through 4-14 summarise the safety-related piping OPEX. The following observations are made with respect to the ability of the CODAP event database to support material degradation issues:

- Figure 4-6: The OPEX is a summary of the piping OPEX for plants that have been in operation for 25 years or longer. The data is organised by plant type, safety class and nominal pipe size. For BWRs and PWRs the bulk of the OPEX is associated with safety class 3 piping systems, which mainly comprise reactor auxiliary systems (e.g. component cooling and spent fuel pool cooling) systems and support systems (e.g. instrument air and essential service water).
- Figure 4-7: BWR and PWR primary pressure boundary piping (safety class 1) OPEX. Two key observations are: First, there is a decline in medium- and large-diameter pipe degradation and failure of BWR-piping since the early-1990s, and this is well understood; Reference [21]. Second, small-diameter fatigue-induced pipe failures continue to cause operational impacts (e.g. plant shutdowns).
- Figure 4-8: Same event population as in Figure 4-7 but with the pre-1995 OPEX screened out. The OPEX is organised by time period in which a failure was observed and by material degradation mechanism. Noteworthy are the seemingly increasing trends in high-cycle fatigue (HCF) and thermal fatigue (TF) failures.
- Figure 4-9: Same event population (safety class 1) as in Figure 4-7. The event population has been normalised against the plant population that produced the failures. The uptick noted from plant age > 39 years is almost exclusively due to small-diameter (≤ DN50) pipe failures.
- Figure 4-10: Summary of the OPEX on safety class 2 piping located inside the containment.
- Figure 4-11: Summary of the safety-related support system piping OPEX. Support systems at each plant are unique. Extensive operating experience exists for the essential service water (ESW) system. In some plants important safety functions (e.g. safeguard equipment room coolers, emergency diesel generator coolers, component cooing water heat exchangers) are cooled directly by service water (i.e. raw water). Other plants have important safety functions served by a closed loop component cooling water system.
- Figure 4-12: Same data as in Figure 4-10, but normalised against the 1995-99 time period to show the temporal trends in the pipe failure data.
- Figure 4-13: Summarised in this chart is the high-cycle fatigue OPEX, which in this case encompasses all safety-related systems; safety classes 1 through 3. The event population has been normalised against the plant population that produced the failures.
- Figure 4-14: Summarised in this chart is pipe failure OPEX associated with emergency diesel generators (EDGs). The OPEX includes two classes of piping system boundaries: 1) diesel fuel oil supply piping; and 2) lubricating oil piping.

**Figure 4-6: The safety related piping OPEX by plant type, safety class and pipe-diameter vs. plant age at the time of pipe failure**




Note: Small-diameter: ∅ ≤ 50 mm; medium-diameter: 50 < ∅≤ 250 mm; large-diameter: ∅ > 250 mm.

**Figure 4-7: The BWR and PWR primary piping system OPEX by event date**

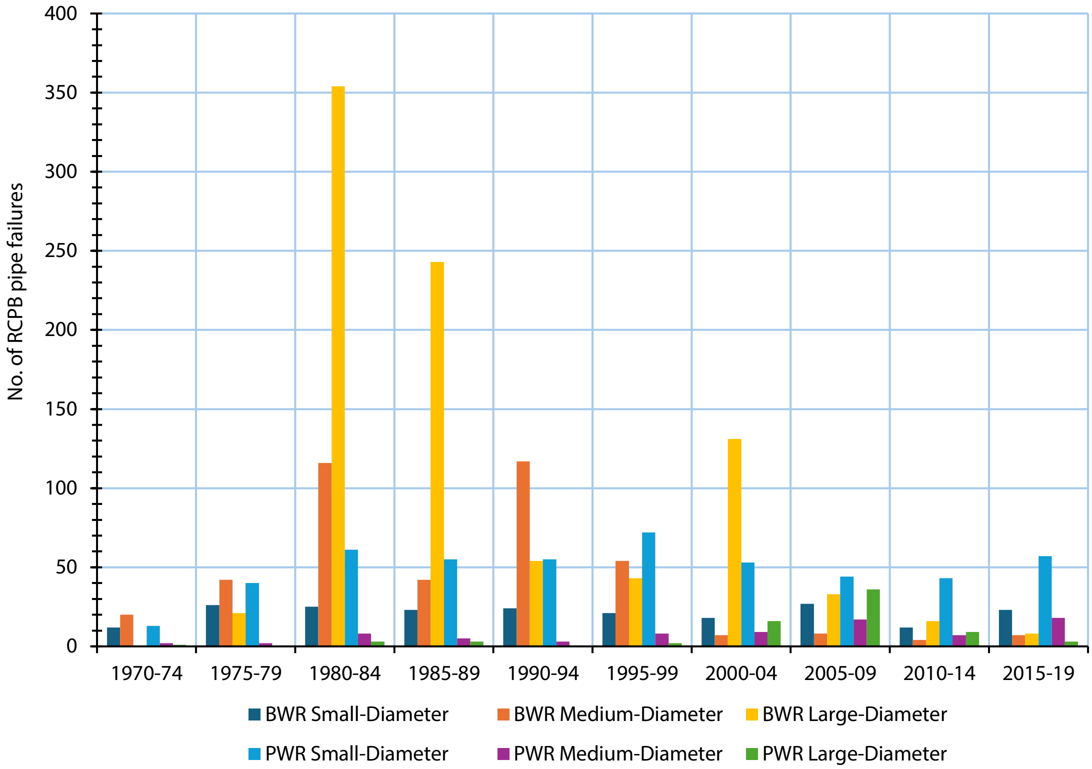



**Figure 4-8: RCPB piping OPEX by degradation mechanism and time period**

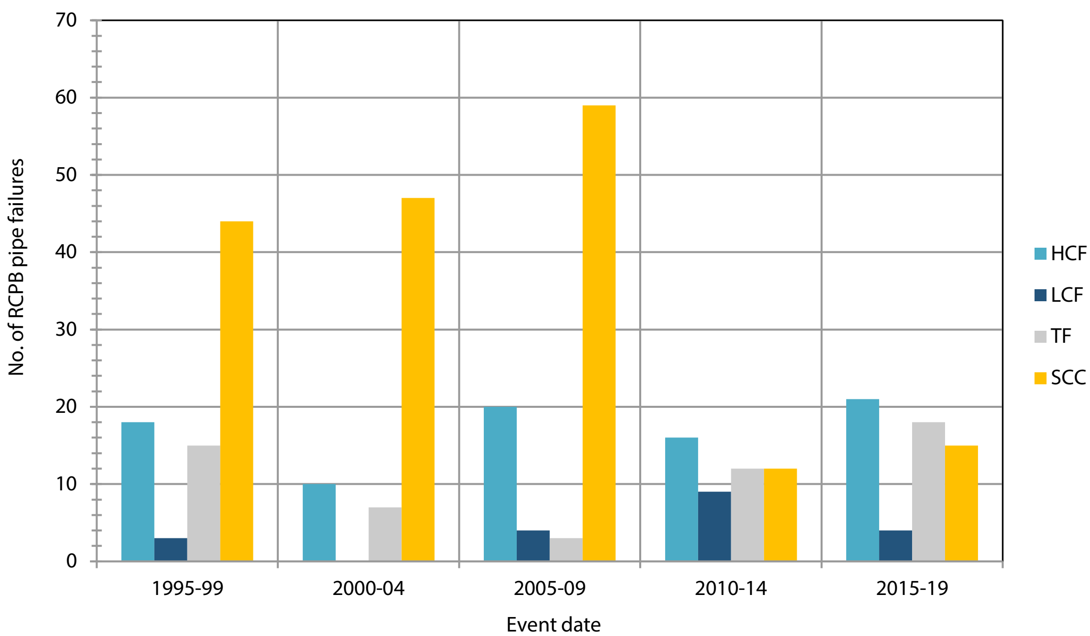


**Figure 4-9: Normalised RCPB piping OPEX**

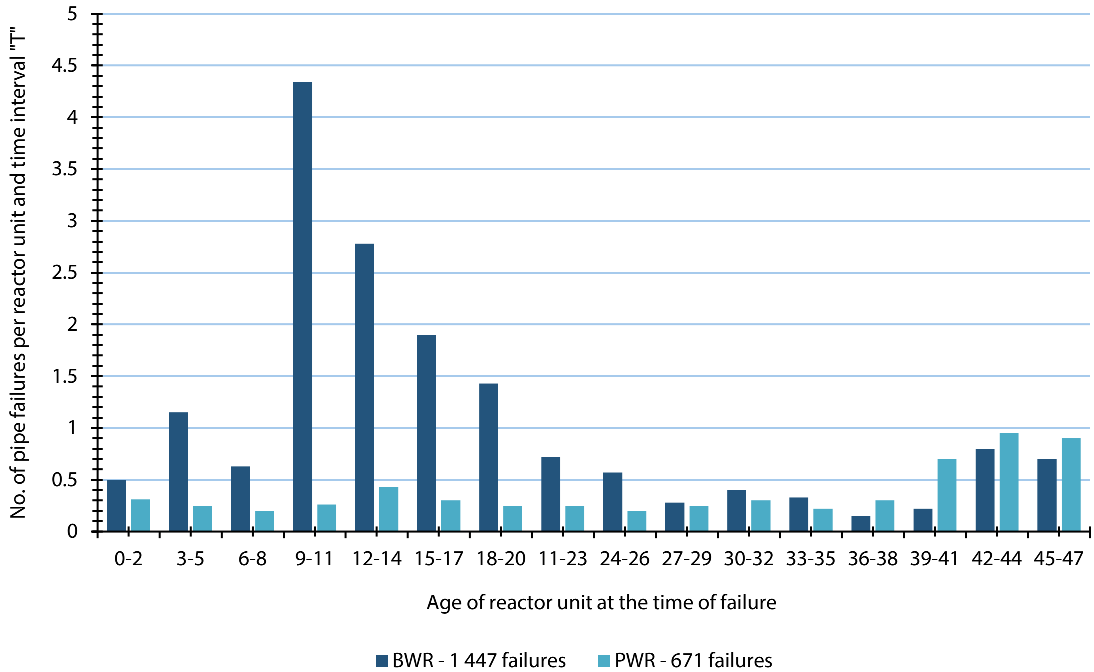
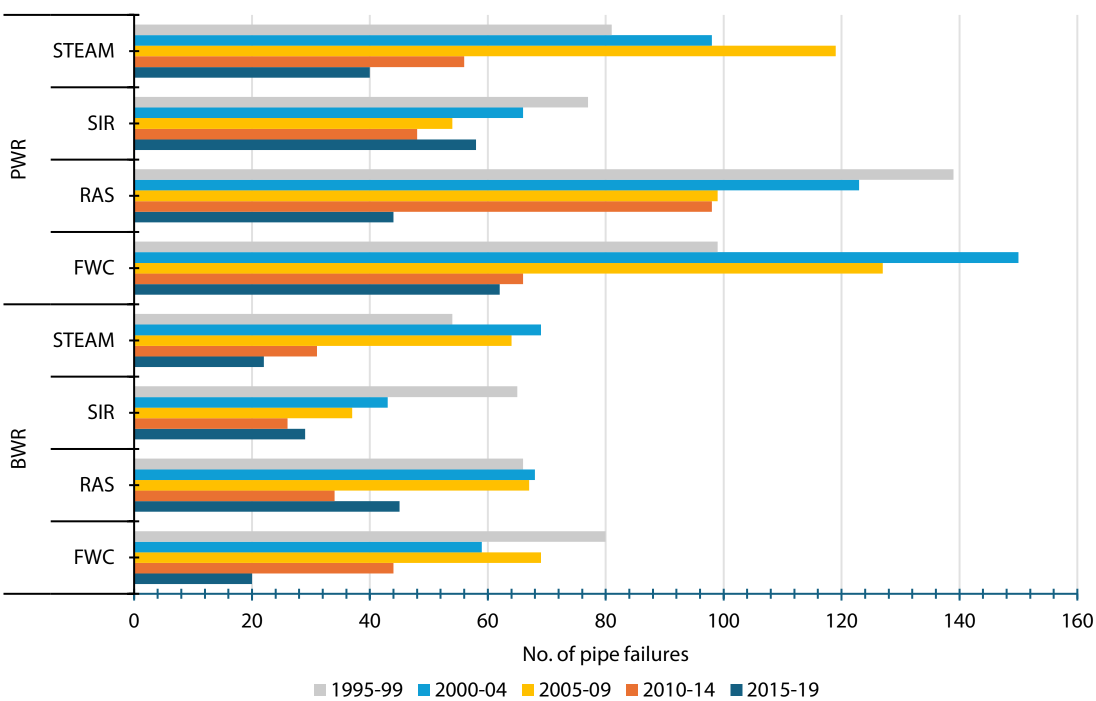
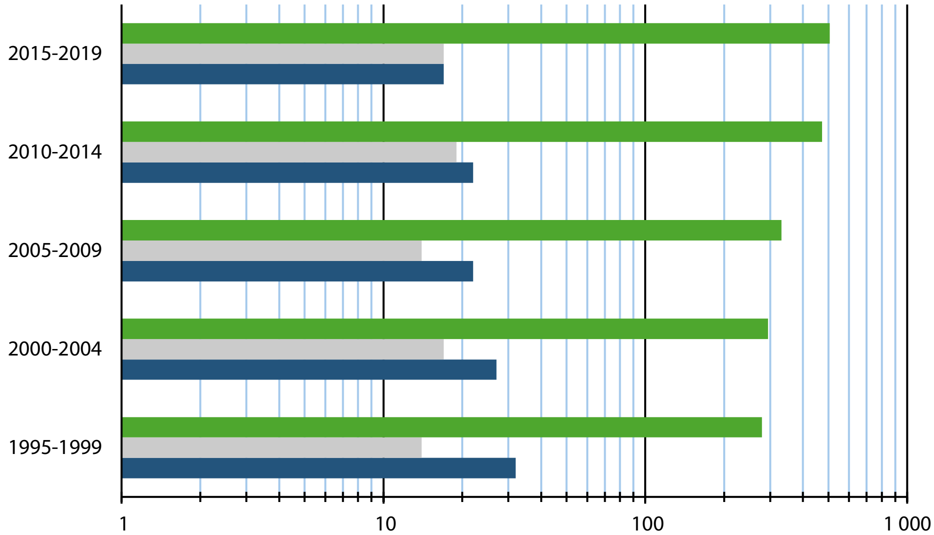



