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CODAP Topical Report

Flow Accelerated Corrosion (FAC) of Carbon Steel and Low Alloy Steel Piping in Commercial Nuclear Power Plants

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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 34 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, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Republic of Korea, the 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.

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NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 31 countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, the Republic of Korea, the Russian Federation, 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 friendly and economical use of nuclear energy for peaceful purposes, as well as

– to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information.

The NEA Data Bank provides nuclear data and computer program services for participating countries. In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

Within the OECD framework, the NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made of senior scientists and engineers, with broad responsibilities for safety technology and research programmes, as well as 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 insofar as they affect the safety of such installations.

The committee's purpose is to foster international co-operation in nuclear safety amongst the NEA member countries. The CSNI's main tasks are to exchange technical information and to promote collaboration between research, development, engineering and regulatory organisations; to review operating experience and the state of knowledge on selected topics of nuclear safety technology and safety assessment; to initiate and conduct programmes to overcome discrepancies, develop improvements and research consensus on technical issues; and to promote the co-ordination of work that serves to maintain competence in nuclear safety matters, including the establishment of joint undertakings.

The clear priority of the committee is on the safety of nuclear installations and the design and construction of new reactors and installations. For advanced reactor designs the committee provides a forum for improving safety related knowledge and a vehicle for joint research.

In implementing its programme, the CSNI establishes co-operate mechanisms with the NEA's Committee on Nuclear Regulatory Activities (CNRA) which is responsible for the programme of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with the other NEA's Standing Committees as well as with key international organisations (e.g., the IAEA) on matters of common interest.

EXECUTIVE SUMMARY

Structural integrity of piping systems is important for plant safety and operability. In recognition of this, information on degradation and failure of piping components and systems is collected and evaluated by regulatory agencies, international organisations (e.g., OECD/NEA and IAEA) and industry organisations worldwide to provide systematic feedback for example to reactor regulation and research and development programmes associated with 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.

Several NEA Member Countries have agreed to establish the OECD/NEA "Component Operational Experience, Degradation & 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. The Project is organised under the OECD/NEA Committee on the Safety of Nuclear Installations (CSNI).

CODAP is the continuation of the 2002–2011 "OECD/NEA Pipe Failure Data Exchange Project" (OPDE) and the Stress Corrosion Cracking Working Group of the 2006–2010 “OECD/NEA SCC and Cable Ageing project” (SCAP). OPDE was formally launched in May 2002. Upon completion of the 3rd Term (May 2011), the OPDE project was officially closed to be succeeded by CODAP. SCAP was enabled by a voluntary contribution from Japan. It was formally launched in June 2006 and officially closed with an international workshop held in Tokyo in May 2010. Majority of the member organizations of the two projects were the same, often being represented by the same person. In May 2011, thirteen countries signed the CODAP 1st Term agreement (Canada, Chinese Taipei, Czech Republic, Finland, France, Germany, Korea (Republic of), Japan, Slovak Republic, Spain, Sweden, Switzerland and United States of America). The 1st Term work plan includes the preparation of Topical Reports to foster technical cooperation and to deepen the understanding of national differences in ageing management. The Topical Reports constitute CODAP Event Database and Knowledge Base insights reports and as such act as portals for future in-depth studies of selected degradation mechanisms. This, the first Topical Report addresses flow accelerated corrosion (FAC) of carbon steel and low alloy steel piping.

FAC involves wall thinning of carbon and low-alloy steel high-energy piping due to turbulent and fast flowing water or wet steam that wears away the oxide layer (protective film) and leads to continued corrosion of the underlying metal. FAC is a chemical effect that is primarily influenced by pH, hydrodynamics, oxygen content, and temperature. The geometric aspects of the system design and piping layout play a key role in the occurrence of FAC-induced wall thinning and potential major structural failures. FAC has caused sudden ruptures (break-before-leak, BBL) in high and moderate energy piping systems, resulting in plant transients and affecting safety/non safety related equipment by leaking steam and water (spatial effects). FAC also poses an occupational safety hazard. All reactor types have experienced some type of FAC related events in their piping systems. As a result, FAC management programmes have been implemented to monitor and mitigate pipe wall thinning.

The CODAP Topical Report on "FAC of Carbon Steel and Low Alloy Steel Piping" includes a primer on the environmental and operational factors affecting FAC-susceptibility, and evaluates service experience data. Also included in the report are descriptions of the national FAC management programme approaches and a summary of other information collected in the CODAP Knowledge Base. The report has been prepared by the CODAP Project Review Group, with support from the CODAP Operating Agent and the CODAP Knowledge Base Coordinator.

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ABBREVIATIONS & ACRONYMS

AEC	Atomic Energy Council (Chinese Taipei)
AFCEN	French Association for Design, Construction and In-Service Inspection Rules for Nuclear Island Components
AFW	Auxiliary Feedwater
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials (now "ASTM International"
AVT	All Volatile Treatment
B&PVC	Boiler & Pressure Vessel Code
BBL	Break-Before-Leak
BOP	Balance-of-Plant
CHEC	Chexal-Horowitz-Erosion-Corrosion
CHUG	CHECWORKS® User Group
CODAP	Component Operational Experience, Degradation and Ageing Program
COG	CANDU Owners Group
CNSC	Canadian Nuclear Safety Commission
COMSY	Condition Oriented Ageing and Plant Life Monitoring System
CRP	Coordinated Research Project (by IAEA)
CRP	Conditional Rupture Probability
CSA	Canadian Standards Association
CSN	Consejo de Seguridad Nuclear
CSNI	Committee on the Safety of Nuclear Installations
DEGB	Double-Ended Guillotine Break
DN	Nominal Diameter [mm]
DO	Dissolved Oxygen
E/C	Erosion-Corrosion
E-C	Erosion-Cavitation
EBS	Equivalent Break Size
ECP	Electrochemical Corrosion Potential
EdF	Electricité de France
ENIQ	European Network for Inspection and Qualification

EPIX	Equipment Performance Information Exchange System
EPR	European Pressurized Reactor
FAC	Flow Accelerated Corrosion
FEA	Finite Element Analysis
GALL	Generic Aging Lessons Learned
H-AVT	High AVT
HBD	Heat Balance Diagram
HELB	High Energy Line Break
HP	High Pressure
HPCI	High Pressure Coolant Injection
HPH	High Pressure Feedwater Heater
HWC	Hydrogen Water Chemistry
IAEA	International Atomic Energy Agency
ID	Inside Diameter
INPO	Institute of Nuclear Power Operations
ISI	In-Service Inspection
JSME	Japan Society of Mechanical Engineers
KB	Knowledge Base
KEPCO	Kansai Electric Power Company
KHNP	Korea Hydro & Nuclear Power
KKM	Kernkraftwerk Mühleberg
LBB	Leak-Before-Break
LCF	Line Correction Factor
LDIE	Liquid Droplet Impingement Erosion
LP	Low Pressure
LPH	Low Pressure Feedwater Heater
MBM	Moving Blanket Method
MSR	Moisture Separator Reheater
NDE	Non-Destructive Examination
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NWC	Normal Water Chemistry
OAR	Owner's Activity Report
OBE	Operating Basis Earthquake
NWC	Normal Water Chemistry
PHTS	Primary Heat Transport System
PSA	Probabilistic Safety Assessment
RCPB	Reactor Coolant Pressure Boundary
RF	Refuelling Cycle
RI-ISI	Risk Informed In-Service Inspection
SCAP	Stress Corrosion Cracking and Cable Ageing Project
SDP	Significance Determination Process
SEAS	Slovenské elektrárne, a.s
SGBD	Steam Generator Blowdown
SNCT	Syndicat National de la Chaudronnerie, de la Tôlerie et de la Tuyauterie Industrielle

