Nuclear Safety
NEA/CSNI/R(2019)7
April 2019
www.oecd-nea.org

NEA Component Operational Experience, Degradation and Ageing Programme (CODAP)

Second Term (2015-2017) Status Report

OECD BETTER POLICIES FOR BETTER LIVES

NEA NUCLEAR ENERGY AGENCY

Nuclear Energy Agency

NEA/CSNI/R(2019)7

Unclassified **English text only**

4 April 2019

NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

NEA Component Operational Experience, Degradation and Ageing Programme (CODAP): Second Term (2015-2017) Status Report

November 2018

Please note that this document is available in PDF format only.

JT03445868

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 36 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

The OECD member countries are: Australia, Austria, Belgium, Canada, Chile, the Czech Republic, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Korea, Latvia, Lithuania, Luxembourg, Mexico, Netherlands, New Zealand, Norway, Poland, Portugal, Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation's statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 33 countries: Argentina, Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Korea, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, Romania, Russia, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission also takes part in the work of the Agency.

The mission of the NEA is:

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

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

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The Committee on the Safety of Nuclear Installations (CSNI) is responsible for the Nuclear Energy Agency (NEA) programmes and activities that support maintaining and advancing the scientific and technical knowledge base of the safety of nuclear installations.

The Committee constitutes a forum for the exchange of technical information and for collaboration between organisations, which can contribute, from their respective backgrounds in research, development and engineering, to its activities. It has regard to the exchange of information between member countries and safety R&D programmes of various sizes in order to keep all member countries involved in and abreast of developments in technical safety matters.

The Committee reviews the state of knowledge on important topics of nuclear safety science and techniques and of safety assessments, and ensures that operating experience is appropriately accounted for in its activities. It initiates and conducts programmes identified by these reviews and assessments in order to confirm safety, overcome discrepancies, develop improvements and reach consensus on technical issues of common interest. It promotes the co-ordination of work in different member countries that serve to maintain and enhance competence in nuclear safety matters, including the establishment of joint undertakings (e.g. joint research and data projects), and assists in the feedback of the results to participating organisations. The Committee ensures that valuable end-products of the technical reviews and analyses are provided to members in a timely manner, and made publicly available when appropriate, to support broader nuclear safety.

The Committee focuses primarily on the safety aspects of existing power reactors, other nuclear installations and new power reactors; it also considers the safety implications of scientific and technical developments of future reactor technologies and designs. Further, the scope for the Committee includes human and organisational research activities and technical developments that affect nuclear safety.

Table of contents

List of tables

List of figures

Executive summary

Several Nuclear Energy Agency (NEA) member countries have agreed to establish the Component Operational Experience, Degradation and Ageing Programme (CODAP) to encourage multilateral co-operation in the collection and analysis of data relating to degradation and failure of metallic piping and non-piping metallic passive components in commercial nuclear power plants. The scope of the data collection includes service-induced wall thinning, part through-wall cracks, through-wall cracks with and without active leakage, and instances of significant degradation of metallic passive components, including piping pressure boundary integrity. This joint database project is organised under the NEA Committee on the Safety of Nuclear Installations (CSNI).

CODAP is the continuation of the 2002-2011 "NEA Pipe Failure Data Exchange Project" (OPDE) and the Stress Corrosion Cracking Working Group of the 2006-2010 "NEA SCC and Cable Ageing project" (SCAP). A key accomplishment of CODAP is the creation of a framework for the systematic collection and evaluation of service-induced degradation and failure of passive metallic components. The online event database facilitates data entry as well as database interrogation. Currently the database includes about 4 900 event records from 324 commercial nuclear power plants.

This report describes the status of the CODAP Project at the conclusion of its second term (2015-2017). It gives a high-level overview of the passive metallic component operating experience as documented in the CODAP event database, including trends-and-patterns, material degradation mitigation effectiveness, and experience with different non-destructive examination techniques. During the second term, three public domain Topical Reports were prepared by the CODAP project review group (now management board).

The Radiation and Nuclear Safety Authority (STUK) of Finland and the Authority for Nuclear Safety and Radiation Protection (ANVS) of the Netherlands have indicated their intention to join the project during its third term (2018-2020). After that, the 13 members of CODAP are as follows: Canada, the Czech Republic, Finland, France, Germany, Korea, Japan, the Netherlands, the Slovak Republic, Spain, Switzerland, Chinese Taipei and the United States. Sigma-Phase Inc. from the United States works as the Operating Agent of the CODAP project.

The CODAP project and its management board intend to actively support proposals to arrange an international benchmark exercise concerning the use of operating experience data to quantify piping reliability parameters for input to a standard problem application; e.g. risk-informed operability determination.

In 2014, the CSNI programme review group recommended that the CODAP project implement operating procedures and processes whereby future national data submissions are commensurate with the number of operating reactors. This target was already taken into account during second term of CODAP, but during third term more work will be done to achieve a more "balanced" event database. In addition, a decision was made by the management board to expand the scope of the event database to address degradation and failure of high-density polyethylene (HDPE) piping.

The third term of the project places an emphasis on the following aspects of operating experience data exchange and analysis:

1. an improved web interface for data submitters;
2. an enhanced database query ability through the web interface;
3. active data submissions by the PRG membership;
4. continued database applications will be pursued through an expanded programme to develop topical reports.

List of abbreviations and acronyms

ADS	Automatic depressurisation system
AEC	Atomic Energy Council
AFW	Auxiliary feed water
AFCN	Agence fédérale de contrôle nucléaire (Belgium)
AMP	Ageing management programme
ANS	American Nuclear Society
ANSI	American National Standards Institute
ANVS	Radiation and Nuclear Safety Authority (The Netherlands)
ASME	American Society of Mechanical Engineers
ASN	Autorité de sûreté nucléaire (France)
BOP	Balance-of-plant
BMI	Bottom mounted instrument
BPIG	Buried Pipe Integrity Group (EPRI)
BWR	Boiling water reactor
CADAK	Cable Ageing Data and Knowledge Project (NEA)
CC	Component cooling
CFR	Code of federal regulation
CG	Coding guideline
CL	PWR Reactor coolant system cold leg
CODAP	Component Operational Experience, Degradation and Ageing Programme
CSN	Nuclear Safety Council (Spain)
CSNI	Committee on the Safety of Nuclear Installations (NEA)
CSV	Character separated values
CY	Calendar year
CCW	Component cooling water
CRDM	Control rod drive mechanism
CS	Containment spray

CSA	Canadian Standards Association
CW	Circulating water
DB	Database
DEGB	Double-ended guillotine break
DM	Degradation mechanism
DMW	Dissimilar metal weld
DN	Nominal diameter [mm]
DWD	Demineralised water distribution
E/C	Erosion-corrosion
E-C	Erosion-cavitation
ECC	Emergency core cooling
EDF	Electricité de France
EDX	Energy-dispersive X-ray spectroscopy
EFPY	Effective full power years
EMDA	Expanded material degradation assessment
ENIQ	European network for inspection and qualification
EOC	End-of-cycle
EPIX	Equipment performance information exchange
EPRI	Electric Power Research Institute (United States)
ESF	Engineered safeguards features
ESW	Essential service water
FAC	Flow accelerated corrosion
FC	Fuel channel
FFSG	Feeder fitness for service guidelines
FIRE	Fire Incidents Records Exchange Project (NEA)
FPS	Fire protection system
FW	Feedwater
GALL	Generic ageing lessons learned
GAO	Government Accountability Office (United States)
GL	Generic letter (of US NRC)
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
GSKL	Group of Swiss nuclear power plant managers
HDPE	High-density polyethylene
HELB	High-energy line break

HL	PWR reactor coolant system hot leg
HT	Heat transport
HVAC	Heating, ventilation and air conditioning
IAEA	International Atomic Energy Agency
IAGE	CSNI working group on integrity and ageing of components and structures (NEA)
ICDE	International common-cause failure data exchange project (NEA)
ICES	INPO consolidated events database
ID	Inside diameter
IDSCC	Interdendritic SCC
IF-PSA	Internal flooding PSA
IGALL	International generic ageing lessons learned
INER	Institute of Nuclear Energy Research (Taiwan)
INPO	Institute of Nuclear Power Operations (United States)
IR	Inspection report
IRSN	Institut de radioprotection et de sûreté nucléaire (France)
ISG	Interim staff guidance
ISI	In-service inspection
IWC	Intermediate water cooler
KAERI	Korea Atomic Energy Research Institute
KB	Knowledge base
KEPCO	Korea Electric Power Corporation
KHNP	Korea Hydro & Nuclear Power
KINS	Korea Institute of Nuclear Safety
LTA	Less-than-adequate
LCO	Limiting condition for operation
LDI	Liquid droplet impingement
LOCA	Loss-of-coolant accident
LTCC	Low-temperature creep cracking
MB	Management board
MCP	Main reactor coolant pump
MCR	Main control room
MELB	Moderate-energy line break
MIC	Microbiologically influenced corrosion

MINOS	Materials innovation for nuclear optimised systems
MS	Main steam
NDE	Non-destructive examination
NEA	Nuclear Energy Agency
NISA	Nuclear and Industrial Safety Agency (Japan)
NLUT	Nonlinear ultrasonic testing
NPP	Nuclear power plant
NPS	Nominal pipe size
NPSAG	Nordic PSA Group
NRC	Nuclear Regulatory Commission (United States)
OD	Outside diameter
OE	Operating experience
OECD	Organisation for Economic Co-operation and Development
OLC	Operational limits and conditions (Technical specifications)
OPDE	OECD Pipe failure Data Exchange project
PARENT	Programme to assess the reliability of emerging non-destructive techniques
PCSG	Pipe crack study group
PDA	Performance demonstration administrator
PDI	Performance demonstration initiative
PFM	probabilistic fracture mechanics
PHTS	Primary heat transport system
PLGS	Point Lepreau Nuclear Generation Station (Canada)
PMDA	Proactive materials degradation assessment
PRG	Project review group
PSA	Probabilistic safety assessment
PSR	Periodic safety review
PVC	Polyvinyl chloride
PWS	Potable water system
PWR	Pressurised water reactor
PWSCC	Primary water SCC
RC	Reactor coolant
RCIC	Reactor core isolation cooling
RCPB	Reactor coolant pressure boundary

RF	Refuelling cycle
RHR	Residual heat removal
RI-ISI	Risk informed in-service inspection
RIM	Reliability and integrity management
RL	Reference level
RPS	Reactor protection system
RT	Radiographic testing
RV	Relief valve
RPVI	Reactor pressure vessel internals
SCAP	Stress Corrosion cracking and Cable Ageing project (NEA)
SCC	Stress corrosion cracking
SD	Significance determination
SDP	Significance determination process
S/G	Steam generator
SRM	Staff requirements memorandum
SRV	Safety relief valve
SSC	Systems, structures and components
SSM	Radiation and Nuclear Safety Authority (Sweden)
STUK	Radiation and Nuclear Safety Authority (Finland)
SW	Service water
TPC	Taiwan Power Company
TPR	Topical peer review
TRM	Technical requirements manual
UHS	Ultimate heat sink
UT	Ultrasonic testing
VHP	Vessel head penetration
VT	Visual inspection technique
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)
XML	Extensible mark-up language

1. Introduction

Structural integrity of piping components and systems and non-piping passive components such as the reactor pressure vessel and internals is important for plant safety and operability. In recognition of this, information on degradation and failure of metallic piping and non-piping passive components is collected and evaluated by regulatory agencies, international organisations (e.g. NEA and IAEA) and industry organisations worldwide to provide systematic feedback for example to reactor regulation and research and development programmes associated with ageing phenomena, non-destructive examination (NDE) technology, in-service inspection (ISI) programmes, leak-before-break evaluations, risk-informed ISI, and probabilistic safety assessment (PSA) applications involving passive component reliability.

Since 2002, the NEA has operated an event database project that collects information on passive metallic component degradation and failures of the primary system, reactor pressure vessel internals, main process and standby safety systems, and support systems (i.e. ASME Code Class one, two and three, or equivalent), as well as non-safety-related (non-code) components with significant operational impact. With an initial focus on piping systems and components (the OPDE project), the scope of the project in 2011 was expanded to also address the rector pressure vessel and internals as well as certain other metallic passive components that are susceptible to environmental degradation. In recognition of the expanded scope, the project review group approved the transition of OPDE to a new, expanded Component Operational Experience, Degradation and Ageing Programme (CODAP).

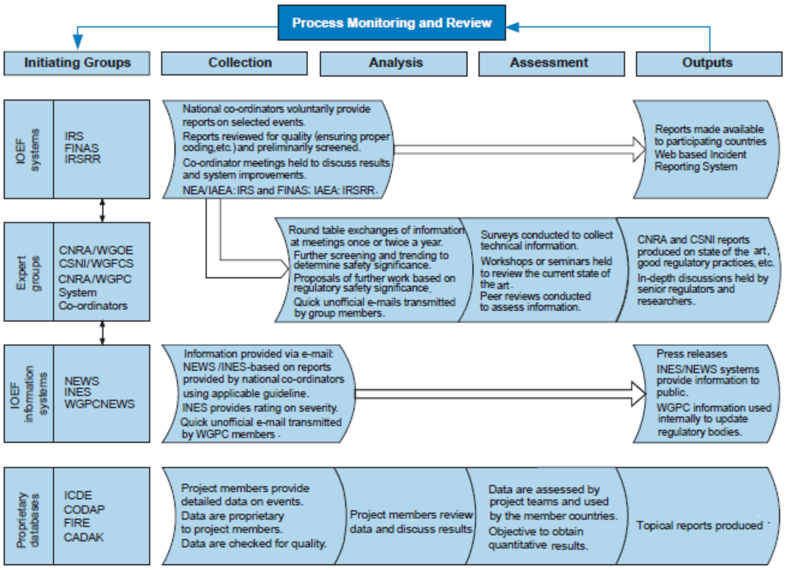
1.1. CODAP origin and project history

Reviews of service experience with safety-related and non-safety-related piping systems have been ongoing ever since the first commercial nuclear power plants came on line in the 1960s [1] [2]. In 1975, the US Nuclear Regulatory Commission established a Pipe Crack Study Group (PCSG) charged with the task of evaluating the significance of stress corrosion cracking in boiling water reactors (BWRs) [3] and pressurised water reactors (PWRs) [4]. Service experience review was a key aspect of the work by the PCSG. Major condensate and feed water piping failures (e.g. Trojan and Surry-2 in the US, Loviisa-1 in Finland and Mihama-3 in Japan) due to flow accelerated corrosion (FAC) resulted in similar national and international initiatives to learn from service experience and to develop mitigation strategies to prevent recurrence of pipe failures [5] [6] [7]. Early indications of the significance of thermal fatigue phenomena evolved in the 1970s, and, again, systematic reviews of the service experience enabled the introduction of improved piping design solutions, NDE methods, and operating practices [8].

The team of analysts responsible for the seminal reactor safety study (WASH-1 400) [9] performed a limited evaluation of nuclear and non-nuclear power plant piping reliability based on field experience data on pipe failures. This evaluation was aimed at estimating loss-of-coolant accident (LOCA) frequencies for input to the two probabilistic safety assessment (PSA) models of WASH-1400. After the publication of WASH-1 400 many other R&D projects have explored the roles of structural reliability models and statistical evaluation models in providing acceptable input to PSA. Furthermore, during the past 20 years' efforts have been directed towards establishing comprehensive pipe failure event databases as a foundation for exploratory research to better understand the capabilities and limitations of today's piping reliability analysis frameworks.

Assessment of passive component service experience data has been an integral element of regulatory and industry programmes to address long-term operation and nuclear plant licence renewal. Examples of such programmes include the proactive materials degradation assessment (PMDA), expanded materials degradation assessment (EMDA), generic ageing lessons learned (GALL), generic ageing lessons learned for subsequent licence renewal (GALL-SLR), and international generic ageing lessons learned (IGALL)[1]. A common feature of these four programmes is the acknowledgement of systematic reviews of the accumulated service experience data as one of several inputs to the development of a technical basis for practical ageing management of metallic passive components. A joint NEA and International Atomic Energy Agency (IAEA) perspective on the international operating experience exchange processes is illustrated in Figure 1.1.

Figure 1.1. The NEA and IAEA International Operating Experience Exchange[2]



1.1.1. Origin of CODAP

The CODAP international collaboration has its origins in the piping reliability R&D sponsored by the Swedish Nuclear Power Inspectorate (SKI)[3] in the early 1990s and in

1. Detailed information on PMDA, EMDA GALL and GALL-SLR are available at www.nrc.gov. The IGALL is summarised in IAEA-TECDOC-1736 (April 2014), which is available at www.iae.org.
2. Adapted from NEA News, Vol. 26:1, 2008. www.oecd-nea.org/pub/newsletter/2008/International%20Operating%20Experience.pdf
3. In July 2008, SKI and the Swedish Radiation Protection Institute (SSI) were merged to form the Swedish Radiation Safety Authority (SSM); www.ssm.se.

response to the so-called 1992 “Barsebäck-2 strainer event.” On 28 July 1992, a steam line pressure boundary breach occurred when a safety relief valve (SRV) inadvertently opened in Barsebäck-2 nuclear power plant, a third generation Swedish boiling water reactor (BWR) design. At the time of the event, the plant was returning to service after an annual refuelling and maintenance outage. With the reactor at about 2% power and 3.2 MPa pressure, a leaking pilot valve caused a depressurisation of the main safety relief valve, which then opened. When the main valve opened a rupture disc, with design pressure of 3 MPa, broke causing an opening into the containment drywell. The resulting steam jet stripped fibrous insulation from adjacent pipework. Part of that insulation debris was transported to the suppression pool and subsequently clogged the intake strainers for the containment vessel spray system about one hour into the event sequence. The 1992 ‘‘strainer event” confirmed some of the safety concerns that had been raised about two decades earlier. Specifically, this generic safety issue was concerned with the impact of dynamic effects of a primary pressure boundary breach such as a pipe break on the operability of emergency core cooling systems. While there had been a number of strainer ‘‘precursors” events in the 1970s, 1980s and 1990s, it was the 1992 strainer event that prompted an extensive and still on-going response by the international nuclear safety community.

The Swedish regulatory and industry response to the strainer event involved the establishment of R&D efforts that focused on physical phenomena associated with containment sump clogging issues, pipe break debris generation, debris transport and the technical basis for more realistic loss of coolant accident (LOCA) frequency assessment. In part, the latter aspect of this broad R&D effort consisted of a five-year R&D effort to explore the viability of establishing an international database on the operating experience with piping in commercial nuclear power plants. An underlying objective behind this five-year programme was to investigate the different options and possibilities for deriving pipe failure rates and rupture frequencies directly from service experience data as an alternative to, for example, probabilistic fracture mechanics (PFM). The R&D programme culminated in an international piping reliability seminar in the fall of 1997 [10] and the completion of a plant-specific LOCA frequency assessment pilot study in 1998 [11].

One outcome of the aforementioned R&D programme was the decision by SKI to transfer the pipe failure database including the lessons learned [12] [13] to an international cooperative effort under the auspices of the NEA. After a series of information exchange and planning meetings organised by the NEA in September 2000 and April 2001, the OECD pipe failure data exchange project (OPDE) was officially launched in May 2002 [14].

1.1.2. CODAP project history

During the three OPDE project terms (2002-2011), the event database was maintained and distributed as a Microsoft® Access database. This database was distributed on a CD to the national co-ordinators twice per calendar year. Towards the end of the first project term, a web-based database format was developed to facilitate data exchange. The web-based OPDE resided on a secure server at the NEA Offices. With the 2011 transition from OPDE to CODAP, a new and enhanced web-based database format was implemented. Since mid-2012, the entire CODAP event database resides on a secure server at NEA Offices. Provisions exist for online database interrogation (e.g. event review, QA, queries) as well as downloading queries (in CSV- or XML-file format) and selected event records or entire database (in XML-file format) to a local computer or computer network.

With an initial focus on piping systems and components (the OPDE Project), the scope of the project was expanded in 2011 to also address the rector pressure vessel and internals as well as certain other metallic passive components that are susceptible to environmental degradation. In recognition of the expanded scope, the project review group approved the transition of OPDE to a new, expanded Component Operational Experience, Degradation and Ageing Programme (CODAP). Access to the CODAP event database is restricted to participating organisations.

The current version of the CODAP event database consists of 110 data fields and about 800 data filters. A basic premise of the use of narrative information is to preserve the original event information as recorded in root cause evaluation reports and reportable occurrence reports. The "related tables" include information on material, location of damage or degradation, type of damage or degradation, system name, safety class, dimensional data, etc. The event database structure, database field definitions and data input requirements are defined in a coding guideline, which is central to the project, including database maintenance, data validation and quality control. The database design has benefitted from a multidisciplinary approach involving chemistry, metallurgy, non-destructive examination, structural integrity and PSA. Each event record relates to a uniquely defined component boundary.

1.2. Transition from OPDE to CODAP

The number of commercial nuclear power plants approaching or in an extended period operation is increasing in NEA member countries. Accordingly, maintenance programmes, in-service inspection and testing of structures, systems and components important to safety have been implemented to ensure that levels of reliability and effectiveness remain in accordance with the design assumptions. This is often being done using an integrated ageing management strategy.

Ageing effects, especially material degradation, have been experienced worldwide and progressively since the start of nuclear power plant operation. Material degradation is expected to continue as plants age and operating licences are extended. It is clear that unanticipated and unmanaged structural degradation could result in significant loss of safety margins, undermining public confidence and straining the resources of both regulatory authorities and the operators.

For regulatory authorities, it is also important to verify the adequacy of the ageing management methods applied by the licensees, based on reliable technical evidence. Two subjects – stress corrosion cracking (SCC) and degradation of cable insulation – were selected as the focus of the SCC and cable ageing project (SCAP) due to their relevance for plant ageing assessments and their implication on nuclear safety. Fourteen NEA member countries joined the project in 2006 to share knowledge about the different types of SCC mechanisms. The project was financed by a Japanese voluntary contribution. Japanese technical institutions also actively co-operated in the project under the co-ordination of the Nuclear and Industrial Safety Agency (NISA) of Japan.

When establishing the SCAP project it was realised that in the limited time available that ageing management could not be addressed in detail over a large range of topics. Stress corrosion cracking has been, and continues to be, a serious problem; cable ageing has been identified as an area requiring more attention from both the regulators and the industry. The failures in both areas continue to offer periodic surprises. These two topics were therefore chosen for specific study in the SCAP project as being examples of an area in which ageing management has been applied for many years and one in which ageing management still needs to be developed in an internationally co-ordinated study which was anticipated could yield greater insights into the management of these failures.

Following the completion of the SCAP project [15], SCC Working Group participants were interested in some form of continuation and discussions were initiated to explore possible alternatives. It was recognised that there were many aspects very similar to those existing in OPDE and the concept of a new project was envisaged to combine the two projects into the Component Operational Experience, Degradation and Ageing Programme (CODAP).

1.3. Report structure

Section 2 describes the CODAP objective and scope. The CODAP project organisation is described in Section 3. The event database and selected operating experience insights are summarised in Section 4. Section 5 summaries the conclusion and recommendation of the three topical reports that were produced by the Management Board (MB) during the second term of the project. Database accessibility is addressed in Section 6. Conclusions and future plans are addressed in Section 7. Finally, a list of references is included in Section 8. Annex A includes a CODAP-MB activity report. Annex B is tribute to the late Dr Karen Gott, the OPDE and SCAP-SCC project chair. Annex C includes a piping system cross-reference table. Annex D is a glossary of terms. Finally, Annex E is an OPDE/CODAP bibliography including references to database applications performed or sponsored by OPDE/CODAP member organisations since 2002.

2. CODAP objective and scope

CODAP is the continuation of the 2002-2011 NEA Pipe Failure Data Exchange Project (OPDE) and the work by the stress corrosion cracking working group of the 2006-2010 NEA SCC and cable ageing project (SCAP). The scope of the CODAP is based on a combination of the concepts from the two projects. Thus, it encompasses service experience data on metallic piping and non-piping passive components as in SCAP as well as the full range of material degradation mechanisms as in OPDE.

2.1. Data collection methodology

The CODAP project exchanges data on passive component degradation and failure, including service-induced wall thinning, non-through wall cracking, leaking through-wall cracks, pinhole leakage, leakage, rupture and severance (pipe break caused by external impact). For non-through wall cracks the CODAP scope encompasses degradation exceeding design code allowable for wall thickness or crack depth as well as such degradation that could have generic implications regarding the reliability of in-service inspection (ISI) techniques. The following failure modes are considered[4]:

- Non-through wall defects (e.g. cracks, wall thinning) interpreted as structurally significant and/or exceeding design code allowable;
- Loss of fracture toughness of cast austenitic stainless steel piping. The loss of fracture toughness is attributed to thermal ageing embrittlement.
- Through-wall defects without active leakage (leakage may be detected following a plant operational mode change involving depressurisation and cool-down, or as part of preparations for non-destructive examination, NDE);
- Small leaks (e.g. pinhole leak, drop leakage) resulting in piping repair or replacement;
- Leaks (e.g. leak rates within limits of the Operational Limits and Conditions (OLC) (US term is technical specifications);
- Large leaks (e.g. flow rates in excess of OLC limits);
- Major structural failure (pressure boundary "breach" or "rupture").

In other words, the CODAP event database collects data on the full range of degraded conditions from "precursors" to major structural failures. The types of failures included in the CODAP Event Database are:

4. Annex E of the CODAP "Coding Guideline" [23] documents the different national reporting thresholds.

- Event-based failures that are attributed to damage mechanisms and local pipe stresses. Examples include high-cycle vibration fatigue due to failed pipe support, and hydraulic transient (e.g. steam or water hammer) acting on a weld flaw (e.g. slag inclusion).
- Failures caused by environmental degradation such as stress corrosion cracking due to combined effects of material properties, operating environment (e.g. corrosion potential, irradiation) and loading conditions.

The CODAP event database is a web based, relational database consisting of ca. 100 uniquely defined data fields. It is a blend of free-format fields for detailed narrative information and fields defined by drop-down menus with key words (or data filters) or related tables. A basic premise of the use of narrative information is to preserve original event information as recorded in root cause evaluation reports and reportable occurrence reports. The "related tables" include information on material, location of damage or degradation, type of damage or degradation, system name, safety class, etc. The event database structure, database field definitions and data input requirements are defined in a coding guideline, which is central to the project, including database maintenance, data validation and quality control. The database design has benefitted from a multidisciplinary approach involving chemistry, metallurgy, non-destructive examination, structural integrity and PSA. Each event record relates to a uniquely defined component boundary.

2.2. How CODAP relates to NPP ageing management

The sharing and evaluation of the international operating experience concerning material degradation and its effects on metallic passive component integrity are important elements of nuclear power plant ageing management. As stated in Volume 2, Annex B of NUREG-2 191 [16]:

- *"Operating experience is a crucial element of an effective ageing management program (AMP). It provides the basis to support all other elements of the AMP and, as a continuous feedback mechanism, drives changes to these elements to maintain the overall effectiveness of the AMP. Operating experience should provide objective evidence to support the conclusion that the effects of aging are managed adequately so that the structure- and component-intended function(s) will be maintained during the subsequent period of extended operation. Pursuant to Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," Section 21(a)(3), of Title 10 of the Code of Federal Regulations [10 CFR 54.21(a)(3)], license renewal applicants are required to implement programs for the ongoing review of operating experience (OE), such as those established in accordance with Item I.C.5, "Procedures for Feedback of Operating Experience to Plant Staff," of NUREG–0 737, "Clarification of TMI Action Plan Requirements."*[5]
- *"The systematic review of plant-specific and industry OE concerning ageing management and age-related degradation confirms that the SLR AMPs are, and will continue to be, effective in managing the aging effects for which they are credited. The AMPs should either be enhanced or new AMPs developed, as appropriate, when it is determined through the evaluation of OE that the effects of*

5. www.nrc.gov/docs/ML0514/ML051400209.pdf

ageing may not be adequately managed. AMPs should be informed by the review of OE on an ongoing basis, regardless of the AMP's implementation schedule."

In evaluating potential ageing effects, a differentiation is made between short-term and long-term effects. Short-term aging effects (e.g. equipment wear-out) tend to be highly predictable and, hence, pose a less challenging analysis problem than long-term aging effects for which there is limited service experience data available to support statistical analysis for trends. An aging effect can be defined as:

- Age-dependent change in a passive system, structure, or component (SSC) performance caused by an active degradation mechanisms or synergistic effects of multiple degradation mechanisms.
- Change in physical or chemical properties resulting from one or more active degradation mechanisms.