**Figure 4-10: Safety class 2 piping OPEX by system group and time period**




Note: FWC = feedwater system; RAS = reactor auxiliary systems (e.g. chemical and volume control, steam generator blowdown, reactor water cleanup); SIR = safety injection and recirculation; STEAM = main steam.

**Figure 4-11: Safety class 3 piping OPEX by system group and time period**




| | 1995-1999 | 2000-2004 | 2005-2009 | 2010-2014 | 2015-2019 |
|---|---|---|---|---|---|
| SW - Service water | 279 | 294 | 331 | 473 | 506 |
| IA - Instrument air | 14 | 17 | 14 | 19 | 17 |
| CC - Component cooling | 32 | 27 | 22 | 22 | 17 |

No. of pipe failures in indicated period

**Figure 4-12: Temporal trends in safety class 3 piping OPEX**

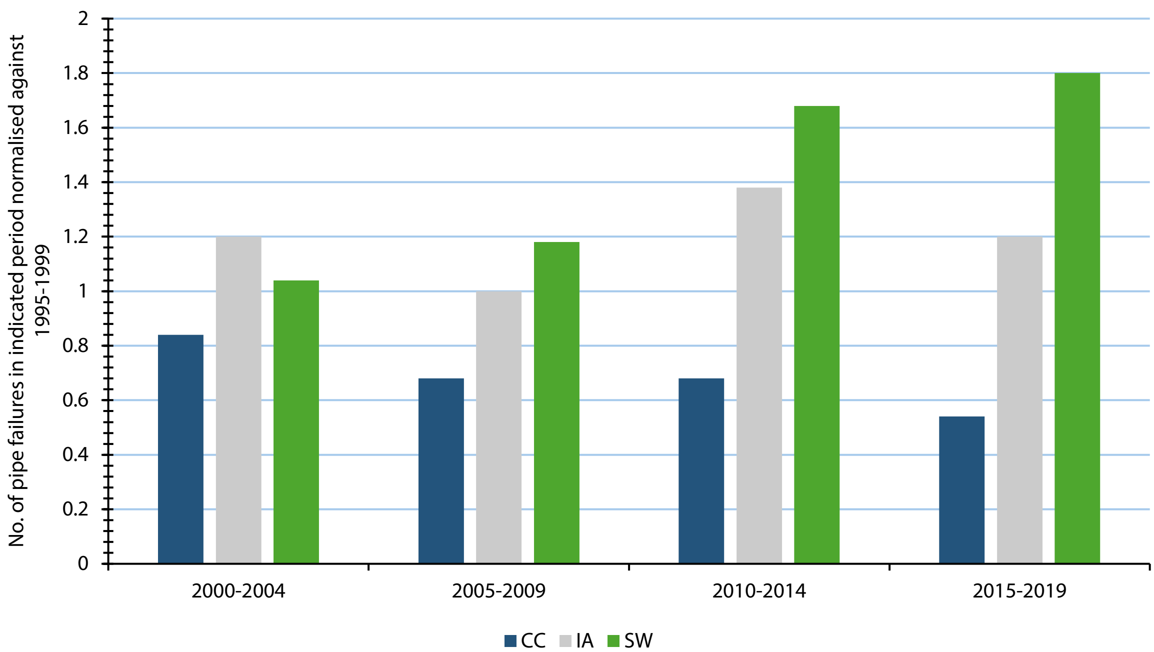


**Figure 4-13: OPEX on HCF in safety-related piping**

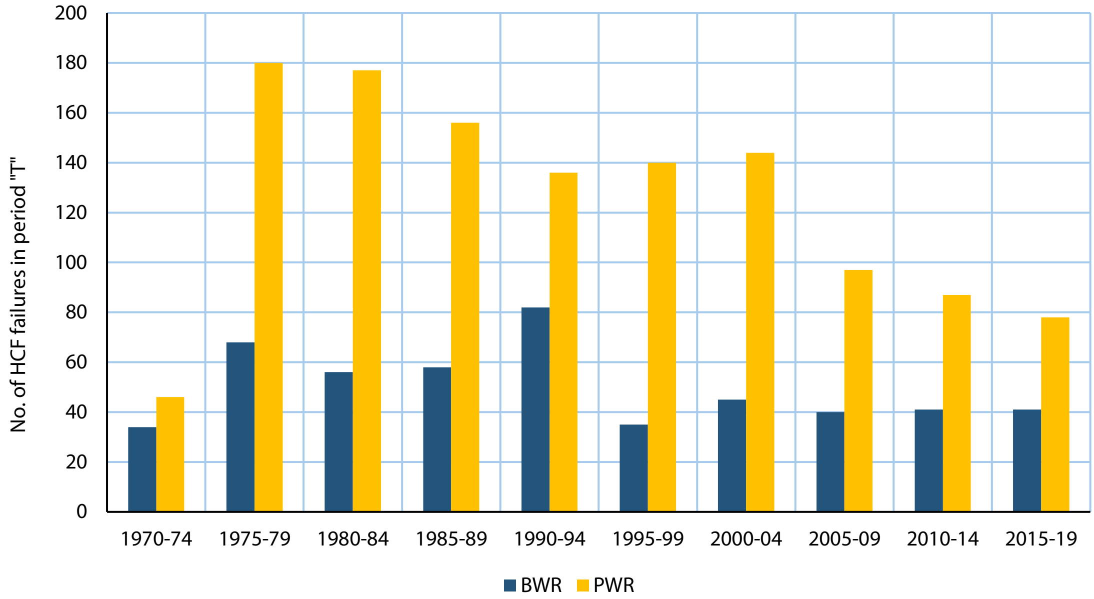
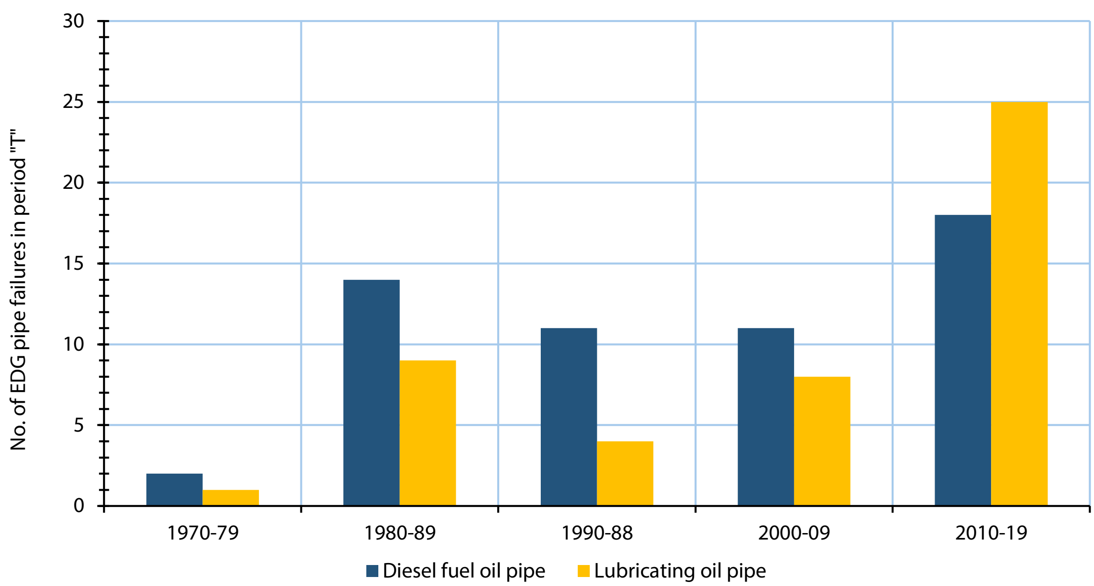


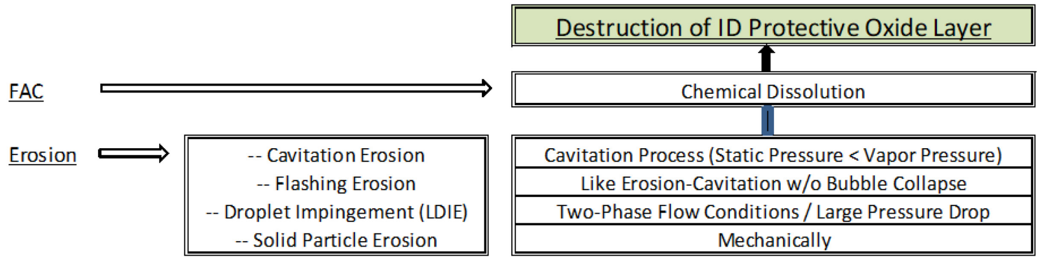

**Figure 4-14: Emergency diesel generator fuel oil and lubricating oil pipe failure OPEX**




## 4.5 PEO/LTO-OPEX data analysis insights – non-safety-related piping

This section addresses OPEX associated with the "balance-of-plant" (BOP) piping, which in this report includes the steam cycle piping systems within the turbine building. The second Topical Report [30] summarised the operating experience with flow-accelerated corrosion (FAC; Figure 4-15) in the CODAP member countries/economies. The Topical Report covered the OPEX for the period 1970 through 2012. Figure 4-16 summarises the FAC failure experience through the end of calendar year 2020.