SSC	Systems, Structures, and Components
SSE	Safe Shutdown Earthquake
SSM	Swedish Radiation Safety Authority
SOL	Safe Operating Life
TG	Turbine Generator
TGSCC	Transgranular Stress Corrosion Cracking
TOFD	Time of Flight Diffraction
TPC	Taiwan Power Company
UT	Ultrasonic Inspection Technique
VGB	Vereinigung der Großkraftwerk-betreiber
VT	Visual Inspection Technique
WPB	Weldable (carbon steel) Pipe Grade B

1. INTRODUCTION

Flow assisted material degradation occurs in a variety of carbon steel and low alloy steel piping systems. There are two types of flow assisted material degradation: 1) erosion of pipe wall caused by physical processes such as high liquid velocities, impinging flows or solid particle impacts; and 2) flow-accelerated corrosion (FAC). The FAC degradation mechanism can cause thinning of large areas (or very gradual local thinning) of piping that can lead to sudden failure of a piping pressure boundary. This Topical Report on FAC is an information resource developed by the CODAP Project Review Group. The report summarizes an evaluation of the FAC-specific service experience data included in the CODAP Event Database and information from the CODAP Knowledge Base.

1.1 Technical Scope of CODAP Topical Report #1

This CODAP Topical Report #1 addresses FAC of carbon steel and low-alloy steel piping systems. FAC is one of many flow assisted degradation mechanisms that cause global or highly localized wall thinning and through-wall flaws; Figure 1. The report summarizes insights from the analysis of service experience data as included in the CODAP Event Database. In analyzing the event data, special database screening criteria have been applied to screen out wall thinning mechanisms such as erosion-cavitation, erosion-corrosion, liquid droplet impingement erosion, and solid particle erosion.

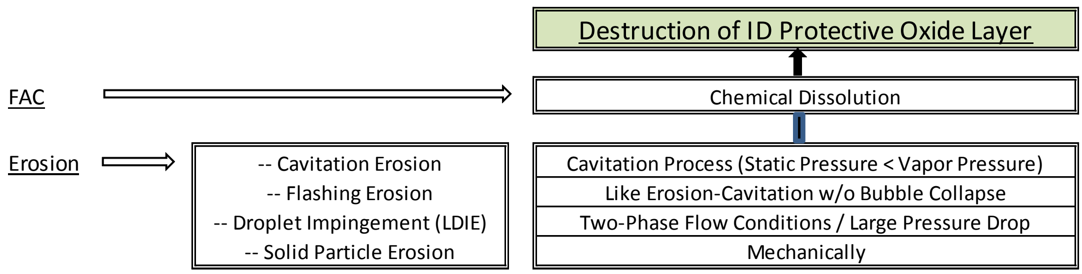


Figure 1: Examples of Flow-Assisted Pipe Wall Thinning Mechanisms

1.2 Report Structure

This report is structured as follows. Section 1 includes a historical perspective on FAC and a summary of the safety significance of FAC. Section 2 describes the CODAP Event Database and Knowledge Base. It elaborates on the failure definitions used by CODAP, with specific reference to FAC events. Section 3 is a primer on FAC theory addressing the different environmental variables that control FAC. Section 4 addresses the different FAC mitigation strategies. Section 5 summarizes the CODAP FAC event content and Section 6 includes a FAC event data analysis. Section 7 includes a summary and conclusions. Finally, a list of references is included in Section 8. Appendix A includes descriptions of selected significant FAC events. Finally, Appendix B is a glossary of terms.

1.3 Historical Perspective

As described explained by Robinson and Drews [1], FAC and erosion-corrosion (E/C) have been used interchangeably to describe similar material degradation processes. Both types of damage involve destruction of a protective oxide film on the inside pipe wall. The removal of the oxide film is generally referred to as the erosion process. This is followed by electrochemical oxidation, or corrosive attack of the underlying metal. The differences between FAC and E/C involve the mechanism by which the protective film is removed from the metal surface. In the E/C-process the film is removed mechanically from the surface. In contrast, in the FAC-process the oxide is dissolved or prevented from forming, allowing corrosion of the unprotected metal. FAC occurs in two-phase flow conditions (e.g., water droplets in steam or steam bubbles in water) as well as single-phase flow conditions.

Nuclear power plant operators have been monitoring FAC since early plant life and have performed repairs and replacements as warranted by pipe wall wear rates and pipe failures. In CODAP, one of the earliest recorded FAC events occurred in 1972 when a 6-inch elbow in a High Pressure Coolant Injection (HPCI)[1] system piping developed a through-wall leak. In CODAP the earliest recorded pipe rupture is from May 1976, when a BWR plant experienced an Auxiliary Feedwater (AFW) bypass line rupture during unit start-up.

In the early 1980s, FAC was considered to be mainly a problem in two-phase flow (e.g., wet steam) systems. The first case of single-phase FAC induced pipe failure was reported in 1985. On March 9, 1985, Trojan Nuclear Power Plant[2] was operating at 100% power when a DN350 (14-inch diameter) heater drain pump discharge pipe made of SA-106 Grade B carbon steel failed catastrophically [2]. The failure caused the release of a steam-water mixture of approximately 180°C into the turbine building. In addition to the fire suppression system actuation by heat sensors in the turbine building and damaged secondary plant equipment, one member of the operating staff received first and second degree burns on 50% of his body from the high temperature fluid.

A second case of single-phase FAC-induced pipe failure was reported in 1986. On December 9, 1986, a DN450 (18-inch diameter) suction line to the main feedwater pump for Surry-2[3] failed in a catastrophic manner [3]. The line temperature at this location is approximately 185°C, with a pressure of approximately 2.55 MPa. The ruptured elbow was made of ASTM A-234 Grade WPB carbon steel. Water flashing from the severed pipe engulfed equipment and personnel in the area. Several workers were seriously burned. Four of the eight men working nearby on another pipe were killed during the event. Within minutes of the pipe rupture, portions of the automatic fire protection system activated, opening 62 sprinklers to cool the atmosphere in the area of the rupture. The water from the sprinklers seeped into electrical panels, shorted out several electrical circuits that control other fire suppression equipment, and activated some systems containing carbon dioxide. The carbon dioxide combined with other fire retardants and seeped into the control room.

These two events are of historical significance. They demonstrated that significant FAC-induced pipe wall thinning can occur not only in wet steam lines (two-phase flow conditions) but also under single phase flow conditions. From a FAC management perspective, the two events raised questions about the effectiveness of the then existing non-destructive examination (NDE) programmes to monitor piping integrity for wall thinning and prevention of pipe failure. In July 1987, the U.S. Nuclear Regulatory Commission (NRC) issued U.S. licensees Bulletin No. 87-01, "Thinning of Pipe Walls in Nuclear Power Plants." The Bulletin requested all licensees for nuclear power plants holding an operating license or construction permit to submit information concerning their programmes for monitoring pipe wall thickness. In summary, the following information was requested:

- The codes and standards to which piping in condensate, feedwater, steam, and connected high-energy piping systems, including all safety-related and non-safety-related carbon steel piping systems is designed and fabricated.