The prospects for developing phenomenological ageing models hinge on a clear definition of what constitutes an aging effect as opposed to readily identifiable, well understood temporal changes in equipment performance and the evolution of non-destructive examination techniques. Access to high quality data that reflect several decades of plant operation is one important element in the evaluation of potential ageing effects.

The physical degradation of metallic passive reactor components involves a complex interaction of material properties (e.g. chemical compositions, fracture toughness), operating environment (e.g. local flow conditions, pressure, temperature, water chemistry), and loading conditions. The effects of a certain degradation mechanism can be mitigated or eliminated through the applications of pro-active ageing management, including in-service inspection, stress improvement, and chemical treatment of process media. The CODAP database structure is a reflection of the physics of material degradation, and the database captures the subtleties of the many factors that contribute to material degradation and failure. Therefore, by utilising the tools and techniques for querying the event records that are included in CODAP a basis exists for in-depth evaluation of temporal changes in the failure data, including positive and negative trends in passive component performance.

2.3. How CODAP relates to PSA

Held in December 2004 and hosted by the Korea Institute of Nuclear Safety (KINS) and Korea Atomic Energy Research Institute (KAERI), a workshop [17] was organised to discuss applications of the OPDE database. By answering two basic questions that structured the workshop, valuable insights helped improving the database structure and educated participants:

- How has the OPDE event database been used?
- What can the OPDE event database be used for?

From the outset, the OPDE/CODAP PRG membership has consisted of a multi-disciplinary group, including material scientist, structural integrity engineers, nuclear safety specialists and PSA practitioners. As anticipated, the 2004 workshop produced a very broad list of potential applications, including the following PSA applications:

- internal flood risk assessment; development of pipe failure rates and rupture frequencies for internal flooding initiating event frequencies;
- high-energy line break (HELB) analysis;

- moderate-energy line break (MELB) analysis;
- loss-of-coolant-accident frequency assessment;
- loss of ultimate heat sink analysis;
- significance determination (SD) assessment;
- accident precursor analysis;
- risk-informed in-service inspection (RI-ISI) programme development;
- reliability and Integrity Management (RIM) programme development.

After a protracted inception process lasting several years, a first major application of the database was initiated in 2007 with the objective to produce a “handbook of pipe failure rates and rupture frequencies” (“R-Book”) and to make it available to PSA practitioners. Sponsored by the Swedish Radiation Safety Authority (SSM) and the Nordic PSA Group (NPSAG), the final product was issued in 2010 [18]. Limited to ASME Code Class one and two piping components, the first edition of the proprietary handbook consists of a password protected CD with input/output files, a summary report, theory manual, and system-specific degradation mechanism analyses and relevant operating experience data summaries; total disc size is ca. 75 Mb. Noteworthy is the fact is that data input to this effort was a non-proprietary 2007 version of the OPDE database.

In 2011, the CSNI “Working group on risk” (WGRISK) performed an international survey of the uses of the NEA database project products in PSA [19]. The CODAP-MB supported the survey and the evaluation of survey results. It also participated in the 2012 “Workshop on the Use of OECD Data Project Products in Probabilistic Safety Assessment.”

2.3.1. Database attributes for PSAs

The ability of an event database such as CODAP to support practical applications is closely linked to its completeness and comprehensiveness. Equally important is the knowledge and experience of the analyst in interpreting and applying a database given typical project constraints. Achievement of database “completeness” and “comprehensiveness” is driven by an in-depth understanding of application requirements. The presence of sustained institutional functions that promote the sharing of operating experience data is critical to the database completeness and comprehensiveness.

There are three general types of CODAP database applications: 1) high-level, 2) risk-informed, and 3) advanced applications. Extensive experience now exists with PSA-oriented database applications. This experience has been synthesised into a guideline for how to structure and perform a well-qualified piping reliability analysis [20] The guideline identifies pipe failure event database infrastructure considerations and the requirements on database integrity, nomenclature, damage and degradation knowledgebase, and high-level and supporting requirements for piping reliability analysis.

Data specialisation is an intrinsic aspect of all PSA oriented applications. This encompasses several specific analysis tasks such as review and assessment of applicability of industry-wide service experience data to a plant-specific piping design (e.g. material, dimension, and operating environment), development of *a priori* failure rate distribution parameters reflective of unique sets of piping reliability attributes and influence factors, and Bayesian update of *a priori* distributions. The update may encompass consideration of different degradation mechanism (DM) mitigation strategies.

Five types of metrics are considered in quantitative piping reliability analysis in support of PSA: 1) failure rate, 2) conditional failure probability, 3) inspection effectiveness, 4) DM mitigation effectiveness, and 5) ageing factors. A pipe failure event database cannot support failure rate estimation unless the database also includes extensive piping system design information that yield information on the total piping component population that has produced the failure observations; i.e. exposure term data. Relative measures of piping reliability such as conditional failure probabilities can be generated by querying an event database. The statistical robustness of such relative measures is correlated with the completeness of the event population.

Completeness and comprehensiveness of a service experience database should be ensured through a sustained and systematic maintenance and update process. Completeness is an indication of whether or not all the data necessary to meet current and future analysis demands are available in the database. The comprehensiveness of a service experience database is concerned with how well its structure and content correctly capture piping reliability attributes and influence factors. A clear basis should be included for how to classify "failures."

The inherent latency in structured data collection efforts is on the order of five years. This means that ca. five years could elapse before achievement of high confidence in data completeness. In other words, around 2020 the data mining for the previous decade (2006-2015) would be expected to approach saturation (as in high confidence in completeness of a database). Could "cliff-edge effects" (e.g. small change in input parameter resulting in large results variation) affect an analysis due to database infrastructure factors? It depends on the maturity of inspection programs and our state-of-knowledge concerning certain degradation mechanisms. Considerations about the use of up-to-date failure data is intrinsically assumed to be factored into any analysis task.

The design of and infrastructure associated with a service experience database should be commensurate with application demands and evolving application requirements. In PSA, the completeness of a relevant event population should be validated, either independently or assured through a sustained maintenance effort. The CODAP project has established such an infrastructure.

To achieve the objectives defined for a database, a coding format should be established and documented in a coding guideline [23]. Such a guideline is built on recognised pipe failure data analysis practices and routines that acknowledge the unique aspects of piping reliability in commercial nuclear power plant operating environments. For an event to be considered for inclusion in the database it must undergo an initial screening for eligibility. An objective of this initial screening is to go beyond abstracts of event reports to ensure that only pipe degradation and failures according to a certain work scope definition are included in the database.

The term "data quality" is an attribute of the processes that have been implemented to ensure that any given database record (including all of its constituent elements, or database fields) can be traced to the source information. The term also encompasses "fitness-for-use", that is, the database records should contain sufficient technical detail to support database applications.

Correlating an event population with the relevant plant and component populations that produced these failure events enables the estimation of reliability parameters for input to a calculation case. The information contained in a database must be processed according to specific guidelines and rules to support reliability parameter estimation. A first step in this data processing involves querying the event database by applying data filters that address the conjoint requirements for pipe degradation and failure. These data filters are integral part of a database structure. Specifically, these data filters relate to unique piping reliability attributes and influence factors with respect to piping system design characteristics, design and construction practice, in-service inspection (ISI) and operating environment. A qualitative analysis of service experience data is concerned with establishing the unique sets of calculation cases that are needed to accomplish the overall analysis objectives and the corresponding event populations and exposure terms.

Most, if not all database applications are concerned with evaluations of event populations as a function of calendar time, operating time or component age at time of failure. The technical scope of the evaluations includes determination of trends and patterns and data homogeneity, and assessment of various statistical parameters of piping reliability. Therefore, an intrinsic aspect of practical database applications is the completeness and quality of an event database. Do the results of an application correctly reflect the current field experience, effectiveness of in-service inspection, ageing management, and/or water chemistry programmes?

A typical database application tends to be computationally intense. In order to derive input to PSA model, several calculation cases must be defined to cover the appropriate range of degradation mechanisms and consequences of a pipe failure. Some examples of this are as follows:

- In one example study, a calculation case is defined by a unique set of pipe rupture frequency versus consequence of a certain, well defined magnitude usually characterised by either the size of a pressure boundary breach and/or through-wall flow rate. In support of a HELB analysis a total of 24 calculation cases were defined [21] A failure rate and rupture frequency distribution had to be developed for each case, and, hence a total of 48 parameter distributions were generated in this example study.
- In another example, in developing a location-specific LOCA frequency model for a pressurised water reactor (PWR) [22] a total of 45 unique analysis cases were defined and a total of 462 parameter distributions were generated.

A carefully crafted analysis tool is needed to manage the calculation of piping reliability parameter distributions. With the advancements in analysis methods and techniques follow new challenges in how to review and validate parameter distributions and the propagation of uncertainties. The entire process, from definition of calculation cases, definition of pipe failure database queries, definition of prior distributions, and performing calculations must be traceable and transparent to ensure efficient review processes.

3. Project organisation

This section describes the CODAP project organisation. The operating procedures, a controlled document, describes the project organisation, infrastructure and additional guidelines for the project to achieve the objectives as stated in the CODAP terms & conditions for project operation.

3.1. Responsibilities of project participants

All power for the CODAP resides with the signatory countries bound by a legal agreement "terms & conditions." The management board (MB), formed by the representatives of the signatories (normally the national co-ordinator), holds all the power to make decisions on running the project. Signatories may involve other bodies in their countries by separate operational agreements.

3.2. The NEA Secretariat

NEA is responsible for administering the project according to NEA rules. This means secretarial and administrative services in connection with the funding of the project such as calling for contributions, paying expenses incurred in connection with meetings, the operating agent, and keeping the financial accounts of the project. NEA appoints the project secretariat.

3.3. CSNI

The NEA Committee for Safety of Nuclear Installations (CSNI) acts as the umbrella committee of CODAP. Prior to publication of project-related results, all public domain reports prepared by the CODAP project review group undergo independent review by the CSNI.

The CSNI is an international committee made up of senior scientists and engineers, with broad responsibilities for safety technology and research programmes, and representatives from regulatory authorities. It was set up in 1973 to develop and co-ordinate the activities of the NEA concerning the technical aspects of the design, construction and operation of nuclear installations that affect the safety of such installations.

The main mission of the Working Group on Integrity and Ageing of Components and Structures (WGIAGE) is to advance the current understanding of those aspects relevant to ensuring the integrity of structures, systems and components (SSC) under design and beyond design conditions. In addition, the main mission is to provide guidance in choosing the optimal ways of dealing with respective challenges to operating and new nuclear power plants as well as other nuclear facilities, and to make use of an integrated approach to design, safety and plant life management. In this context, CODAP is improving the quality of data obtained relating to piping and non-piping passive component degradation experience, and, in turn, rendering such data more useful in predicting structural component degradation and failure.

3.4. CODAP management board

The CODAP management board (MB) runs the project, with assistance from the NEA project secretary and the operating agent. The MB meets at least once per year. The MB responsibilities include but are not limited to the following types of decisions:

- secure the financial and technical resources necessary to carry out the project;
- nominate the CODAP project chairperson;
- define the information flow (public information and confidential information);
- approve the admittance of new members;
- nominate project task leaders (lead countries) and key persons for the MB tasks;
- define the priority of the task activities;
- monitor the progress of the project and task activities;
- approve and monitor the work of the operating agent and quality assurance.

3.5. The operating agent

To assure consistency of the event database data contributed by the national co-ordinators the project operates through an operating agent. The operating agent, Sigma-Phase Inc., verifies whether the event information provided by the national co-ordinators complies with the CODAP coding guidelines (CG); CODAP-PR01 [23] It also verifies the completeness and accuracy of the data and assigns the quality index jointly with the respective national co-ordinator who has provided such data.

The CODAP applications handbook (CODAP-AH) [24] includes guidelines for extracting insights from the event database about material degradation, including failure trends and event population data for input to statistical parameter estimation tasks. It includes descriptions of the data processing steps that are needed to facilitate statistical evaluations of operating experience with metallic piping components and non-piping passive components. Whereas the CODAP coding guideline (CODAP-CG) defines database structure and data submission requirements, the CODAP-AH includes guidelines for creating database queries and associated data processing steps. CODAP-AH is a companion document to CODAP-CG.

4. CODAP event database

The CODAP event database is a web based, relational database consisting of ca. 100 uniquely defined data fields. It is a mixture of free-format fields for detailed narrative information, fields defined by drop-down menus with key words (or data filters) or related tables, and hyperlinks to additional background information (e.g. photographs, root cause evaluation reports). The "related tables" include information on material, location of damage or degradation, type of damage or degradation, system name, safety class, etc. At the end of the second term the CODAP event database included ca. 4 900 records on degraded and failed metallic piping and non-piping passive components.[6] Section four presents the scope of the event database and summarises the database structure and main features of the online event database.

4.1. Scope of the event database

The event database scope and structure, database field definitions and data input requirements are defined in the coding guideline, which is central to the project, including database maintenance, data validation and quality control. The database design has benefitted from a multidisciplinary approach involving chemistry, metallurgy, structural integrity and PSA expertise.

The CODAP event database collects service experience data on the full range of degraded conditions, from "precursors" to major structural failures involving metallic piping components and non-piping metallic passive components. According to the IAEA safety glossary [25], a passive component is defined in the following way:

- A passive component is "component whose functioning does not depend on an external input such as actuation, mechanical movement or supply of power.
  - A passive component has no moving part, and, for example, only experiences a change in pressure, in temperature or in fluid flow in performing its functions. In addition, certain components that function with very high reliability based on irreversible action or change may be assigned to this category.
  - Examples of passive components are heat exchangers, pipes, vessels, electrical cables and structures. It is emphasised that this definition is necessarily general in nature, as is the corresponding definition of active component.

6. At the end of the first project term (2011-2014) the database included close to 4 700 failure records.

- Certain components, such as rupture discs, check valves, safety valves, injectors and some solid state electronic devices, have characteristics which require special consideration before designation as an active or passive component."

With the above definition as a basis and building on the OPDE and SCAP-SCC project experience, recent operating experience and associated regulatory actions, the Project Review Group made further refinements and specialisations to arrive at a scope definition as summarised in Table 4.1. Consistent with the Operating Procedures, the scope definition is revisited and periodically updated. In Table 4.2, the column "Metallic, Non-Piping Passive Components" captures the BWR and PWR internals as documented and evaluated in IAEA-TECDOC-1471 [26] and IAEA-TECDOC-1119 [27], respectively.

Table 4.1. Scope of CODAP Event Database[7]

METALLIC PASSIVE COMPONENTS				
PIPING COMPONENTS			NON-PIPING PASSIVE COMPONENTS	
Piping - Below Ground / Concealed			Reactor Pressure Vessel (RPV)	
	Pipe - Concrete Encased Pipe			Vessel Head Penetration - PWR
	'Bonna' Pipe			Bottom Mounted Instrument (BMI) Nozzle - PWR
	Pipe - External Coating			RPV Head Thermocoupling (T/C) Housing - PWR
Ex-RPV - In-Plant Piping (Accessible)				RPV Head T/C Nozzle - PWR
	Pipe - Base Metal		Pressurizer	
	Pipe - Cement Lined			Pressurizer Heater
	Pipe - Epoxy Lined			Pressurizer Manway Diaphragm Plate
	Pipe - Rubber Lined			Pressurizer Nozzle
	Bend			Pressurizer Relief/Safety Valve Nozzle
	Blind Flange		RPV Internals	
	Branch-Connection - Socket Welded			Baffle-Former Assembly Bolt - PWR
	Branch-Connection - Stub-in Weld			Core Shroud Access Hole Cover Weld
	Cap / End-Cap			Core Shroud Head Bolt - BWR
	Elbow			Core Shroud Weld - BWR
	Elbow - Long-Radius			Core Shroud Tie Rod - BWR
	Elbow - 45-Degree			Core Shroud Support - BWR
	Elbow - 90-Degree			Core Spray Sparger - BWR
	Expander			In-Core Instrument Tube
	Expansion Joint			Jet Pump Hold-Down Beam
	Fitting			Jet Pump Riser
	Mixing Tee			Jet Pump Support Brace
	Reducer			Steam Dryer - BWR
	Socket Weld		Pump	
	Tee			Pump Casing
	Weld - Butt Weld			RCP Turning Vane Bolt
	Weld - Dissimilar Metal Weld		Valve	
	Weld - Girth Weld (Full Penetration Weld)			Valve Body
	Weld			

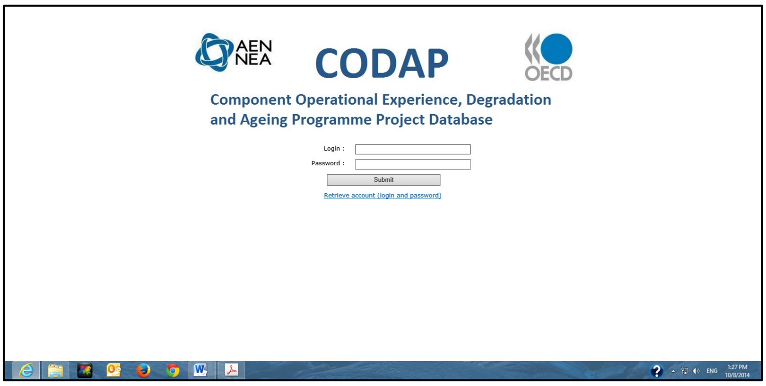
7. See Reference [23] for a complete listing of passive component types in CODAP.

In CODAP the term "failure" covers the full spectrum of degraded conditions, from rejectable flaws requiring repair or replacement to major structural failures. As an example, ASME Section XI, Article IWA-3000 [28] defines acceptance standards for flaws that are discovered during non-destructive examinations (NDEs). Flaws determined to be rejectable (i.e. not fit for continued operation) according to relevant NDE code are required to be repaired or replaced.

4.2. Working with the event database

CODAP is a SQL (relational) database.[8] The data entry is managed via input forms, tables, roll down menus and database relationships. Figure 4.1 shows the online opening screen and Figure 4.2 shows the online main work area. The online version is accessible via a secure server at the Nuclear Energy Agency headquarters. User names and passwords are provided by NEA IT-department upon written request by the national co-ordinator. The online version includes help menus. The project members' work area includes a FAQ area as wells as tutorials on data input, database interrogation.

Figure 4.1. CODAP Database Opening Screen



The Main Menu page is displayed in Figure 4.3. There are a total of seven sub-meus: 1) Records, 2) Create, 3) Search, 4) Statistics, 5) Delete, 6) Export, and 7) Help. The online user-interface consists of the following tabs:

- **Records.** This menu provides an overview of the entire database content and provides basic information such as plant name, event date, when records was first created, when a record was last updated as well as QA status (i.e. draft, validation pending, validation completed. This menu supports database management.
- **Create.** The data input format is equivalent to that presented in Section five. Each record added to the database is assigned a unique "EID/key number". "create new record" opens the data input form. A partially filled in form can be saved and retrieved and the information modified as needed. Use the "tab key" to move from form one field to another. The user may append attachments (e.g. drawings, photographs, PDF files) to a data record.

8. According to the American National Standards Institute (ANSI), SQL is the standard language for relational database management systems.

- **Search.** This tab includes two areas: 1) Search Criteria, and 2) Result Column. The entire database can be searched and filtered by a very large number of attributes.
- **Statistics.** This tab supports basic database queries. The query results can be exported to a local computer as a CSV (character-separated value) file or XML (extensible mark-up language) file. Selecting "CSV" automatically generates an Excel-file with the tabular query results. Additional post-processing of data may be performed on a local computer.
- **Delete.** This tab enables deletion of individual records or a set of records. Only NEA-IT and the operating agent can execute this function and only upon pre-approval by the respective national co-ordinator.
- **Export.** Downloading records from the online version is straightforward. Pressing the "export" button returns a listing of all records. Selected records or the entire database can be exported to a local computer. The online version creates a zip-file ("export file") that can be opened or saved to a local disc. The data records are converted to a XML file format that is compatible with Microsoft® Office programs (e.g. Access, Excel, Word).
- **Help.** This tab provides user support. It includes abbreviated versions of the coding guidelines and applications handbook.

Figure 4.2. CODAP Main Menu

4.3. Data submissions 2015-2017

Respective national co-ordinators are responsible for data submissions. The preferred method for submitting new data to the database is via the web-based interface. Data submissions may also be handled by e-mail with event information attached in Microsoft® Access, Excel or Word file format. The CODAP "terms and conditions" and "operating procedures" define the expectations regarding data submissions. Respective national co-ordinators have overall responsibility for data submissions. Organised by MB member country, Table 4.2 is a summary of data submissions (2002 up to now).

Table 4.2. Data Submission Summary

Project Members		Number of Data Submissions through CY 2014[9]	Data Submissions 2015-2017
Country	**Status**		
BE - Belgium	Member of OPDE Terms 1 & 2 (2002-2008)	8	N/A
CA - Canada	Member since 2002	187	37
CH - Switzerland	Member since 2002	95	4
CZ – Czech Republic	Member since 2002	25	6
DE - Germany	Member since 2002	350	8
ES - Spain	Member since 2002	50	6
FI - Finland	PRG Member through end of 2014[10]	56	N/A
FR - France	Member since 2002	148	21
JP - Japan	Member since 2002	287	--
KR – Korea (Republic of)	Member since 2002	69	12
MX – Mexico	Member of SCAP-SCC Project (2006-2010)	3	N/A
SE - Sweden	PRG Member through end of 2014	365	N/A
SK – Slovak Republic	Member since 2011	5	5
TW – Chinese Taipei	Member since 2011	15	11
US – United States of America	Member since 2002	3035	113
CODAP Event Database Content - No. of Records:		4698	223 (4921)

4.4. High-level database summary[11]

In its present form the online version of the database facilitates data submissions, various search and sort functions, and database interrogation functions. The latter are performed in the "statistics" area of the database. There are four database application facilities: 1) Records Management, 2) SEARCH, 3) Database Query Function, and 4) Export Function. The data queries are performed across 28 fields and with the aid of 668 pre-defined data filters[12] (or keywords). An abbreviated, high-level summary of the CODAP event database content is given by selected queries as illustrated in Figures 4.3 through 4.18. Additional operating experience insights are addressed in Section 4.5 and Section 5.

9. The starting point was an in-kind contribution by the Swedish Radiation Safety Authority (SSM); a SQL database on piping failures consisting of 2 291 failure records for the period 1970-1998.

10. Returning as a CODAP PRG member for the third project term (2018-2020)

11. Data collection is ongoing for the period 2012 to date. This summary of the database content is current as of 31 December 2017.

12. For example, plant system, degradation mechanism, pipe size, type of material and piping component type (e.g. bend, 90-degree elbow, reducer, socket weld), etc.

4.4.1. Ageing effects

The CODAP event database structure includes two fields that address the age of a passive component at the time of an observed failure. A "default age" is calculated, which is the equivalent to the number of reactor operating years (from initial reactor criticality to the time of an observed passive component failure) and the corresponding effective full power years (EFPY). According to the coding guideline, it is expected that rather than using a calculated default value an effort be made to assess the actual age of a failed passive component. Such information is normally included in ASME Form NIS-2 ("Owner's Report for Repair/Replacement Activity") or equivalent document. In theory it is therefore feasible to generate passive component failure rates as a function of component age.

An abbreviated, high-level summary of age-dependent passive component failure rates is given by selected queries as illustrated in Figures 4.19 and 4.20. Figure 4.19 shows the number of pipe failures per nuclear reactor and as a function of all "categories" (e.g. safety class and system) and "types" (e.g. material, nominal diameter and degradation mechanism). Restricted to U.S. operating experience, Figure 4.20 shows the number of service water piping failures per BWR unit and PWR unit and as a function of age, respectively.

Figure 4.3. Evolution of the Event Database – Data Submissions by Event Date[13]

OPDE

CODAP 1st Term

CODAP 2nd Term

PRG Member	Number of Event Data Record Submittals by Event Date - As of 31-December-2017																
	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017	Total
BE	1	--	--	--	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	1
CA	--	11	3	3	16	22	10	3	10	9	4	2	5	5	6	3	112
CH	1	4		6	3	5	1	7	2	2	4	--	2	1	--	--	38
CZ	1	1	2	--	1	--	1	--	1	--	1	1	1	3	--	--	13
DE	17	9	21	21	19	20	7	17	5	3	4	4	1	2	--	--	150
ES	1	4	3	8	1	--	--	1	--	2	--	2	1	3	1	--	27
FI	--	1	--	--	1	--	4	--	1	1	3	--	1	N/A	N/A	N/A	12
FR	7	4	7	4	6	5	5	3	2	6	3	1	2	4	--	1	60
JP	111	39	14	6	14	10	13	1	4	1	--	--	--	--	--	--	213
KR	7	1	--	1	--	3	2	2	2	4	3	5	5	--	--	--	35
SE	1	7	4	1	1	1	--	--	--	1	--	--	--	N/A	N/A	N/A	16
SK	1	1	1	--	--	--	--	--	2	--	--	2	--	--	--	--	7
TW	--	1	1	--	1	--	1	2	--	4	3	2	2	5	--	--	22
US	56	74	63	68	29	22	26	32	28	28	36	37	37	32	35	18	621
	203	157	119	118	92	88	70	68	57	61	61	56	57	55	42	18	1327

SCAP-SCC

13. Note the differences between Figure 4.3 and Figure 4.4. In Figure 4 the database content is organised by the calendar year in which an event occurred. Figure 4.4 is summary of data submissions made in a given calendar year. For a given country the yearly data submissions may consist of events that have occurred in different time periods; e.g. submissions made in 2017 may include events that occurred in 2012, 2013, etc.

Figure 4.4. Evolution of the Database – Data Submissions by Calendar Year[14]

CODAP DATA SUBMISSIONS BY CALENDAR YEAR (CY)								
PRG Member	**CY 2011**	**CY 2012**	**CY 2013**	**CY 2014**	**CY 2015**	**CY 2016**	**CY 2017**	**Total as of 31-December-2017**
Canada	No data submittals - the work scope focused on finalizing DB structure, developing & implementing the Web-Based Event Database	1	25	8	18	6	13	70
Chinese Taipei		--	6	9	2	6	3	26
Czech Republic		--	1	3	--	6	--	10
Finland		--	--	9	1	N/A	N/A	10
France		--	--	17	5	10	6	38
Germany		8	4	10	3	1	4	30
Japan		--	1	1	--	--	--	2
Korea (Republic of)		--	17	1	1	7	4	30
Slovak Republic		1	--	4	--	5	--	10
Spain		--	3	2	--	3	3	11
Sweden		--	--	--	1	N/A	N/A	1
Switzerland		1	5	1	1	--	3	11
USA		33	61	56	45	39	29	263
	Totals:	44	123	121	77	83	65	513

14. Note the differences between Figure 4.3 and Figure 4.4. In Figure 4.3 the database content is organised by the calendar year in which an event occurred. Figure 4.4 is summary of data submissions made in a given calendar year. For a given country the yearly data submissions may consist of events that have occurred in different time periods; e.g. submissions made in 2017 may include events that occurred in 2012, 2013, etc.