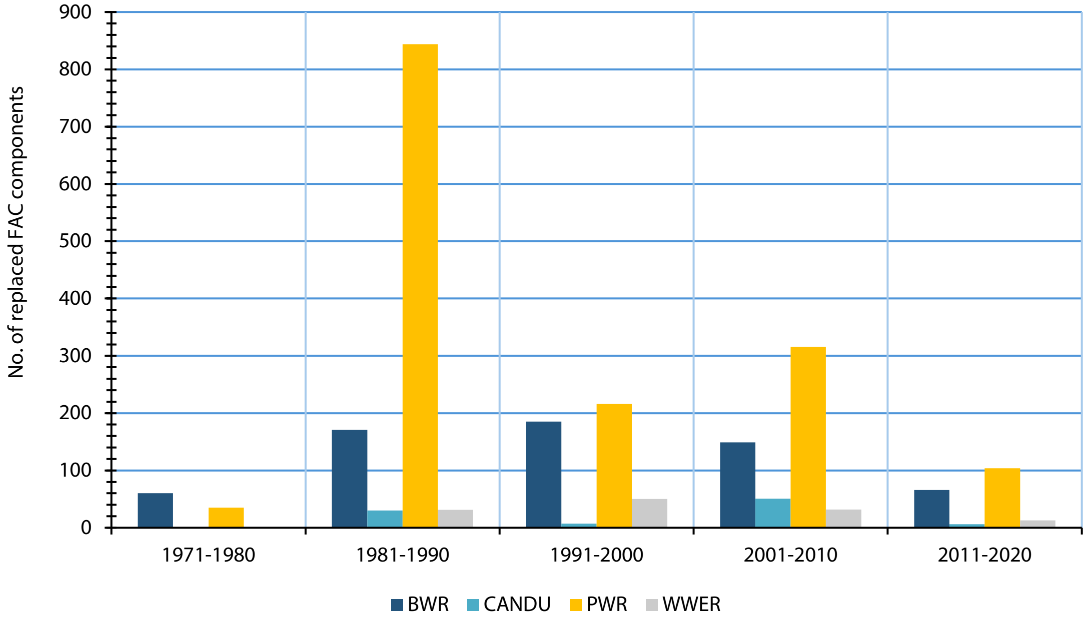
**Figure 4-15: Flow-assisted pipe wall thinning mechanisms**




All plant operators have implemented a long-term FAC monitoring programme to prevent piping failures in high energy (two-phase and single-phase) carbon steel piping systems. The programmes are developed by each utility using plant-specific conditions, industry-wide operating experience, engineering judgement, NDE techniques and computer analysis of high energy carbon steel systems.

The FAC operating experience data is recorded in different types of information systems such as condition reports, work orders, and databases of FAC monitoring programmes. Reportable occurrence reports or licensee event reports capture the major structural failures. In addition, information notices (or equivalent documents) are issued by regulators to inform licensees about generic issues.[3] Members of the CHECWORKS® Users Group (CHUG) exchange operating experience data biannually. CHUG does not operate a formal FAC OPEX database, however.

**Figure 4-16: FAC operating experience**

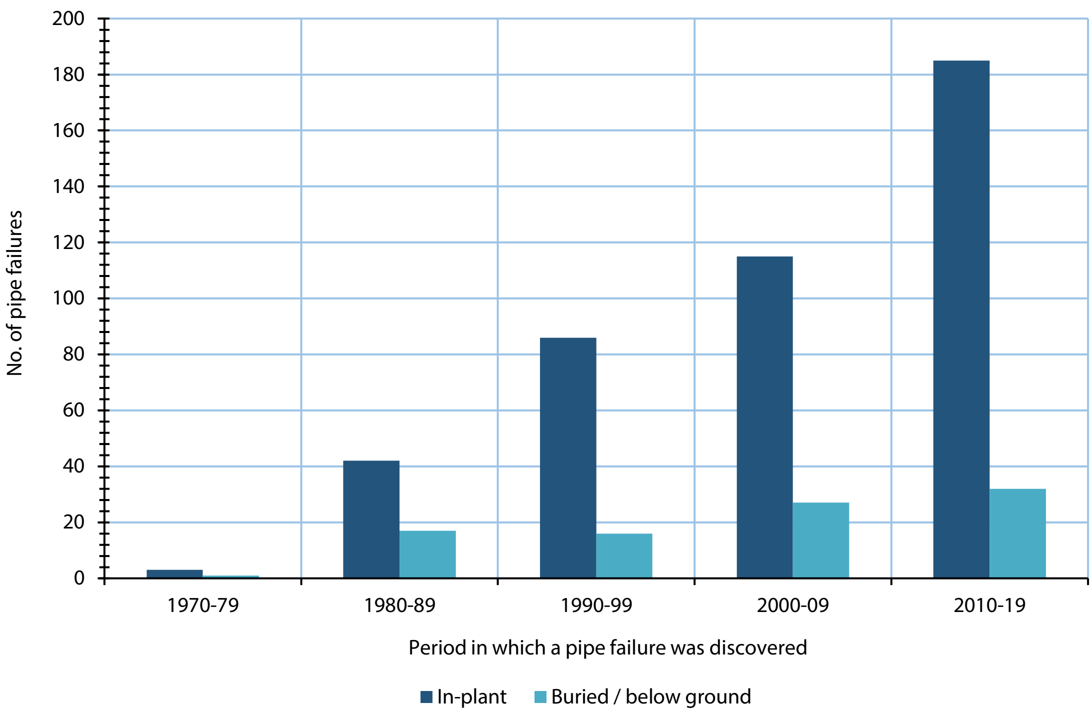



3. In 2019 the US Nuclear Regulatory Commission issued Information Notice 2019-08, Flow-Accelerated Corrosion Events, www.nrc.gov/docs/ML1906/ML19065A123.pdf.

## 4.6 PEO/LTO-OPEX data analysis insights – fire water system piping

The fire protection (FP) water system is important to plant safety. The assessment of the structural reliability of the piping is an integral aspect of internal flooding probabilistic safety assessments. Major leaks and breaks can cause internal flooding and spraying of, for example, vital electrical equipment. The source of fire water can be the ultimate heat sink (e.g. untreated raw water from a lake or river) or domestic water stored in a tank.[4] The fire water system piping OPEX is summarised in Figure 4-17 and it is limited to through-wall defects that have resulted in minor to significant pipe through-wall flow rates. CODAP does not include OPEX on blockages of FP piping due to build-up of corrosion products.

**Figure 4-17: Fire protection water system pipe failure experience**




Most plants have implemented "MIC monitoring programmes" which include the use of NDE techniques, chemical treatment of fire water, and periodic high-velocity flushing of certain pipe sections. In terms of length of piping and in-piping water volume, the typical fire water piping system is very extensive. A typical piping system consists of a large-diameter ring header (pipe which encircles the plant buildings), large-diameter in-plant distribution header, medium-diameter riser piping and small-diameter sprinkler head piping. Except for piping directly associated with the FP pump (including keep-fill pump) suction, discharge and recirculation lines, the predominant degradation mechanism acting on the FP piping is microbiologically influence corrosion (MIC), pitting and crevice corrosion and uniform corrosion. Pinhole leaks are relatively frequent occurrences. Unless there is a plant operational impact, there is no process for external reporting of such occurrences.

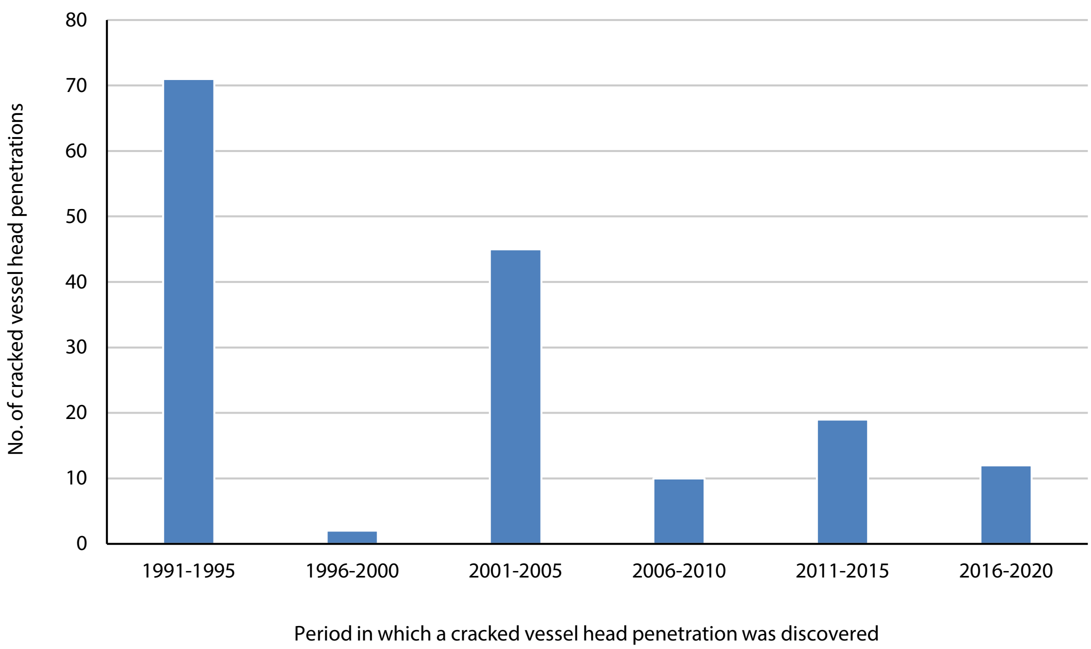
4. Domestic water (or potable water) is used for indoor and outdoor household purposes (cleaning, washing). It refers to water that is self-supplied (e.g. from a well) or water withdrawn directly from a municipal water distribution system.

Depending on the FP piping system design, water hammer may be a concern for the piping integrity. Some plants systems have experienced severe water hammer pressure pulses [107]. These severe water hammer events have occurred in so-called pre-activation systems where certain piping sections are normally kept empty and only filled upon a demand. Many plants have implemented design modifications to prevent water hammer in these systems. Examples of such design modifications include the installation of vacuum breakers at the top of critical piping system risers, and replacement of electric FP pump bypass valves with relief valves in accordance with the US National Fire Protection Association (NFPA) standard NFPA-25 [108].

## 4.7 PEO/LTO-OPEX data analysis insights – non-piping passive components

Except for PWR vessel head penetration (VHP) ageing degradation (Figure 4-18), in CODAP, the population of non-piping passive component failure events is limited to "selected representative events" (SREs)[5]. Based on the scarce data in CODAP it is therefore not possible to develop any generalised OPEX-based conclusions regarding the age-dependency of the various material degradation scenarios for reactor internals, or other types of passive components.