[1] The HPCI system includes a turbine-driven pump. The steam supply/discharge piping to/from turbine-driven pump is susceptible to FAC.

[2] Trojan NPP, an early 4-loop PWR designed by Westinghouse, was permanently shutdown in November 1992.

[3] A 3-loop PWR designed by Westinghouse, the unit entered into commercial operation in May 1973.

- Description of the scope and extent of programmes for ensuring that pipe wall thicknesses are not reduced below the minimum allowable.

- For liquid-phase systems, identify the factors that are considered in establishing criteria for selecting points at which to make thickness measurements.

- Chronologically list and summarize the results of all inspections that have been performed, which were specifically conducted for the purpose of identifying pipe wall thinning.

- Description of any plans for revising the present or for developing new or additional programmes for monitoring pipe wall thickness.

All licensees responded to the bulletin and the NRC staff completed its review of the responses in December 1987. Furthermore, at the end of September 1988, the NRC staff completed inspection of 10 plants to assess the licensees' efforts toward implementing their FAC monitoring programmes. NUREG-1344 [4] summarizes the responses to Bulletin 87-01.

The Electric Power Research Institute (EPRI) in 1985 [5] issued a guideline for FAC inspection of wet steam lines. The proposed FAC inspection programme addressed the parameters affecting FAC and a technical basis for prioritizing the examinations. Consideration was given to using the empirical relationship of pressure, velocity, and moisture content and Keller's Equation [6] as methods for predicting the rate of FAC in wet steam piping.

Subsequent to the above mentioned FAC failures in single-phase systems, EPRI developed the CHEC® family of computer codes as predictive tools to assist plant operators in planning inspections and evaluating the inspection data to prevent pipe failures caused by FAC. In March 1987, the Nuclear Management and Resources Council (NUMARC, now the Nuclear Energy Institute, NEI) established a working group on FAC. The group developed a recommended industry programme to address single-phase and two-phase FAC. NUMARC and EPRI developed a recommended inspection plan to monitor pipe wall thinning problems. That programme is documented in "Recommendations for an Effective Flow-Accelerated Corrosion Program," NSAC-202L-R3 [7].

In May 1989, the NRC issued Generic Letter 89-08 [8], requesting all licensees to provide assurances that the NUMARC programme or another equally effective program had been implemented and that the structural integrity of all high-energy (two phase as well as single phase) carbon steel systems was maintained. In case a FAC programme had not yet been implemented a scheduled implementation date was to be provided.

The international nuclear safety community at large has been extensively involved in basic research towards the development of FAC monitoring strategies. As an example, in 1982 Heitmann and Kastner [9] published the results of theoretical work on FAC phenomena and experiments. This work resulted in the development of the WATHEC computer code for the prediction of wall thinning in single- and two-phase water/steam systems. A first version of this computer code was released in 1986. In addition, a computer code DASY was developed for recording, managing, evaluating and documenting the data obtained from non-destructive examination of individual piping components. In 1998 the WATCHEC and DASY codes were combined and further developed into the COMSY software program as an integrated tool for ageing and plant management of mechanical components. In France, FAC-related research by EdF produced the BRT-CICERO™ computer code for FAC wear rate prediction [10]. Since 2000, the use of this code is mandatory for all operating nuclear power plants in France and applies to carbon steel piping > DN100.

In September 1994, the OECD/NEA organized the "Specialist Meeting on Erosion and Corrosion of Nuclear Power Plant Materials" [11]. This meeting addressed FAC experience and the different FAC monitoring and mitigation approaches. The International Atomic Energy Agency (IAEA) has organized specialists meetings on FAC [12], and at the April 2009 meeting in Moscow [13] the Agency announced a new Coordinated Research Project (CRP) entitled "Review and Benchmark of Calculation Methods of Piping Wall Thinning due to Erosion-Corrosion in Nuclear Power Plants." Research work is currently being performed in several countries including Japan.

1.4 Safety Significance of FAC Induced Pipe Failures

FAC-induced major structural pipe failures are energetic and sudden. The failures have break-before-leak (BBL) rather than leak-before-break (LBB) characteristics. FAC-susceptible piping poses an occupational safety hazard and can result in power reduction, manual or automatic reactor shutdown. The spatial effects of FAC-induced pipe failure, including collateral equipment damage and area spraying/flooding, can be considerable. Even a small steam leak can adversely affect the fire protection system and electrical equipment adjacent to or in relatively close proximity to the leak location. Therefore consideration of FAC is an integrated part of the current internal flooding probabilistic safety assessment (PSA) practice and modelling of high-energy line breaks (e.g., main feedwater line break initiating event frequency). Significance determination process (SDP) assessments are concerned with determining the risk significance of degraded/failed piping, including FAC-induced failures using PSA methodology. Risk-informed high-energy line break (HELB) analyses can be used to strengthen an existing FAC programme.

Despite the progress with implementation and maintenance of comprehensive inspection programmes, FAC-induced pipe failures continue to cause power reductions and forced outages. These failures can be indicative of programmatic weaknesses as well as the challenges in identifying vulnerable locations. Also, plant modernization and power uprate projects involving piping design changes may result in new regions being susceptible to FAC.

The FAC event data collected by CODAP supports the full range of probabilistic evaluations of piping reliability. The event population and exposure term data provide input to pipe failure rate and rupture frequency calculations. An objective of this CODAP Topical Report is to summarize the FAC service experience data and the national FAC inspection programmes. It is intended as a resource document for future applications of the CODAP event database.

2. CODAP OBJECTIVE AND SCOPE

CODAP is the continuation of the 2002 – 2011 “OECD/NEA Pipe Failure Data Exchange Project” (OPDE) and the work by the Stress Corrosion Cracking Working Group of the 2006 – 2010 “OECD/NEA SCC and Cable Ageing Project” (SCAP). OPDE was formally launched in May 2002. Upon completion of the 3rd Term in May 2011 the OPDE project was officially closed. SCAP was enabled by a voluntary contribution from Japan. It was formally launched in June 2006 and officially closed with an international workshop held in Tokyo in May 2010. Most of the members of the two projects were the same, often being represented by the same person. 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 and well as a Knowledge base as in SCAP but the full range of failure mechanisms as in OPDE.

2.1 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 1960’s. In 1975 the U.S. Nuclear Regulatory Commission established a Pipe Crack Study Group (PCSG) charged with the task of evaluating the significance of stress corrosion cracking (SCC) in boiling water reactors (BWRs) and pressurized water reactors (PWRs). Service experience review was a key aspect of the work by the PCSG. Major condensate and feedwater piping failures (e.g., Trojan and Surry-2 in the U.S.) due to FAC resulted in similar national and international initiatives to learn from service experience and to develop mitigation strategies to prevent the recurrence of pipe failures. 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.