Figure 4.5. Cumulative Number of Failure Records

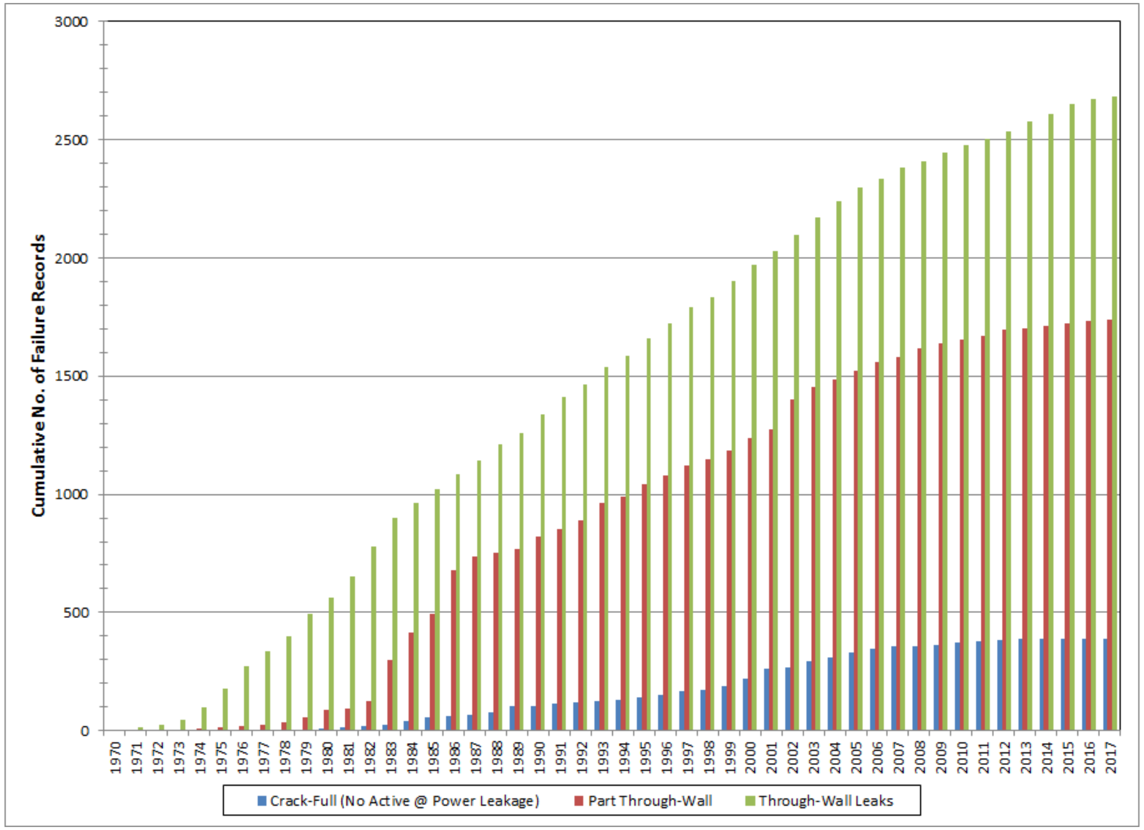
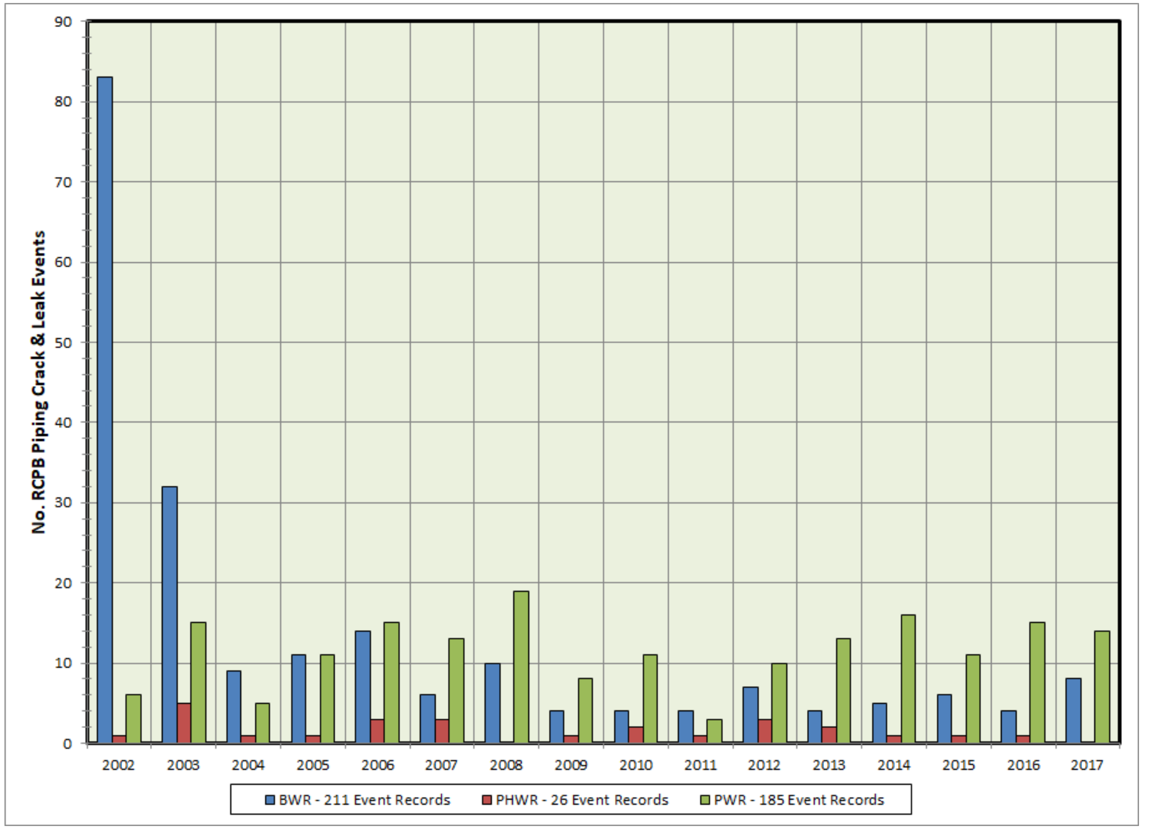
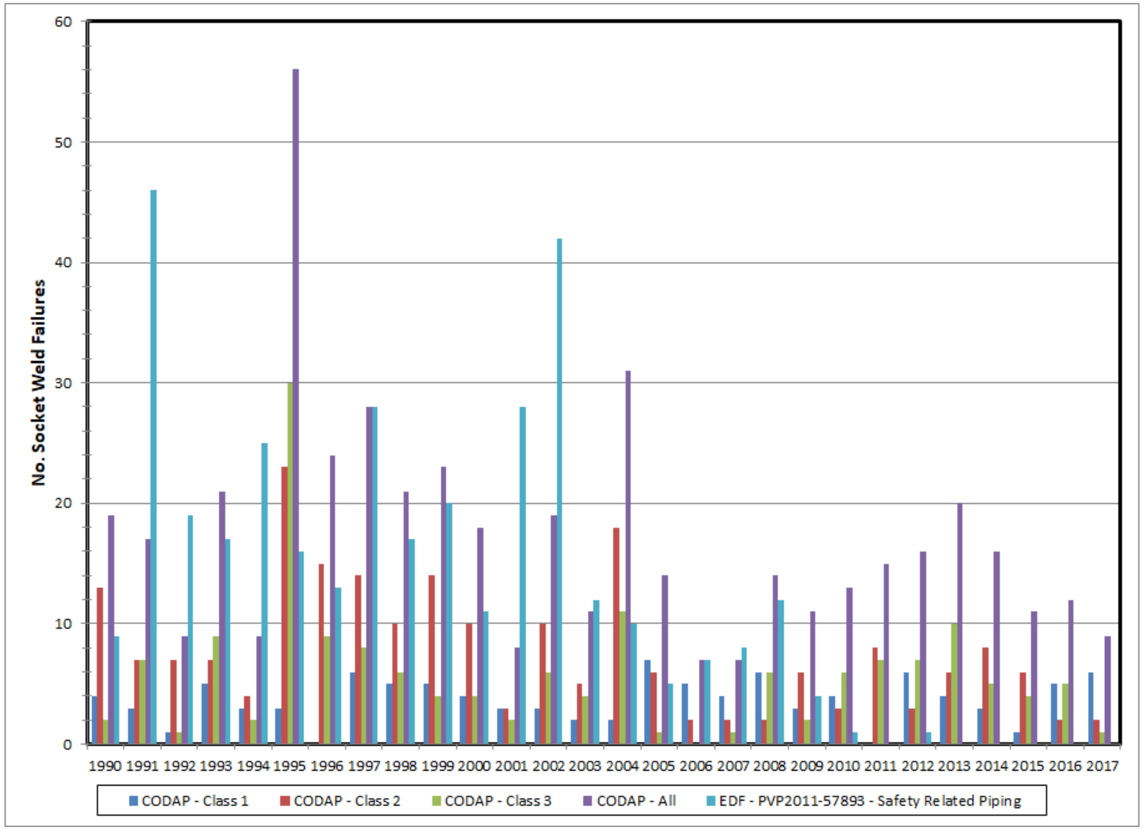


Figure 4.6. RCPB Crack and Leak Event Data[15]



15. In this chart the event data is organised by the year in which an event occurred.

Figure 4.7. Socket Weld Failure Data[16]



16. The EDF-data set is extracted from a technical paper presented at the ASME2011 Pressure Vessels and Piping Conference [29].

Figure 4.8. Operational Impact of Pipe Failure by Pipe Size (NPS)

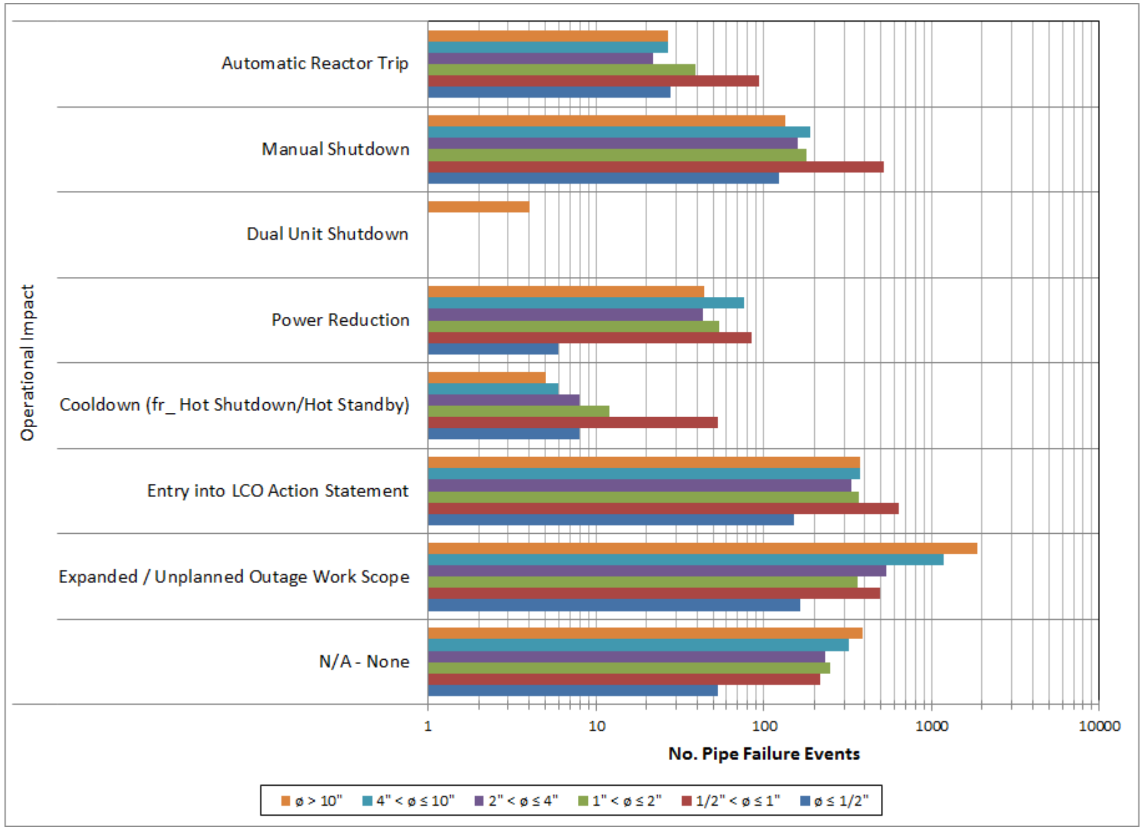


Figure 4.9. Pipe Failure by Fatigue Mechanism

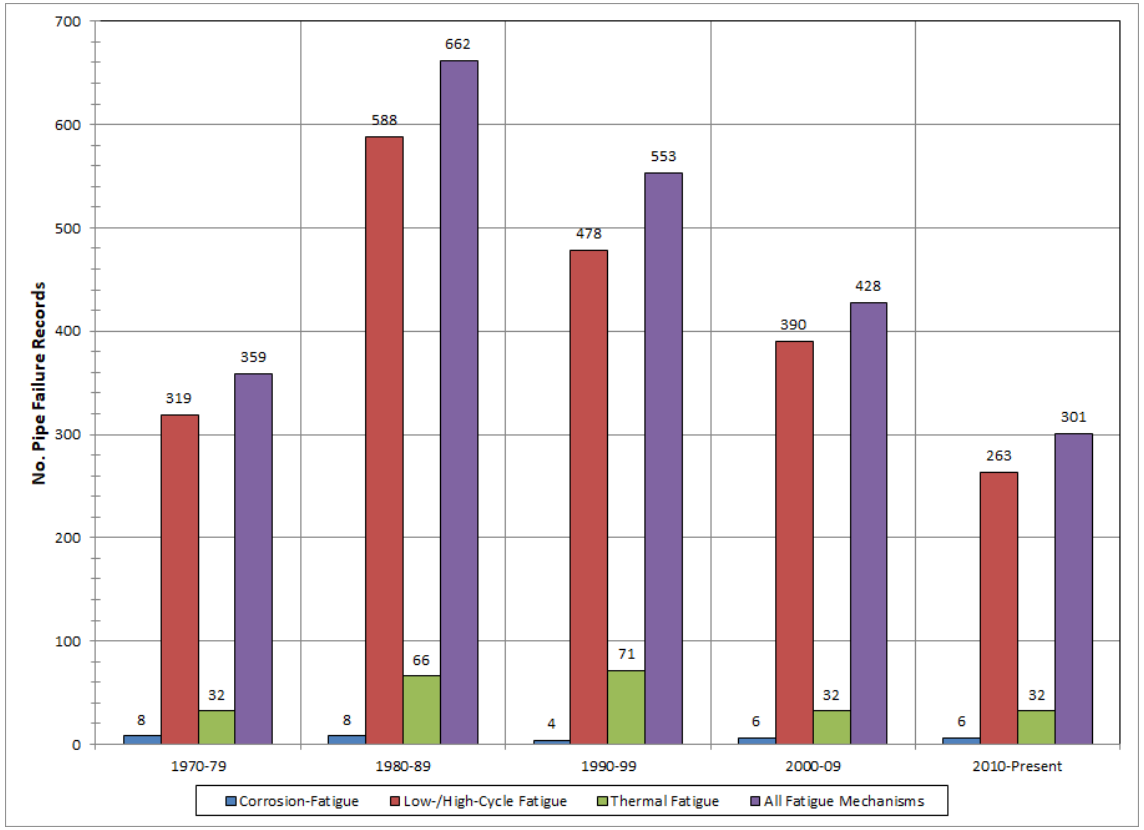


Figure 4.10. PWSCC Events in CODAP

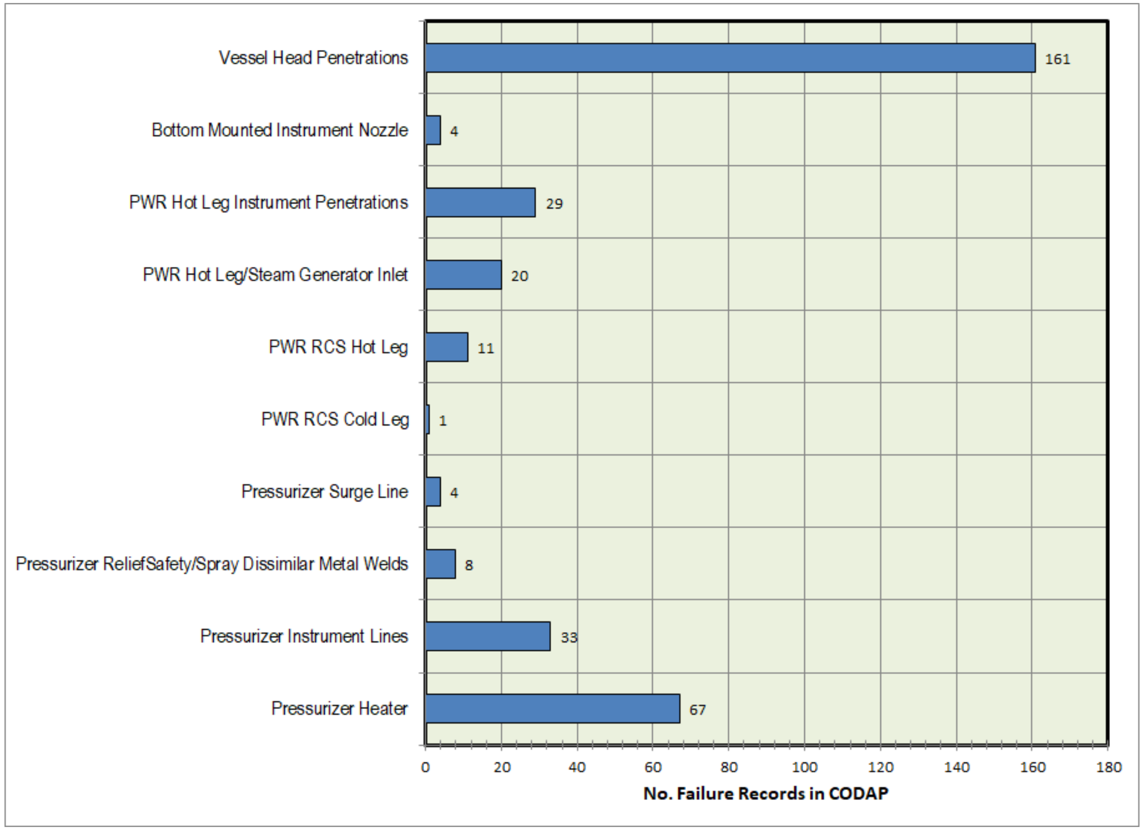


Figure 4.11. Database Content by Passive Component Type

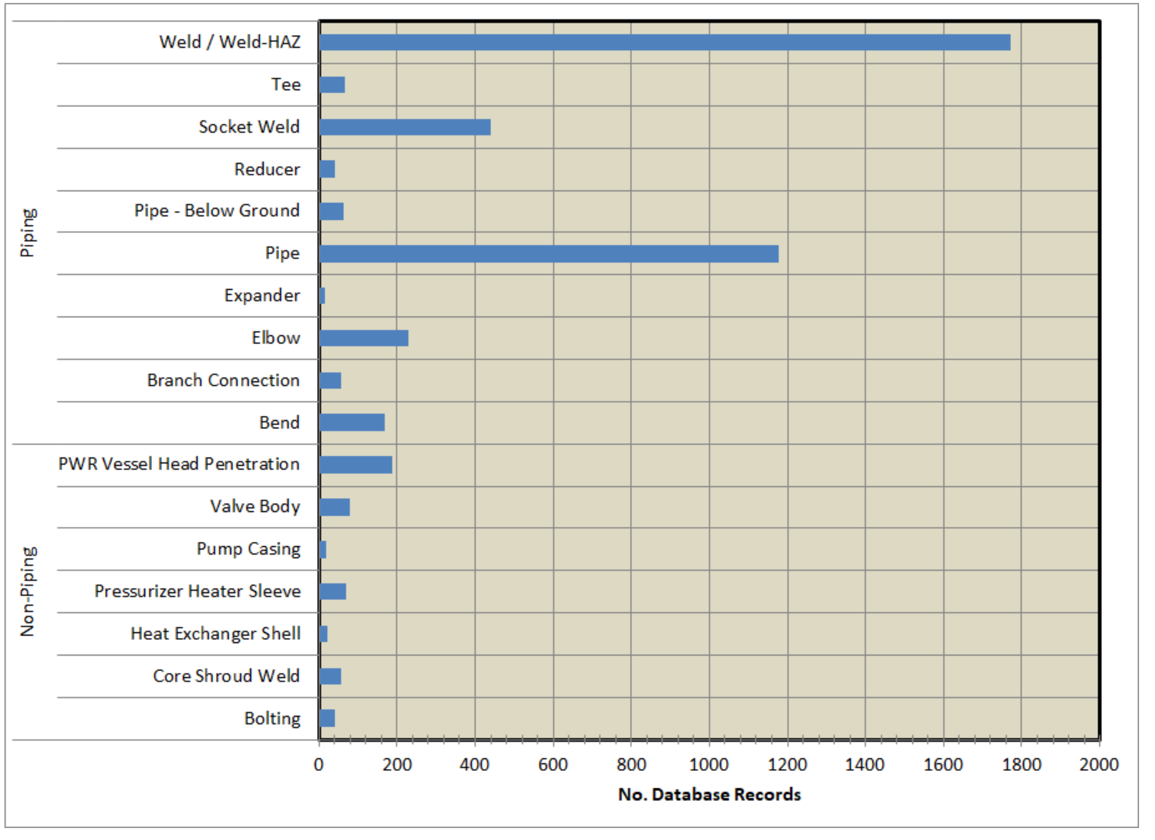


Figure 4.12. Database Content by Material Type and Mode of Failure

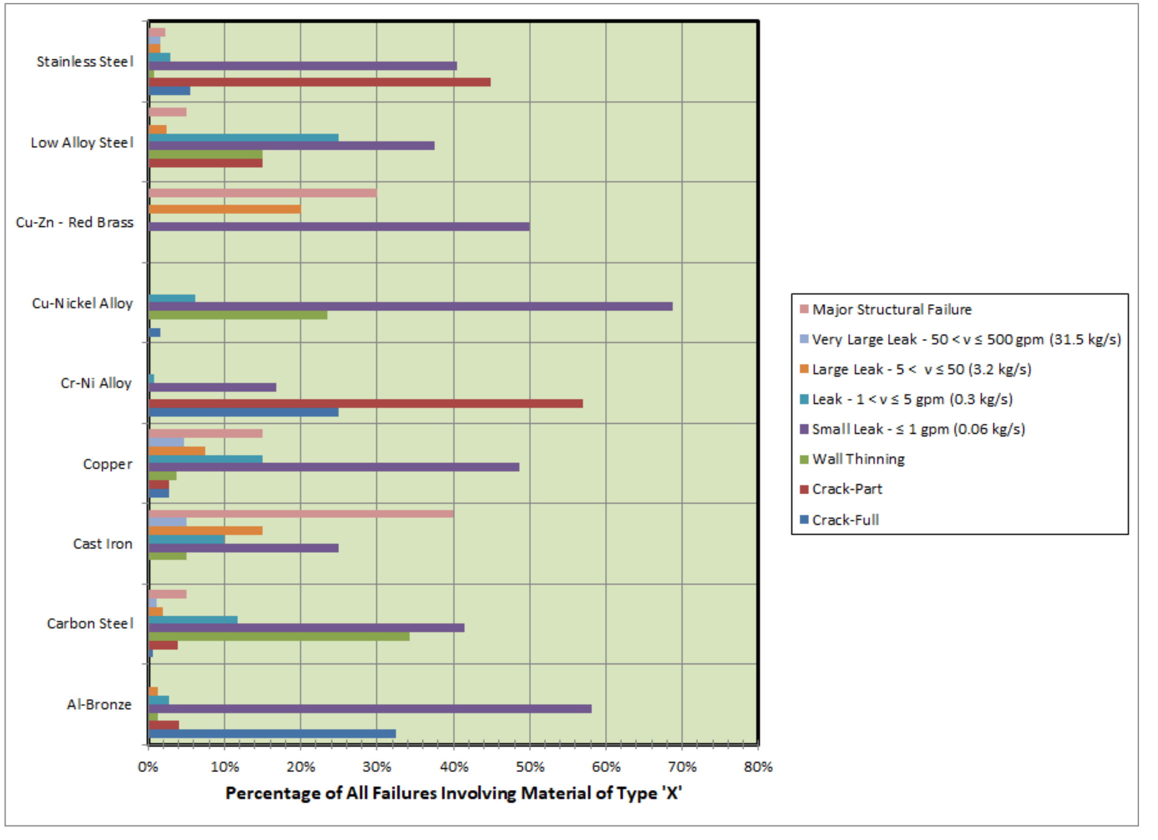
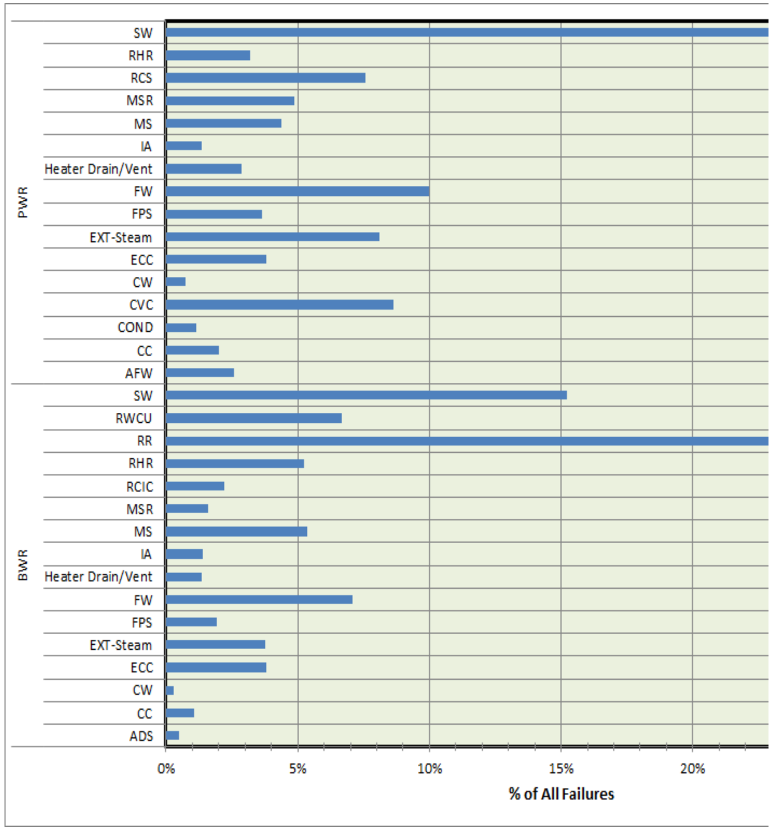


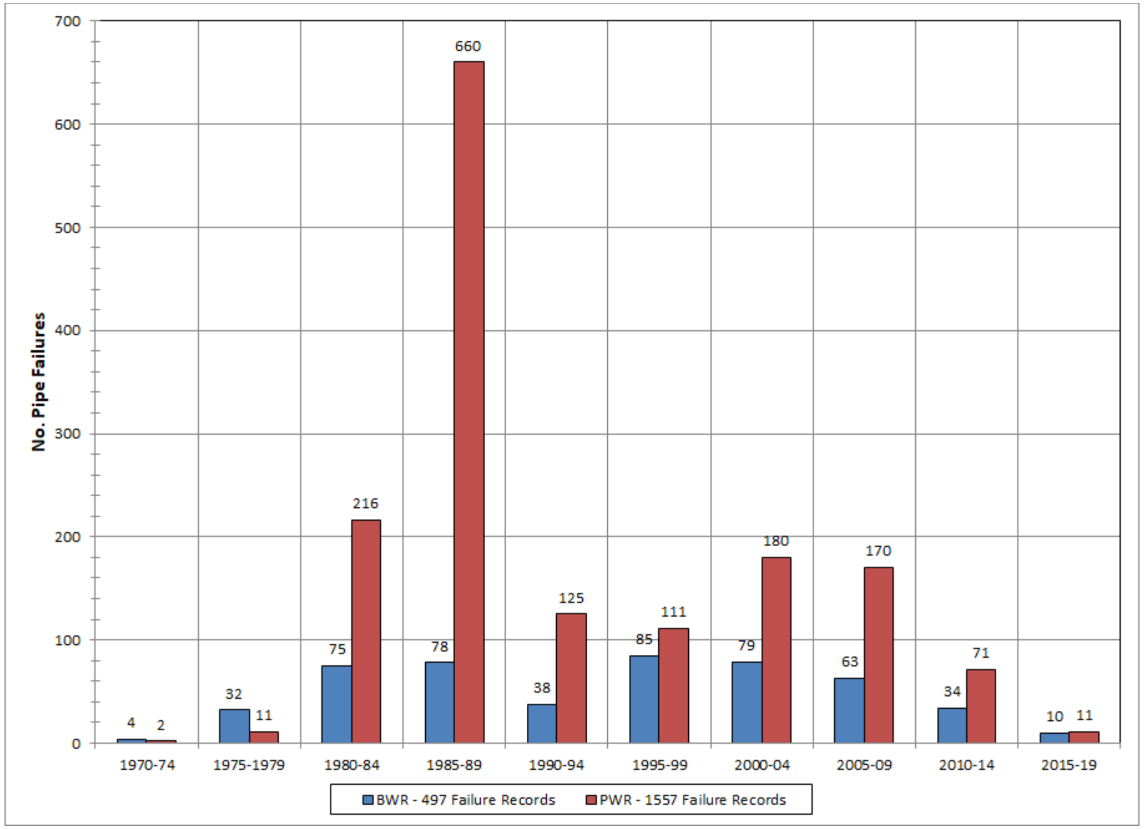
Figure 4.13. Light Water Reactor Piping OperatingExperience by Plant System