**Figure 4-18: PWR vessel head penetration OPEX**




5. In the CODAP project the term "SRE" has a specific meaning. The term is used to indicate that a degraded condition of specific and well characterised attributes also applies to multiple degraded locations (i.e. component boundaries) within a specific system. A single record for which all relevant database fields have been filled in also is representative of all the other degraded/flawed locations within that system. As an example, during in-service inspection 24 reactor vessel head penetrations were found to have recordable flaw indications. Following a complete flaw size characterisation using a qualified NDE technology and structural evaluation, 18 of the 24 vessel head penetrations were determined not to be fit for continued operation. Instead of creating 18 records, only a single SRE is created leaving it up to a future database user to expand the database content as needed.

There are two groups of SREs: 1) Extensive legacy OPEX available and dating from the 1970s but not generally accessible for detailed analysis, and 2) recent first-of-a-kind OPEX for which it is known that degradation progresses slowly for a specific combination of component, material and environment. An example of the former is baffle-former bolt (BFB) degradation. The population of damaged and replaced BFBs is in the many thousands, and it would therefore require a formidable effort to establish a comprehensive and (in a statistical sense) complete BFB OPEX database. In the current version of the database there are only a handful of records on BFB failures, including:

- a single SRE on alloy X-750 BFB failures discovered in the 1970s;
- a single SRE on Ti-stabilised austenitic stainless steel BFB failures in the early 2000s;
- a few SREs on unstabilised austenitic stainless steel BFB failures post-2005.

An example of the latter SRE-type would be material flaws discovered in the outside-diameter PWR core barrel (refer to Section 2.5 of this report). The current (mid-2021 version of the CODAP event database) includes a single first-of-a-kind event. This material ageing degradation discovery was made after 33.7 EFPYs of operation. To ensure a more rigorous "treatment" of reactor internals in CODAP, this type of event could potentially be a benchmark for the expanded consideration of reactor internals in the CODAP project.

# Chapter 5. **Conclusions and recommendations**

The objective of this report was to document observations from evaluations of the operating experience with passive metallic components in commercial nuclear power plants during periods of extended operation (PEO) and long-term operation (LTO). Specifically, this report addresses trends and patterns in material degradation as a function of plant age The data collection period for this report ended in 2020, and only the validated data was used for this report. It is a topic that for a very long time has received considerable attention from the point of view of reactor regulations with respect to ageing management programmes, codes, and standards with respect to reliability and integrity management, and the many supporting research programmes. In preparing this report, simple visual examinations of graphical plots were made to reach preliminary insights about the different material degradation mechanisms as a function of plant age. Summarised below are the conclusions and recommendations of this study.

## 5.1 Main conclusions

A five-step approach was used to reach conclusions about age-dependent material degradation mechanisms during PEO/LTO. The CODAP event database content was compared and contrasted against the results of two expert panels on material degradation [17][27], a questionnaire on long-term operation by the Working Group on Integrity and Ageing of Components and Structures (WGIAGE) [32], and the 2019 "International Workshop on Age-Related Degradation of Reactor Vessels and Internals" [39]. Below are the conclusions on how the technical scope and content of the CODAP event database addresses age-dependent passive component material degradation.

### *WGIAGE vs. CODAP*

Several respondents to the NEA WGIAGE questionnaire assigned a "high priority" to the following "technical areas of mutual interest related to ageing-dependent degradation of materials during long-term operation of nuclear power plants":

- irradiation embrittlement of reactor pressure vessel materials;
- irradiation embrittlement of reactor internals.

The CODAP project collects information on material degradation of reactor internals. So far, the data exchange has focused on selected representative events. The project does recognise the importance of irradiation assisted, age-dependent degradation of reactor internals. The complexity in accessing the relevant OPEX is also recognised. Invariably the reactor internals' non-conforming conditions are discovered during in-service inspections and the resulting findings and material flaw evaluations are generally not readily accessible for third party review; see recommendations below.

Several of the respondents to the WGIAGE questionnaire also assigned a "high priority" to "pitting and crevice corrosion near deposits, metallic contact points and stagnant conditions, including internal/external corrosion of below ground piping." This class of material degradation impacts the pressure boundary integrity of raw water-cooling systems, e.g. ESW and FP water systems. Both system groups are important for safe plant operation. Although CODAP includes a sizeable population of OPEX for these system groups, the technical content is skewed in that it mainly consists of US OPEX, thus reflecting the associated national routines and requirements for reporting and evaluating corrosion failures. This could potentially impede the ability to reach generic conclusions about the importance of ESW and FP age-dependent degradation that are applicable across all CODAP member countries/economies and plant types. It could be that the US OPEX is an "outlier".

Two of the respondents assigned high-to-medium priority to flow-accelerated corrosion (FAC) in high-energy piping. From a CODAP-perspective it is unclear why this is so; see recommendations below.

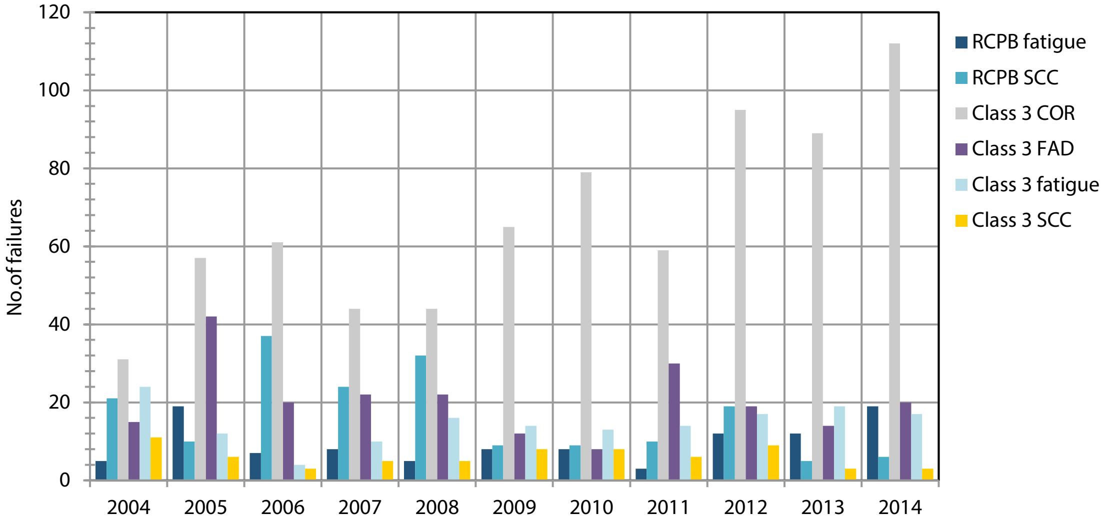
### *PMDA/EMDA/WGIAGE vs. CODAP*

The US NRC Office of Nuclear Regulatory Research has convened two expert panels on age-dependent material degradation. The proactive material degradation assessment (PMDA; 2004 to 2006) [17] identified the different material degradation scenarios that could affect plant systems for up to 40 years of operation. The expanded material degradation assessment (EMDA; 2012 to 2014) [27] extended the analysis time frame from 40 years to 80 years. In comparing the PMDA/EMDA and WGIAGE questionnaire results with CODAP, four questions were raised and the associated observations are listed below:

1. What was the passive component operating experience during the 2004-to-2014 time period; the start date of the PMDA expert panel and the conclusion of the EMDA expert panel?
   - A small sample of the OPEX for the given period is summarised in Figure 5-1. It shows a small but noticeable uptick in fatigue failures and an increase in corrosion failures.
2. Are the apparent and underlying causes of material degradation during PEO/LTO similar to those experienced during early plant life? If not, what are the main differences?
   - Low-cycle fatigue, high-cycle mechanical fatigue and thermal fatigue continue to cause operational challenges. Hence, there is no fundamental difference between early life and PEO/LTO. An increasing failure trend is observed for safety class 3 raw water-cooling water piping [1]. In contrast, flow-accelerated corrosion (FAC) monitoring programmes and stress corrosion cracking (SCC) mitigation processes appear to have a positive impact on the longer-term material performance. CODAP has published two reports on these topics [30][21]. New OPEX should determine the need for producing Topical Report updates.
   - WWER reactors are in operation in three of the current CODAP member countries/ economies (Czechia, Finland and the Slovak Republic). In the WGIAGE questionnaire, FAC acting on high-energy piping was assigned a "high" priority by one country. CODAP includes only very limited OPEX on FAC in WWER plants, however.[2]
3. Is it feasible to validate the WGIAGE questionnaire results [32] against the CODAP event database? In other words, to what extent are the material degradation issues identified in the Questionnaire Results also acknowledged and tracked by the CODAP project?
   - It is unclear whether the respondents to the WGIAGE questionnaire were aware of or had access to the CODAP event database. Nevertheless, the purpose of CODAP is to collect field experience data on degradation mechanisms that have produced failures. CODAP does not collect data on minor RPV fabrication flaws.
4. Based on the operating experience evaluation results that are documented in this Topical Report, are there any gaps in the lists of degradation scenarios that were identified by the PMDA and EMDA expert panels or the WGIAGE Questionnaire?
   - No gaps identified.

1. Also referred to as essential service water (ESW) or nuclear service water (NSW).
2. A possible explanation relates to the extensive use of field segmentation and welding of pipe elbows for installation in WWER secondary-side piping systems. Segmented pipe elbows are more prone to FAC-induced wall thinning than is the case for the more common prefabricated cold- or hot-formed elbows with smooth interior pipe wall surface.

**Figure 5-1: CODAP OPEX for the PMDA/EMDA period (2004-2014)**[3]




### *The "Workshop" vs. CODAP*

CODAP is concerned with degradation mechanisms that have produced failures. The intent of the "Workshop" was to exchange information on ongoing research on age-dependent degradation of reactor vessels and internals. The CODAP project exchanges reactor internals OPEX on selected representative events.

## 5.2 CODAP activities for the fourth term (2021-2023) and beyond

CODAP activities for the fourth term fall into three categories: 1) improvements to data exchange protocols; 2) validation of the data analysis observations; and 3) benchmarking and research topics.

### *Data exchange protocol*

Since the 1960s the exchange and analysis of nuclear power plant operating experience has been a point of focus for nuclear regulatory authorities, technical support organisations and the nuclear industry at large. The processes and systems for collecting and disseminating operating experience data continue to evolve. To paraphrase Y. Benoist[4], OPEX analysis and feedback is an ongoing activity that is technically challenging and workload intensive. For the year 2021 and beyond, the CODAP Management Board (MB) is considering the following activities to potentially enhance the data collection process:

- Identifying the current national routines for recording and submitting material degradation information, including OPEX access, especially for those "issues" that do not cause a direct impact (e.g. reactor trip/turbine trip) on routine plant operation. Information on events that result in reactor trip is recorded in readily available and

3. The PMDA Expert Panel was convened during 2004 to 2006 and the EMPDA Expert Panel was convened during 2012 to 2014. COR = corrosion mechanisms (various), FAD = flow-assisted degradation mechanisms (various), SCC = stress corrosion cracking.
4. From a presentation made at the 6th EUROSAFE Forum, 8-9 November 2004. The theme of the conference was "Learning from Experience – A Cornerstone of Nuclear Safety." More information is available at www.eurosafe-forum.org/eurosafe2004.

accessible licensee event reports. Annex B summarises examples of how material degradation OPEX is documented in the United States.