The team of analysts responsible for the seminal Reactor Safety Study (WASH-1400) [14] performed a limited evaluation of nuclear power plant piping reliability based on service experience from the then (early 1970s) approximately 150 U.S. commercial nuclear reactor operating years. This evaluation was aimed at estimation of loss-of-coolant-accident (LOCA) frequencies for input to the two PSA models of WASH-1400. After the publication of WASH-1400 in 1975 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 establishment of 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.

In parallel with these efforts to evaluate service experience data and to correlate the occurrence of material degradation with piping design and operational parameters, initiatives have been presented to establish an international forum for the systematic collection and exchange of service experience data on piping. An obstacle to the use of the database by other countries of national qualitative and quantitative pipe failure information is that criteria and interpretations applied in the collection and analysis of events and data differ among the various countries. A further impediment is that the descriptions of reported events and their root causes and underlying contributing factors, which are important to the assessment of the events, are usually written in the native language of the countries where the events were observed.

To overcome these obstacles, the preparation for the OECD Pipe Failure Data Exchange (OPDE) Project was initiated in 1994 by the Swedish Nuclear Power Inspectorate (SKI)[4]. In 1994 SKI launched a 5-year R&D project to explore the viability of creating an international pipe failure database and a related

[4] Swedish Radiation Safety Authority (SSM) as of July 1, 2008.

analytical basis for deriving reliability parameters for use in PSA. During this period SKI hosted meetings to present results of the R&D and to discuss the principles of database development and maintenance.[5] In September 2000 and, again in April 2001, the OECD/NEA organized preparatory meetings to explore the feasibility and interest in forming an international cooperative effort to systematically collect, evaluate and exchange service experience data.

Since May 2002, the OECD/NEA has formally operated the project under the coordination of the Committee on the Safety of Nuclear Installations (CSNI). The starting point for the Project was an in-kind contribution by SKI in the form of an international pipe failure database in Microsoft® Access. This database included pipe failure data for the period 1970 to 1998, and it contained approximately 2,300 records. During the first term of OPDE the emphasis was on validating the content of the SKI in-kind contribution, improving and streamlining the database structure and data input format, and populating the database with new failure data for the period 1999 to the present, as well as with pre-1998 records. The data validation benefitted from multi-disciplinary considerations, including material science, structural integrity and PSA. The first term of the Project covered the years 2002-2005, the second term covered the period 2005-2008 [15], and the final term covered the period 2008-2011 [16].

In 2006 the SCC and Cable Ageing Project (SCAP) was established under the auspices of the OECD/NEA to assess, due to their implication on nuclear safety and their relevance for plant ageing management, two subjects: stress corrosion cracking (SCC) and degradation of cable insulation. The project ran successfully from June 2006 to June 2010 [17].

Following the completion of the SCAP project, SCC Working Group participants were interested in some form of continuation and discussions were initiated to explore possible alternatives. It was recognized that there are 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 & Ageing Programme” (CODAP). 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 standby safety systems, and support systems (i.e., ASME Code Class 1, 2 and 3, or equivalent). It also covers non safety-related (non-Code) components with significant operational impact. It is intended that CODAP will also include information on age-related degradation of buried tanks and plastic piping.

In May 2011 the Project Review Group (PRG) approved the transition of OPDE to a new, expanded "OECD-NEA Component Operational Experience, Degradation & Ageing Program (CODAP)." A first CODAP National Coordinators Meeting was held at NEA Headquarters in November 2011. The CODAP PRG Membership corresponds to that of the OPDE (eleven member countries), with two additional member countries (Slovak Republic and Chinese Taipei). The CODAP project builds on the success of OPDE and a related OECD-NEA data project, the SCAP-SCC Working Group.

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 Coordinators 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 Headquarters. With the 2011 transition from OPDE to CODAP, a new and enhanced web-based database format was implemented. As of mid-2012, the entire CODAP event database resides on a secure server at NEA Headquarters. Provisions exist for online database interrogation (e.g., reviews, edits, QA, queries, validation) as well as downloading selected event records or the entire database to a local

[5] In September 1996 SKI organized the “Initial Meeting of the International Cooperative Group on Piping Performance” with participants from thirteen countries. Again, in September 1997 SKI organized the “Seminar on Piping Reliability” (SKI Report 97:32); this time with participants from eleven countries.

computer or computer network. The event database structure also includes a provision for uploading of event-specific information such as photographs, isometric drawings and root cause analysis reports. In addition to the event database, CODAP includes a web-based Knowledge Base (KB) that contains relevant national and international reference material on passive metallic component damage and degradation mechanisms. Included in the KB are codes and standards, R&D results, regulatory frameworks, and country-specific aging management programmes. As is the case for the event database, the KB also resides on a secure server at NEA Headquarters.

2.2 Data Collection Methodology

The CODAP Project exchanges data on passive component degradation and failure, including service-induced wall thinning, non-through wall crack, leaking through-wall crack, pinhole leak, leak, 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:

- 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 [18].

- Through-wall defects without active leakage (leakage may be detected following a plant operational mode change involving depressurization 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 Technical Specification limits);

- Large leaks (e.g., flow rates in excess of Technical Specification 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 structural integrity of a pressure boundary is determined by multiple and interrelated reliability attributes and influence factors. Depending on the conjoint requirements for damage and degradation, certain combinations of material, operating environment, loading conditions together with applicable design codes and standard, certain passive components are substantially more resistant to damage and degradation than others. As an example, for stabilized austenitic stainless steel pressure boundary components, there are no recorded events involving active, through-wall leakage. By contrast, for unstabilized austenitic stainless steel, multiple events involving through-wall leakage have been recorded, albeit with relative minor leak rates. Flow-accelerated corrosion (FAC), if unmonitored, is a relatively aggressive degradation mechanism that has produced major structural failures, including double-ended guillotine breaks (DEGB). The types of pipe failure included in the CODAP Event Database are:

- 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 mix of free-format fields for detailed narrative information and fields defined by drop-down menus with key words (or data filters) or related tables. 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, structural integrity and PSA.

2.3 FAC Failure Definitions

The CODAP Event Database includes FAC failure events involving non-through-wall and through-wall conditions. For non-through-wall event records, the minimum measured (t_{Meas}) pipe wall thickness must be equal to or less than the minimum allowable (t_{Min}) thickness as defined in FAC Program Plans. Pipe replacement should be performed when (or before) t_{Min} has been reached. Periodic FAC inspections are performed in order to estimate the FAC wear rate of the piping. Three methods are used to calculate the wear rate: 1) band method, 2) point-to-point method, and 3) moving blanket method (MBM). The "band method" calculates wear rates by taking a band around the circumference of the pipe and subtracting the minimum wall thickness reading in the band from maximum wall thickness reading in the band. The "point-to-point" method calculates wear rates by subtracting the measurement taken at a grid point during a current refuelling outage from the measurement taken at the same grid point during a previous refuelling outage. The MBM is a relatively new analysis procedure. It involves splitting the pipe wall thickness data into smaller zones, averaging the wall thicknesses, and comparing against the other zones, with the zone with the maximum calculated difference approximating the measured wear.