Legends:
ADS = Automatic Depressurisation System
AFW = Auxiliary Feedwater
CC = Component Cooling
COND = Condensate
CVC = Chemical & Volume Control
CW = Circulating Water
ECC = Emergency Core Cooling
HPCI & LPCI for BWR
HPSI & LPSI for PWR
HPCI = High Pressure Coolant Injection
HPSI = High Pressure Safety Injection
LPCI = Low Pressure Coolant Injection
LPSI = Low Pressure Safety Injection
FPS = Fire Protection Water System
FW = Feedwater
IA = Instrument Air
MS = Main Steam
MSR = Moisture Separator Reheater
RCIC = Reactor Core Isolation Cooling
RCS = Reactor Coolant System
RHR = Residual Heat Removal
RR = Reactor Recirculation
RWCU = Reactor Water Cleanup
SW = Service Water

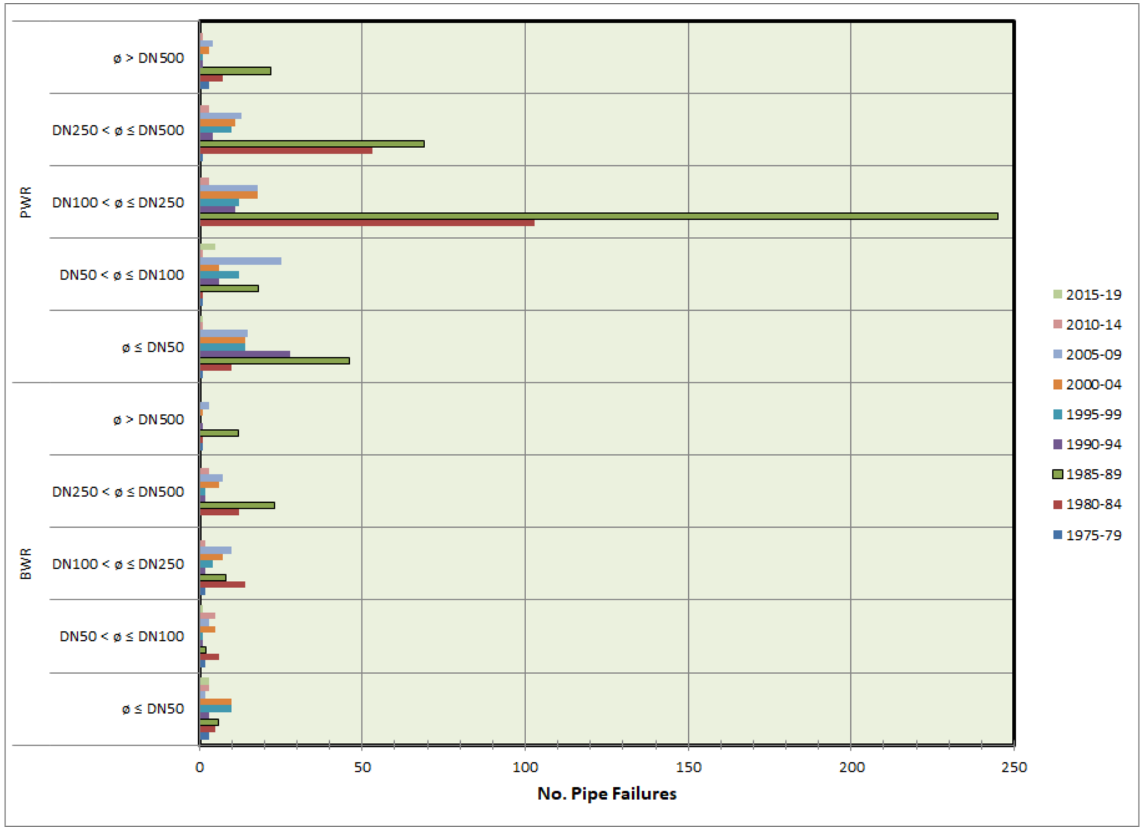
See Appendix C for a plant system cross-reference table.

Figure 4.14. Pipe Failures Attributed to Flow-Accelerated Corrosion – Part 1[17]



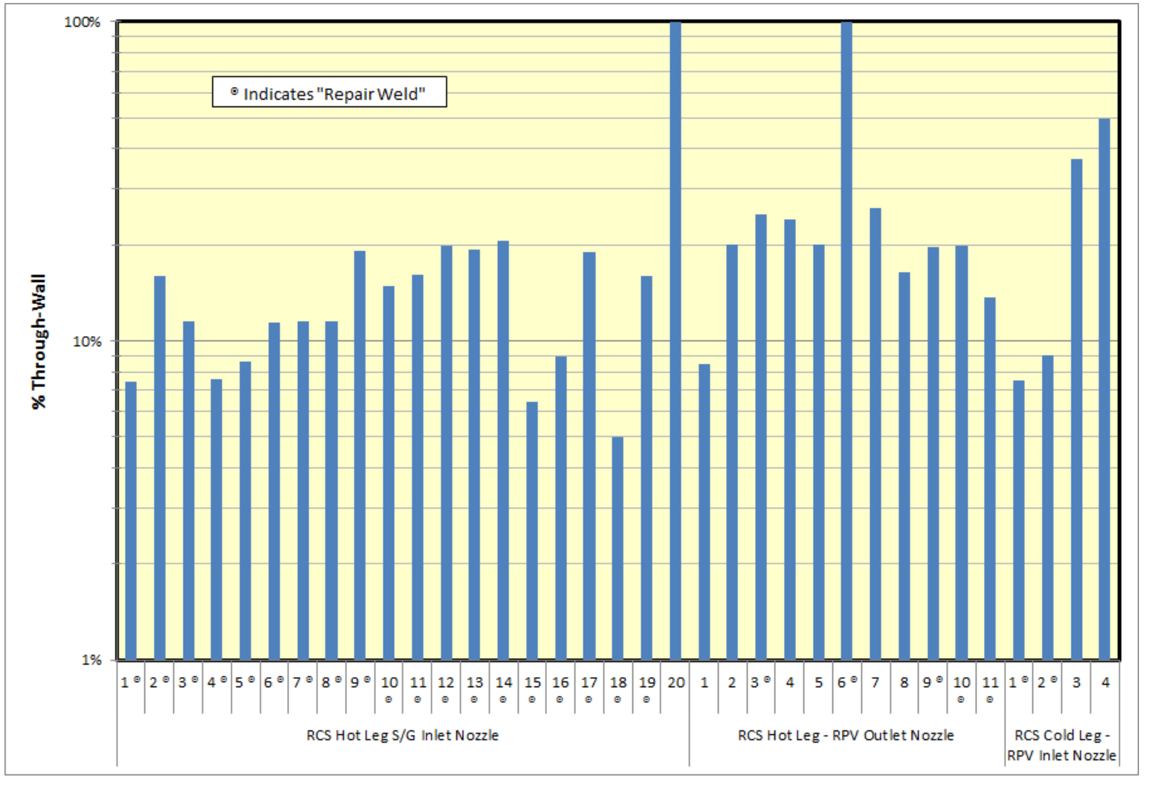
17. Refer to Reference [30] for an explanation of the "1985-1989" peak.

Figure 4.15. Pipe Failures Attributed to Flow-Accelerated Corrosion – Part 2[18]



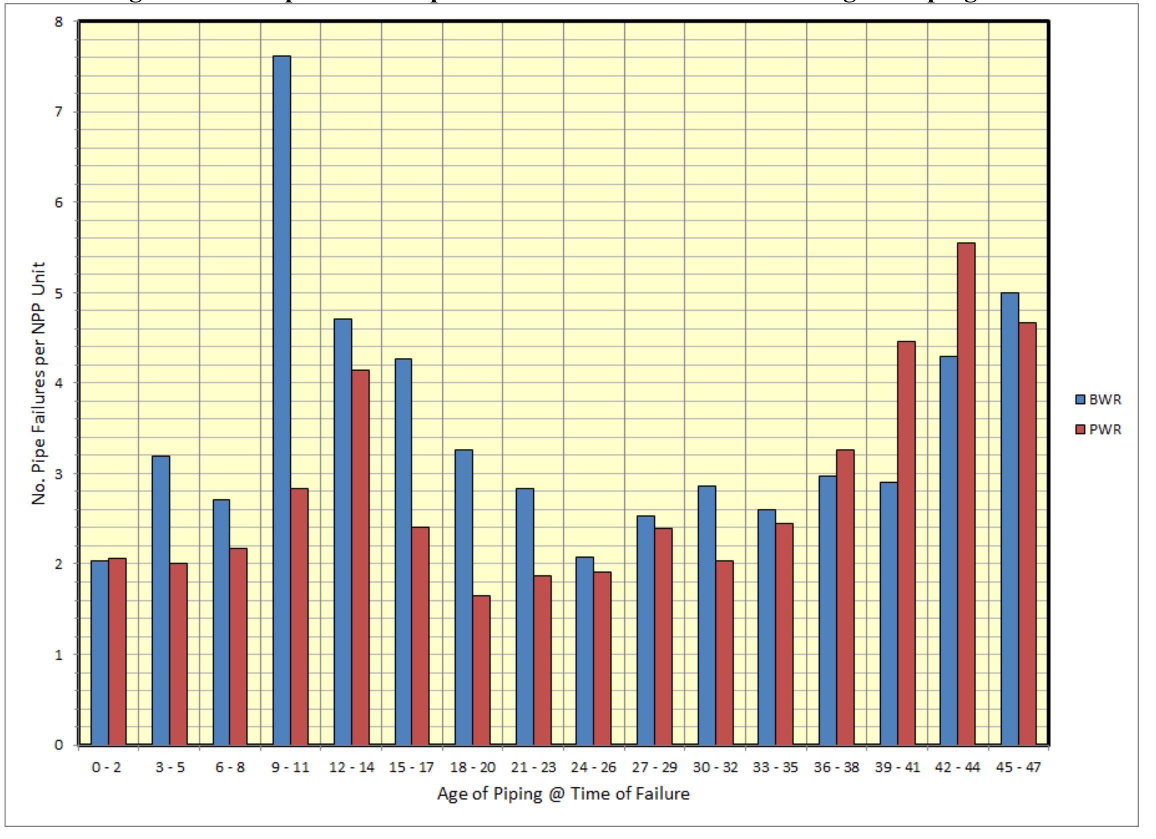
18. Refer to reference [30] for an explanation of the "1985-1989" peak.

Figure 4.16. Pressurised Water Reactor PWSCC Operating Experience Data[19]



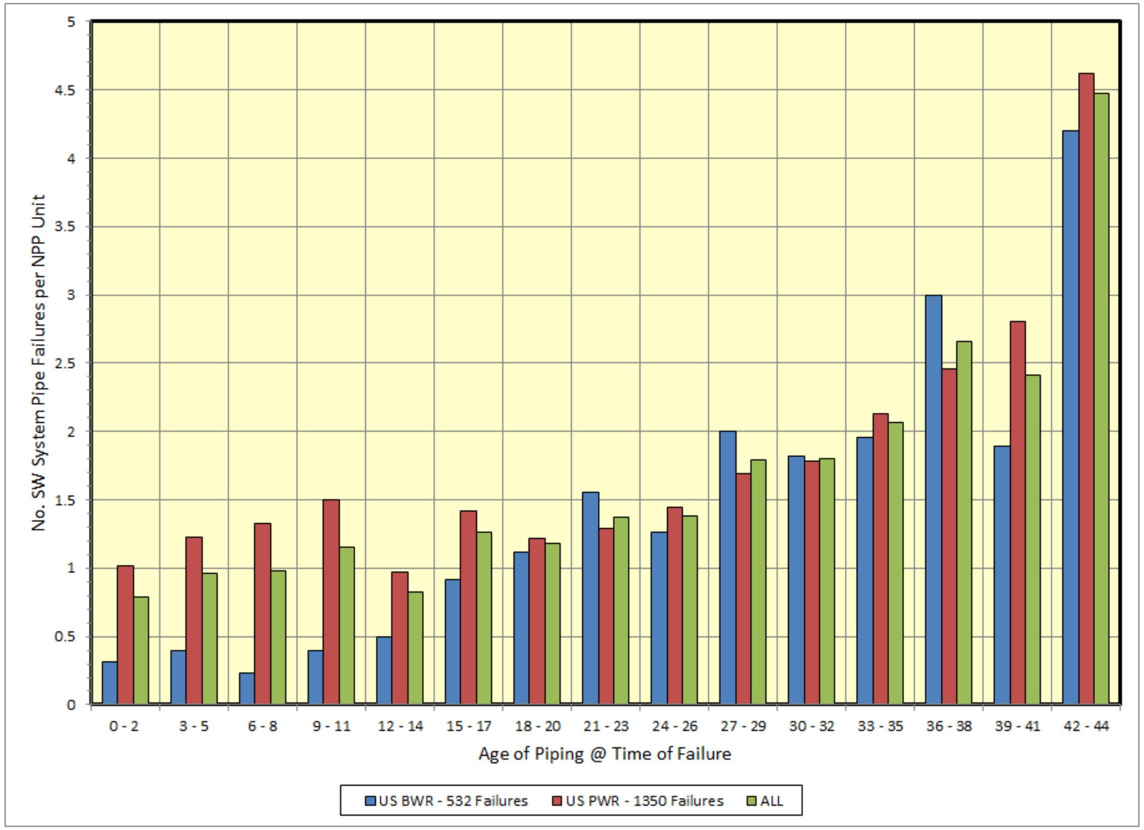
19. This chart is a summary of specific PWSCC events. As one example, to date there have been 20 events (one through 20) involving PWSCC in Reactor Coolant System Hot Leg Steam Generator inlet bimetallic welds. Full descriptions are found in the CODAP event database and by using the following query definition: PWR – RCS Hot Leg – Bi-metallic Weld – PWSCC – Crack Depth.

Figure 4.17. A High-Level Perspective on Pipe Failure Rate as Function of the Age of Piping at Time of Failure[20]



20. This chart displays the global BWR and PWR operating experience data. It reflects all the different environmental degradation mechanisms that have produced failures. To fully appreciate the progression in this operating experience some basic knowledge about the specifics of these degradation mechanisms is required. As one example, the peak for BWRs and the interval nine to 11 years reflects the very high IGSCC incident rate observed during the mid-1970s to mid-1980s. IGSCC has since been largely mitigated through various ageing management programs (e.g. enhanced primary water chemistry control, use of improved materials, stress improvement).

Figure 4.18. A High Level Perspective on Ageing Effects on Service Water System Piping



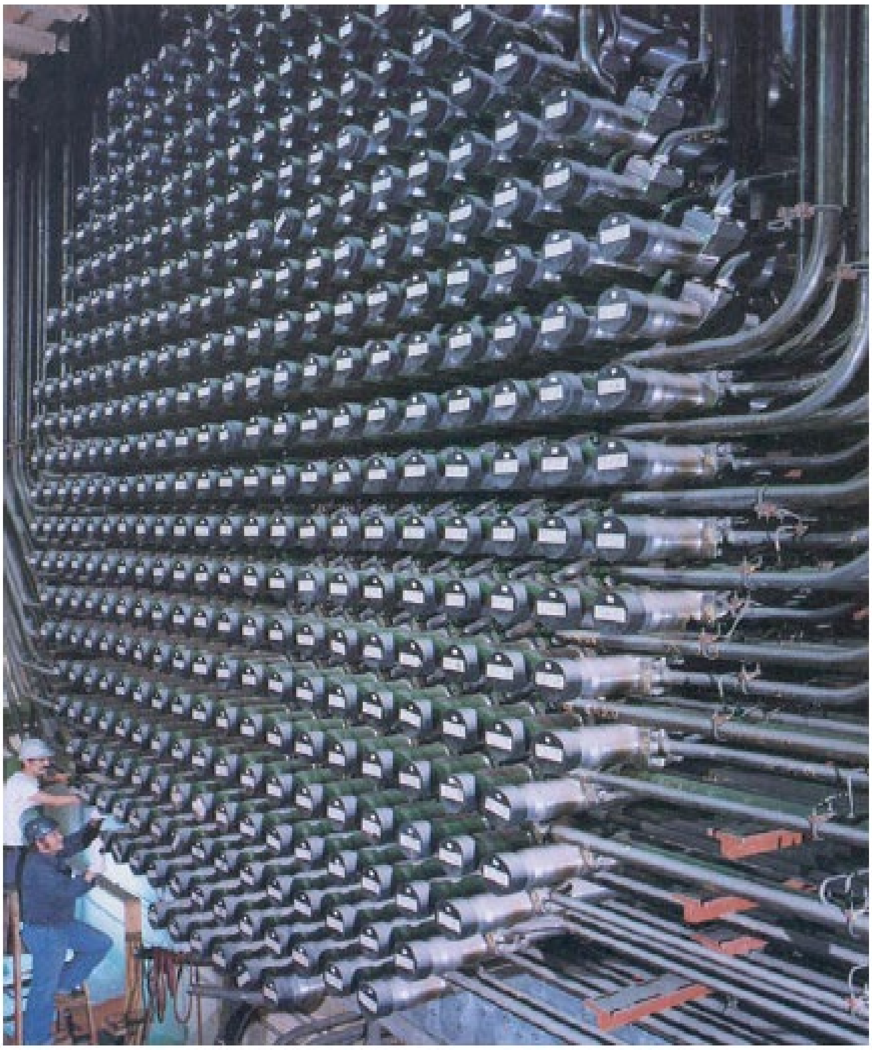
4.5. 2015-2017 Selected material degradation issues

Included in this section are country-specific overviews of selected material degradation issues, including significant operational events, the corresponding regulatory actions and relevant R&D programmes.

4.5.1. Canada

Feeder pipes are part of the CANDU (Canadian Deuterium Uranium) reactor primary heat transport (PHT) system, as shown in Figure 4.19. An essential function of the feeder pipe is to transport heavy water (D2O) coolant to and from fuel channels (FCs) to cool the fuel bundles in the pressure tubes. The feeder pipes are designed as nuclear class one piping and are subject to in-service inspections in accordance with Canadian Standards Association (CSA) N285.4, periodic inspection of CANDU nuclear power plant components [31].

Figure 4.18. Feeders on the Reactor Face in the Feeder Cabinet



The total number of feeders per reactor varies from 760 (i.e. 380 inlet feeders and 380 outlet feeders) to 960 (i.e. 480 inlet feeders and 480 outlet feeders) depending on the power rating of the reactor (i.e. 600 MWe to 900 MWe). Each feeder contains one or two tight radius bend(s)/elbow(s) in the lower portion of the feeders. In addition, each feeder has large radius bends and a swaged reducer for accommodating changes in pipe diameters, and may contain pipe fittings such as a flow measurement element, etc. In short, the various feeder components are connected by welds. Feeders were originally fabricated from a SA-106 Grade B carbon steel piping with the size in diameter ranging from 38.1 mm (1.5″) to 101.6 mm (3.5″) with pipe Schedule 80 for extra wall thickness compared to pipe Schedule 40 for standard thickness used for non-nuclear power piping systems.

The design temperature for inlet feeders ranges from 251oC (483oF) up to 279oC (535oF) and the design pressure ranges from 10.5 MPa (1523 psig) to 12.7 MPa (1835 psig) depending on the design or power rating of the reactor. The design temperature for outlet feeders ranges from 299oC (571oF) up to 321oC (609oF) and the design pressure ranges from 10.0 MPa (1451 psig) to 11.3 MPa (1635 psig) [32].

Feeders are nuclear class one piping components which transport heavy water (D2O) coolant to remove heat from nuclear fuel bundles located inside the pressure tubes. Failure of the feeders could result in radiological conditions that exceed the health and safety limits for normal operation as stated in the safety report. Thus, feeders are inspected for aging effects to ensure the structural integrity. Wall thinning due to flow accelerated corrosion and cracking have been identified as major aging degradation mechanisms based on the operating experience and inspection activities. Research programs and reviews have identified stress corrosion cracking (SCC) and hydrogen assisted low temperature creep cracking (LTCC) as plausible causes of the cracking mechanisms [33]. Several parameters (i.e. high residual stress resulting from a bend fabrication process, cyclic loadings during operation period, elevated material hardness due to cold work, chemical environment, and FAC generated hydrogen) could also affect feeder cracking.

At the design stage of CANDU-6 feeders, laboratory testing indicated that wall thinning rates under conditions similar to normal operating conditions in CANDU units were below 0.01 mm/year [33] [34]. However, in the mid-1990s thinning rates for some feeders in the Point Lepreau Nuclear Generation Station (PLGS) were estimated to be up to 0.15 mm/year. Ultrasonic techniques were used to measure wall thickness at the tight radius bend in accordance with CSA N285.4 Periodic Inspection of CANDU nuclear power plant components [35] Subsequently, more measurements obtained from other Canadian utilities identified higher wall thinning rates than expected at the tight radius bend as well. Moreover, in 2004, blunt flaws were found near the elbow weld during the destructive examination for multiple removed outlet feeders from one Canadian utility. The blunt flaw, which is volumetric in nature, has a finite root radius, such as a fretting flaw or a pit-like flaw, and is typically treated as a local stress concentration because of its small size. Inspection tools were developed to discover the blunt flaw, and the blunt flaw has not been detected since 2004. The occurrence of wall thinning and the likely presence of blunt flaws were the main driving forces to initiate and develop the feeder fitness for service guidelines (FFSG).

Operating experience and inspection results of feeders show that wall thinning at the tight radius bend/elbow and the Grayloc weld region in outlet feeders is the most susceptible/critical location which could affect the operating life of feeders. Wall thinning rates in inlet feeders are much lower than those in outlet feeders due to the lower operating temperature and iron-saturated coolant at the inlet.

It is known that operating temperature of internal flow, iron solubility, turbulent flow at the Grayloc hub and a rapid direction change of the internal flow due to the configuration of the tight radius bend/elbow results in the acceleration of wall thinning. The total length of a feeder is approximately 17 m (=56.1 ft) on average. However, the length from the Grayloc hub to the end of the tight radius bend/elbow, which is the region susceptible to higher wall thinning rates, is less than 0.5 m (=1.7 ft). Thus, the wall thickness measurement is intensively carried out to detect the wall thinning trend in the tight radius bend/elbow and the Grayloc weld region. On the contrary, wall thinning rates at the other feeder regions (i.e. straight portions, long radius bends, and nearby pipe fittings) are considerably lower than those at the tight radius bend and the Grayloc weld region because there is no abrupt geometrical change that leads to turbulent flow and rapid momentum change of internal flow.

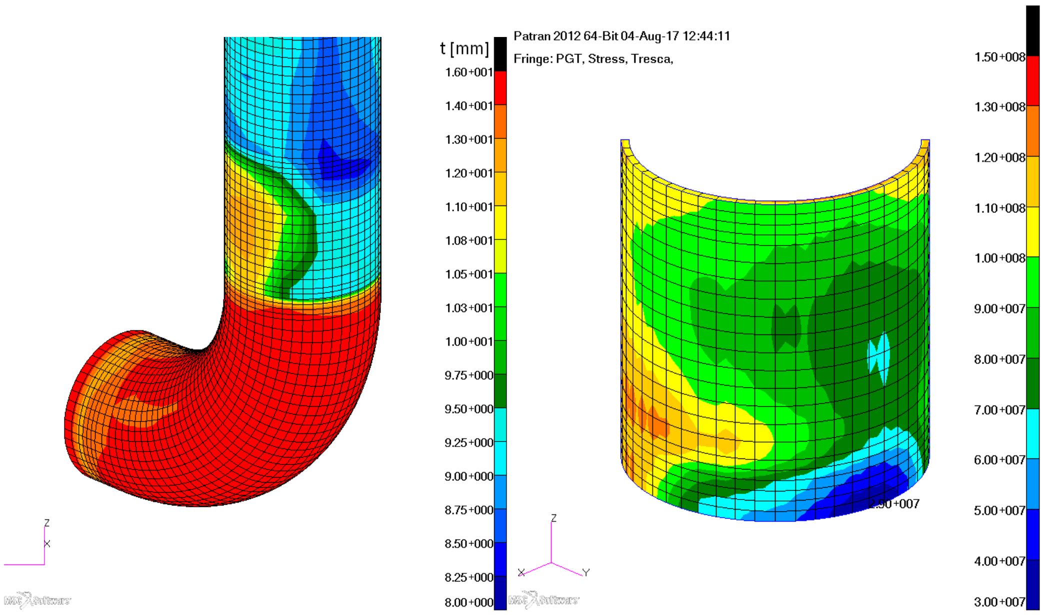
To reduce susceptibility to FAC, feeders in refurbished plants in Canada were replaced with an ASME SA-106, Grade C steel having a chromium content of minimum 0.3%. This chromium content is higher than the typical 0.03% chromium content in the original feeder material of SA-106 Grade B. This higher chromium content in SA-106 Grade C is expected to reduce FAC induced wall loss approximately by half. In addition, feeder replacements have been conducted and planned during outages. In the recent practice, a portion of the feeder is replaced with the SA-106, Grade C material that has higher strength than Grade B. The operating experience to date has confirmed that the FAC rates in the replaced feeder sections are lower than in the original feeder section.

4.5.2. Czech Republic

This section includes selected examples of current structural integrity management activities. All bodies of the primary circuit relief valves of NPP Temelin have been replaced to resolve leak tightness issues. The area of concern has been the flange connections between the relief valves and piping. Also, all bodies of relieves valves of the primary circuit of NPP Dukovany have been replaced due to wearing of seating surfaces.

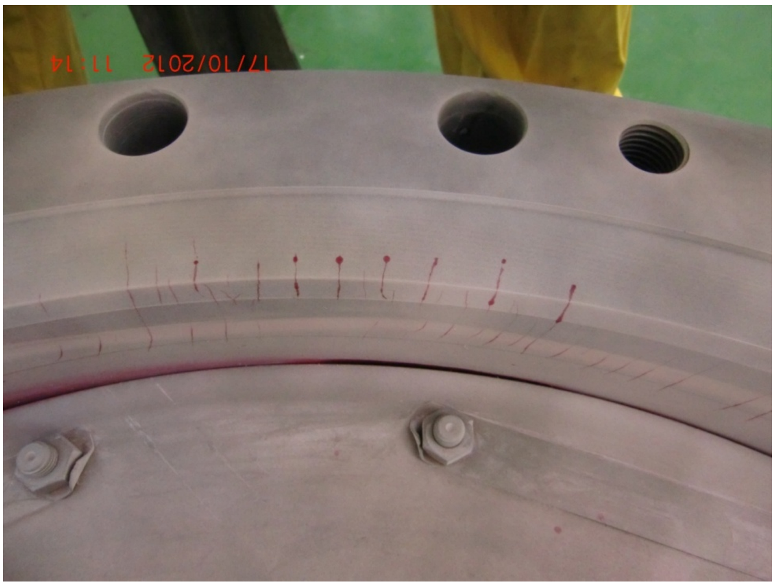
The flow accelerated corrosion (FAC) program activities are continuing unabatedly at all reactor units in the Czech Republic. On average five to six inspection locations with localised reduction of wall thickness are subjected to detailed evaluations in order to gain full understanding of wear patterns and rates; Figure 4.21. The results of these evaluations are incorporated in the FAC programme plan in an effort to optimise the non-destructive examination effort and to minimise the need for component replacements, especially in hermetic zone.

Figure 4.19. Examples of Measured Wall Thickness and Calculated Stresses



There is a continuing process at Dukovany of repairing degraded areas of the main reactor coolant pump (MCP) diffusers. The diffuser function, where cracks have been identified, is to protect the main bearings against primary circuit medium temperature; non-pressure retaining part of the MCP. The cracked areas of respective diffusor have been removed by machining. The probable cause of material degradation is a stress corrosion cracking (SCC) mechanism.

Figure 4.20. Local Cracks on MCP Diffusor Surface line.