- Reporting on the current national regulatory and industry routines/practices and requirements for performing operability determinations or fitness for service assessments. In particular, the question of how these evaluations provide relevant OPEX data for potential submission to CODAP should be elaborated.
- Developing an enhanced list of BWR, PWR and WWER reactor internal piece parts and considering the development of a plan to conduct the future reactor internals OPEX exchange.

The national reporting routines and requirements for documenting passive component degradation are extensive and multifaceted (see for example Annex B). Subject matter expertise (SME) is needed to extract and evaluate the extensive national material degradation and failure information. While data access limitations may exist, the time needed to perform data mining and evaluations and the language barriers (i.e. translating non-English highly technical documents into English) could be the critical path inhibitors from the point of view of populating the CODAP event database with new information.

### *Validating the observations of this Topical Report*

The report investigated age-dependent material degradation of piping systems and reactor internals. It is not clear if the identified trends and patterns are representative across all CODAP member countries/economies. One way of validating the conclusions of this study is to:

- Select a single “system-material-environment” grouping for a limited and recent time period (e.g. 2015-2019 ) to determine:
  - The applicable country/economy-specific ageing management programme requirements, regulations and operability determination procedures;
  - The country/economy-specific OPEX;
  - The country/economy-specific regulatory inspection programmes and findings from site inspections.
- Evaluate the CANDU- and WWER-specific FAC OPEX to determine if and how it differs from the corresponding PWR-specific OPEX.
- The CODAP event database contains a significant population of FP and SW pipe failures in US plants. Possible technical issues to address include:
  - Determining if this “bias” is attributed to unique US piping system designs, material selections, operating environments, regulations, reporting, and/or material ageing management programmes, or whether it is more a reflection of the data exchange process.
  - For plants that remain in extended periods of layup, addressing how the integrity of piping systems is verified prior to a possible plant restart.

### *Benchmarking and research topics*

The evaluation of possible age-dependent material degradation was done in a high-level manner using simple “visual tests” of graphical plots. It is a first step in a more rigorous analysis of the OPEX. The CODAP Management Board will consider continuing the work done in producing this Topical Report as a formalised benchmark involving multiple TSOs under an “umbrella project”, e.g., by developing a CSNI Activity Proposal Sheet (CAPS) entry that involves CODAP/WGIAGE/WGRISK oversight and guidance. This could set the stage for widened CODAP database access and usage and could encourage and engage early career material scientists, fracture mechanics specialists and PSA engineers in the study of passive component integrity.

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# Annex A
# Glossary of technical terms

*Annealing (or recovery annealing)*. Annealing is a heat treatment process used mostly to increase the ductility and reduce the hardness of a material. This change in hardness and ductility is a result of the reduction of dislocations in the crystal structure of the material being annealed. Annealing is often performed after a material has undergone a hardening or cold working process to prevent it from brittle failure or to make it more formable for subsequent operations.

*ASME Section XI, Division 2*. Issued in July 2019, this Division provides the requirements for the creation of the reliability and integrity management (RIM) programme for advanced nuclear reactor designs. The RIM programme addresses the entire life cycle for all types of nuclear power plants. It requires a combination of monitoring, examination, tests, operation and maintenance requirements that ensures each structure, system and component (SSC) meets plant risk and reliability goals that are selected for the RIM programme.

*Backfitting*. According to NUREG-1409 Revision 1 (2010), backfitting occurs when the NRC imposes new or changed regulatory requirements or staff interpretations of the regulations or requirements on nuclear power reactor licensees, certain nuclear power reactor applicants, or select nuclear materials licensees. Backfitting is an integral part of the regulatory process and may be needed when the staff addresses safety or security issues. The NRC may take a backfitting action only after conducting a formal, systematic review to ensure that the action is defined and justified. This process ensures discipline, predictability and optimal use of NRC and licensee resources. The backfitting requirements are found in 10 CFR 50.109, "Backfitting," 70.76, "Backfitting," 72.62, "Backfitting," and 76.76, "Backfitting." Provisions analogous to the backfitting requirements, referred to as issue finality provisions, are set forth in 10 CFR Part 52, "Licences, Certifications, and Approvals for Nuclear Power Plants."

*Below ground piping*. Buried piping in contact with soil or concrete and underground piping that is below grade but is contained within a tunnel or vault such that it is in contact with air and is located where access for inspection is restricted.

*Boat sampling technique (BST)*. A technique developed for obtaining samples from the surface of any operating component. The technique is a non-destructive surface sampling technique as it does not cause any plastic deformation or thermal degradation of the operating component. BST can be used, remotely and in water-submerged condition, with the help of a handling mechanism. The samples are boat-shaped, having three mm maximum thickness and require about three hours to get scooped from a location. The samples are used for metallurgical analysis to confirm the integrity of the operating component. BST incorporates mainly sampling module, handling mechanism and electric and pneumatic sub-systems.

*BONNA® pipe*. A thin steel pipe embedded in reinforced concrete. It has rebar or a heavy wire mesh embedded in the OD concrete.

*Buried piping*. Piping that is below grade and in direct contact with soil. Buried piping is provided with corrosion protection such as coating and cathodic protection.

*BWR core shroud access hole cover (AHC)*. Jet pump BWRs are designed with access holes in the shroud support plate, which is located at the bottom of the annulus between the core shroud and the reactor vessel wall. Each reactor vessel has two such holes that are located 180 degrees apart. These holes are used for access during construction and are subsequently closed by welding a plate over the hole. The covers and shroud support ledge are Inconel Alloy 600 material. The connecting weld material is also Inconel 600 (Alloy 182 or 82). In the worst case, access hole cover cracking could progress through wall and cause the cover to detach partially or completely. A substantial flow path from the bottom head into the annulus region would be created, impacting core flow distribution during normal operation. The distribution would be detectable at significant levels. Such cracking would impact the boundary which assures 2/3 core coverage following a LOCA event. The consequence of cracking is high.

*BWRVIP reactor internals AMP*. A set of ageing management guidance provided by the EPRI BWRVIP issue programme for managing age-related degradation in BWR reactor internals. The set of applicable BWRVIP guidance includes all NEI 03-08 "Mandatory" and "Needed" guidance elements applicable to BWR reactor internals except for repair-based guidance. It also includes administrative controls and procedures ensuring that guidance is revised as needed to address new OE and other relevant data. Annex A of BWRVIP-315NP (2019) provides a list of the applicable BWRVIP guidance documents associated with the BWRVIP reactor internals AMP.

*Calandria*. The "calandria" is the cylindrical reactor vessel that contains the heavy water moderator. It is penetrated from end-to-end by hundreds of calandria tubes, which accommodate the pressure tubes containing the fuel and coolant.

*Carbon fibre reinforced polymer (CFRP) repair*. A large-diameter carbon steel pipe repair/ rehabilitation technique for degraded (e.g. thinned) large-diameter carbon steel pipe. Specially designed layers of carbon fibres and glass fibres are bonded to the internal surface of the piping.

*Cathodic protection (CP)*. A corrosion protection technique in which the potential difference is applied to buried piping from an external power source or a more anodic material (sacrificial anode) for the purpose of making the piping behave in a cathodic manner. Using CP, the corrosion rate is normally reduced to an acceptable level.

*Cavitation*. Cavitation damage may occur when there is a flowing liquid stream that experiences a drop in pressure followed by a pressure recovery. Such a pressure drops (i.e. the difference between the upstream pressure and the downstream pressure) can occur in valve internals where the flow has to accelerate through a small area. As the fluid moves through the restricted area, the fluid velocity increases, and the pressure decreases as shown by the momentum equation (i.e. Bernoulli's theorem). If the local pressure passes below the vapour pressure at the liquid temperature, small bubbles are formed. When the downstream pressure rises above the vapour pressure, these bubbles collapse. The collapse of the bubbles causes high local pressures and very high local water jet velocities. If the collapsing bubbles are close enough to a solid surface, damage to that surface will occur. The collapse of the numerous bubbles generates noise and vibration. Most often, cavitation causes most of its damage by vibration (e.g. cracked welds, broken instrument lines, loosened flanges). The erosion caused by cavitation also generates particles that contaminate the process fluid.

*CFRP system*: A buried piping rehabilitation and repair technique. It consists of high strength carbon fibre fabrics and/or glass fibre fabrics, fully saturated in a two-part 100% solids epoxy matrix. These laminates are bonded both longitudinally and circumferentially to the interior surface of the pipe forming a structural lining within the pipe. This lining can be designed to replace the degraded portions of the existing system without reliance on the degraded piping for the life of the repair, except at the terminal ends of the repair.

*Charpy impact test*. Also known as the Charpy V-Notch test, it is a standardised high strain-rate test which determines the amount of energy absorbed by a material during fracture.

*Concrete encased piping (CEP)*. Below ground piping that is embedded in concrete. The piping is not easily extracted nor is the interior pipe surface readily accessible for inspection. The CEP category also includes piping recessed in plant building floors.

*Crevice corrosion*. Crevice corrosion occurs in a wetted or buried environment when a crevice or area of stagnant or low flow exists that allows a corrosive environment to develop in a component. It occurs most frequently in joints and connections, or points of contact between metals and nonmetals, such as gasket surfaces, lap joints and under bolt heads. Carbon steel, cast iron, low alloy steels, stainless steel, copper, and nickel base alloys are all susceptible to crevice corrosion. Steel can be subject to crevice corrosion in some cases after lining/cladding degradation.

*Cumulative usage factor (CUF)*. The design basis for Class 1 nuclear power plant components subjected to cyclic service is in ASME Section III. Transient events that the plant might credibly experience are evaluated to establish a design basis for plant equipment. The fatigue analyses rely on the definition of design basis transients that envelope the expected cyclic service and the calculation of a cumulative usage factor (CUF). The CUF is the ratio of the number of cycles experienced by a structure or component divided by the number of allowable cycles of that structure or component. In accordance with ASME Section 111, Subsection NB, the cumulative usage factor must be less than 1.0.

*Cured-in-place-pipe (CIPP)*. A BP temporary repair method. A resin-saturated felt tube made of polyester, fibreglass cloth or a number of other materials suitable for resin impregnation, is inverted or pulled into a damaged pipe. It is usually done from the upstream access point (manhole or excavation).

*De-alloying (selective leaching)*. As defined by NACE, "de-alloying" or "selective leaching" refers to the selective removal of one element from an alloy by corrosion processes. A common example is the dezincification of unstabilised brass, whereby a weakened, porous copper structure is produced. The selective removal of zinc can proceed in a uniform manner or on a localised (plug-type) scale. It is difficult to rationalise dezincification in terms of preferential Zn dissolution out of the brass lattice structure. Rather, it is believed that brass dissolves with Zn remaining in solution and Cu replating out of the solution. Graphitic corrosion of grey cast iron, whereby a brittle graphite skeleton remains following preferential iron dissolution, is a further example of selective leaching. During cast iron graphitic corrosion, the porous graphite network that makes up 4-5% of the total mass of the alloy is impregnated with insoluble corrosion products. As a result, the cast iron retains its appearance and shape but is weaker structurally. Testing and identification of graphitic corrosion is accomplished by scraping through the surface with a knife to reveal the crumbling of the iron beneath.