Different wall thickness criteria are used to determine an inspected piping component's safe operating life (SOL). Determination of SOL is based on the calculated FAC wear rate. The "Owner-defined" FAC programmes include SOL criteria to determine whether continued operation is acceptable or if a repair or replacement must be implemented prior to return to service. Typical wall thickness criteria are:

- Components with the calculated minimum wall thickness (t_{Min}) above the minimum manufacturing tolerances (t_{Nom}) are considered adequate for continued service beyond one refuelling cycle.

$$t_{\mathrm{Min}} > 0.875 \times t_{\mathrm{Nom}} \Rightarrow \text{Adequate Wall Thickness} \quad (1)$$

- Components with the measured wall thickness, t_{Meas}, at or below t_{Nom} require further evaluation. This evaluation will relate the measured wall thickness with the wall thickness required for the component design pressure and temperature (t_{Min}).

$$t_{\mathrm{Meas}} < 0.875 \times t_{\mathrm{Nom}} \Rightarrow \text{Further Evaluation Required} \quad (2)$$

The criterion of “87.5% of nominal wall thickness” originates from ASME Code Case N-480[6]. A technical basis for calculating the minimum wall thickness is included in national codes and standards for

[6] ASME Boiler and Pressure Vessel Code, Code Case N-480: "Examination Requirements for Pipe Wall Thinning Due to Single Phase Erosion and Corrosion, Section XI, Division 1," approved May 10, 1990, in 1992 Code C ases: Nuclear Components, p. 787, July 1992.

piping design. As examples, Section 104 of ASME "Code for Pressure Piping, B31.1" [19] and Standard RD EO -571-2006 [20] provide formulas for minimum wall thickness calculations.

2.4 CODAP Knowledge Base

The CODAP Knowledge Base has been established to reflect basic international technical information of relevance to the project in a systematic manner. The KB is password protected and resides on a secure server at NEA Headquarters in Paris, France. The KB is intended to provide a source of information on technical issues related to all the failure mechanisms covered by the Event Database. The type of information collected includes regulations/ codes and standards, inspection/ monitoring/ qualification, preventive maintenance/ mitigation, repair/ replacement, safety assessment, and R&D. The information is both of a general nature and also more specific for the different degradation mechanisms. The KB is intended to provide a source of systematically organised information for members, as well as input to the topical reports the project is intending to prepare. There is a search function to facilitate retrieval of information.

The KB is a web-based area of the CODAP project domain. It is organised as a hierarchical system of folders for general information, degradation specific information and a country folder for each project member. The country folders have two purposes: to upload files for inclusion in the common KB and to provide a means of organising documents of national interest and relevance. In the latter case documentation can be in the language of the country, with a title in English. In the other folders all documents are in English.

3. FLOW ACCELERATED CORROSION PRIMER

This chapter contains a short description of the major parameters and factors affecting FAC. It is not intended to be a complete state-of-the-art documentation of the mechanism but to highlight important variables which should be considered when evaluating FAC.

Flow accelerated corrosion (FAC, also termed flow-assisted corrosion, and sometimes wrongly erosion-corrosion) leads to wall thinning (metal loss) of steel piping exposed to flowing water or wet steam. The wall thinning is the result of the dissolution of the normally protective oxide layer formed on the surfaces of carbon and low alloy steel piping. The rate of metal loss depends on a complex interplay of several parameters including water chemistry, material composition, and hydrodynamics, but based on operating experience the metal loss can be as high as 3 mm/yr. Carbon steel piping components that carry wet steam are especially susceptible to FAC and represent an industry wide problem. The most dominant variables are temperature, fluid velocity, fluid pH, the water amine, oxygen content, steam quality, void fraction of the fluid, piping geometry, and the pipe material composition. This section describes the different variable effects. It is important that FAC degradation is diagnosed correctly so that the correct mitigation methods can be implemented. Historically the terminology was ambiguous since erosion-corrosion was used for both the chemical mechanism now known as FAC and the mechanisms in which the oxide is broken down mechanically by the impingement of particles, solids or gaseous bubbles. There are also differences in the surface morphology of FAC and erosion-corrosion. Single phase FAC has a scalloped or orange-peel appearance and two phase damage often has a characteristic pattern known as tiger striping. These surface features are absent in surfaces damaged by erosion mechanisms. Another difference is that FAC is often more widespread than the localised erosion damage. It should also be noted that most of the codes developed to predict wall thinning do not distinguish between FAC and erosion-corrosion. A complete glossary of the different mechanisms which can be confused can be found in Appendix B.

FAC mainly affects the secondary circuit of pressurized water reactors, but also BWR feedwater piping is susceptible to single phase FAC induced damage. In BWRs, several main steam line sub-systems, including the high-pressure turbine exhaust piping, the turbine crossover piping, the extraction steam lines, and certain straight portions of the steam lines are susceptible to two-phase FAC, Shah and MacDonald [21]. The moisture content in the main steam leaving the reactor pressure vessel is about 0.1 % and increases as the steam reaches the main turbines. The high moisture content in the steam extraction and exhaust lines and turbine crossover lines makes these lines particularly susceptible to FAC. The main steam line pipes are not susceptible to FAC unless moisture is present.

3.1 Effect of Temperature

An important variable affecting the FAC resistance of carbon and low alloy steels is temperature. Most of the reported cases of FAC damage under single-phase conditions have occurred within the temperature range of 80 to 230 °C, whereas the range is displaced to higher temperatures (140 to 260 °C) under two-phase flow. The exact location of the maximum wear rate changes with pH, oxygen content, and other environmental variables. Experience has shown that the wear rate is highest at around 150^{o}C and increases with fluid velocity. Furthermore, FAC can occur in low temperature single phase systems under unusual and severe operating conditions.

3.2 Effect of Flow Velocity

Flow rate of the liquid has been found to have a linear effect on the FAC wear rate. As higher velocities are experienced, higher wear rates are expected. Since the enhanced mass transfer associated with turbulent flows is the fundamental process in the accelerated dissolution of the pipe wall protective oxide layer, the effect of flow is best described in terms of the mass transfer coefficient, which is a function of flow velocity and geometry. Local flow velocities can differ by a factor of 2 – 3 from the bulk flow velocity.

3.3 Effect of Fluid pH

FAC wear rates are strongly dependent on pH. In general, increasing the pH value reduces the wear. The FAC wear rate of carbon steels increases rapidly in the pH range of 7 – 9, and drops sharply above pH 9.2 Wu [4]. As the fluid becomes more acidic, more pipe wall losses are expected. The pH value can be affected by the choice of control agents (e.g., morpholine or ammonia) and by impurities in the water. In two-phase flows the critical parameter is the pH of the liquid phase. This can be significantly affected by the partitioning of the control agent between the steam and liquid phase. There is no adjustment of pH performed in BWR plants.

3.4 Effect of Oxygen

FAC rates are inversely affected by the amount of dissolved oxygen (DO) in the feedwater, and too low an oxygen level is harmful to carbon steel piping. The FAC rate decrease rapidly when the water contains more than 20 ppb oxygen [21], but the precise oxygen level required to prevent FAC depends on other factors such as pH and the presence of contaminants.