4.5.3. France

This section is concerned with the internal flooding PSA (IF-PSA) study development at the IRSN. The reference level (RL) O1.1 proposed by the Western European Nuclear Regulators Association (WENRA) about the scope and content of Probabilistic Safety Assessments (PSA) studies for existing reactors states [36]: "For each plant design, a specific PSA shall be developed for level one (determination of cored damage frequency) and level two (determination of large early release frequency), considering all relevant operational states, covering fuel in the core and in the spent fuel storage and all relevant internal and external initiating events."

In line with this requirement and in the frame of the fourth periodic safety review (PSR) of the 900 MWe pressurised water reactors (PWRs), the French licensee EDF developed a PSA study to evaluate the core damage frequency taking into account the internal flooding risk. IRSN reviews the EDF study to ensure that the prevention and mitigation measures proposed by EDF are sufficient to provide an acceptable level of safety with respect to internal flooding risks. To do so, IRSN is developing an internal flooding PSA study focused on the most important plant buildings.

In defining the IF-PSA study scope the IRSN decided to focus on potential flood sources within the fuel building, the nuclear auxiliary building and within the electrical buildings (including control, connecting and shared electrical buildings of the 900 MWe twin units), beginning by the former buildings. Diesel buildings have not been studied because no initiating events caused by internal flooding scenarios have been identified in these buildings. The following are the main limitations and scope of the IRSN IF-PSA study:

- consideration of an as built plant without taking into account existing non-conformities;
- evaluation of core damage frequency for full-power operational states;
- selection of flooding incidents only leading to initiating events existing in IRSN level one PSA scenarios.

IRSN developed its internal flooding PSA study based on the EPRI IF-PSA guidelines [37]. The main steps identified by IRSN to apply the aforementioned methodology consist of:

1. identifying areas that can be affected by flooding events in the selected buildings. All potential flood areas, their features (floor area, elevation level, ceiling height, mitigating features, etc.) and their interconnections, such as doors, openings, heating, ventilation and air conditioning (HVAC) conducts, etc., have been listed;
2. identifying all flood sources, mechanisms, paths and systems, structures and components (SSCs) that could be affected by these flooding events;
3. assessing the flooding scenarios, their consequences and the associated flooding initiating events frequencies for the areas selected at step one. Simplified hydraulic models were used to evaluate flow rates through doors, HVAC conducts and other openings;
4. assessing the flood mitigation measures and analysing human reliability in the particular context of flooding events;
5. modelling the accidental sequences and quantifying the PSA model to compute the core damage frequency with respect to internal flooding events, to evaluate importance measures and to perform sensitivity analyses.

In evaluating the relevant operating experience, the IRSN reviewed and processed all flooding events that have been recorded for French nuclear power plants; from the first commercial PWR commissioning in 1978 (Fessenheim-1) to 2015, representing about 1 700 reactor operating years. These events cover a large spectrum of flooding incidents from actuations of fire extinguishing systems up to passive component failure (mainly pipes break) and human induced floods. About 190 flooding events were retained for further analysis. These events are mainly due to spurious actuations of fire extinguishing systems which lead to minor consequences. However, due to the difficulties inherent to the determination of representative piping rupture frequencies, IRSN might investigate the following options:

- using CODAP database to gain insight on piping rupture frequencies thanks to the important operational experience gathered on PWRs;
- complementing its own data with international data such as Nordic PSA Group (NPSAG) data [38] [39].

IRSN determined that the potential important flooding risks (in terms of core damage or spent fuel pool uncover frequency) could be induced by flooding sources localised in:

- the electrical building, due to the possibility of losing both electrical safety divisions, instrument and control (I&C) rooms, the main control room (MCR) or rooms containing the auxiliary feedwater (AFW) motor-driven pumps and turbine-driven pump.
- The fuel building, due to the presence of both low pressure safety injection (LPSI) and containment spray (CS) pumps in adjacent rooms leading to the risk of losing both trains of these safety systems (LPSI and CS systems). Furthermore, it should be stated that the fuel building houses both spent fuel pit (SFP) cooling pumps.
- The nuclear auxiliary building, due to the proximity of train A and B component cooling water (CCW) pumps and high pressure safety injection (HPSI) pumps.

Finally, IRSN first findings point out that the global hourly internal flooding frequency appears to be higher in shutdown states than in full-power operational states. Moreover, based on the events frequencies evaluated by IRSN, electrical buildings are the most susceptible to internal flooding mainly because they host about hundred rooms from which about 60 contain flooding sources.

4.5.4.1 A representative internal flooding event

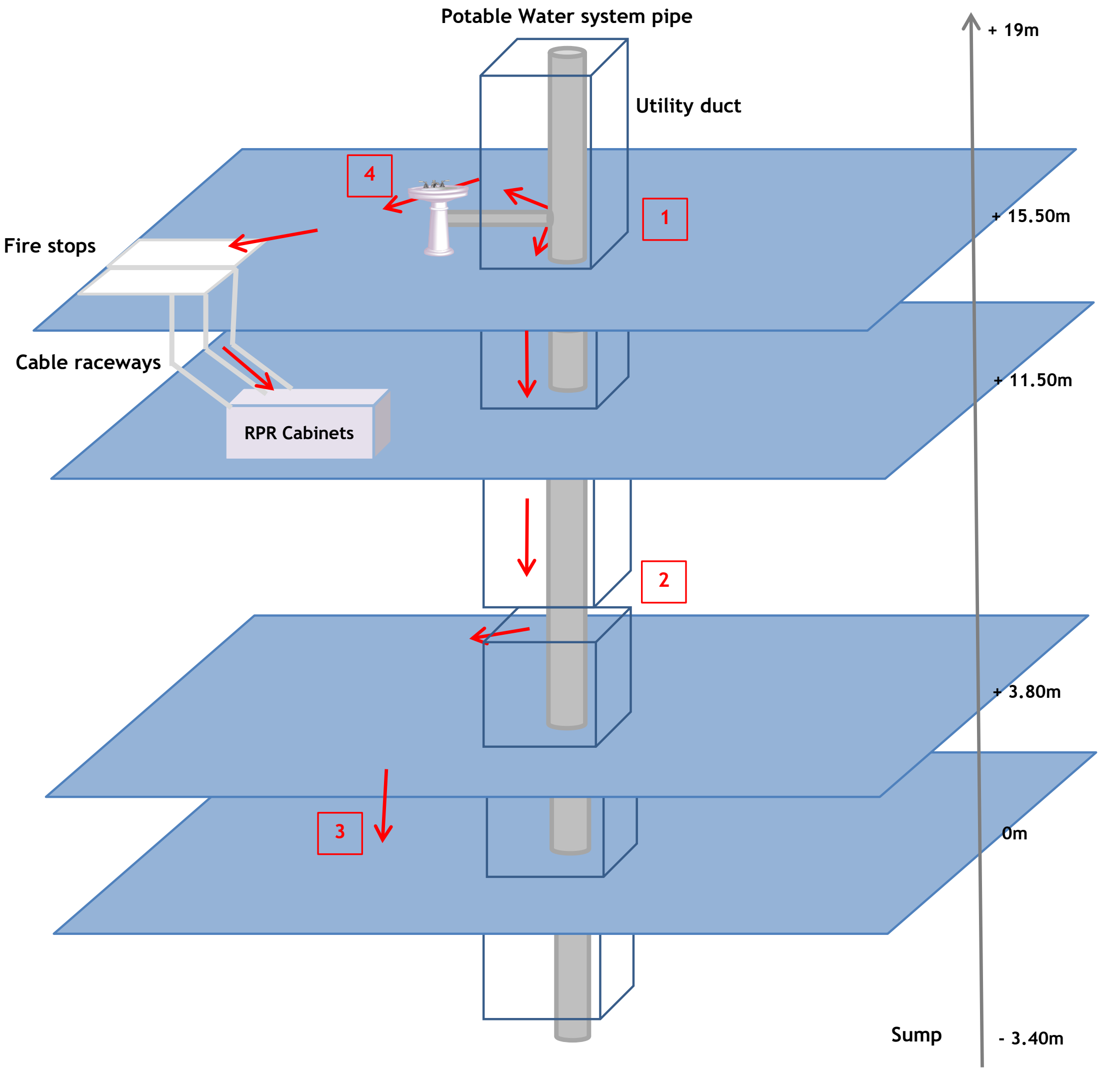
In September 2012 a leak in the potable water (PW) system caused a reactor shutdown at Blayais unit one. The potable water system, which supplies potable water to the whole facility, runs through the electrical buildings of nuclear reactors. At the 900 MWe Blayais-1 PWR, the potable water system pipe is in a fire resistant utility duct in the electrical building, which means that any leaks flow down to lower levels, from level +19m to level -3.40m, where they are collected in a sump (see Figure 4.23).

In September 2012, EDF received a control room alert due to an insulation fault alarm on a 200V AC power supply and distribution switchboard. Investigations led to the discovery of a water leak from the PW system pipe at level +15.50m of the electrical building (Point one in Figure 4.23). The water flowed down inside the utility duct to level +3.80m, where it spilled out onto the floor of the room through an opening in the utility duct at this level (Point two in Figure 4.23). It then flooded the access corridor leading to the cold changing room in the nuclear auxiliary building (Point three in in Figure 4.23).

After spending several minutes seeking to detect and isolate the leak, the licensee closed a valve in the radiation-controlled area that supplies potable water to all rooms in reactors one and two and the waste auxiliary building. In the meantime, the water, which had broken through the utility duct at level +15.50m and spread through the adjoining room, had trickled through a fire stop on cable raceways to reach the RP system electrical switchboards on level +11.50m (Point four in Figure 4.23).

The licensee received a second insulation fault alarm on the 48V DC power supply and distribution switchboard for train A of the LCA system. Malfunctions were reported on various sensors, including a low-low steam generator level sensor that is part of the protection system. As a result of these faults, the licensee reported one of the two logical trains of the RP system to be unavailable. In compliance with the rules on equipment unavailability, the licensee initiated a normal reactor shutdown, reactor being cooled by the residual heat removal (RHR) system, within an hour.

Figure 4.23. Internal Flood Pathway



During the shutdown process, the insulation fault on the LCA switchboard disappeared, allowing it to be declared available once more. However, a quick repair of the RP system could not be guaranteed in the 24-hour period required under the operating technical specifications when the reactor is in normal shutdown, being cooled by the RHR system. The licensee therefore continued reactor shutdown to maintenance outage mode, which does not require the RP system to be available.

After the flooded rooms had been dried out, the licensee carried out diagnostics on the equipment in those rooms and replaced 34 relay PCBs in RP system cabinets that showed signs of oxidation or humidity. With the multiple faults on RP system train A, the licensee considered train A of the system to be totally unavailable, even though only a few relay PCBs had actually been impacted by the flow of water. The function of the RP system is to monitor physical variables and to trigger reactor protection actions such as reactor trip and initiation of safeguard systems if necessary. The flow of water could have generated spurious protection orders.

Nevertheless, train B remained available because the two RP system trains are separated both electrically and physically, in compliance with the safety analysis report. Reactor protection actions thus remained available during this event.

The pipes of the PW system were designed in carbon steel. Under an equipment modification, the original pipes were replaced by polyvinyl chloride (PVC) pipes. A screw-on copper coupling was fitted to the pipe at level +15.50m to supply water to a basin and a water heater. In addition, the PVC pipe between level +19m and level +15.50m and the branch connection on level +15.50m was not supported by any bracket. The screw-on coupling was therefore affected by system vibrations when it was put into service or during maintenance operations, and became damaged. The coupling failed and caused a leak that eventually went through the utility duct, at a place which had non-watertight fire-resistance insulation made of plaster.

No modification file was put together for the aforementioned equipment modification, and no internal flooding risk analysis was carried out. The modification was implemented more than ten years ago, when regulations did not require a modification file when the equipment modification did not affect systems involved in reactor operation. Regulations have now changed.

The leak on the PW system led to a critical insulation fault on a 48V switchboard (LCA system), which the licensee therefore reported as unavailable. The LCA 48V electrical switchboard is involved in 48V DC power supply and distribution, powering the RP system and I&C, train A. This function is kept operational in the event of total loss of electrical power supply via the 48V LCA batteries (battery life at least one hour). The flooding could have caused loss of I&C and loss of electrical power to train A equipment.

Following this event, the branch connection in question was removed from the PW system on reactors one and three. The PW system only runs through odd-numbered units at the site, so corrective actions were not required on reactors two and four. The licensee did not replace the pipe for reactor three, because it was made of a composite and did not have the same fragility. The attachment supports were repaired, however. The PVC pipe on reactor one PW system was secured pending a definitive solution involving replacement with a high-density polyethylene pipe from level +19m down to level -3.40m. The licensee also checked that there were no other PVC branch connections on the PW system pipe.

In order to comprehensively deal with this issue, the licensee carried out a survey of internal flooding risks in electrical equipment rooms based on site inspections and analysis of operating feedback. The licensee listed and located all previously non-listed valves on the PW system or the demineralised water distribution (DWD) system, in order to enable shift crews to quickly isolate any leaks on these two systems. The PIDs and databases were also updated. In particular, an instruction was drafted to enable teams to quickly identify which valves should be closed in the event of a leak on the PW system or DWD systems in electrical equipment rooms. Post-earthquake operating instructions were also amended to include shutdown of the PW system.

This event illustrates the safety risks that can be caused by equipment not involved in reactor operation but installed in an industrial area of the plant. Such equipment is not bound by the same maintenance and modification requirements as equipment in the nuclear island and conventional island. As a consequence, any modification in an industrial room will now require a modification file that always includes an assessment of the impact of this modification, particularly with respect to the risk of internal flooding.

4.5.4. Germany

During the CODAP term two (2015-2017), a number of reportable events occurred in German NPPs within the scope of the CODAP project. The root cause/degradation mechanism was clarified or is part of an ongoing analysis. Upon known root cause, suitable corrective measures have successfully been undertaken. In conclusion, the safety significance of all events was considered to be low. Yet, in some cases there was evidence of a generic issue, which gave rise to the issuing of an information notice by GRS. The most significant events regarding CODAP are briefly described below.

During a planned outage, the yearly functional test was performed in the area of the main steam safety and relief valves. At one valve, the test had to be stopped due to an unexpected steam leakage in a draining line. Based on a root cause analysis, the damage was attributed to a uniform corrosion attack of the inner surface of the draining line leading to significant wall thinning over time. Accordingly, the failure of the residual wall was due to pressurisation of the draining line (usually unpressurised, except for functional tests) in the course of the test. It was assumed that the corrosion attack was favoured by condensate accumulation in the draining line, which has not been taken into account in the design of the line. The accumulation of condensate was caused by humidity which entered the draining line from the outside during the monthly functional test of the main steam relief control valve. The affected draining line was indeed laid with gradients as required by the specifications. However, complete draining of the line was prevented by a draining valve having an offset between the inlet and outlet opening. Inspections of the other draining lines revealed wall thinning, partly below the required minimum wall thickness. In the following, the safety consequences of a postulated systematic failure of the draining line in all redundancies were discussed. As a result of further analyses it could be confirmed that even in this case all protection goals are fulfilled although temperature and pressure of the primary side are different from the case without steam leakage. It also turned out that this disturbed plant condition is covered by instructions in the operating manual. In an information notice on this event, recommendations were given dealing with identification, testing and – as required – replacement of comparable and potentially affected small-bore pipes.

In the course of inspections during a planned refuelling outage, crack indications were found on thermal sleeves in two of the four injection nozzles of the residual heat removal system into the main coolant line cold leg. The pressure retaining boundary was free of indications. All indications are in the weld region of the thermal sleeve to the nozzle. Crack initiation and propagation was driven by mechanical fatigue due to the coolant flow passing the thermal sleeve favoured by minor manufacturing deficiencies of the safe end of the injection nozzle. There was no evidence of the involvement of any corrosion phenomena on crack initiation and growth. The safety significance of the event was considered to be rather low because the thermal sleeve did not loosen. Even in the case of a loosened thermal sleeve, the coolant flow is not constrained as the thermal sleeve is hindered to leave the nozzle by crimps and wipers. In an information notice on this event, recommendations were given dealing with the qualification of appropriate inspection techniques and the concept of in-service inspections on this type of nozzles.

In the framework of preparatory activities for a pressure test of the shell side of an intermediate water cooler (IWC), a blind flange cover of the flushing nozzle was opened for the first time since plant start-up in 1984. It was found that the blind flange was manufactured from a different material than specified, without the specified hard rubber lining, and with a lower thickness than specified. Due to the missing hard rubber lining, the wall thickness was further reduced by corrosion in the course of operation, but it was still a factor of 2.5 above the required minimum wall thickness. At a second IWC, the corresponding blind flange cover was also made from a different material than specified and without the required hard rubber lining the wall thickness however was as specified. All other IWCs and heat exchangers from the same manufacturer were without comparable findings. Identified deviations are manufacturing defects indicating a systematic error during the manufacturing process, but the root cause could not be clarified any more. Since the blind flange covers were not in the scope of the recurrent visual inside inspection of the IWC, the deviations have remained undetected since 1984. Remedies included the exchange of the affected flange covers by new specification-compliant covers and the expansion of the scope of the visual inspections to include the inside surface of the flange covers.

During power operation, the leakage monitoring system provided evidence of a primary coolant leakage inside the reactor building containment. After manual shutdown of the reactor, a very small leakage was found on a differential pressure measuring instrument line which branches off of loop four near the corresponding nozzle of the main coolant line. During inspections, a crack was found beginning in the nozzle-to-pipe weld and propagating into the base metal of the pressure sensing line.

The direct cause of the leakage was mechanical low cycle fatigue due to vibrations induced during start-up and shutdown operation of the main coolant pump. In the meantime, the main coolant pump casing including the shaft sealing housing has been modified. The resulting vibration frequency does no longer cause any relevant fatigue load in the pressure sensing pipe.

During regular eddy current inspections of steam generator tubes, dozens of tubes with indications were found in one steam generator, some of them being beyond the registration level. All indications were on the cold side (outlet side) of the steam generator. Most indications are in the regions with deposits. Tubes with indications exceeding 30% wall thickness were plugged. The root cause analysis is still ongoing. Besides, it is still under investigation why only one steam generator is affected as there are no obvious differences between the four steam generators.

4.5.5. Japan

Overviews of the post-Fukushima materials research can be found in reference [40] as well as in the "E-Journal of Advanced Maintenance" (www.jsm.or.jp/ejam/). Except for a single event, no Japanese operating experience data have been entered into the CODAP event database in the second term. A primary reason for this is that almost all Japanese nuclear power plants have been off-line since mid-2011.

4.5.6. Korea

In commercial nuclear power plants, the service water piping system carries cooling water to various heat exchangers. It removes the heat from such auxiliary systems as component cooling heat exchangers, emergency diesel generators, containment coolers, lube oil coolers, room coolers, and chiller condensers. It consists of an intake structure for the cooling water, a distribution system within the plant and an outlet structure. Suction is taken from the ultimate heat sink (e.g. ocean, river, lake, or reservoir), heat is removed via various heat exchangers, and the water is discharged back to the ultimate heat sink (UHS) or self-contained UHS (e.g. spray cooling pond or cooling tower). This section includes an overview of the recent Korean operating experience with the safety-related service water piping.

On 19 October 2017, a pin-hole was discovered in the essential service water system (ESW) in Shin-Wolsung unit two[21]. The pin-hole was located on a pipe connected to an orifice downstream of a flow control valve from the train A component cooling heat exchanger. The bounding size of the pin-hole was estimated to be a 30 mm in diameter. On 7 December 2017, the licensee of Shin-Wolsung-1 also found wall thinning that did not meet the minimum required wall thickness of the ESW pipe connected to an orifice downstream of a flow control valve from the train B heat exchanger. The location of thinned area was similar to the location where the pin-hole was discovered in Shin-Wolsung-2.

The damaged lines are 24-inch pipes made of carbon steel with 2-mm glass flake reinforced polyester/vinylester (Archcoat) lining. The cause of the pin-hole and wall thinning was determined to be the damage of interior Archcoat lining and the subsequent exposure of base metal to sea water. It could be a result of turbulence flow occurred in the pipes downstream of the orifices and vibration of the pipes. The affected pipes were replaced. A research project has been conducted to identify the root cause and the needs for design modification of the affected piping.

A review to identify past similar events in domestic NPPs was performed and a total of 20 repair and replacement cases were identified for the ESW system during the 2005-2017 timeframe; Table 4.3. Of the 21 cases, seven cases (No. 1, 3, 6, 11, 15, 16, 18) were identified as similar to the base metal degradation discovered at Shin-Wolsung units one and two.

21. A 2-loop second generation PWR (OPR-1 000) designed by KHNP and KEPCO.

Table 4.3. Recent Cases of ESW Pipe Degradation

Event No. 1)	Plant/Location	Event Type	Pipe Information
1	Shin-Wolsung-1: Pipe downstream of flow control valve from the train B CCW heat exchanger	Wall thinning	• Material: A106-B • Size: 24”, 3/8”t • Interior Coating: Archcoat
2	Hanbit-4: Debris filter backwash outlet header in the ESW system train B	Pin-hole leak	• Material: A106-B • Size: 6”, SCH40 • Interior Coating: Rubber lining
3	Shin-Wolsung-2: Pipe downstream of flow control valve from the train A CCW heat exchanger	Pin-hole leak	• Material: A106-B • Size: 24”, 3/8”t • Interior Coating: Archcoat
4	Hanul-2: Tee connected to downstream of the pump in the ESW system train A	Pin-hole leak	• Material: TU42C 2) • Size: 28”, 7.92t (Tee) • Interior Coating: Rubber lining
5	Hanbit-2: Essential chiller condenser outlet seawater pipe in the ESW system train B	Pin-hole leak	• Material: A106-B • Size: 10”, SCH40 • Interior Coating: Rubber lining
6	Hanbit-3: Pipe downstream of flow control valve from the train A CCW heat exchanger	Pin-hole leak	• Material: A672-B60 CL22 3) • Size: 36” • Interior Coating: Archcoat
7	Hanul-2: Reducer upstream of the ESW system train A pump	Pin-hole leak	• Material: TU42C • Size: 20X28”, 7.92t (Reducer) • Interior Coating: Rubber lining
8	Hanul-4: Debris filter 02A backwash outlet pipe in the ESW system	Pin-hole leak	• Material: A106-B • Size: 6”, SCH40 • Interior Coating: Archcoat
9	Hanul-4: Debris filter 02B backwash outlet pipe in the ESW system	Pin-hole leak	• Material: A106-B • Size: 6”, SCH40 • Interior Coating: Archcoat
10	Hanul-2: Elbow downstream of the train B CCW heat exchanger	Pin-hole	• Material: TU42C • Size: 16”, 8.8t (Elbow) • Interior Coating: Archcoat
11	Hanbit-5: Pipe downstream of the orifice in the ESW system train A	Pin-hole leak	• Material: A672-B60 CL22 • Size: 36” • Interior Coating: Archcoat
12	Shin-Wolsung-1: Debris filter 01B drain line in the ESW system train B	Pin-hole leak	• Material: A106-B • Size: 6”, 0.065”t • Interior Coating: Archcoat
13	Hanul-2: Pipe downstream of the train B heat exchanger	Pin-hole	• Material: A42CP • Size: 28”, 7.92t • Interior Coating: Rubber lining
14	Hanul-3: Pipe downstream of the orifice connected to the pump 03A discharge in the ESW train A	Pin-hole leak	• Material: SB-165 4) • Size: 2”, SCH40
15	Hanul-4: Reducer downstream of the valve from the train B CCW heat exchanger	Pin-hole leak	• Material: SA-234 5) • Size: 24”X36”, 3/8”t (Reducer) • Interior Coating: Archcoat
16	Hanul-1: Pipe downstream of the train B CCW heat exchanger	Pin-hole	• Material: TU42b • Size: 16”, SCH20F

Event No. 1)	Plant/Location	Event Type	Pipe Information
17	Hanul-1: Elbow downstream of the train B CCW heat exchanger	Pin-hole	• Material: TU42C • Size: 16”, SCH20F
18	Hanul-1: Elbow downstream of the train B heat exchanger	Pin-hole	• Material: TU42C • Size: 28”
19	Hanul-3: Pipe weld downstream of the orifice in the ESW system train B	Pin-hole leak	• Material: A672-B60 CL22 • Size: 36”, SCH20 • Interior Coating: Archcoat
20	Hanul-3: Traveling screen line in the ESW system	Pin-hole	• Material: SB-165 • Size: 2”, SCH40
21	Hanul-5: Pipe downstream of the flow control valve in the ESW system train B	Pin-hole	• Material: A106-B • Size: 24”, 3/8”t • Interior Coating: Archcoat

1) The events appear in chronological order; most recent (2017) to oldest (2005)
2) TU42C is a low-alloy steel
3) Seam-welded carbon steel piping with material chemical composition equivalent to ASTM A-106 GrB
4) SB-165 (Monel or ALLOY400) is a nickel-base material that contain between 29 and 33% copper.
5) SA-234 is low-alloy steel

Regarding research projects related to component aging and degradation, for the period 2013 to 2018 the Korea Institute of Nuclear Safety (KINS) has been engaged in Nuclear Safety Research projects in order to:

- assess emerging techniques of nondestructive evaluations;
- develop confirmatory flaw evaluation methods for major components in nuclear power plants;
- develop regulatory guides for environmental fatigue;
- develop regulatory infrastructure for probabilistic fracture mechanics applications.

4.5.7. 4.5.8 Slovak Republic

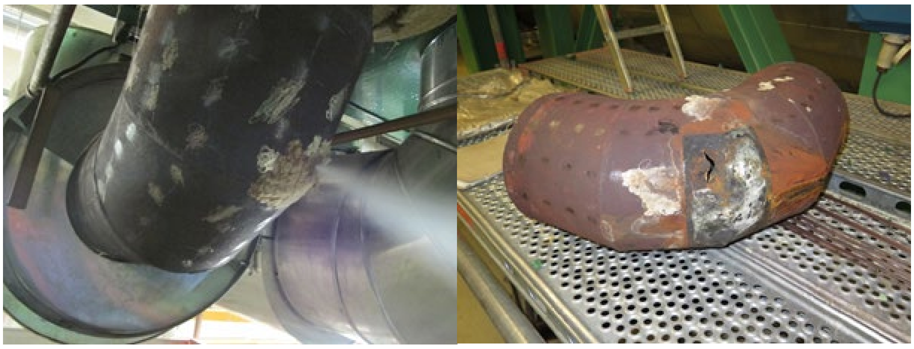
During the CODAP term two (2015-2017) one reportable event occurred in Slovak NPPs within the scope of CODAP project. Event occurred on steam pipeline of secondary circuit and there was no consequence on nuclear safety and operation of NPP. As a root cause the flow accelerated corrosion was assigned. Damaged component was replaced by the new component and supplementary measurements were done on the equivalent components of steam pipeline to other deaerators. A brief description of event is given in the following text.

- Event at NPP Bohunice unit three occurred on the secondary circuit during the normal power operation. Leakage was located on the second segment of the elbow (Figure 4.24) downstream the motor operated valve on steam pipeline to feedwater tank deaerator. Leakage was identified after removing of pipeline insulation. The operational parameters of steam during the event were following: maximum pressure 0.7 MPa, maximum temperature 164°C. Dimensions of component ø 273 x 6.5 mm, material: 12 022.1[22]. Minimal allowed thickness of this component is 2.1 mm. The leakage was temporarily stopped by the Furmanite™ method until the

22. Material of Type 12.022.1 is equivalent to DIN St 45.8 low alloy steel.

beginning of planned outage. Damaged component was replaced by the new component during planned outage in 2015. Flow accelerated corrosion was determined as the degradation mechanism in this case. After completion of the repair wall thickness measurements were performed on the equivalent components of steam pipeline to other deaerators.

Figure 4.21. Photo of Steam Leakage and Detail of Damaged Mitered Elbow



4.5.8. Spain

The Spanish Nuclear Safety Council (CSN) was an early adopter of the US Nuclear Regulatory Commission's generic letters (GL) 89-13 [41] and 90-05 [42]. In summary, "generic letters" request licensee actions and/or information to address issues regarding emergent or routine matters of safety, and require a written response by licensees. GL 89-13 is concerned with material degradation of raw water cooling piping and GL 90-05 is concerned with the management of the structural integrity of moderate-energy piping systems. To summarise GL 90-05, whenever a degraded condition such as significant wall thinning or through-wall leakage is discovered a licensee has the option of either performing a controlled plant shutdown to perform a "code repair" per regulatory requirements or perform a temporary repair to enable continued operation and defer repair actions until the nearest planned outage of sufficient duration. In order to perform a temporary repair a licensee must first submit a formal relief request with details on the extent of degradation, root cause, augmented inspection results, and structural integrity. As a consequence of this regulatory requirement, whenever temporary repairs are being proposed the reporting of degraded conditions tend to be quite detailed. Examples of the Spanish experience with GL 90-05 are included in the CODAP event database, and two examples of this experience are given below.