*Delayed hydride cracking (DHC)*. Delayed hydride cracking is a subcritical crack growth mechanism occurring in zirconium alloys as well as other hydride-forming materials that requires the formation of brittle hydride phases at the tip of a crack and subsequent failure of that hydride resulting in crack extension. Hydrogen in solution in the zirconium alloy is transported to the crack tip by diffusion processes where it precipitates as a hydride phase. When the precipitate attains a critical condition, related to its size and the applied stress intensity factor, $K_I$, fracture ensues, and the crack extends through the brittle hydride and arrests in the matrix. Each step of crack propagation results in crack extension by a distance approximately the length of the hydride.

*Double-walled pipe*. A double-walled pipe is a secondary contained piping system. It is a pipe-within-a-pipe, or encased in an outer covering, with an annulus (interstitial space) between the two diameters. The inner pipe is the primary or carrier pipe and the outer pipe is called the secondary or containment pipe.

*Enhanced visual examination (EVT-1)*. The EVT-1 method is intended for the visual examination of surface breaking flaws. Any visual inspection for cracking requires a reasonable expectation that the flaw length and crack mouth opening displacement meet the resolution requirements of the observation technique. The EVT-1 specification augments the VT-i requirements to provide more rigorous inspection standards for stress corrosion cracking. EVT-1 is also conducted in accordance with the requirements described for visual examination (i.e. VT-1) with additional requirements (such as camera scanning speed). Any recommendation for EVT-1 inspection will require additional analysis to establish flaw-tolerance criteria, which must consider potential embrittlement due to thermal ageing or neutron irradiation. Acceptance criteria methodologies to support plant-specific augmented examinations are documented in WCAP-17096-NP[1].

1. Westinghouse Electric Company LLC (2009), *Reactor Internals Acceptance Criteria Methodology and Data Requirements*, WCAP-17096-NP, Cranberry Township, PA.

*Erosion cavitation (E-C)*. This phenomenon occurs downstream of a directional change or in the presence of an eddy. Evidence can be seen by round pits in the base metal and is often wrongly diagnosed as FAC (see below). Like erosion, E-C involves fluids accelerating over the surface of a material; however, unlike erosion, the actual fluid is not doing the damage. Rather, cavitation results from small bubbles in a liquid striking a surface. Such bubbles form when the pressure of a fluid drops below the vapour pressure, the pressure at which a liquid becomes a gas. When these bubbles strike the surface, they collapse or implode. Although a single bubble imploding does not carry much force, over time, the small damage caused by each bubble accumulates. The repeated impact of these implosions results in the formation of pits. Also, like erosion, the presence of chemical corrosion enhances the damage and rate of material removal. E-C has been observed in PWR stainless steel decay heat removal and charging system piping.

*Erosion/corrosion (E/C)*. "Erosion" is the destruction of metals by the abrasive action of moving fluids, usually accelerated by the presence of solid particles or matter in suspension. When corrosion occurs simultaneously, the term "erosion-corrosion" is used. In the CODAP event database, the term "erosion/corrosion" applies only to moderate energy carbon steel piping (e.g. raw water piping).

*Equivalent break size (EBS)*. The calculated size of a hole in a pipe given a certain through-wall flow rate and for a given pressure.

*Fatigue usage factor*. The ratio of the number of cycles anticipated during the lifetime of a component to the allowable cycles.

*Flashing*. Flashing occurs when a high-pressure liquid flows through a valve or an orifice to a region of greatly reduced pressure. If the pressure drops below the vapour pressure, some of the liquid will be spontaneously converted to steam. The downstream velocity will be greatly increased due to a much lower average density of the two-phase mixture. The impact of the high velocity liquid on piping or components creates flashing damage.

*Flow accelerated (or assisted) corrosion (FAC)*. FAC is "a process whereby the normally protective oxide layer on carbon or low-alloy steel dissolves into a stream of flowing water or water-steam mixture."[2] It can occur in both single phase and two-phase regions. The cause of FAC is a specific set of water chemistry conditions (e.g. pH, level of dissolved oxygen), and there is no mechanical contribution to the dissolution of the normally protective iron oxide (magnetite) layer on the inside pipe wall.

*Full structural weld overlay (FSWOL)*. A structural reinforcement and stress corrosion cracking (SCC) mitigation technique through application of a SCC-resistant material layer around the entire circumference of the treated weldment. The minimum acceptable FSWOL thickness is 1/3 the original pipe wall thickness. The minimum length is $0.75\sqrt{(R \times t)}$ on either side of the dissimilar metal weld to be treated, where R is the outer radius of the item and t is the nominal thickness of the item.

*Fusion (or heat fusion)*. In the context of HDPE piping design and installation, fusion techniques are used to join pipes together. It is a welding process used to join two different pieces of a thermoplastic. This process involves heating both pieces simultaneously and pressing them together. The two pieces then cool together and form a permanent bond.

*Grayloc region*. An inspection area at or near the mechanical connection between a PHTS feeder tube and a fuel channel of the CANDU primary system. Grayloc® is a trade name of a type of mechanical connector that provides metal-to-metal seals in piping systems.

*Harvesting (of irradiated materials)*. Accessing and segmentation of irradiated metallic materials from commercial nuclear reactors that have ceased operations. The harvesting is done according to detailed research plans. Detailed information on the research planning and its

2. Dooley, R.B and V.K Chexal (2000), "Flow-accelerated corrosion of pressure vessels in fossil plants", *International Journal of Pressure Vessels and Piping*, Volume 77, Issues 2-3, https://doi.org/10.1016/S0308-0161(99)00087-3.

practical implementation is documented in PNNL-27120 Revision 1 ("Criteria and Planning Guidance for Ex-Plant Harvesting to Support Subsequent Licence Renewal," Pacific Northwest National Laboratory, 2019) and ORNL/TM-2016/240 ("Light Water Reactor Sustainability Program: Report on the Harvesting and Acquisition of Zion Unit 1 Reactor Pressure Vessel Segments," Oak Ridge National Laboratory, 2016).

*Holiday in pipe coating*. A holiday is a hole or void in the coating film which exposes buried piping to corrosion.

*Hydrostatic pressure test*. A pressure test conducted during a plant or system shutdown at a pressure above nominal operating pressure or system pressure for which overpressure protection is provided.

*INCEFA+ (INcreasing Safety in Nuclear Power Plants by Covering Gaps in Environmental Fatigue Assessment)*. The INCEFA+ project is intended to deliver new experimental data and new guidelines for assessment of environmental fatigue damage to ensure safe operation of European nuclear power plants. Austenitic stainless steels will be tested for the effects of mean strain, hold time and material roughness on fatigue endurance. Testing will be in nuclear light water reactor environments. The three experimental parameters were selected in the framework of an in-kind project during which the current state of the art for this technical area was developed. The data obtained will be collected and standardised in an online fatigue database with the objective of organising a Comité Européen de Normalisation (CEN) workshop on this aspect. The gaps in available fatigue data lead to uncertainty in current assessments. The gaps will be targeted so that fatigue assessment procedures can address behaviour under conditions closer to normal plant operation than is currently possible. The project term is 1 July 2015 until 30 June 2020.[3]

*Inclusion*. An "inclusion" is a non-metallic impurity such as slag, oxide and sulphide that is present in the original ingot. During rolling of billets into bar stock, impurities are rolled in a lengthwise direction. These direction-oriented inclusions in the finished product are generally referred to as non-metallic inclusions or "stringers". These stringers may be surface or subsurface and are usually short in length and parallel to the grain flow.

*Indication*. The definition of the term "indication" as it applies to NDE is: "A response or evidence of a response disclosed through NDE that requires further evaluation to determine its true significance."[4]

*In-service pressure test*. A system pressure test conducted to perform visual examination VT-2 while the system is in service under operating pressure.

*Latent failure*. A degraded material condition that may lie dormant for a long period before leading to a visible flaw (e.g. through-wall crack, active leakage).

*Leak detection system*. Instrumentation and controls that use various temperature, pressure, level and flow sensors to detect water and steam leakages in selected reactor systems and to initiate annunciation and provide isolation signal (in certain cases) to limit leakage from the reactor coolant pressure boundary when limiting leakage conditions exists.

*Leakage pressure test*. A system pressure test conducted during operation at nominal operating pressure, or when pressurised to nominal operating pressure and temperature.

*Lead factor*. The ratio of the fast neutron fluence at the centre of the surveillance capsule to the peak fast neutron fluence at the inside surface of the reactor pressure wall.

*Less-than-adequate (LTA)*. In the context of a root cause analysis of RIM programme deficiencies, the term "less-than-adequate" (LTA) is used to characterise a procedure that lacks something essential to successfully to perform an activity.

3. https://cordis.europa.eu/project/id/662320.
4. NRC (2012), 0444-E306 Nondestructive Examination (NDE) Technology and Codes, www.nrc.gov/docs/ML1214/ML12146A160.html.

*Limiting condition for operation (LCO)*. According to Technical Specifications[5], a LCO is the lowest functional capability or performance level of a piece of equipment required for safe operation of a nuclear plant. When a LCO cannot be met, the reactor must be shut down or the licensee must follow any remedial action permitted by the Technical Specifications until the condition can be met.

*Liquid droplet impingement (LDI)*. Liquid droplet impingement is caused by the impact of high velocity droplets or liquid jets. LDI usually occurs when a two-phase stream experiences a high-pressure drop (e.g. across an orifice on a line to the condenser). When this occurs, there is an acceleration of both phases with the liquid velocity increasing to the point that, if the liquid strikes a metallic surface, damage to the surface will occur. The main distinction between flashing and LDI is that in flashing the fluid is of lower quality (mostly liquid with some steam), and with LDI, the fluid is of higher quality (mostly steam with some liquid).

*Liquid penetrant examination*. Liquid penetrant examination (LPT) uses liquids to detect cracks in materials. In the mid and late 1930s, Robert and Joseph Switzer worked with processes incorporating visible coloured dyes in the penetrant to give better contrast. In 1941 they introduced processes using fluorescent dyes which, when viewed under a black light, produced contrasts superior to those obtainable with the visible dyes. The fluorescent method was quickly accepted by the military for aircraft part examination. Since then, the use of both colour-contrast and fluorescent penetrants has spread to practically all fields of manufacturing, and new and improved PT products are constantly being developed.

*Low-frequency electromagnetic testing (LFET)*. This technique measures the changes in electromagnetic fields while the scanner passes over the metal. Defects and corrosion maps are calculated, and video displayed in real-time, high resolution, 3-D colour graphics that can be saved for further data analysis or permanent record archiving. Very low frequency magnetic signals are not affected by iron oxide or any non-magnetic surface deposits, which allows for accurate testing on base metals in piping.

*Low-leakage core (LLC)*. A reactor core loading pattern in which the more burned fuel is placed on the periphery of the core to achieve lower radial neutron leakage and to reduce fuel pin peak burnup.

*Marshall Study Group*. A group of experts chaired by Dr W. Marshall (U.K. Atomic Energy Authority) and established to review the factors determining the integrity of LWR pressure vessels. A first report, "An Assessment of the Integrity of PWR Pressure Vessels," was published in 1976. A second, updated report was published in 1982.

*Master curve (MC)*. The MC approach is used to assess irradiated reactor pressure vessel material fracture toughness parameters; for details see for example IAEA Technical Report Series No. 429, Guidelines for Application of the Master Curve Approach to Reactor Pressure Vessel Integrity in Nuclear Power Plants (2005).