In BWRs, hydrogen water chemistry (HWC) can be applied with the main intention to suppress intergranular stress corrosion cracking (IGSCC) susceptibility and crack growth rate. The FAC rate has been measured in a laboratory test to be higher for a time period of 8 months after starting HWC. After this time the FAC rate appears to be similar to that in a reference normal water chemistry (NWC) environment. General Electric guidelines consider an oxygen level of 20 to 50 ppb desirable for hydrogen for hydrogen water chemistry. Some plants must add oxygen in their feedwater when using HWC, while others do not [21]. The effects of higher hydrogen levels under NWC conditions are plant specific and must be taken into account as for HWC conditions. The use of noble metals to reduce the quantities of hydrogen required to establish HWC conditions has to date not had a more pronounced effect on FAC than the application of HWC itself.

Main steam lines made of carbon steel are susceptible to FAC in the steam phase because most of the oxygen, being a gas, remains in the steam phase and does not partition to the liquid. For the same reason injection of oxygen into the wet steam will not prevent FAC. Injection of hydrogen peroxide has been explored as a possible mitigation for FAC because most of the hydrogen peroxide partitions to the liquid phase and spontaneously decomposes into oxygen and water and thus, enriches the liquid phase with oxygen. However, although the FAC rate is decreased, hydrogen peroxide injection is not as effective as a remedy towards FAC as replacement of materials to low alloy steel (alloyed with chromium) or the presence of a stainless steel coating.

3.5 Effect of Alloy Additions

The FAC rate is highest in carbon steel piping with very low levels of alloying elements. The presence of chromium, copper and molybdenum, even at low percentage levels, reduces the FAC rate considerably. The relative corrosion rate of steels is reduced by 80 % at a chromium content as low as 0.2 %. The FAC rate is decreased by a factor of 4 with the steel type 2-1/4 % Cr and 1 % Mo (2-1/4 Cr- 1 Mo steel). Austenitic stainless steels are virtually immune to FAC [4].

3.6 The Entrance Effect

In the 1990s a new FAC wear effect was documented. This effect has been called the “leading edge effect” or the “entrance effect.” This effect occurs when flow passes from a FAC-resistant material to a non-resistant (susceptible) material, which causes a local increase in the corrosion rate. This effect is normally manifested by a groove up- or downstream of the attachment weld between the corroding and the resistant material. In one, relatively recent example significant wear was detected in an expander. The area in question consisted of a valve followed by a 150 mm by 200 mm expander attached to another 200 mm by 400 mm expander. A number of FAC inspections had been performed before and after replacement of the upstream expander with resistant material due high FAC wall thinning. An almost linear thinning rate of 1.77 mm per cycle was noted over the subsequent refuelling cycles (RF11-RF15); the downstream expander was replaced during the RF15 outage. The CODAP Event Database includes several examples of pipe damage caused by the entrance effect.

The effect of piping and piping component geometry is also a contributing factor to the occurrence of FAC. The general layout of the piping such as the positioning of elbows, Tees and inner surface geometry such as reduction of the internal diameter, surface finish of weld roots, flow changes in valve bodies, orifices, pressure reducers, areas where flow, pressure and temperature are measured, and regions where the inner surface finish or geometry change over short distances, are all contributing factors to the occurrence of FAC.

4. FAC MANAGEMENT & FAC MITIGATION STRATEGIES

This section summarizes the FAC management and FAC mitigation strategies that have been implemented by the CODAP member countries. Computer programs for tracking and predicting wear include BRT-CICERO^{TM}, CHECWORKS, COMSY, and WATHEC. Where computer codes are used to monitor pipe wall thinning rate, the feedback of operating experience and inspection data is necessary to improve the accuracy in the FAC wear predictions. The conjoint requirements for FAC provide the basis for the formulation of mitigation strategies; e.g., changing the water chemistry or material composition. This section summarizes the different national approaches to FAC mitigation.

4.1 Canada

Following the 1986 Surry FAC failure event, the Canadian Nuclear Safety Commission (CNSC) requested the Canadian utilities to implement monitoring and inspection programs for FAC. Following the Mihama incident, utilities ensured that their inspection programs would address degradation such as experienced at Mihama. The 2009 edition of the Canadian Standards Association (CSA) Standard N285.4 introduced a new requirement (7.4.7) for assessment of flow accelerated corrosion and consequent identification of inspection sites resulting from assessment. These requirements are intended to access flow accelerated corrosion mechanisms that result in loss of piping or component wall thickness, either locally or over a large area. Standard N285.4 requires that the assessments be reviewed at intervals not exceeding 5-years. This shall take into consideration equipment operational history and any modifications, repair or replacements. Feeder piping and steam generator tubing are addressed separately in component specific periodic inspection as well as life cycle (commonly referred to as ageing) management programmes. Furthermore, Standard CSA N285.4 specify the parameters that shall be considered in the assessment, i.e., composition of component material (primarily ferrite material), operating temperatures, hydrodynamic conditions, component geometry, coolant quality, chemistry, and operating time at adverse conditions.

In Canada, the FAC programs are based on CANDU Owners Group (COG) and EPRI guidelines in NSAC-202L “Recommendations for an Effective Flow Accelerated Corrosion Program.” The general approach to FAC management involves:

- Assess susceptibility using CHECWORKS and conduct monitoring inspection at selected sites

- Monitoring of flow accelerated corrosion in feeders in accordance with requirements of CSA N285.4 (Clause 13)

- Document and report findings in accordance with either CSA N285.4 (for nuclear grade piping) or Canadian Regulatory Document RD-99.

The FAC programmes vary between utilities but in general they are intended to predict, detect, and monitor wall thinning in feeders, piping, tubing, fittings, valve bodies, feedwater heaters and heat exchangers, among safety significant systems and components. CHECWORKS, analytical stress analyses and periodic examinations of locations that are most susceptible to wall thinning due to flow accelerated corrosion are used to predict the amount of wall thinning. The FAC programmes include analyses to determine critical locations, baseline inspections to determine the extent of thinning at those critical locations, and follow-up inspections to confirm the predictions. Inspection are performed using range on non-destructive techniques ranging from eddy-current, ultrasonic, radiographic to visual and any other industry recognized reliable testing techniques capable of detecting wall thinning. Repairs and component replacements are performed on as needed basis.

Operating experience reviews indicate that to date there have been no incidents at Canadian nuclear power plants involving catastrophic failures due to FAC. Current FAC programs as implemented by all utilities provide reasonable assurance that wall thinning aging effects are adequately managed so that intended functions of systems and components are maintained consistent with the current licensing basis and internationally accepted practices.

4.2 Chinese Taipei

In 1989 the US NRC issued GL 89-08 [8] requiring all nuclear power plants in the United States to assess the impact of erosion-corrosion on carbon steel piping. The same year Taiwan Atomic Energy Council (AEC) requested the Taiwan Power Company (TPC) to inspect wall thickness of carbon steel piping during the refuelling outage. TPC submitted a long term wall thickness inspection plan for carbon steel piping to AEC in June 1989. In 1994 TPC introduced the EPRI CHEC family of computer codes (encompassing CHECMATE, CHEC-NDE and CHEC-T) and also adopted EPRI recommendations for the selection and assessment of piping susceptible to FAC. The piping which is susceptible to FAC under normal operating conditions can be evaluated using CHECMATE. The calculated FAC wear rate and the remaining life of piping are taken as the major parameters to screen piping to be inspected during the refuelling outage. For carbon steel piping not addressed by CHEC the effect of FAC can be obtained by wall thickness measurements based on past inspection records, plant operation and repair history as well as plant specific and international experience feedback. In 1995 TPC formally applied these tools for the inspection of carbon steel piping in Chinshan Unit 2. In 1998 TPC upgraded CHECMATE from the DOS version to a Windows based version. Currently, CHECWORKS 3.0 sp 2 is being used to plan inspections for FAC.