- Leak due to Internal Corrosion of a Pipe Between Flanges in Line 22Z02 of the Service Water System (VE)[23]. In May 2014 a through-wall pipe flaw was discovered in a straight section of the VE pipe (3m) between two valves, one of which was a non-return valve. The inside water was maintained stagnant; moving only when the VE pump 40D001 was operated once per month. Under such conditions, losses of thickness occurred in the form of undercuts/sinkholes. The perforation was circular shape of 4-6 mm diameter. Water in the SW VE system is treated with corrosion inhibitors, and hypochlorite circulates inside the pipeline at

23. The components of the VE system are mainly made of carbon steel, and a few of stainless steel.

a temperature that could reach up to 35ºC during accident conditions. The 610 mm diameter seam welded piping is of carbon steel material, DIN St-37-3). Since outwardly there were no signs of wear or any other defect that could have induced the perforation, it was decided to replace the section between valves VE22S005/10 and to remove an 800 mm long piece of the degraded pipe spool and to a laboratory for metallographic examination as part of the root cause determination. The inside surface of the pipe was covered with a compact layer of oxides and/or deposits of blackish and ochre colour, partially exfoliated. The laboratory carried out visual inspection and reception controls, as well as Energy-Dispersive X-ray spectroscopy (EDX) and chemistry analysis of oxides and deposits, macrographic examinations and hardness measurements. The upper pipe wall had a high density of tubercles, while the lower pipe wall had few tubercles but in its place there were corrosion undercuts, among which was the pore that gave rise to the leak. The corrosion products revealed by the EDX leave no doubt about that the conditioned or retained water was being treated with zinc, frequently used as an inhibitor of oxygen corrosion. The acceleration of the undercuts has been partly due to the differential aeration that provokes the galvanic effect and the acid pH of the cavities (samples with pH around four).

- Microbiologically influenced corrosion of essential service water (ESW) Piping. During a system walkdown in August 2013, a pinhole leak was discovered on a 24-inch service water train A pipe line. The leak location was in a bypass of a motorized filter of train A of unit two of the service water (SW) system. The through-wall flaw was characterised as a 2 mm diameter pinhole. In response to this discovery, the SW train A bypass line in its entirety was subjected to 100% UT surface examination. The unit two train B bypass line was also inspected. Similarly, both bypass lines of both trains A and B of Almaraz unit one were also inspected by UT. In the bypass line of train B of unit one was also detected an area with a remaining thickness of 1.9 mm, higher than the allowable minimum thickness (1.81 mm), but less than the assessed minimum thickness (2.07 mm). A "non-code" repair was performed by welding reinforcement in the area where the pinhole was detected. Reinforcement plate obtained from a pipe of the same diameter and of the same material as the existing pipes, SA-106 Gr B, was installed over the degraded area. The size of the reinforcement was big enough to cover the affected zone until thickness of the material reached nominal design. Design according to Code Case N-789. A permanent repair was performed at the next refuelling outage, November 2013. The affected zones were cut/removed and replaced by new sections.

With reference to concealed piping systems, for the fire protection water system (FPS) the aging management strategy in Spain [43] is based on AMP.XI.M27 (of the NRC GALL report[24]) and LR-ISG-2012-02[25] , and with the objective to ensure that no significant ageing mechanism acts on the interior component surfaces as a result of general and micro-bacterial corrosion in the FPS, for which periodic blowdown flushing operations and

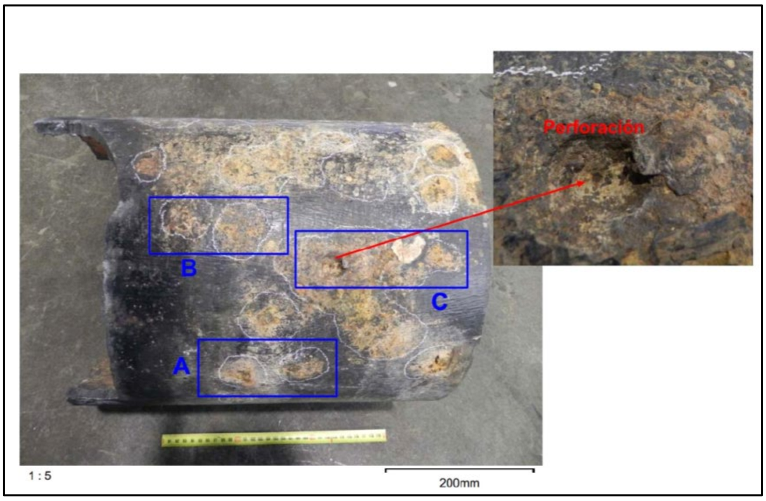
24. www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1801/

25. Aging Management of Internal Surfaces, Fire Water Systems, Atmospheric Storage Tanks, and Corrosion Under Insulation, www.nrc.gov/docs/ML1322/ML13227A361.pdf

functional tests are performed in accordance with the recommendations of NFPA-25 [26]. In the Spanish plants the system works with pre-treated water, and hence, piping degradation due to MIC is unlikely. The system is considered susceptible to general corrosion and tuberculation. Ageing management of the exterior of buried pipes is in accordance with AMP.XI.M41 "Buried and Underground Piping and Tanks", and with further elaborations per LR-ISG-2015-01[27]. Spain has made several data submissions to CODAP pertaining to degraded buried FPS piping, including the following:

- At one of the Spanish NPPs, in the course of 2014-2015 multiple leaks were discovered in the buried section of a FPS DN300 ring header; Figure 4.25. Since the first discovery in December 2014, the licensee has been evaluating the causes of the piping failure and taking measurements to improve the inspection program of the FPS buried pipes. It was determined that the pipe wall perforation was caused by degraded pipe coating causing localised external pipe wall corrosion. After a thorough root cause evaluation, the licensee developed a comprehensive FPS aging management program consisting of the following programmatic elements: 1) collection of soil samples to determine its corrosivity, 2) perform an assessment of the substrate including the type and extent of backfill used, 3) assess the need for installing cathodic protection, and 4) perform an assessment of the integrity of the external coating material.

Figure 4.22. Detail of the Area with the Most Extensive Pipe Wall Deterioration



4.5.9. Switzerland

At the end of 2016 the Swiss Federal Nuclear Safety Inspectorate (ENSI) decided that Switzerland would take part in the first Topical Peer Review (TPR) of the European Union on "Ageing management in nuclear power plants". The licensees of the Swiss nuclear

26. NFPA 25: Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems, National Fire Protection Association, www.nfpa.org/

27. Changes to Buried and Underground Piping and Tank Recommendations www.nrc.gov/docs/ML1530/ML15308A018.pdf

power plants were requested to submit reports with the information in accordance with the specification of the Western European Nuclear Regulators Association (WENRA) [44].

The ageing management programme (AMP) implemented in the Swiss nuclear power plants is based on a long-standing development process that has been closely followed by ENSI. The licensees of the Swiss nuclear power plants had already been required to introduce an AMP for safety-relevant structures, systems and components (SSCs) by the former Swiss regulatory authority (now ENSI) at the end of 1991. Consequently, the licensees organised in the group of Swiss nuclear power plant managers (GSKL) started a common project to develop an AMP on the basis of the already existing maintenance programmes. Within the framework of the GSKL project the fundamental documents for the introduction of the AMP in the electrical engineering, civil engineering and mechanical engineering fields were developed. The implementation of the AMP is done by specially-founded ageing management expert teams in each nuclear power plant. From ENSI's point of view, the structure and organisation of ageing management in the Swiss nuclear power plants and its integration into established plant programmes via existing quality and plant management systems has proven its value over time.

The requirements for ageing management are specified at the guideline level based on experience gained in the implementation and updating of the AMP in Swiss nuclear power plants. In 2004, general and field-specific requirements for ageing management were incorporated in a regulatory guideline for the first time. This guideline related in particular to the basic elements of a systematic AMP and to the scope of the SSCs to be included in ageing management for each field. In the guideline ENSI-B01 [45] the scope of the safety-relevant SSCs to be included in the AMP has been expanded and, for the first time, concrete requirements for the proof of the resistance of the pressure vessel to brittle fracture as well as for the scope and evaluation of the fatigue monitoring have been included.

Section four of the 2017 Swiss topical peer review report [46] summarises the approach to AMP for below-ground (concealed) piping. According to the investigations performed by the four licensees, there are only a few areas in which there are safety-relevant concealed pipework. In individual cases, preventive measures have been implemented in these areas. Irrespective of this, from ENSI’s point of view it is important to check whether concealed pipework areas are systematically included in the existing plant programmes for ageing management.

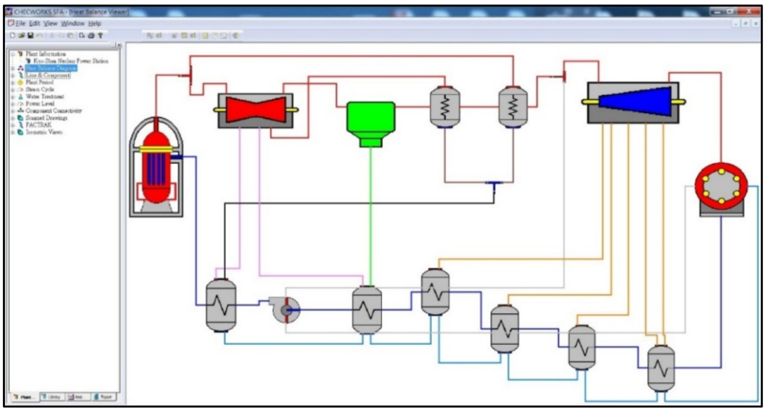
4.5.10. Chinese Taipei

All nuclear utilities have programmes in place to protect pipes from flow-accelerated corrosion (FAC), including damage caused by erosion. The most common forms of erosion encountered—cavitation, flashing, liquid droplet impingement, and solid particle erosion—have caused material loss, leaks, and ruptures and resulted in unplanned shutdowns. Repair and replacement of damaged piping and equipment have been a continuing expense. Additionally, noise and vibration caused by cavitation or flashing have posed control and maintenance problems.

In 1986, one high-pressure condensate line in Surry power station ruptured, which was induced by FAC and resulted in several fatalities and injuries. In 1989, the Nuclear Regulatory Commission (NRC) accordingly required all of the nuclear power plants in the US to implement the FAC programme focusing on carbon steel pipes. In the same year, a long-term wall-thickness inspection plan for carbon steel pipes proposed by the Taiwan Power Company (TPC) was implemented after being approved by Taiwan Atomic Energy Council (AEC). Since 1994, the engineering software CHECWORKSTM developed by

Electric Power Research Institute (EPRI) has been introduced to evaluate pipe wall thickness for nuclear power plants in Taiwan in aid of the piping inspection program. Figure 4.26 shows the interface of CHECWORKSTM.

Figure 4.23. CHECWORKSTM User Interface



TPC follows the EPRI recommendations to evaluate the carbon-steel pipes which are susceptible to FAC under normal operations by using CHECWORKSTM. The calculated FAC wear rate and the remaining life of the pipes form a basis for screening the inspection locations before refuelling outages. For carbon-steel pipes which could not be evaluated by CHECWORKSTM, past inspection data, operating experiences and maintenance records could be used to trace the condition of the carbon-steel pipes.

For analysis of inspection data, a software named "Pipe Thickness Inspections Processing and Evaluation System" (PIPES) developed by Institute of Nuclear Energy Research (INER) is used to calculate wear rate of pipes to help engineers have a profound understanding of pipe degradation. Before the beginning of an outage, engineers have to update a list showing which carbon steel pipes are to be inspected. In Maanshan nuclear power plant (a PWR plant), one week before reactor shutdown, an amine measurement should be performed to obtain the input data for CHECWORKSTM. If there is a pipe whose wall thickness is greater than minimum acceptance, it could be used based on its calculated inspection period or it will be replaced by a new one.

During each outage about 250 locations are inspected. For example, during the EOC-24 outage of Kuosheng unit one (BWR), a carbon steel pipe whose wall thickness was detected being less than 0.368 inch, and the diameter of thinning region was about one inch; Figure 4.27. This indicates TPC has the ability to detect the thinning tendency of wall thickness and takes necessary precautions against it.

Figure 4.24. Demonstration of the FAC pipe thinning region.

In Maanshan nuclear power plant, pipes of the "Feedwater Heater Extraction Drains" are susceptible to degradation because it was made of carbon steel. According to "Recommendations for an Effective Flow-Accelerated Corrosion Program (NSAC-202L-R3)", feedwater heaters, including the shell, nozzles, internals (e.g. tube sheets, stay bars, etc.) and drain coolers should be taken into consideration during inspection. Thus feedwater heaters should be prioritised in the inspection programme.

However, for most small-bore pipes, TPC has replaced them with stainless steel pipes and stainless steel is known to be more resistant to FAC.

In order to reinforce FAC inspection work, TPC issued "Guideline of Inspection of Carbon Steel Pipes in Nuclear Power Plants" in 2015, which including personnel qualification, pipes monitoring, precautions and lesson learned from international experiences.

On account of the countermeasures taken by TPC against the piping that are suspected of being degraded, there has been no pipe rupture event happened since TPC began to carry out the inspection work to evaluate degradation degree of piping and its remaining life. In conclusion, TPC has a general understanding on the condition of carbon steel pipes to prevent the nuclear power plants from unplanned shutdown caused by unanticipated FAC-induced pipe ruptures.

4.5.11. United States

This section describes efforts by the US Nuclear Regulatory Commission (NRC) and its licensees to address buried piping leakage, particularly leakage that involves the discharge of tritiated water. In 2009, buried piping leaks released water that contained low levels of tritium at Dresden, Oyster Creek and Peach Bottom nuclear plants. The levels of tritium did not exceed any NRC on-site limits. Furthermore, after additional dilution and decay that would occur as a natural consequence of migration toward the site boundary, the tritium level would not exceed any NRC offsite limit. Although these leaks did not exceed any NRC limits, either on-site or offsite, the level of tritium triggered the licensees to initiate voluntary communications with local and state officials.

In response to these leakage events the NRC performed a collective significance review of buried piping degradation. In September 2009, the NRC chairman issued a tasking memorandum that required the NRC staff to review the adequacy of regulations, codes, standards and industry activities related to degradation of buried piping. In response to the chairman tasking memorandum, the NRC staff prepared SECY 09-0174, "Staff Progress in Evaluation of Buried Piping at Nuclear Reactor Facilities," [47] which concluded that no immediate regulatory changes were necessary to address degradation of buried piping because a) leakage from buried piping was of low safety significance with respect to structural integrity of the piping, and b) the amount of radioactive material that has been released has been a small fraction of regulatory limits. Additionally, SECY 09-0174 described a number of ongoing NRC staff codes and standards, and industry activities. The NRC staff identified plans to review operating experience to continue to validate conclusions in the SECY paper and indicated it would continue its participation in codes and standards organisations efforts to incorporate changes in the state-of-the-art with respect to maintenance and evaluation of buried piping. On 14 September 2010, the NRR action plan on buried piping (i.e. the action plan) [48] was issued which outlined a plan to track the action items in SECY 09-0174. The action plan also tracked interaction with industry to understand whether, by 2015, their buried piping and underground piping and tanks integrity initiatives (discussed below) ultimately reduced the incidence of degradation and leaks.

Independent of the NRC actions, in November 2009, the nuclear industry issued their "buried piping integrity initiative," an executive level inter-utility agreement to address degradation of buried piping. The NRC staff identified actions necessary to understand the breadth of implementation and effectiveness of this initiative. Additionally, the NRC staff identified actions related to licence renewal, new reactors, and the need to communicate about buried piping issues with licensees and other stakeholders

In September 2010, the industry developed the "underground piping and tanks integrity initiative," which extended the objectives and actions in the buried piping integrity initiative to all buried and underground piping and tanks that are not inside buildings regardless of whether or not they are in direct contact with soil. Actions in this plan that previously applied only to the buried piping integrity initiative applied to the full scope of both industry initiatives. The NEI report, "Guideline for the Management of Underground Piping and Tank Integrity" [49] establishes the goals and requirements of the underground piping and tanks integrity initiative (i.e. the initiative). The goals of the initiative are to (1) proactively assess and manage the condition of in-scope piping and tanks, (2) share operating experience within the industry, (3) guide the development of technologies that improve available inspection and analysis techniques and (4) "Improve regulatory and public confidence in the industry's management of the material condition of its underground tanks and piping systems." To meet the goals of the Initiative, industry agreed to implement certain measures deemed as requirements and others deemed as recommendations outlined in four elements:

Procedures and oversight: (a) ensures clear roles and responsibilities including senior level accountability and (b) Develops and maintains an underground piping and tank integrity programme document and implementing procedures:

1. risk ranking and/or prioritisation: (a) requires risk ranking of in-scope items and provides risk ranking categories, and (b) prioritises items for inspection;
2. inspection plan: requires the development and maintenance of an inspection plan that provides reasonable assurance of piping and tank integrity, and;
3. asset management plan: requires the development of an asset management plan based on inspection results.

A plan of action and milestones was established to complete the four elements of the initiative above, including the execution of the inspections plans. As of 2016, all four elements were completed by all operating plants and the Initiative has transitioned into ongoing plant asset management programs based on the asset management plans developed in element 4 above [50].

During 2010-2011, the Government Accountability Office (GAO) performed a review of NRC activities related to buried and underground piping. On 3 June 2011, the GAO issued GAO-11-563, "Oversight of Underground Piping Systems Commensurate with Risk, but Proactive Measures Could Help Address Future Leaks" [51], which contained a recommendation for the NRC staff to keep abreast of emerging inspection technology. A milestone was added to revision two of the action plan in November 2011 to specifically address the GAO recommendation.

On 15 August 2011, the commission issued a staff requirements memorandum (SRM) for SECY 2011-0019 [52] that approved the NRC staff's continued efforts to work with industry initiatives and consensus standards organisations. This SRM also stated "if, based on its participation in consensus standard activities the staff determines that revisions to the agency's regulations are necessary to incorporate changes to the ASME codes related to groundwater protection, the staff should seek commission approval via a notation vote paper." An action item was added to revision two of the action plan to address this requirement. Furthermore, the NRC completed activities associated with temporary instruction TI-182 and verified all plants were following the industry's buried piping integrity initiative and underground piping and tanks integrity initiatives [53]. By November 2015, the NRC had completed all action items and closed the action plan.

Over the course of the six years while the action plan was in place leakage associated with buried piping and underground tanks, when it has occurred, a) has been of low safety significance with respect to structural integrity, and b) the amount of radioactive material that has been released has been a small fraction of regulatory limits. Furthermore, over that time period rates of significant leakage events as tracked by the Institute of nuclear power operations initially increased and has since exhibited a decreasing trend consistent with improved maintenance and inspection practices. Reported significant leaks, those in safety related piping or in piping containing environmentally hazardous material, increased from eight to 15 from 2009 to 2010, but have since decreased to eight in 2011, five in 2012, four in 2013, and three in 2014. Reporting of buried piping degradation and failure is done through:

- ASME XI ISI owner activity reports. Include information on repair/replacement activities associated with buried Code Class three piping.
- NRC inspection reports (IRs).
- NRC event notification reports.
- Licensee event reports.
- INPO ICES/EPIX databases. This information is proprietary. Data is input by each utility in the US. The amount of data provided varies by utility.
- Licence renewal process. Extensive operating experience data available on buried piping, but with focus on buried condensate system piping, fire protection water system piping, and service water system piping.

The industry continues to share buried piping performance information and develop inspection technologies through the Buried Piping Integrity Group (BPIG) established in 2008 by EPRI. The objectives of the BPIG are to (1) exchange buried piping experience among utilities, (2) sponsor technical investigations, (3) inform plant owners of relevant technologies, (4) support BPWORKS software, (5) provide training and (6) "develop and support industry standard approaches to deal with buried piping." [28] The NRC continues to monitor buried piping performance and engage stakeholders through its various reporting activities mentioned above, ASME Code activities, and public meetings.

4.5.12. International programmes

Leakage events due to primary water stress corrosion cracking (PWSCC) or interdendritic stress corrosion cracking (IDSCC) have been recorded in the United States and internationally. This cracking has been observed at several weld locations in reactor coolant systems including penetrations to the reactor vessel (e.g. control rod drive mechanism (CRDM) penetrations, bottom-mounted instrumentation (BMI) penetrations, and nozzle penetrations), and nozzle penetrations on steam generator and pressuriser components. In-service inspections (ISI) are conducted at nuclear power plants to detect cracks before leakage occurs. The effectiveness of ISI is dependent on several factors such as the frequency with which periodic examinations occur, human factors, the performance capability of the non-destructive examination (NDE) procedures and techniques used, etc. Leakage events, both domestic and international, have indicated a need for additional research to evaluate the performance of NDE procedures and techniques for the detection and sizing of PWSCC and IDSCC flaws in reactor components.

In February 2012, the US Nuclear Regulatory Commission executed agreements with organisations in Finland, Japan, Republic of Korea, Sweden, Switzerland, and the United States to establish the programme to assess the reliability of emerging non-destructive techniques (PARENT) to investigate the performance of current and emerging NDE techniques to find flaws in nickel-alloy welds and base material. This assessment was performed by conducting a series of open and blind international round-robin tests on a set of component mock-ups. The project was divided into open and blind testing to separate the evaluation of novel techniques implemented by nonqualified teams from the evaluation of more established techniques implemented by commercial inspection service providers. The objective of the blind test was to obtain quantitative empirical estimates of the performance of contemporary NDE inspection procedures and techniques used within the industry to determine which of these may be more reliable for detecting and accurate sizing of PWSCC or IDSCC flaws. The objective of the open testing was to evaluate the performance of novel NDE procedures and techniques that have not yet reached the maturity appropriate for field testing.

The PARENT blind test results provide quantified estimates for the performance of various NDE procedures as applied to large-bore and small-bore dissimilar metal welds (DMW) test blocks with crack defects [54]. The data generated from PARENT provides empirical evidence of the impact of test block size and procedure variables on NDE performance. Although the test conditions were less challenging than field conditions, data were collected for all test blocks and procedures under consistent conditions. Thus, conclusions may be derived regarding relative performances that should also be applicable under field conditions. The results generated from blind testing can also be used to inform analyses of the effectiveness of NDE and ISI performed in nuclear power plants. The data generated by the PARENT blind testing provides insight into capabilities of current non-destructive

28.. EPRI Buried Piping Integrity Group Meeting Facebook Page: www.facebook.com/events/427080333999196/ accessed on 26 February 2018 methods used to detect cracks in reactor components and the data from the open testing will provide insight into capabilities of more experimental non-destructive methods. These insights can be used in developing regulatory positions and to help direct additional research activities.

The PARENT open test phase assessed several non-standard NDE procedures and techniques such as nonlinear ultrasonic testing (NLUT) and advanced phased array UT (ADVPAUT). According to NUREG/CR-7 236 [55] the ADVPAUT and NLUT procedures exhibited more consistent depth sizing error over the range of flaw depths sampled in comparison to established PAUT procedures. According to NUREG/CR-7 236, "the results obtained in open testing can be considered optimistic to what would be anticipated under blind test conditions or field conditions. Thus, the PARENT open test results have illustrated that NDE procedures and techniques are limited to less than ideal performance by their fundamental capability and that overall ISI effectiveness could benefit from NDE technology advancements vi that improve upon fundamental capability."

5. Topical reports

During the second term of CODAP the PRG produced three topical reports. These reports constitute CODAP event database insights reports and are intended as "portals" for future database application projects and in-depth studies of selected degradation mechanisms. The three reports have been approved by the CSNI and are in the public domain; www.oecd-nea.org/nsd/docs/indexcsni.html.

5.1. Passive component reliability and integrity management (RIM)

Effectiveness of reliability and integrity management (RIM) practices was selected as the subject of the third CODAP Topical Report [56]. The report addresses selected international practices with respect to pressure testing, leak detection, in-service inspection including non-destructive examination (NDE), and performance demonstration initiatives to improve the reliability of NDE techniques. The purpose of RIM is to prevent the occurrence of piping through-wall leaks as well as to monitor passive metallic component degradation. RIM programmes can utilise risk insights to augment or enhance existing deterministic integrity management programmes. Through a systematic examination of the operating experience as recorded in the CODAP event database, the field experience with the different RIM strategies has been evaluated in order to primarily draw qualitative insights about integrity management reliability.

The report documents how RIM strategies are accounted for in the CODAP event database. According to the CODAP coding guideline that has been prepared by and adopted by the PRG, for each record an evaluation is performed of the various RIM-influences that have played a role in preventing or contributing to a structural failure. Hence, the database includes a significant volume of in-service inspection (ISI) information from which valuable insights about RIM performance issues can be drawn. It is quite clear that RIM very significantly contributes to a high level of structural integrity. The operating experience insights also point to RIM implementation challenges.

The CODAP event database captures instances of less-than-adequate (LTA) RIM, including failures in detecting pre-existing flaws before exceeding acceptance criteria. CODAP uses a broad definition of LTA-RIM in that the term is defined as events where degradation has progressed beyond acceptable limits in systems, structures or components (SSCs) that have a RIM programme. These LTA-RIM events have some safety significance. In this topical report the LTA-RIM definition is broadened to also include events where a RIM programme has resulted in a "false positive"; that is, it has identified degradation that either didn't exist or was not close to violating acceptance criteria. While such events needlessly expend resources and could be considered LTA-RIM from an economic perspective, they do not have any safety significance. In the database passive component failure information is recorded in a tiered manner. All data submissions undergo verification for technical accuracy and completeness in accordance with procedures and protocols established by the CODAP PRG. First, basic failure information is recorded to address the most fundamental information about an event and this includes a free-format event narrative that describes the sequence of events, including plant response, consequence, in-plant location of failed component, dimensional data, and component type. This is followed by recording the known ISI history, including the date when the failed component was last inspected, the method of NDE qualification if a qualified method had been used, and any NDE performance deficiencies or failures. Finally, details about the service environment (e.g. water chemistry, stresses, pressure and temperature) are recorded as a lead-in to details from root cause evaluations (flaw data, chemical composition of material, results of metallographic examinations, apparent and underlying causes of material degradation). When RIM fails, one or more of the following factors are often present:

- Accepting a rejectable flaw indication for continued operation. This could be due to misinterpretation of NDE results.
- Rationalising away detected defects.
- Using improperly qualified or modified NDE techniques or not selecting the correct procedure to implement.
- Poorly implementing qualified procedures.
- Poorly implementing owner-defined inspection programs.
- Not identifying the correct location to inspect.
- Missing a flaw with a qualified procedure. A procedure may not sufficiently document the basis for the examination details used to inspect for a specific, previously observed degradation mechanism. There may also be inadequate administrative controls for augmented inspections and disposition of inspection results.
- Experience from examinations in the field allows the conclusion that the sensitivity of advanced UT examination techniques is so high that the detection of material flaws, among them also crack-like flaws, does not pose a problem in general. The difficulty lies rather more in the characterisation and assessment of the indications, especially if these are actually outside the validity scope that has been defined for the examination techniques by their qualification. Examples from volumetric examinations performed in the filed show that there can be a tendency to put greater emphasis on assessing the examining technician's results in the "impermissible scope", which may then lead to large discrepancies between the determined and the actual dimensions of the flaw.

According to the high-level data analysis, the number of instances of LTA-RIM has remained largely the same over the past three decades. The rate of LTA-RIM is a function of the number of such instances versus the overall number of examinations that have been performed, however. There has been a very significant evolution in RIM practices and requirements, and therefore qualified statistical insights concerning the reliability of RIM programmes necessitates an in-depth analysis of the field experience data as collected by CODAP.