*MEACTOS (*Mitigating environmentally assisted cracking through optimisation of surface condition). Environmentally assisted cracking (EAC) is one of the major failure modes occurring in light water reactors (LWRs). The condition of surfaces exposed to the primary coolant plays a main role in the susceptibility of components to EAC. However, many national and international guidelines and standards do not address surface condition of critical components in nuclear power plants. The goal of the MEACTOS project is to improve the safety and reliability of Generation II and III nuclear power plants by improving the resistance of critical locations, including welds, to EAC through the application of optimised surface machining and improved surface treatments. The project end date is 31 August 2021.[6]

*Mechanical Stress Improvement Process (MSIP®)*. A patented process that was invented, developed, and first used in 1986 by NuVision Engineering Inc. for mitigating stress corrosion cracking in nuclear plant weldments. MSIP® works by using a hydraulically operated clamp which contracts

5. "Betriebshandbuch" in German.
6. See https://cordis.europa.eu/project/id/755151.

the pipe on one side of the weldment. A typical tool design consists of a specially designed hydraulic box press for bringing the clamp halves together. By contracting the pipe on one side of the weldment, the residual tensile stresses are replaced with compressive stresses.

*NDE qualification*. In the context of NDE, qualification includes technical justification, which involves assembling all the supporting evidence for inspection capability (results of capability evaluation exercises, feedback from site experience, applicable and validated theoretical models, physical reasoning), and may include practical trials using deliberately defective test pieces.

*NUREG-1061 (Report of the US Nuclear Regulatory Commission Piping Review Committee)*. This report deals with six topical areas: event combinations, response combinations, stress limits and dynamic allowables, water hamer loadings, relief valve opening and closing loads and piping vibration loads. Recommendations prepared by the staff were based on consultant position papers and industry comments and treat revisions to present NRC requirements and directions for future research. Foreign information was obtained from sources in Belgium, Canada, France, Italy, Japan, Sweden and the Federal Republic of Germany. In addition, the report contains qualitative value impacts for the proposed recommendations. This report was developed over a period of approximately one year, and partially fulfils and complies with the requirements of the 13 July 1983 memorandum from the Directors of the Offices of Nuclear Reactor Regulation and Nuclear Regulatory Research to NRC's Executive Director for Operations.[7]

*Optimised weld overlay (OWOL)*. A subset of the full structural weld overlay (FSWOL) process. It has been developed for larger geometries (e.g. RCS hot and cold leg nozzles) where FSWOL application becomes too time consuming for a typical refuelling outage. The optimised weld overlay thickness is less than that of a full structural weld overlay in order allow completion in the time available in a typical refuelling outage for the larger geometries.

*Phenomena identification and ranking technique (PIRT)*. PIRT is a systematic way of gathering information from experts on a specific subject, and ranking the importance of the information, to meet some decision-making objective. It has been applied to many nuclear technology issues including material degradation assessment to help guide research or develop regulatory requirements.

*Probability of detection (POD)*. It is the probability that a flaw of a certain size will be detected and is conditional on factors such as wall thickness, NDE personnel qualifications and flaw orientation.

*Radiographic examination*. A non-destructive testing (NDE) method of inspecting materials for hidden flaws by using the ability of short wavelength electromagnetic radiation (high energy photons) to penetrate various materials.

*Reactor vessel embrittlement*. Embrittlement caused by the reactor vessel material exposure to high energy neutron flux from the reactor core. The area most likely to be affected by neutron embrittlement is the beltline region of the reactor vessel due to its proximity to the core. Embrittlement is characterised by a gradual reduction in the pressure vessel's fracture toughness. If this reduction in toughness were to continue to progress, and a crack existed, it is possible that a brittle fracture of the vessel could occur under postulated events such as pressurised thermal shock (PTS) and low temperature overpressurisation (LTOP).

*Red brass*. An alloy of copper, tin and zinc. Proportions vary but 88% copper, 8-10% tin, and 2-4% zinc is an approximation.

*Reliability and integrity management (RIM)*. Those aspects of the plant design and operational phase that are applied to provide an appropriate level of reliability of SSCs and a continuing assurance over the life of the plant that such reliability is maintained. The most recent edition of the ASME Boiler and Pressure Vessel Code Section XI was issued in 2019. Division 2 of the 2019 Edition provides the requirements for the creation of the reliability and integrity management (RIM) programme for advanced nuclear reactor designs. The RIM programme addresses the entire life cycle for all types of nuclear power plants. It requires a combination of

7. See www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1061/#abstract.

monitoring, examination, tests, operation, and maintenance requirements that ensures each structure, system, and component (SSC) meets plant risk and reliability goals that are selected for the RIM programme[8].

*Selective leaching*. Also referred to as de-alloying, demetalification, parting and selective corrosion, selective leaching is a corrosion type in some solid solution alloys, when in suitable conditions a component of the alloys is preferentially leached from the material. The less noble metal is removed from the alloy by a microscopic-scale galvanic corrosion mechanism. The most susceptible alloys are the ones containing metals with high distance between each other in the galvanic series, e.g. copper and zinc in brass.

*Sockolet*. The weldolet (see below) and the sockolet both have a "run end" and a "branch end". The run end connects to the larger diameter pipe, and the branch end connects to the smaller diameter pipe. The connection to the run pipe is a fillet weld for both the weldolet and the sockolet. They differ, however, at the connection to the branch pipe. The weldolet has a bevelled end to butt weld to the branch pipe (which will also have a bevelled end), whereas the sockolet has a socket into which a plain end branch pipe is inserted, then fillet welded together.

*Solid particle erosion (SPE)*. SPE is damage caused by particles transported by the fluid stream rather than by liquid water or collapsing bubbles. If hard, large particles are present at sufficiently high velocities, damage will occur. In contrast to LDI, the necessary velocities for SPE are quite low. Surfaces damaged by SPE have a very variable morphology. Manifestations of SPE in service usually include thinning of components, a macroscopic scooping appearance following the gas/particle flow field, surface roughening (ranging from polishing to severe roughening, depending on particle size and velocity), lack of the directional grooving characteristics of abrasion, and in some but not all cases, the formation of ripple patterns on metals.

*Stainless steel grade 405*. A ferritic high-alloy stainless steel with chromium content of about 12%.

*Stainless steel grade 409*. A ferritic steel that offers good mechanical properties and high-temperature corrosion resistance. It is commonly considered as a chromium stainless steel, with applications that demand weldability. Grade 409 steels are also available in highly stabilised forms, such as grades S40930, S40920 and S40910. The stability of these grades is provided by the presence of niobium, titanium, or both, in the composition of steels.

*Subsequent licence renewal (SLR)*. A licensing process by the US NRC that would authorise nuclear power plants to operate beyond the 60 years of the initial licence and the first licence renewal. Subsequent licence renewals would also be for 20 years. The NRC has developed guidance for staff and licensees specifically for the subsequent renewal period.

*Super emergency feedwater system (SEFW)*. The SEFW system is specific to WWER-type reactors. It is designed to respond to total loss of heat sink accident scenarios (i.e. total loss of normal and auxiliary feedwater). The system injects feedwater directly into the steam generator secondary side by separate stainless-steel feedlines from demineralised water tanks.

*Thermal ageing embrittlement (TAE)*. A time-and temperature-dependent change in the microstructure of the material which leads to a reduced ductility and deterioration of the fracture toughness and the impact properties. The material will show an increased embrittlement over time.

*Thermal striping*. Incomplete mixing of high temperature and low temperature fluids near the surface of structures with subsequent fluid temperature fluctuations giving rise to thermal fatigue damage to wall structures.

*Thermal treatment (TT) process*. Steam generator tubing is thermally treated at about 705°C for 15 hours to relieve fabrication stresses and to further improve the microstructure. The thermal treatment process promotes carbide precipitation at the grain boundaries and diffusion of chromium to the grain boundaries.

8. See www.ndt.net/article/NDE-Nuclear2019/papers/2.A.01_ASME_Section_XI,_Division_2_Rewrite.pdf.

*Time-limited ageing analysis (TLAA)*. In the context of nuclear power plant licence renewal, a TLAA is formal engineering analysis of the effect of ageing on the structural integrity of systems, structures and components.

*Tinel coupling.* A pipe coupling made of Tinel material (half titanium and half nickel). The coupling is machined to predetermined dimensions with the I.D. smaller than the tube O.D. Passing a mandrel through the coupling, I.D. while submerged in liquid nitrogen at cryogenic temperature will expand the coupling. It will remain as expanded while submerged in liquid nitrogen. When the coupling is installed to couple two pieces of tubing together, it will automatically return to its predetermined dimensions when exposed to room temperature. The shape memory is the result of a change in the crystal structure of the alloy known as reversible austenite to martensite phase transformation.

*Trepan.* Trepanning removes a solid cylindrical core of material by cutting around it.

*Tritium*. Tritium is a naturally occurring radioactive form of hydrogen that is produced in the atmosphere when cosmic rays collide with air molecules. As a result, tritium is found in very small or trace amounts in groundwater throughout the world. It is also a byproduct of the production of electricity by nuclear power plants. Tritium emits a weak form of radiation, a low-energy beta particle like an electron. The tritium radiation does not travel very far in air and cannot penetrate the skin.

*Tritium in nuclear power plants*. Most of the tritium produced in nuclear power plants stems from a chemical, known as boron, absorbing neutrons from the plant's chain reaction. Nuclear reactors use boron, a good neutron absorber, to help control the chain reaction. Toward that end, boron either is added directly to the coolant water or is used in the control rods to control the chain reaction. Much smaller amounts of tritium can also be produced from the splitting of Uranium-235 in the reactor core, or when other chemicals (e.g. lithium or heavy water) in the coolant water absorb neutrons. Like normal hydrogen, tritium can bond with oxygen to form water. When this happens, the resulting "tritiated" water is radioactive. Tritiated water (not to be confused with heavy water) is chemically identical to normal water and the tritium cannot be filtered out of the water.

*Uniaxial constant load (UCL) Test.* The UCL tests use tensile-type specimens to assure precise applied stress values by direct measurement to determine Young’s modulus, yield stress, ultimate tensile strength and elastic strain energy density.

*Unified numbering system (UNS)*. An alloy designation system in use in North America. It consists of a prefix letter and five digits designating a material composition. For example, a prefix of S indicates stainless steel, C indicates copper, brass or bronze alloys.

*Visual examination.* The oldest and most used NDE method is visual testing (VT), which may be defined as “an examination of an object using the naked eye, alone or in conjunction with various magnifying devices, without changing, altering, or destroying the object being examined.”[9] Per ASME XI, there are three different VT methods; VT-1, VT-2 and VT-3.

*Warm pre-stress (WPS) effect*. The WPS effects were identified in the 1960s. According to research and tests, the apparent fracture toughness of a ferritic steel can be elevated in the fracture mode transition: regime if the specimen is first “pre-stressed” at an elevated temperature, e.g. through a pressurised thermal shock. Once a specimen is subjected to a certain $K_{applied}$ and has not failed, the temperature can be reduced, and the specimen will remain intact despite the fact that the process of reducing the temperature has also reduced the initiation fracture toughness ($K_{Ic}$ or $K_{Jc}$) to values smaller than $K_{applied}$. In the past five decades, the physical mechanisms responsible for the WPS effect have been identified, studied extensively and validated.[10]

9. NRC (2012), “0444-E306 Nondestructive Examination (NDE) Technology and Codes”, www.nrc.gov/docs/ML1214/ML12146A160.html.
10. Section 7.7.1.1 in NUREG-1806, Vol. 1; Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61), 2007.

*Weld inlay*. A mitigation technique defined as application of PWSCC-resistant material (Alloy 52/52M) to the inside diameter of a dissimilar metal weld that isolates the PWSCC-susceptible material (Alloy 82/182) from the primary reactor coolant.