FAC Inspection plan

The principles of pipe screening for the long-term monitoring were based on AEC approval of the FAC inspection plan. The plan was developed by reviewing isometric drawings of the piping, its geometrical configuration and taking into account the feasibility of performing in-situ measurements. The FAC inspection plan consists of the following three parts in order to select the piping segments adequately for inspection:

- Piping addressed by CHEC Code
    - The calculated line correction factor (LCF) using CHEC Code is between 0.5 and 2.5, and the remaining life is less than 26,280 hours.
    - The calculated LCF using CHEC Code is less than 0.5 or greater than 2.5, and the wall thinning rate is greater than expected by CHEC Code.
- Piping not addressed by CHEC Code
    - The residual life of a pipe segment is less than two fuel cycles.
    - Sampling piping segments for long term monitoring of piping systems.
    - Piping replaced during the most recent refuelling outage.
    - Piping segments included in the next inspection cycle based on assessment results from the previous refuelling outage.
    - Experience learned from historical inspection records, international and plant specific operating experience.
    - Items requested by the regulator, or identified by plant specific assessment.

- Large diameter pipe segments: pipes of 30 or 24 inch diameter with no isometric drawings visual inspection (VT) of the inner surface is performed of instead of UT inspection. If wall thinning is detected by VT, a UT measurement is performed.

Long-term inspection strategy for FAC

In response to the request from AEC concerning a long-term inspection strategy and based on the experience gained from 1995 to 1998, TPC prepared the pipe wall thickness long-term inspection strategy (FAC inspection plan) for NPPs in September 1998. Taking into consideration NSAC-202 Revision 2 [22] issued following the Mihama accident, the second edition of the FAC inspection plan was issued. The third edition was later issued to include the Lung-men plant.

Based on inspection records and EPRI recommendations the FAC inspection plan developed by TPC included the pre-screening of piping which exempts the following piping systems from the FAC inspection plan.

- Piping systems which are non-safety related and are not operating during normal operation.

- The sea-water, fire water, sampling lines, plant ventilation, floor and equipment drains, gas, oil and fuel, and waste systems which are not located in the reactor building, turbine building or reactor auxiliary building.

- Piping with a diameter equal or larger than 3 inches and which operates during off-normal conditions and operates at a temperature less than 93^{o}C and a pressure of less than ~2 MPa.

- Raw water, chemical control or vendor supply piping systems which will not affect the safe operation of the plant.

- Piping with a diameter equal or larger than 3 inches and not made of carbon steel or low alloy steel.

The purpose of the FAC long-term inspection plan is to manage the FAC trending of piping systems in an effective manner by appropriate and adequate selection of piping for examination during refuelling outages. Basically there are two classes of piping systems that must be monitored and included in the plan:

- Class A piping screening criteria significantly affected by FAC: steam quality $\leq$ 99.5 %; operating temperature $\geq$ 93^{o}C; per cent of operating time $\geq$ 2 %; material composition Cr $<$ 1.25 %; incomplete design and operational data necessary for the evaluation software.

- Class B piping screening criteria that meets any of the following requirements: operating temperatures below 93^{o}C; per cent of operating time less than 2%; piping containing superheated steam extracted from the low pressure turbine; incomplete design and operational data.

In addition, the plant specific and other experience feedback and design changes are evaluated and can lead to a revision of the FAC inspection plan. Feedback from the inspection records are also taken into account.

4.3 Czech Republic

FAC events must be reported to the National Regulatory body of the Czech Republic (SÚJB) in the following instances:

- Failure in safety class piping

- Reduction of NPP power
- General information concerning replacement during an outage.

The FAC management programme in the Czech Republic is based on the EPRI program CHECWORKS and on actual thickness measurements. The critical value for component thickness, t_{mA}, is based on a simple analytical evaluation of the thickness. It is calculated according to The Czech Normative Codes, similar to the minimum value of piping thickness according to ASME, Section III – Division 1 –NC 3641 Class 2 components:

$$t_{mA} = \frac{pD_0}{2(S+Py)} + A \qquad (3)$$

Where:

D_o = Outside diameter

S = Allowable stress at temperature

P = Internal design pressure

y = Additional thickness

A = Coefficient.

The critical thickness of the piping is based on fulfilling structural integrity stress conditions for all defined loadings. A stress analysis using piping programs is usually used. The actual thickness of the piping is usually larger than the design dimensions of pipes, valves and other components installed in a system. In some cases of local thinning and complicated replacements a detailed analysis is performed to determine if there are sufficient safety margins. The following procedure is used:

- If the thickness measurement is close to the critical value, which is based on a simple analytical evaluation of the thickness, a more detailed analysis is performed if the thinning is local. The analysis must include all regions with significant thickness reductions.
- Detailed analysis of the stress limits are performed in accordance with the Czech Normative Codes.
- Piping programs are normally used to determine the influence of piping forces and moments on the areas with reduced thickness.
- Detailed finite element analysis (FEA) of the areas with reduced thickness is performed using the actual thickness measurements. For this detailed analysis the stress limits as defined in the Czech Normative Codes are used. All limits for all stress categories must be fulfilled.

Particular attention is paid to uncertainties with regard to significant thinning that has been measured or is expected to occur such as locations with pipe supports, wall penetrations, pipe-whip restraints, T-joints with reinforcing pads and generally inaccessible regions of piping.

4.4 Finland

The Finnish Nuclear Safety Authority (STUK) prepares guidelines for design and operation of nuclear power plants, but the responsibility for safe operation lies with the licensee. The YVL guidelines, which can be found on the STUK website (www.stuk.fi), are under revision at the moment, thus some references in the following are to drafts of a guideline. Concerning wall thinning, i.e., flow-accelerated corrosion (in the guidelines also named flow-assisted corrosion) and erosion-corrosion, the following is stated:

- "Susceptibility to erosion-corrosion shall be limited by proper selection of materials and by avoiding flow discontinuities and exceptionally high flow rates. The design shall also consider phase changes of the fluid and the accumulation of non-condensable gases in the piping." (In Guideline YVL.3.3, Nuclear Facility Piping [23]).

- When considering the need for updating the stress analysis, this is needed also in the following situations if: "1) modifications leading to unfavourable changes of pressure equipment dimensions, material properties, supports, loadings or other factors; 2) design pressure or temperature rise or operational change giving rise to increased loadings; 3) loading increase, wall thinning, unexpected fracture toughness decrease or any other deviation from input data which tends to reduce the safety margins." (In Guideline YVL 3.5, Ensuring the Strength of Nuclear Power Plant Pressure Equipment [24]).

- Concerning nuclear aging management: "Aging means physical or technological aging which may occur in nuclear plant structures, systems and components. Appendix A presents typical aging mechanisms appearing in nuclear plants", and "Erosion-corrosion – In the case of erosion-corrosion, the fluid flow rate that exceeds a critical value removes the oxide film protecting the metal surface, thereby speeding up the corrosion." (In Guideline YVL A.8, Nuclear Plant Aging Management, Draft [25]).