With respect to the continued database development and maintenance (i.e. data submissions and validation) it is recommended that the following items be considered in the ongoing active data submission activities by the CODAP PRG Members as well as in the current programme for an enhanced version of the online database:

- Encourage the PRG membership to more actively share RIM experience insights. As a standing action, future working group meetings should focus on technical discussions regarding how to utilise CODAP and how to share data analysis insights with the nuclear safety community.
- Expand the sharing of operating experience data within the PRG. Future working group meetings should include as a standing action, national overviews of recent operational events, including the findings of root cause analyses.
- For the PRG member states that have implemented RI-ISI, add appropriate database fields that indicate events that involve reactor components that are included in a RI-ISI programme. Having access to this information would be highly beneficial to future database applications so that the database content and inspection programme can be correlated. It is noted that the European network for inspection and qualification (ENIQ) has undertaken an evaluation of lessons learned from the application of risk-informed ISI (RI-ISI) to European nuclear power plants. The PRG membership is encouraged to review this ENIQ effort and to determine how conclusions by ENIQ correspond to the field experience data as recorded in CODAP.
- Similarly, add appropriate database fields that indicate the presence of an augmented inspection programme. The basis for this recommendation is as follows:
  - Embedded in the database are examples where an augmented inspection program is in place with the provision that a 100% volumetric examination of a given component boundary is to be performed. Yet, through-wall defects have occurred. The underlying contributing factors include use of non-qualified NDE technique, or application of too coarse UT-scanning matrix. Having the ability to quickly and reliably identify such events in the database would greatly enhance the level of user friendliness.
- Based on the results of the evaluations of the CODAP database content, the number of through wall leakages could be decreased by the following actions:
  - periodic review and independent validation of UT-scanning matrices used in inspection piping components;
  - RIM programme optimisation on the basis of probabilistic and risk-informed methodologies.

5.2. Below ground piping operating experience

The Fourth CODAP topical report [57] is a summary of the operating experience with below ground piping systems in commercial nuclear power plants. Through an examination of the operating experience as recorded in the CODAP event database, the field experience with the different below ground piping systems is evaluated in order to draw qualitative and quantitative insights about the damage and degradation mechanisms and their potential plant operability and safety impacts.

Consequences of a below ground pipe failure on plant operation can be direct impacts (as in flow diversion and loss of the affected train or system or an initiating event as analysed in probabilistic safety assessment studies) or indirect impacts (e.g. the failure results in depletion of a tank and loss of the systems supplied by the tank). The operating experience as recorded in the CODAP event database includes examples of below ground pipe failures that have had multi-unit impacts as well as caused flooding of equipment areas and utility tunnels. An example of an initiating event could be loss of service water or dual-unit loss of service water. Another example of a significant buried piping failure includes the loss of fire protection water due to a fire protection header break coincident with a fire suppression demand.

Since its inception in 2002, operating experience with below ground piping has been an intrinsic aspect of the technical scope of the CODAP database project. Specifically, CODAP collects data on below ground pipe failures with operational impacts as well as potential safety impacts. The scope of the CODAP event database is to collect, evaluate and exchange operating experience data on metallic passive components. In the database the earliest recorded buried pipe failure dates from April 1976 when a significant (ca. 3 kg/s) through-wall leak developed in a buried Service Water system pipe line at a US BWR plant.

The report includes an example of how the CODAP event database can be used to obtain quantitative estimates of below ground piping reliability. Specifically, this example addresses the reliability of buried (or inaccessible) essential service water (ESW) piping and includes a quantitative comparison of inaccessible versus accessible ESW piping reliability.

Most, if not all commercial nuclear power plants have extensive below ground piping systems that transport cooling water to and from the plant, fire protection system water, emergency diesel generator fuel oil, instrument air, and water containing radioactive isotopes (e.g. tritium). The amount and type of below ground piping systems vary significantly among nuclear power plants. As nuclear power plants age, their below-ground piping systems tend to corrode, and since these systems are largely inaccessible it can be challenging to determine their structural integrity. The report includes the results of a survey of below ground piping systems in CODAP-PRG member countries.

Some CODAP member countries (e.g. Canada, France, Spain and the US) have implemented a risk-ranking methodology to identify the specific below ground piping locations that are most susceptible to degradation and failure. This risk-ranking methodology has been developed by EPRI with support from plant operators and ASME. A software implementation of EPRI's risk-ranking methodology was released in 2008.

With respect to the continued database development and maintenance (i.e. data submissions and validation) it is recommended that the following items be considered in the ongoing data submission activities by the CODAP PRG Members as well as in the current program for an enhanced version of the online database (see Section nine for details):

- Encourage the PRG Membership to more actively share below ground piping operating experience insights. As a standing action, future Working Group meetings should expand the focus on technical discussions regarding how to utilise CODAP and how to share data analysis insights with the nuclear safety community.
- Within the PRG Membership, share insights from ageing management programme audits with focus on below ground piping, including the associated NDE experience.
- On the basis of the CODAP event database, the PRG membership should consider how to perform risk categorisation of below ground piping systems, conditional on

different degradation susceptibilities and different reliability and integrity management (RIM) strategies.

- Expand the sharing of operating experience data within the PRG. Future working group meetings should include as a standing action, national overviews of recent operational events, including the findings of root cause analyses; the technical as well as organisational factors contributing to material degradation and failures.
- The Working Group on Risk Assessment (WGRISK) of the Committee on the Safety of Nuclear Installations (CSNI) is planning the "Joint Workshop on Use of NEA Data Project Operating Experience Data for Probabilistic Risk Assessment." It is recommended that the CODAP PRG Membership actively support this initiative and present insights from database application such as the buried ESW piping reliability assessment as documented in Section 4.4 of the fourth Topical Report.

5.3. Basic principles of collecting OE data on passive metallic components

The fifth CODAP topical report [58] documents the CODAP event database structure and the underlying principles of collecting operating experience data on metallic passive components. The report represents a summary of the CODAP operating procedures, the CODAP event database coding guideline, and the CODAP applications handbook. An event database on passive component degradation and failure has been operated since May 2002 in NEA. The objective of CODAP is to collect information on passive metallic component degradation and failures of the primary system, reactor pressure vessel internals, main process and safety systems, and support systems. It also covers non-safety-related components with significant operational impact. An effort is underway to systematically evaluate the database content and to make a series of database insights reports available to material scientists as well as risk management practitioners. Data exchange among participating organisations enables comparisons of the different national practices regarding reliability and integrity management of passive components.

The CODAP database improvement plan to be implemented in two phases over an 18-month period. Specifically, the CODAP Database Improvement involves certain subtle modifications to the existing software to improve the user friendliness and an effort to produce an advanced, state-of-the-art database user interface. The CODAP PRG faces two important future challenges. Firstly, while efforts have been made to promote CODAP and associated data project products to the nuclear safety community at large, there remain programmatic issues relative to how to make the restricted CODAP event database available to PSA practitioners as well as material scientists. Secondly, work remains to be done relative to the development of PSA-centric database application guidelines and associated analytical infrastructure (i.e. piping reliability analysis techniques and tools).

In the context of nuclear plant ageing management, structural integrity assessments and probabilistic safety assessment (PSA), a fundamental objective of an event database such as CODAP is to provide complete and comprehensive information on the field experience so that independent and accurate "measurements" of material performance can be obtained, including the identification of adverse trends.

With respect to the continued database development and maintenance (i.e. data submissions and validation) it is recommended that the following items be considered in the ongoing active data submission activities by the CODAP PRG members as well as in the current programme for an enhanced version of the online database:

- Encourage the PRG Membership to more actively share metallic passive component operating experience insights. As a standing action, future working group meetings should focus on technical discussions regarding how to utilise CODAP and how to share data analysis insights with the nuclear safety community.
- Expand the sharing of operating experience data within the PRG. Future working group meetings should include as a standing action, national overviews of recent operational events, including the findings of root cause analyses.

6. Database accessibility

The CODAP terms and conditions contain statements on the use of data within or outside the CODAP project and on the handling of proprietary information. The event database is a restricted database and its access is limited to participating organisations that provide input data. The restricted database is available on the internet via a secure server located at the OECD-NEA headquarters.

It has been recognised by the management board that many member organisations will want to pass on the CODAP database to their consultants for use in specific projects, and suchlike. For this purpose, a non-confidential version of the CODAP database will be made available for use by consultants for a limited period of time. Before supplying a non-confidential version, the member organisation making the request must provide the national co-ordinator with written proof that the intended recipient of the non-confidential version of the database has agreed to comply with the confidentiality terms and conditions of the project.

The planned software upgrades (see Section 7.2) will provide new facilities for accessing the database by technical support organisations as well as other non-member organisations.

7. Conclusions and future plans

The second term of the Component Operational Experience, Degradation and Ageing Programme (CODAP) project officially commenced on 1 February 2015 with 11 Nuclear Energy Agency (NEA) member countries ageing to exchange metallic passive component failure data. Finland and Sweden decided to no longer participate in the project. Based on the second term accomplishments, summarised below are the conclusions and recommendations of the CODAP project review group.

7.1. Conclusions and recommendations

The objectives of the second term of the CODAP project were to:

- Collect and analyse information on passive metallic component degradation and failures to promote a better understanding of underlying causes, impact on operations and safety, and prevention.
- Analyse the information collected in the event database to develop topical reports on degradation mechanisms. Objectives and schedules for the topical reports will be developed for each calendar year of project operation. CODAP will actively seek technical input from the NEA CSNI Working Group on Integrity and Ageing of Components and Structures (WGIAGE). In addition, the project review group will communicate and co-ordinate as needed with WGIAGE concerning technical matters of mutual interest.
- Develop and implement an enhanced web-based event database that supports the creation of standard and custom reports on certain aspects of the database content. Building on the experience with the existing web-based event database, the new development will address user-friendliness, improved database structure, and analysis tools that enable advanced statistical analyses of the database content.
- Provide ageing management programme support that addresses current operability determination practices, performance of new materials in the field (e.g. dual-certification stainless steels, super-austenitic stainless steels, alloy 690, alloy 52/152), and commendable practices of licence renewal and long-term operation.
- Facilitate the exchange of the existing and future information amongst the participating organisations as a way to improve the quality of decisions made about components material degradation, ageing management and operability determination. The CODAP database along with other relevant information collected will be used for applications of service experience data with an emphasis on observed trends-and-patterns, past and current degradation mechanism mitigation practices, and risk characterisation of passive component failure events.

These objectives were largely met. While an effort was made to reach a consensus on a new software specification, the actual programming effort was deferred to the third term of the project. As indicated in Sections four (CODAP event database) and five (topical reports), the current content of the database has a strong US bias. A recommendation for the third term (2018-2020) of the project is to put in place operating procedures and processes whereby future national data submissions are commensurate with the number of operating reactors. Furthermore, in-depth database applications will be pursued to investigate the correlations between reported degradation and failure events versus piping system design modifications, degradation mitigation practices and NDE qualification.

It is equally important to put in place a process to capture legacy information concerning significant events. In the context of CODAP, the term "significant" implies to both significant unexpected structural degradation or failure and events that have prompted significant regulatory action. Database completeness strongly affects the possibilities to perform advanced database applications.

The CODAP MB faces two important future challenges. Firstly, while efforts have been made to promote CODAP and associated data project products to the nuclear safety community at large, there remain programmatic issues relative to how to make the restricted CODAP event database available to PSA practitioners. Secondly, work remains to be done relative to the development of PSA-centric database application guidelines and associated analytical infrastructure (i.e. piping reliability analysis techniques and tools). Two initiatives are under consideration by the MB to address the stated challenges. The Working Group on Risk Assessment (WGRISK) of the Committee on the Safety of Nuclear Installations (CSNI) is planning the "Joint Workshop on Use of NEA Data Project Operating Experience Data for Probabilistic Risk Assessment." The CODAP MB intends to actively support this joint workshop. Additionally, a proposal has been made for an international benchmark exercise concerning the use of operating experience data to quantify piping reliability parameters for input to a standard problem application; e.g. risk informed operability determination.[29]

7.2. Planned activities beyond 2017

The management board recognises that there are a multitude of future challenges concerning the response to environmental degradation of passive components in heavy water and light water reactor operating environments. It is important to ensure that the almost five decades of operating experience insights are preserved and made readily available to future generations of material scientists, structural engineers and PSA engineers. In planning for activities beyond 2017 questions concerning the effectiveness of degradation mitigation processes and NDE reliability need to be addressed. One way of doing so is to actively monitor any perceived or actual trends and patterns in the worldwide operating experience data that is fed back to the CODAP event database. The scope of the event database will be expanded to also address degradation and failure of high density polyethylene (HPDE) piping.

The CSNI programme review group in 2014 recommended that the CODAP project implements operating procedures and processes whereby future national data submissions are commensurate with the number of operating reactors. This target was already taken into

29. The topic of an international benchmark exercise has been under discussion since the inception of the OPDE/CODAP project.

account during second term of CODAP, but during third term more work will be done to achieve a more “balanced” event database.

The Management Board has prepared Terms and Conditions for the 3rd Term (2018-2020) of CODAP. Prior to the CODAP14 meeting (October 2017) the Radiation and Nuclear Safety Authority (STUK) of Finland indicated its intent to re-join the project in 2018. Similarly, the Authority for Nuclear Safety and Radiation Protection (ANVS) of the Netherlands has indicated its intent to join the project as a new member. The third term of the project places an emphasis on two aspects of operating experience data exchange and analysis. First, to encourage active data submissions by the MB membership, an improved web-based database structure will be implemented. Second, continued database applications will be pursued through an expanded programme to develop topical reports.

8. References

[1] Gibbons, W.S. and Hackney, B.D., Survey of Piping Failures for the Reactor Primary Coolant Pipe Rupture Study, GEAP-4574, Atomic Power Equipment Department, General Electric, San Jose, CA, 1964.

[2] Kustu, O., Identification of Significant Problems Related to Light Water Reactor Piping Systems, ALO-89, Sandia National Laboratories, Albuquerque, NM, 1980.

[3] U.S. Nuclear Regulatory Commission, Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants, NUREG-75/067, Washington, DC, 1975.

[4] U.S. Nuclear Regulatory Commission, Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors, NUREG-0691, Washington, DC, 1980.

[5] Brown, E.J., Erosion in Nuclear Power Plants, AEOD/E4-16, Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, Washington, DC, 1984.

[6] Cragnolino, G., Czajkowski, C., and Shack, W.J., Review of Erosion-Corrosion in Single-Phase Flows, NUREG/CR-5156, U.S. Nuclear Regulatory Commission, Washington, DC, 1988.

[7] IAEA, Corrosion and Erosion Aspects in Pressure Boundary Components of Light Water Reactors, Proc. Specialist Meeting Organized by the IAEA, IWG-RRPC-88-1, Vienna, Austria, 1990.

[8] OECD Nuclear Energy Agency, Experience with Thermal Fatigue in LWR Piping Caused by Mixing and Stratification, Specialist Meeting Proceedings, NEA/CSNI/R(98)8, Boulogne-Billancourt, France, 1998.

[9] U.S. Nuclear Regulatory Commission, Reactor Safety Study. An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400 (NUREG-75/014), Washington, DC, 1975.

[10] Swedish Nuclear Power Inspectorate, Seminar on Piping Reliability, SKI Report 1997:32, Stockholm, Sweden, 1997.

[11] Lydell, B., Failure Rates in Barsbäck-1 Reactor Coolant Pressure Boundary Piping, SKI Report 1998:30, Swedish Nuclear Power Inspectorate, Stockholm, Sweden, 1999.[30]

[12] Swedish Nuclear Power Inspectorate, Reliability of Piping System Components. A Resource Document for PSA Applications, SKI Report 1995:58, Swedish Nuclear Power Inspectorate, Stockholm, Sweden (1995).

[13] Swedish Nuclear Power Inspectorate, Reliability of Piping System Components. Framework for Estimating Failure Parameters from Service Data, SKI Report 1997:26, Stockholm, Sweden (1997).

[14] OECD Nuclear Energy Agency, OECD/NEA Piping Failure Data Exchange Project (OECD/NEA OPDE), Final Report, NEA/CSNI/R(2012)16, Boulogne-Billancourt, France, 2012.

30. All cited SKI-reports are available from www.ssm.se.

[15] OECD Nuclear Energy Agency, Technical Basis for Commendable Practices on Ageing Management – SCC and Cable Ageing Project (SCAP), Final Report, NEA/CSNI/R(2010)5, Boulogne-Billancourt, France, 2010.

[16] U.S. Nuclear Regulatory Commission, Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR Report, NUREG-2191, Washington, DC, 2017.

[17] OECD Nuclear Energy Agency, OECD-NEA Piping Failure Data Exchange Project (OPDE) Workshop on Database Applications, OPDE/SEC(2004)4, Boulogne-Billancourt, France, 2005.

[18] Nordic PSA Group, 'R-Book' – Reliability Data Handbook for Piping Components in Nordic Nuclear Power Plants, 1st Edition, Project 04-007, Stockholm, Sweden.[31]

[19] OECD Nuclear Energy Agency, Use of OECD/NEA Database Project Products in Probabilistic Safety Assessment, NEA/CSNI/R(2014)2, Boulogne-Billancourt, France, 2014.

[20] Lydell, B.O.Y., Piping Reliability Analysis Guidelines for PSA Practitioners, Proc. 2013 Nordic PSA Conference, Stockholm, Sweden, 2013.

[21] Dominion Energy Kewaunee, Inc., Kewaunee Power Station Flooding Significance Determination Process Risk Assessment Report, Serial No. 05-746, Kewaunee, WI, 2005. NRC ADAMS Accession No. ML053180483.

[22] Fleming, K.N. and Lydell, B.O.Y., "Insights into Location Dependent Loss-of-Coolant-Accident (LOCA) Frequency Assessment for GSI-191 Risk-Informed Applications," Nuclear Engineering and Design, **305**:433-450, 2016.

[23] OECD Nuclear Energy Agency, OECD/NEA Component Operational Experience, Degradation & Ageing Programme, Database Coding Guideline & User Manual, Revision 1, CODAP-PR01, Boulogne-Billancourt, France, 2013.

[24] OECD Nuclear Energy Agency, OECD/NEA Component Operational Experience, Degradation & Ageing Programme, Database Applications Handbook, CODAP-PR05, Boulogne-Billancourt, France, 2013.

[25] IAEA, Safety Glossary. Terminology Used in Nuclear Safety and Radiation Protection, STI/PUB/1290, Vienna, Austria, 2007.

[26] IAEA, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: BWR Pressure Vessel Internals, IAEA-TECDOC-1471, Vienna, Austria, 2005.

[27] IAEA, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: PWR Pressure Vessel Internals, IAEA-TECDOC-1119, Vienna, Austria, 1999.

[28] American Society of Mechanical Engineers, 2007 ASME Boiler and Pressure Vessel Code, Section XI: Rules for Inservice Inspection of Nuclear Power Plant Components, 2008a Addenda, New York, NY, July 2008.

[29] Economou, J., Thebault, Y. and Costes, P-E., "Small Bore Pipe Branch Connections Fatigue," PVP2011-57893, Proc. ASME Pressure Vessels and Piping Conference, American Society of Mechanical Engineers, New York, NY, 2011. http://proceedings.asmedigitalcollection.asme.org/proceeding.aspx?articleid=1628106

31. Available in electronic form on a CD; access to the Handbook is restricted to NPSAG member organisations. For additional information, go to www.npsag.org/home/

[30] OECD Nuclear Energy Agency, CODAP Topical Report on Flow Accelerated Corrosion of Carbon Steel and Low Alloy Steel Piping in Commercial Nuclear Power Plants, NEA/CSNI/R(2014)6, Boulogne-Billancourt, France, 2014.

[31] Canadian Standards Association (CSA) Document, N285.4-05, “Periodic inspection of CANDU nuclear power plant Component”, 2005.

[32] Eom, S., Liu, S., Jin, J., (2016), “Overview of Aging Management Programme for CANDU Feeders – Canadian Regulatory Perspectives”, Paper PVP2016-63545, Proceedings of the ASME 2016 Pressure Vessels and Piping Conference - PVP2016, 17-21 July 2016, Vancouver, British Columbia, Canada.

[33] John P. Slade and Tracy S. Gendron, “Flow Accelerated Corrosion and Cracking of Carbon Steel Piping in Primary Water - Operating Experience at the Point Lepreau Generating Station”, Proceedings of the 12th International Conference on Environmental Degradation of Materials in Nuclear Power System – Water Reactors, 2005.

[34] D.H. Lister, J. Slade, and N. Arbeau, “The Accelerated Corrosion of CANDU Outlet Feeders – Observations, Possible Mechanisms and Potential Remedies”, Proceedings of 18th Annual the Canadian Nuclear Society Conference, Toronto (1997).

[35] John C. Jin, Raoul Awad and Tomas Viglasky, “Fitness for Service assessment of degraded feeder piping – Canadian regulatory expectations”, Transactions, SMiRT 19, Toronto, August 2007 Paper # D02/3.

[36] Western European Nuclear Regulators Association, Safety Reference Levels for Existing Reactors, 2014 (www.wenra.org/)

[37] Fleming, K.N. and Lydell, B.O.Y., Guidelines for Performance of Internal Flooding Probabilistic Risk Assessment, 1019194, Electric Power Research Institute, Palo Alto, CA, 2009.

[38] B. Lydell, A. Olsson, Reliability Data for Piping Components in Nordic Nuclear Power Plants “R-Book” Project Phase 1, Report 2008:1, Swedish Nuclear Power Inspectorate, Stockholm, Sweden, 2008. (http://ssm.se)

[39] V-H. Swaling, A. Olsson, Reliability Data Handbook for Piping Components in Nordic Nuclear Power Plants – R Book, Phase 2, Report 2011:06, Swedish Radiation Safety Authority, Stockholm, Sweden, 2011. (http://ssm.se)

[40] Shoji, T., Materials Ageing Degradation Programme in Japan and Proactive Ageing Management in NPP, MINOS Workshop on Materials Innovation for Nuclear Optimized Systems, Commissariat à l'énergie atomique, Saclay Nuclear Research Centre, Saclay, France, 2012. http://dx.doi.org/10.1051/epjconf/20135104001

[41] U.S. Nuclear Regulatory Commission, Service Water System Problems Affecting Safety-Related Equipment, Supplement 1, Washington, DC, 1990.

[42] U.S. Nuclear Regulatory Commission, Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2 and 3 Piping, Generic Letter 90-05, Washington, DC, 1990.

[43] Consejo de Seguridad Nuclear, Aging Management and Long-Term Operation of Nuclear Power Plants, Instruction IS-22, Madrid, Spain, 2009.

[44] Western European Nuclear Regulators Association, Topical Peer Review 2017: Ageing Management Technical Specification for the National Assessment Reports, 2016. www.wenra.org/media/filer_public/2017/05/15/2016-12-21_tpr_specification_final.pdf

[45] Eidgenössisches Nuklearsicherheitsinspektorat, Ageing Management, Guidelines for the Swiss Nuclear Power Plants, ENSI-B01, Brugg, Switzerland, 2012.

[46] Eidgenössisches Nuklearsicherheitsinspektorat, Swiss National Assessment Report for the Topical Peer Review of Ageing Management in Nuclear Power Plants, Brugg, Switzerland, 2017.

[47] Leeds, E.J., Staff Progress in Evaluation of Buried Piping at Nuclear Reactor Facilities, SECY 09-0174, U.S. Nuclear Regulatory Commission, Washington, DC, 2009.

[48] U.S. Nuclear Regulatory Commission, NRR Action Plan, Revision 1, Buried Piping, Washington, DC, 2010 (NRC-ADAMS Accession No. 2010, ML102590171).

[49] Nuclear Energy Institute, Guideline for the Management of Underground Piping and Tank Integrity," NEI 09-14 (Rev. 4), Washington, DC, 2015.

[50] Richter, Mark, "Industry Programme to Address Underground Pipe and Tank Assets," Public Meeting Between the Nuclear Regulatory Commission and Industry Representatives to Discuss Buried and Underground Piping 20 September 2016, Rockville, MD, NRC-ADAMS Accession No. ML16251A145.

[51] United States Government Accountability Office, Nuclear Regulatory Commission Oversight of Underground Piping Systems, GAO-11-563, Washington, DC, 2011.

[52] Hardies, R.O., Summary of Results from NRC Temporary Instruction 2515/182, Review of the Implementation of the Industry Initiative to Control Degradation of Underground Piping and Tanks, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC, 2016.

[53] U.S. Nuclear Regulatory Commission, "Review of the Implementation of the Industry Initiative to Control Degradation of Underground Piping and Tanks," Temporary Instruction 2515/182, NRC Inspection Manual, Washington, DC, 2015.

[54] Meyer, R.M. and Heasler, P.G., Results of Blind Testing for the Program to Assess the Reliability of Emerging Nondestructive Techniques, NUREG/CR-7235, U.S. Nuclear Regulatory Commission, Washington, DC, 2017.

[55] Meyer, R.M., Holmes, A.E. and Heasler, P.G., Results of Open Testing for the Programme to Assess the Reliability of Emerging Nondestructive Techniques, NUREG/CR-7236, U.S. Nuclear Regulatory Commission, Washington, DC, 2017.

[56] Nuclear Energy Agency, CODAP Project Topical Report on Operating Experience Insights into Pressure Boundary Component Reliability and Integrity Management, NEA/CSNI/R(2017)3, Boulogne-Billancourt, France, 2017.

[57] Nuclear Energy Agency, CODAP Project Topical Report on Operating Experience Insights into Below Ground Piping, NEA/CSNI/R(2018)2, Boulogne-Billancourt, France, 2018.

[58] Nuclear Energy Agency, CODAP Project Topical Report on Basic Principles of Collecting OE Data on Passive Metallic Components, NEA/CSNI/R(2018)12, Boulogne-Billancourt, France, 2018.

Annex A. CODAP PRG Activity report

A.1 PRG meetings

During the second term of the CODAP project, the project review group met on six occasions per Table A-1.

Table A-1: 2nd Term Project Review Group Meetings 2014-2017

Meeting	Location	Date(s)
CODAP09, 2nd Term Kick-off Meeting	NEA Headquarters; Issy-les-Moulineaux	11 December 2014
CODAP10, National Co-ordinators Meeting	NEA Headquarters; Issy-les-Moulineaux	5-6 May 2015
CODAP11, National Co-ordinators Meeting	OECD Conference Centre; Paris	23-24 February 2016
CODAP12, National Co-ordinators Meeting	Seoul, Korea (Republic of); Meeting hosted by KINS	10-11 October 2016
CODAP13, National Co-ordinators Meeting	Cologne, Germany; Meeting hosted by GRS	3-4 May 2017
CODAP14, National Co-ordinators Meeting	NEA Headquarters; Boulogne-Billancourt	3-4 October 2017

A.2 CODAP topical reports

During the second term of CODAP the PRG produced three topical reports. These reports constitute "CODAP Event Database and Knowledge Base" insights reports and are intended as "portals" for future database application projects including in-depth studies of selected degradation mechanisms.

- NEA/CSNI/R(2017)3. Operating Experience Insights into Pressure Boundary Component Reliability and Integrity Management.
- NEA/CSNI/R(2018)2. Operating Experience Insights into Below Ground Piping. Approved for publication by the CSNI-PRG on November 8, 2017.
- NRA/CSNI/R(2018)12. Basic Principles of Collecting & Evaluating Operating Experience Data on Metallic Passive Components. This report was finalised in October 2017 and will be submitted to the CSNI-PRG for final review and approval in spring of 2018.

A.3 Conference participation

During the second term of CODAP, the project was represented at the following international conferences:

- Fourth International Conference on Fatigue of Nuclear Reactor Components, Seville, Spain, 28 September – 1 October 2015.
  - “CODAP Project Operating Experience Insights Related to Fatigue Mechanisms,” NEA/CSNI/R(2017)2.
- NACE Corrosion Risk Management Conference, Houston, TX, USA, May 23-25, 2016.
  - “Piping Corrosion Risk Management on the Basis of NEA CODAP Project Database,” Paper No. RISK2016-8327.
- 13th International Conference on Probabilistic Safety Assessment and Management, Seoul, Korea (Republic of), 2-7 October 2016.
  - “The OECD/NEA CODAP Project & Its Contribution to Ageing Management and Probabilistic Safety Assessment,” Paper No. 062, Proc. PSAM13.

Annex B. A tribute to Dr Karen Gott

With immense and lasting sadness the material science community and the OECD NEA OPDE/SCAP-SCC/CODAP working groups lost a very dear friend & highly respected colleague with the passing of Dr Karen Gott in October 2015. Karen touched us all in various and very positive ways.