*Weldolet®*. The most common of all branch connections and is welded onto a larger-diameter pipe. The ends are bevelled to facilitate this process, and therefore the "weldolet" is considered a butt-weld fitting. Weldolets are designed to minimise stress concentrations and provide integral reinforcement.

*XFEM*. The extended finite element method (XFEM) is a finite element analysis (FEA) tool that allows for mesh-independent analysis of discontinuities and singularities and can be used to simulate crack growth in complex geometries.

*Zorita Internals Research Project (ZIRP)*. The nuclear power plant known as José Cabrera (more commonly referred to as "Zorita") was Spain's first nuclear power plant and began operation in July 1965. The 160 MWe Westinghouse-designed pressurised water reactor was permanently disconnected from the national power grid in April 2006 after 38 years of operation. It was a single-loop PWR. In 2013, baffle plate material and core barrel weld material were removed from the nuclear power plant during the decommissioning activities. These materials provide the foundation for several EPRI materials reliability programme (MRP) research efforts, including the Zorita Internals Research Project (ZIRP). The objective of ZIRP is to study stainless steel ageing degradation to generate correlations between the radiation and temperature history of the materials and the changes in mechanical properties.

Annex B

# Examples of reporting systems/routines for passive component material degradation and failure

### Table B-1: Examples of US reporting routines and requirements for communicating/documenting degraded or failed passive components

| Reporting level | Report/database type | Level of technical content from the point of view of material science, PSA applications, etc. | 3rd party accessibility for purposes of R&D and/or operational support |
|---|---|---|---|
| **US Nuclear Regulatory Commission** | | | |
| Regulatory requirements and guidance | Licensee event report | High-level – requires interpretation and classification by a SME - high reporting thresholds. Since the 1990s only a few LERs of relevance to CODAP are issued on an annual basis; < 10 per year | Public domain; covers 1980 to date; https://lersearch.inl.gov/LERSearchCriteria.aspx. A large number of pre-1980 LERs are available from https://adams.nrc.gov/wba/ |
| | Event notification (EN) system | High-level event tracking system for incoming notifications of significant nuclear events with an actual or potential effect on plant safety. An EN may result in a LER. On an annual basis, only a few ENs of relevance to CODAP are issued | Public domain; all ENs from 1 January 1999 onwards are available at: www.nrc.gov/reading-rm/doc-collections/event-status/event/en.html |
| | Generic communications – Generic letters, information notices | Identifies potential generic issues that are safety significant and require technical resolution | www.nrc.gov/reading-rm/doc-collections/gen-comm/ |
| | Generic letter 90-05 Request for temporary repair | Applicable to moderate-energy piping, the operability evaluations that are part of the 90-05 relief request are comprehensive. Excellent source of pipe failure information | Public domain; https://adams.nrc.gov/wba/ |
| | 10 CFR 50.55a(z)(2) – Application of ASME Code Cases | The operability evaluations that are part of the ASME Code Case relief request are comprehensive. Excellent source of pipe failure information | See below under "ASME Code Cases". Any Code Case Relief request that is submitted to the NRC for approval is also available for review by 3rd party. |
| | Biannual Problem Identification and Resolution Inspection Report | Regulatory review of a licensee's corrective action program and the licensee's implementation of the program to evaluate its effectiveness in identifying, prioritising, evaluating, and correcting problems, and to confirm that the licensee is complying with NRC regulations and licensee standards for corrective action programmes | Public domain; https://adams.nrc.gov/wba/ |
| | Quarterly integrated inspection reports | Significant repository for information on passive components failures. Requires interpretation and classification by a SME | Public domain; https://lersearch.inl.gov/IRSearchCriteria.aspx |

**Table B-1: Examples of US reporting routines and requirements for communicating/documenting degraded or failed passive components** *(cont'd)*

| Reporting level | Report/database type | Level of technical content from the point of view of material science, PSA applications, etc. | 3rd party accessibility for purposes of R&D and/or operational support |
|---|---|---|---|
| | **US nuclear industry level** | | |
| Institute of Nuclear Power Operations (INPO) | INPO Consolidated Events Database (ICES). Participating plants submit ARs, CRs, WORs, etc. – mainly free-format narrative information – to a searchable database | Varying - data extraction, interpretation and classification can be time-consuming | Access restricted to ICES participating organisations. With respect to the OECD/NEA ICDE Project, an NRC-INPO MoU has enabled a time-limited database access in support of the project |
| Nuclear Energy Institute (NEI) | NEI 07-07: Industry Ground Water Protection Initiative – Final Guidance Document, 2007 | Reporting of below ground/buried pipe failures causing > 0.38 $m^3$ unintended release | Annual radioactive effluent release report submitted by licensees to the US NRC |
| Electric Power Research Institute (EPRI) | EPRI-CHUG: Presentation material from the biannual CHUG meetings. Informal exchange of FAC information (summary of FAC failures, information on the use of NDE technology) | Varying - data extraction, interpretation and classification can be time-consuming | Access restricted to funding organisation. Non-proprietary versions of EPRI-MRP reports can be downloaded from the NRC public website |
| | EPRI-MRP, multiple programmes – reactor internals, thermal fatigue, ageing management | NRC receives biennial summaries of the BWR and PWR reactor internals inspection results. These summaries are available on the NRC website | |
| | Plant engineering/long-term operations/risk and safety management programmes | Extensive source of material degradation and failure information organised by system, degradation mechanism, etc. These programmes build on active technical input from the sponsoring US and international member organisations. Extensive repository of relevant PEO/LTO material degradation information | Access restricted to funding organisation |
| NRC/EPRI-MRP Annual Technical Information Exchange Meetings | Presentation material summarising R&D status, results of reactor internals inspections, thermal fatigue inspections, etc. | High-level | All presentation material available through the NRC Public Document Room |
| GE service information letter (SIL), and equivalent bulletins from the PWROG | Information on material degradation issues in response to recent OPEX | Chronological lists of events with some information on method of discovery, extent of degradation, flaw size data, etc. Plant identities not revealed | The SILs can be downloaded from the NRC Public Document Room; for example see www.nrc.gov/docs/ML0501/ML050120032.pdf |

**Table B-1: Examples of US reporting routines and requirements for communicating/documenting degraded or failed passive components** *(cont'd)*

| Reporting level | Report/database type | Level of technical content from the point of view of material science, PSA applications, etc. | 3rd party accessibility for purposes of R&D and/or operational support |
|---|---|---|---|
| | | **Plant-level** | |
| During routine operation – Shiftly walkdown inspection | Action request (AR), condition report (CR), corrective action request (CAR), Incident report (IR), Non-conformance report (NCR), problem investigation process (PIP), repair/ replacement plant (RRP), work order request (WOR), operability determination, and structural evaluation in accordance with applicable ASME Code Case | Details on location of flaw/leak with reference to P&ID, isometric drawing – oftentimes accompanied by photographic records, root cause analysis results, NDE results, metallographic data, flaw size data, leak/flow rates, spatial effects, risk significance, safety significance. Piping fabrication data, operational parameters, piping design information (material, dimensional data) | Restricted/proprietary. US NRC Resident Inspectors have access to information on request |
| Leak detection – containment/dry well | Shiftly (once every 8/12-hour shift) RCS inventory mass balance calculation, Chem. Lab trending of inflows to Aux. Building / Reactor Building sumps and holding tanks | Logs, databases (various types) | Restricted/proprietary. US NRC Resident Inspectors have access to information on request |
| | Containment/drywell leak detection alarm (≥ 0.1 gpm/6.3E-3 kg/s) | Control room logs/computer printouts, Excel spreadsheets – holding tank inflow, details of through-wall leak rates as a function of time and location, etc. | Restricted/proprietary. US NRC Resident Inspectors have access to information on request |
| In-service inspection (ISI) per ASME XI – mandatory inspections | List of "recordable indications", operability determinations/fitness-for-service (FFS) evaluations, ASME XI relief requests (for more details see "Regulatory" above. Within 90 days of the conclusion of a refuelling outage, the plant owner is required to submit an ISI summary report ("Owner's Activity Report") to the US NRC. This report itemises all inspections performed (Code Class 1, 2 and 3), any operability determination is documented | The ASME relief requests include all relevant information to meet CODAP event data input requirements. Included are "Abstract of Repair/Replacement Activities Required for Continued Service" and "Items with Flaws or Relevant Conditions That Required Evaluation for Continued Service" | The ISI summary reports are publicly available via the NRC Public Document Room; https://adams.nrc.gov/wba/ |
| Owner defined ISI programmes | FAC, Erosion-Corrosion, MIC and thermal fatigue programmes | NDE reports, "system health reports" | Restricted/proprietary. US NRC Resident Inspectors have access to the information upon request |

### Table B-1: Examples of US reporting routines and requirements for communicating/documenting degraded or failed passive components *(cont'd)*

| Reporting level | Report/database type | Level of technical content from the point of view of material science, PSA applications, etc. | 3rd party accessibility for purposes of R&D and/or operational support |
|---|---|---|---|
| American Society of Mechanical Engineers (ASME) | Numerous Code Case have been developed for the operability determination of degraded piping; e.g. N-513 (Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 3 Piping), N-561 (Alternative Requirements for Wall Thickness Restoration of Class 2 and High Energy Class 3 Carbon Steel Piping), N-562 (Alternative Requirements for Wall Thickness Restoration of Class 3 Moderate Energy Carbon Steel Piping), N-666 (Weld Overlay of Class 1, 2, and 3 Socket Welds | Details on location of flaw/leak with reference to P&ID, isometric drawing – oftentimes accompanied by photographic records, root cause analysis results, NDE results, metallographic data, flaw size data, leak/flow rates, spatial effects, risk significance, safety significance. Piping fabrication data, operational parameters, piping design information (material, dimensional data). Structural integrity evaluation (inputs/results/ interpretation) is part of the required documentation. The ASME relief requests include all relevant information to meet CODAP event data input requirements | The Code Case relief requests that are submitted to the NRC for approval are also available for 3rd party review. Where a Code Case has been approved for plant-specific use, an associated operability determination is kept on file on site and available for review by NRC Resident Inspector. Hence, a 3rd part review is not possible unless the underlying technical information (e.g. fitness-for-service evaluations) has been placed in the NRC Public Document Room |
| System health reports (or system engineers' notebooks) | Access or Excel-based databases that are part of the overall ageing management programme. The system health reports provide summaries of all through-wall leaks | High-level, intended for trending of material degradation issues, effectiveness of degradation mitigation programmes | Restricted/proprietary. US NRC Resident Inspectors may have access to the information upon request |

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# Review of Operating Experience Involving Passive Component Material Degrading in Periods of Extended and Long-Term Operation

This publication addresses trends and patterns in material degradation as a function of nuclear power plant ageing, which is a key topic in the regulation of ageing management programmes, codes and standards, reliability and integrity management, and the many supporting research programmes. It documents insights reached from operating experience involving passive metallic components in commercial nuclear power plants during periods of extended operation and long-term operation. These evaluations are enabled by an NEA joint project entitled the Component Operational Experience, Degradation and Ageing Programme (CODAP). Established in 2002, the project has developed a comprehensive database on the operating experience with material degradation mechanisms that have produced failures of piping and selected non-piping passive components. Examinations of graphical plots were performed to reach insights about the different material degradation mechanisms as a function of plant age. The report also includes recommendations for future work.

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