Karen studied metallurgy and materials science at Imperial College, London. During a professional career that spanned four decades, she was deeply involved in research on:

- Creep crack formation in stainless steels (mechanical testing, electron and light optical metallography).
- Fracture mechanics (corrosion fatigue, residual stress measurement, non-destructive testing).
- Reactor chemistry (PWR and BWR chemistry, activity build-up including field measurements, decontamination).
- Reactor materials (surveillance testing, failure analysis, metallography of Inconel 182).

The CODAP collaborative effort in the area of material degradation will continue and Karen will stay with us. Her insightfulness and leadership during the formative years of the OPDE and CODAP projects will prevail and inspire all PRG members personally and professionally.

Figure B-1: Karen (Front-centre) and Her CODAP PRG Colleagues on 19 May 2011

Annex C. Plant system cross reference table

Table C-1: Plant Systems Cross-Reference Table

CODAP Generic(1)	Description	Czech Republic	France	Germany (7)		Finland (8)/Sweden
				AKZ	KKS	
ADS	BWR Primary Depressurization System (BWR)	N/A	N/A	TK, RA		314
AFW	Auxiliary Feedwater System		ASG	RQ		327
CC	Component Cooling Water System	TF	RRI	TF	LA	711/712
COND	Condensate System			RM, RN	LC	414/430 (3)
CRD	Control Rod Drive (Insert/Removal/Crud Removal)	--	RGL			354
CS	Containment Spray System	TQ	EAS			322
CVC	Chemical & Volume Control System (PWR)		RCV	TA, TC,	KB	334
Make-up Water	Water Inventory Control Function of the CVC System (PWR)		REA			
CW	Circulating Water System/Intake Cooling Water		CRF			443
EHC	Electro Hydraulic Control System					442
EXT	Steam Extraction System		CEX			419/423
FPS	Fire Protection Water System	C-52	JPx		SGA	762
FW	Main Feedwater System		ARE	RL	LA	312/415 (4)
HPCS	High Pressure Core Spray (BWR)	N/A	N/A	TJ		--
HPSI	High Pressure Safety Injection (PWR)	TJ	RIS	TH	JN	--
IA	Instrument Air System	US	CAS			484
LPCI	Low Pressure Coolant Injection (BWR)	N/A	N/A			323 (5)
LPCS	Low Pressure Core Spray (BWR)	N/A	N/A	TK, TM		323
KC	Demineralized Water Storage & Transfer		SED			736
LK	Nitrogen Supply System		RAZ			754
LPSI	Low Pressure Safety Injection (PWR)	TH	RIS	TH	JN	--
MS	Main Steam System		VVP	RA	LB	311/411 (6)
MSR	Moisture Separator Reheater System		GSS	RB	LB	422
RCS	Reactor Coolant System (PWR)		RCP	YA, YB,	JA, JE	313
RHR	Residual Heat Removal System	(2)	RRA	TH	JN	321
RR	Reactor Recirculation System (BWR)	N/A	N/A			313
RPV-HC	RPV Head Cooling System (BWR)	N/A	N/A	TC		326

CODAP Generic[(1)]	Description	Czech Republic	France	Germany [(7)]		Finland [(8)]/Sweden
				AKZ	KKS	
RVLIS	Reactor Vessel Level Indication System (BWR)	N/A	N/A			536
RWCU	Reactor Water Cleanup System (BWR)	N/A	N/A	TC	KB	331
SA	Service Air System	TL	SAT	TL	KL	753
SFC	Spent Fuel Pool Cooling System	TG	PTR	TG	FA	324
S/G Blowdown	Steam Generator Blowdown System (PWR)		APG	RS	LA	337
SLC	Standby Liquid Control System (BWR)	N/A	N/A			351
SW	Service Water System	VF	SEC	VE	PE	712/715

Notes:

1. See IEEE Std 805-1984 (IEEE Recommended Practice for System Identification in Nuclear Power Plants and Related Facilities) for information on system boundary definitions and system descriptions.
2. No dedicated RHR system in WWER-440 (decay heat removal is through natural circulation)
3. 414 for F1/F2/R1/R2/R3/R4 and 430 for O1/O2/O3
4. 312 for O1/O2/O3 and 415 for F1/F2/R1/R2/R3/R4. Also note that 312 is the designation for steam generators in Ringhals-2/3/4
5. Forsmark-3 & Oskarshamn-3
6. 311 for O1/O2/O3 411 for F1/F2/R1/R2/R3/R4
7. AKZ = Anlagen Kennzeichnungs System, KKS = Kraftwerk Kennzeichnungs System.
8. Olkiluoto Units 1 & 2

Annex D. Glossary of technical terms

Boat Sample. The boat sampling technique (BST) has been developed for obtaining samples from the surface of a pressure boundary 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 3 mm maximum thickness and require 180 minutes for getting scooped from a location. The samples are used for metallurgical analysis to confirm the integrity of the component. BST incorporates mainly sampling module, handling mechanism and electric and pneumatic sub-systems.

Bonna Pipe. Piping intended for raw water service. It is fabricated from rolled and seam welded steel plates. It has an internal concrete liner and an external reinforced concrete liner.

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 drop (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, then 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.

Component Boundary. Defines the physical boundary of a component required for system operation. A component boundary definition should be consistent with the parameter database supporting PSA model quantification. Isometric drawings (fabrication isometrics and in-service inspection isometrics) uniquely defines the piping component boundaries.

Damage Mechanism. Excessive internal or external loading conditions that cause physical damage to a component pressure boundary. Examples include, high-cycle vibration fatigue and thermal stratification, as well as pressure shocks from steam/water hammer.

Degradation Mechanism. Phenomena or processes that attack (crack, erode, wear, etc…) a pressure-retaining material over time and might result in a reduction of pressure boundary integrity. Also, includes phenomena that cause changes in material properties (e.g. reduction in fracture toughness).

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).

Fatigue. "Fatigue" refers to an ageing degradation mechanism where components undergo cyclic stress. This mechanism involves either low-load, high frequency stresses or high-load, low frequency stresses generated by thermal cycling, vibration, seismic events, or loading transients. Environmental factors may accelerate fatigue and eventually may result in a component failure.

Fillet Weld. Fillet welding refers to the process of joining two pieces of metal together whether they be perpendicular or at an angle. The weld is triangular in shape and may have a concave, flat or convex surface depending on the welder's technique

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." 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.

Furmanite™ Leak Repair Method. A common temporary pipe leak repair technique involving the placement of a mechanical clamp over the leak area. The enclosed pipe is filled with a sealant.

Grayloc® Connector. A clamp connector used for connecting piping components.

www.oceaneering.com/grayloc/

High-Density Polyethylene (HDPE). A polyethylene thermoplastic made from petroleum. With a high strength-to-density ratio, HDPE is used in the production of corrosion-resistant piping in, for example a raw water environment

High-Energy Piping: A piping system for which the maximum operating temperature exceeds 200 ºF (94.33 ºC) or the maximum operating pressure exceeds 275 psig (1.896 MPa).

Intergranular Stress Corrosion Cracking (IGSCC). IGSCC is associated in particular with a sensitised material (e.g. sensitised austenitic stainless steels are susceptible to IGSCC in an oxidizing environment). Sensitisation of unstabilised austenitic stainless steels is characterised by a precipitation of a network of chromium carbides with depletion of chromium at the grain boundaries, making these boundaries vulnerable to corrosive attack.

Irradiation Assisted Stress Corrosion Cracking (IASCC). IASCC refers to intergranular cracking of materials exposed to ionising radiation. As with SCC, IASCC requires stress, aggressive environment and a susceptible material. However, in the case of IASCC, a normally non-susceptible material is rendered susceptible by exposure to neutron irradiation. IASCC is a plausible ageing mechanism, in particular for PWR internal components (e.g. baffle bolts).

LCO Action Statement. For operating reactors, the technical specifications (in Germany and Switzerland referred to as "Betriebshandbuch" – BHB)[32] define the limiting conditions for operation (LCOs) (operational limits and conditions (OLC) in UK) that specify minimum requirements for ensuring safe operation. The "ACTIONS" associated with an LCO state conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated condition are required action(s) and completion time(s). The completion time is the amount of time allowed for completing a required action. It is referenced to the time of discovery of a situation (e.g. inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the applicability of the LCO. Required actions must be completed prior to the expiration of the specified completion time. An ACTIONS condition remains in effect and the required actions apply until the Condition no longer exists or the unit is not within the LCO applicability.

Liquid Droplet Impingement (LDI). Liquid droplet impingement is caused by the impact of high velocity droplets or liquid jets. Normally, LDI 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).

Moderate Energy Piping. A piping system for which the maximum operating temperature is less than 200 ºF (94.33 ºC) or the maximum operating pressure is less than 275 psig (1.896 MPa

Nominal Pipe Size (NPS). A North American set of standard pipe sizes. Based on NPS and the schedule of a pipe (see below), the pipe outside diameter (OD) and wall thickness can be obtained from reference tables; e.g. ASME/ANSI B36.10M and B36.19M. For example, NPS14 Sch40 has an OD of 14 inches and a wall thickness of 0.437 inches. However, the NPS and OD values are not always equal:

32. www.rskonline.de/sites/default/files/reports/epanlage3rsk447hp.pdf

- For NPS ⅛ to 12, the NPS and OD values are different. For example, the OD of an NPS 12 pipe is actually 12.75 inches.
- For NPS14 and up, the NPS and OD values are equal.

Operational Limits and Conditions (OLC) are briefly defined by IAEA as a set of rules setting forth parameter limits, the functional capability and the performance levels of equipment and personnel approved by the regulatory body for safe operation of a nuclear power plant. In US the similar term is “technical specifications”.

Pipe Schedule: The schedule number (SN) is defined as SN = 1000 □ (P/SE), where P is operating pressure in lb/in2 and SE is allowable stress range multiplied by joint efficiency in lb/in2. Most U.S. pipe failure reports include pipe schedule information.

“Red Brass”: An American term for the copper-zinc-tin alloy known as gunmetal; it is an alloy which is considered both a brass and a bronze. “Red brass” is also an alternative name for copper alloy C23000, which is composed of 14–16% zinc, a minimum 0.05% iron and minimum 0.07% lead content, and the remainder copper. This material is susceptible to IGSCC.

Repair Weld (or Welding). Any welding performed after original construction, but prior to commissioning (e.g. hot functional testing).

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.

Thermal Ageing. Possible effects of elevated temperature service include phase transformations that can adversely affect mechanical properties. Extended time at elevated temperature may permit even very slow phase transformations to occur. This is of particular concern for cast stainless steel components where the formation of a brittle alpha-phase can result in a loss of fracture toughness and lead to brittle failure.

Thermal Stratification. Hot water can flow above cold water in horizontal runs of piping when the flow (hot water into a cold pipe or cold water into a hot pipe) does not have enough velocity to flush the fluid in the pipe. The temperature profiles in the pipe where the top of the pipe is hotter than the bottom causes the pipe to bow along with the normal expansion at the average temperature.

Water Hammer. If the velocity of water or other liquid flowing in a pipe is suddenly reduced, a pressure wave results, which travels up and down the pipe system at the speed of sound in the liquid. Water hammer occurs in systems that are subject to rapid changes in fluid flow rate, including systems with rapidly actuated valves, fast-starting pumps, and check valves.

Annex E. OPDE/CODAP bibliography

OPDE-1. Lydell, B., Mathet, E. and Gott, K., "OECD Pipe Failure Data Exchange (OPDE) Project: A Framework for International Cooperation in Piping Reliability," Proc. American Nuclear Society International Topical Meeting on Probabilistic Safety Assessment, Detroit, Michigan, 2002.

OPDE-2. Colligan, A., Lojk, R., Riznic, J., Lydell, B., "The OECD Pipe Failure Data Exchange Project - Canadian Contribution on Data Validation", Canadian Nuclear Society Bulletin, Vol. 24, No. 3, August 2003.

OPDE-3. Choi, S.Y., Choi, Y.H., Piping Failure Analysis for the Korean Nuclear Power Piping including the effect of In-Service Inspection, Asian Pacific Conference on NDT, APC NDT 2003, Jeju, Korea, 2003.

OPDE-4. Lydell, B., Mathet, E. and Gott, K., "OECD Pipe Failure Data Exchange Project (OPDE) – 2003 Progress Report," ICONE12-49217, Proc. ICONE-12: 12th International Conference on Nuclear Engineering, Arlington, Virginia, 2004.

OPDE-5. Colligan, A., Lojk, R., Riznic, J. "The OECD pipe failure data exchange project - Data validation on Canadian plants", Proceedings of the International Conference on Nuclear Engineering, pp. 65-70, ICONE12-49273, Arlington, Virginia USA, 25-29 April 2004.

OPDE-6. Sun, Y.C., Young, H.C., Yeon, K.C. "Application of a piping failure database to some nuclear safety issues in Korea", Proceedings of the International Conference on Nuclear Engineering, pp. 175-179, ICONE12-49572, Arlington, Virginia USA, 25-29 April 2004.

OPDE-7. Fleming, K. N. and Lydell, B. O. Y, (2004), "Use of Markov Piping Reliability Models to Evaluate Time Dependent Frequencies of Loss of Coolant Accidents," Proc. ICONE-12, the 12th International Conference on Nuclear Engineering. 25-29 April 2004, Arlington, Virginia, U.S.A.

OPDE-8. Lydell, B., Mathet, E., Gott, K., "Piping Service Life Experience in Commercial Nuclear Power Plants: Progress with the OECD Pipe Failure Data Exchange Project", 2004 ASME Pressure Vessels and Piping (PVP) Conference, San Diego, CA, 25-29 July 2004.

OPDE-9. Choi, Y.H. et al, "Safety Evaluation of Socket Weld Integrity in Nuclear Piping," Key Engineering Materials, 270-273:1725-1730, 2004.

OPDE-10. OECD/NEA, OECD/NEA Piping Failure Data Exchange Project, Workshop on Database Applications, OPDE/SEC(2004)4, 2005.

OPDE-11. Choi, S. Y., Choi, Y. H., 2005, "Evaluation of nuclear piping failure frequency in Korean pressurized water reactors", Key Engineering Materials, Vols. 297-300, pp.1645-1651.

OPDE-12. NEA, 2006a. OPDE 2006:2 Coding guideline (OPDE-CG), OPDE PR01, OECD Nuclear Energy Agency, Issy-les-Moulineaux, France.

OPDE-13. NEA, 2006b. OPDE Quality assurance programme (OPDE-QA), OPDE-PR-02, OECD Nuclear Energy Agency, Issy-les-Moulineaux, France.

OPDE-14. Vglasky, T, A. Blahoianu, B. Lydell, J. Riznic, "The OECD Pipe Failure Data Exchange Project – Validation of Canadian Data," ICONE14-89176, Proc. ICONE-14: 14th International Conference on Nuclear Engineering, Miami, Florida, 2006.

OPDE-15. Lydell, B.O.Y., Mathet, E., Gott, K., "OECD Pipe Failure Data Exchange Project: First Term (2002-2005) Results and Insights", International Conference on Probabilistic Safety Assessment and Management- PSAM 8, New Orleans, Louisiana, 14-19 May 2006.

OPDE-16. Olsson, A., Lydell, B. and Kochenhauer, M., 2006. „Development of a Reliability data handbook for piping components in Nordic nuclear power plants", Proc. Int. 8th International Probabilistic Safety Assessment and Management Conference, ASME, New York, NY.

OPDE-17. Reck, H. and Schulz, S., "Internationale Betriebserfahrung mit Rohrleitungen in Kernkraftwerken: Fortschritt des OECD Pipe Failure Data Exchange (OPDE) Datenbank Projektes, " MPA Seminar 32, Stuttgart, Germany, 2006.

OPDE-18. Choi, Y.H. and Choi, S.Y., "Socket Weld Integrity in Nuclear Piping under Fatigue Loading Conditions," Nuclear Engineering and Design, 237:213-218, 2007.

OPDE-19. Lydell, B., "The Probability of Pipe Failure on the Basis of Operating Experience, PVP2007-26281, Proc. 2007 ASME Pressure Vessel and Piping Division Conference, San Antonio, Texas, 2007.

OPDE-20. Lydell, B., Huerta, A. and Gott, K., "Progress with the International Pipe Failure Data Exchange Project: PVP2007-26278, Proc. 2007 ASME Pressure Vessel and Piping Division Conference, San Antonio, Texas, 2007.

OPDE-21. Simonen, F.A., Gosselin, S.R., Lydell, B.O.Y. and Rudland, D.L., "Application of Failure Event Data to Benchmark Probabilistic Fracture Mechanics Computer Codes," PVP2007-26373, Proc. 2007 ASME Pressure Vessel and Piping Division Conference, San Antonio, Texas, 2007.

OPDE-22. Lydell, B., Anders, O., "Reliability Data for Piping Components in Nordic Nuclear Power Plants "R-Book" Project Phase I" Ski Report 2008:01, Swedish Nuclear Power Inspectorate, Stockholm, Sweden, January 2008.

OPDE-23. OECD/NEA, "OECD Piping Failure Data Exchange (OPDE) Project, Terms and Conditions for Project Operation 2008-2011", Paris, France, 2008.

OPDE-24. Nuclear Energy Agency, "OPDE 2008:1 Coding Guideline (OPDE-CG) & User's Guide", OPDE PR01, Version 05, Issy-les-Moulineaux, France, 2008.

OPDE-25. OECD Nuclear Energy Agency, CSNI Workshop on Risk-Informed Piping Integrity Management, CSNI Working Group on Integrity of Components and Structures (IAGE), NEA/SEN/SIN/IAGE(2008)6, Boulogne-Billancourt, France, 2008.

OPDE-26. Pandey, M., "Piping Failure Frequency Analysis using OECD/NEA Data", University of Waterloo, Final Report, CNSC Research and Support Programme. Report RSP-0236, February 2008.

OPDE-27. Olsson, A., Lydell, B. "Development of a reliability data handbook for piping components in Nordic nuclear power plants - Part II", 9th International Conference on Probabilistic Safety Assessment and Management 2008, pp. 8-15, 2008.

OPDE-28. Lydell, B., Huerta, A. and Gott, K., "Characteristics of Damage & Degradation Mechanisms in Nuclear Power Plant Piping Systems," PVP2008-61914, Proc. 2008 ASME Pressure Vessel and Piping Division Conference, Chicago, Illinois, 2008.

OPDE-29. Yuan, X.X., Pandey, M.D., Riznic,J.R., (2008), "A Point Process Model for Piping Failure Frequency Analysis using OPDE Data", ICONE16-48078, Proceedings of the 16th International Conference on Nuclear Engineering ICONE-16, Orlando, Florida, 11-15 May 2008, American Society of Mechanical Engineers.

OPDE-30. Lydell, B. and Riznic, J., "OPDE – The International Pipe Failure Data Exchange Project," Nuclear Engineering and Design, 238:2115-2123, 2008.

OPDE-31. Yuan, X. X., Pandey, M. D., Riznic, J. R., (2009), A Stochastic Model for Piping Failure Frequency Analysis using OPDE Data, *Journal of Engineering for Gas Turbine and Power,* 131, (5), art. 052901, Sept. 2009.

OPDE-32. Riznic, J. "OECD/NEA Pipe Failure Data Exchange (OPDE) 2002-2008 Status Report", Canadian Nuclear Safety Commission, 2009.

OPDE-33. Choi, Y.H. and Choi, S.Y., "Assessment of Socket Weld Integrity in Piping," J. Loss Prevention in the Process Industries, 22:850-853, 2009.

OPDE-34. Koriyama, T., Li, Y., Hamaguchi, Y., Yamashita, M. and Hirano, M., "Study on Risk-Informed In-Service Inspection for BWR Piping," Journal of Nuclear Science and Technology, 46:846-873, 2009.

OPDE-35. Olsson, A., Hedtjärn Swaling, V., Lydell, B. "Reliability data handbook for piping components in nordic nuclear power plants - Part III", 10th International Conference on Probabilistic Safety Assessment and Management 2010, pp. 2337-2348, 2010.

OPDE-36. Grebner, H. et al., Development and Test Applications of Methods and Tools for Probabilistic Safety Analyses, GRS-A-3555, Gesellschaft für Analagen und Reaktorsicherheit, Cologne, Germany, 2010.

OPDE-37. OECD Nuclear Energy Agency, EC-JRC/OECD-NEA Benchmark Study on Risk Informed in Service Inspection Methodologies (RISMET), NEA/CSNI/R(2010)13, Boulogne-Billancourt, France, 2011.

OPDE-38. Lydell, B., Huerta, A., Gott, K. "Insights from PSA Applications of the OECD Nuclear Energy Agency (OECD/NEA) OPDE Database," Proc. Int. Topical Meeting on Probabilistic Safety Assessment and Analysis 2011, pp. 1696-1706, 2011.

OPDE-39. Olsson, A., Swaling, V. H., Lydell, B., "Experiences from Implementation of Updated Reliability Data for Piping Components Using the R-Book," Proc. ANS PSA 2011 International Topical Meeting on Probabilistic Safety Assessment and Analysis, American Nuclear Society, Wilmington, NC, 13-17 March 2011.

OPDE-40. Lydell B., Mosleh A., Chrun D., "Enhanced Piping Reliability Models for Use in Internal Flooding PSA," Proc. ANS PSA 2011 Int.Topical Meeting on Probabilistic Safety Assessment and Analysis, American Nuclear Society, Wilmington, NC, 13-17 March 2011.

OPDE-41. Swaling, V., Olsson, A., Sunde, E. and Lydell, B., "Experience from the Implementation of Updated Reliability Data for Piping Components Using the 'R-Book'," Proc. PSAM11 & ESREL 2012 Joint Conference, Helsinki, Finland, 25-29 June 2012. www.proceedings.com/16286.html

OPDE-42. Cronvall, O., Männistö, I., Kaunisto, K. "On Applications Concerning OECD Pipe Failure Database OPDE," Proc. 11th Int. Probabilistic Safety Assessment and Management Conference and the Annual European Safety and Reliability Conference 2012, pp. 2891-2901, 2012.

OPDE-43. Wood, J., Gonzalez, M., Harris, C., Tobin, M., Coyne, K. "Estimating conditional Failure Probabilities of Observed Piping Degradations," Proc. 11th Int. Probabilistic Safety Assessment and Management Conference and the Annual European Safety and Reliability Conference 2012, pp. 2870-2875, 2012.

OPDE-44. Park, J.S., Choi, Y.H. "Application of Piping Failure Database to Nuclear Safety Issues in Korea," International Journal of Pressure Vessels and Piping, Vol. 90-91, pp. 56-60, 2012.

OPDE-45. OECD Nuclear Energy Agency, OECD/NEA Piping Failure Data Exchange Project (OECD/NEA OPDE), Final Report, NEA/CSNI/R(2012)16, Boulogne-Billancourt, France, 2012.

CODAP-1. Lydell, B., Huerta, A., Gott, K., Riznic, J. "OECD-NEA CODAP Event Data Project on Passive Component Degradation & Failures in Commercial Nuclear Power Plants," Proc. Int. Topical Meeting on Probabilistic Safety Assessment and Analysis 2013, pp. 663-672, 2013.

CODAP-2. Coyne, K., Tobin, M., Siu, N., Roewekamp, M. "Use of OECD Data Project Products in Probabilistic Safety Assessment", Proc. Int. Topical Meeting on Probabilistic Safety Assessment and Analysis 2013, pp. 558-568, 2013.

CODAP-3. Lydell, B., "Incorporation of FAC Considerations in Probabilistic Safety Assessment," Proc. FAC2013: Conference of Flow Accelerated Corrosion, 21-24 May 2013, Avignon, France.

CODAP-4. OECD Nuclear Energy Agency, CODAP Topical Report on Flow-Accelerated Corrosion (FAC) of Carbon Steel & Low Alloy Steel Piping in Commercial Nuclear Power Plants, NEA/CSNI/R(2014)/6, Boulogne-Billancourt, France, 2014.

CODAP-5. Lydell, B., Nevander, O., Gott, K., Riznic, J. "CODAP Project on International Cooperation in the Area of Structural Integrity of NPP", ASME International Mechanical Engineering Congress & Exposition, 2014.

CODAP-6. Breest, A., Gott, K., Lydell, B. and Riznic, J., "OECD/NEA Multilateral Cooperation in the Area of Structural Integrity & Aging Management," Paper PBNC2014-025, Proc. 19th Pacific Basin Nuclear Conference, 24-28 August 2014, Vancouver, British Columbia, Canada.

CODAP-7. Nevander, O., Riznic, J., Gott, K. and Lydell, B., "OECD/NEA Component Operating Experience, Aging & Degradation Project," Paper O-T05-163, Proc. Fontevraud 8 Conference, 15-18 September 2014, Avignon, France.

CODAP-8. OECD Nuclear Energy Agency, OECD/NEA Component Operational Experience, Degradation & Aging Programme (CODAP): First Term (2011-2014) Status Report, NEA/CSNI/R(2015)7, Boulogne-Billancourt, France, 2015.

CODAP-9. Lydell, B., Castelao, C. "CODAP Project Operating Experience Insights Related to Fatigue Mechanisms", 4th International Conference on Fatigue of Nuclear Reactor Components, 28 September – 1 October 2015, Seville, Spain, 2015.

CODAP-10. Dragea, T., Riznic, J. "The Component Operational Experience Degradation and Ageing Programme (CODAP): Review and Lessons Learned (2011-2014)", 23th International Conference on Nuclear Engineering, ICONE23-1001, Chiba, Japan, 17-21 May 2015.

CODAP-11. Rivet, M., Riznic, J., (2015), The Component Operational Experience Degradation and Ageing Programme (CODAP) 2014-2015 Status Report RSP R 144.6, Operational Engineering Assessment Division Report, CNSC, Ottawa, e-docs#4904008, December 2015.

CODAP-12. OECD Nuclear Energy Agency, CODAP Topical Report on Operating Experience Insights into Pipe Failures in Electro-Hydraulic and Instrument Air Systems, NEA/CSNI/R(2015)6, Boulogne-Billancourt, France, 2015.

CODAP-13. Rivet, M., Riznic, J., (2015), The Component Operational Experience Degradation and Ageing Programme (CODAP) 2014-2015 Status Report- RSP R 144.6, Operational Engineering Assessment Division, CNSC, e-docs# 4904008, December 2015.

CODAP-14. Yigitoglu, A.G., A Methodology for Modeling Nuclear Power Plant Passive Component Aging in Probabilistic Risk Assessment under the Impact of Operating Conditions, Surveillance and Maintenance Activities, PhD Thesis, Dept. of Nuclear Engineering, Ohio State University, Columbus, OH, 2016.

CODAP-15. Lydell, B., Nevander, O. and Riznic, C., "Piping Corrosion Risk Management on the Basis of OECD/NEA CODAP Project Database," RISK16-8327, Proc. NACE International Corrosion Risk Management Conference, Houston, TX, 23-25 May 2016.

CODAP-16. Rivet, M., Cormier, K., Riznic, J., "The Component Experience, Degradation and Ageing Programme (CODAP) Project; Canada's Contributions and Benefits to the Canadian Industry," submitted to the 40th Annual CNS/CNA Student Conference, Toronto, Canada, 19-22 June 2016.

CODAP-17. Lydell, B., Nevander, O. and Riznic, J., "The OECD/NEA CODAP Project & Its Contribution to Plant Aging Management & Probabilistic Safety Assessment, Paper 062, Proc. 13th International Conference on Probabilistic Safety Assessment and Management (PSAM 13), Seoul, Korea, 2-7 October 2016.

CODAP-18. Lydell, B., "A Review of the Progress with Statistical Models of Passive Component Reliability," Nuclear Engineering and Technology, 49:349-359, 2017.

CODAP-19. Lydell, B., Nevander, O. and Riznic, J., "The Nuclear Energy Agency Contribution to Nuclear Materials Performance Knowledge Preservation," C2018-10530, Proc. NACE Corrosion Conference, Phoenix, AZ, 15-19 April 2018.

CODAP-20. Lydell, B., Nevander, O. and Riznic, J., "Risk Characterization of Concealed Piping Failures," RISK18-12102, Proc. NACE International Corrosion Risk Management Conference, Houston, TX, 11-13 June 2018.

CODAP-21. Lydell, B., Fleming, K.N. and Roy, J-F., "Analysis of Possible Aging Trends in the Estimation of Piping System Failure Rates for Internal Flooding PRA," Proc. International Conference on Probabilistic Safety Assessment and Management (PSAM 14), Los Angeles, CA, 16-21 September 2018.