- "Pressure vessels and their internals shall be designed in such a manner that the flow rates, flow-induced vibration, as well as phase and temperature changes of the fluid will not cause erosion, corrosion, flow-assisted corrosion, fatigue or other degradation." (In Guideline YVL E.3, Pressure Vessels and Piping in Nuclear Plant, Draft [26]).

- "Strength analyses shall be revised during the service life if the pressure component exhibits load increases, reduction in wall thickness or decreasing fracture toughness values deviating from the design bases. Revision may also become necessary due to an extension of service life, periodic safety assessment or an event affecting safety due account of which could not be taken during design." (In Guideline YVL E.5, Nuclear Pressure Equipment In-Service Inspections by Non-Destructive Testing, Draft [27]).

- "The assessment shall take into account the failure mechanisms specified in ASME B&PV Code, Section XI, Non-Mandatory Appendix R[1], Supplement 2, Table R-S2-1, such as fatigue, stress corrosion cracking, and erosion-corrosion. Water hammer and other exceptional loadings, as well as repairs, shall be considered in the risk assessment. If structural reliability assessment is based on probabilistic fracture mechanics models, they shall be evaluated by expert judgment in conformance with recommended practice ENIQ RP 9." (In Guideline YVL E.5 [27]).

To fulfil the requirements of the YVL guidelines and to secure safe and economic operation of nuclear power plants, the licensees have their own plant life management programs including procedures and technical instructions, as well as a systematic approach for monitoring wall thinning and planning for in-service inspection including NDE. All Finnish in-service inspection programs are based on risk-informed in-service inspection (RI-ISI). The Loviisa power plant has applied AVT secondary-side water chemistry suitable for a not fully copper-free secondary system. The COMSY computer code is used to assist in plant life management. The same code has been selected for the Olkiluoto-3 EPR power plant, which is currently under construction, and where the secondary-side

[1] The Non-Mandatory Appendix R of ASME XI addresses "Risk-Informed Inspection Requirements for Piping."

water chemistry will be H-AVT. The COMSY code has been used by the plant supplier to identify susceptible systems and locations as well as ascertain the material selections. The FAC program includes pre-operational inspections as well as inspections during operation.

4.5 France

In-service inspection of safety related piping is performed according the RSE-M Code (2010 Edition with the 2012 Addendum) issued by the French Association for Design, Construction and In-Service Inspection Rules for Nuclear Island Components (AFCEN). The RSE-M Code takes into account the French operating experience supplemented by some requirements stipulated by French Law. For non-safety-related FAC-susceptible piping, a new "pressure vessel law" was issued by "Syndicat National de la Chaudronnerie, de la Tôlerie et de la Tuyauterie Industrielle" (SNCT) on March 15, 2000. Initially, this law applied to piping > DN100 with an operating pressure > 0.5 bar. It has subsequently been revised to apply to small-bore piping as well. According to the high-level requirements of the pressure vessel law, the integrity of pressure bearing components must be maintained continuously, and the plant operators have the ultimate responsibility for maintaining structural integrity throughout the lifetime of the plant.

After the Mihama event in 2004, EDF re-examined its FAC inspection strategy. A new "National Maintenance Rule" (RNM) for non-Code piping was issued in 2009 [28]. The main principles of the RNM are as follows:

- The selection of inspection locations is based entirely on predictions by the BRT-CICERO™ software developed by EDF.
- Each pipe section modelled in BRT-CICERO™ and predicted to be below design wall thickness at outage N+1 must either be inspected (by chromium and thickness measurements) during outage N, or a written justification for continued operation must be submitted to the Recognized Inspection Service (RIS)[2] for review and approval. A special inspection programme is required for welds that are assessed to be susceptible to FAC (e.g., locations immediately downstream of flow control valves, elbows, reducers).
- For lines susceptible to FAC that are not yet modelled in BRT-CICERO™ (e.g., small-bore piping), an inspection programme must be developed that includes the most sensitive areas according to engineering evaluations that address local flow conditions, service experience, and base metal chromium content.

Weld root areas have been identified as a weak point indicating a need to develop and adapt an inspection method that enables the detection of wall thinning at or near welded pipe connections. Since 2006 the ultrasonic "Time of Flight Diffraction" (TOFD) technique has been tested, and it was qualified in 2009 for locating and characterizing FAC-induced defects in weld root regions. Based on thousands of inspections, it has been determined that weld root areas with misalignments between pipe fittings are susceptible to preferential FAC. References [10] and [29] provide additional details on the FAC management approach at EDF.

4.6 Germany

In comparison to other corrosion mechanisms such as chloride-induced TGSCC or pitting corrosion, flow accelerated corrosion (FAC) is not considered to be a major issue in German NPPs. However, incidents of minor safety significance due to FAC, mainly in the area of the water-steam cycle, have been observed from the very beginning in German PWRs as well as BWRs. Larger dimension piping of the main feedwater and main condensate systems have to date not been seriously affected by FAC. As a typical countermeasure to avoid the recurrence of FAC, affected pipe sections

[2] Independent third party inspection organization.

have been replaced. Depending on the safety relevance of the system, materials with higher resistance against FAC such as austenitic steels have been used.

The positive German operating experience regarding FAC is usually attributed to specific design characteristics on the one hand, and to the implemented water chemistry on the other hand. The most important design features of German NPPs with respect to mitigation of FAC are:

- Lower average flow velocity,
- Wider use of resistant materials (higher alloyed steel),
- Reduction of geometrically-induced turbulence.

Besides design features, a crucial approach to prevent FAC is the chemical treatment of the secondary side. Here, the pH-value of the secondary side of PWRs and the oxygen concentration of the water-steam cycle of BWRs are considered to be key-parameters. Since the mid-80s, all German PRWs have been using the so-called High-AVT Treatment with pH > 9.8. The main prerequisite for this chemical mode is a copper-free secondary side which has been achieved by using condenser tubes made of titanium or stainless steel. If all requirements for High-AVT are met, the occurrence of FAC on the secondary side of PWRs can be excluded to a large extent. In case of BWRs, High-AVT Treatment is not applicable. Here, protection against FAC is achieved by controlling the oxygen concentration in the water-steam cycle. According VGB Guideline R401 J [30], the oxygen concentration in the main feedwater system has to be adjusted between 20-200 μg/kg during normal operating conditions. As an important consequence of the chemical treatment modes outlined above, the number of events due to FAC has decreased over the years.

In terms of in-service inspections, as far as GRS knows, visual inspections and ultrasonic wall thickness measurements are carried out regularly in German NPPs. The spatial and temporal inspection frequency depends on the plant-specific experience and is fixed in the corresponding inspection programmes. Computer codes such as COMSY are also used to perform a plant-wide screening for identification of system areas which are sensitive to degradation mechanisms typically experienced in nuclear power plants (FAC, corrosion fatigue, IGSCC, Pitting, etc.). Important criteria for determination of FAC-relevant test areas are:

- Carbon steel, low alloyed steel,
- Medium temperature between 80 – 250 °C,
- Locally increased flow velocity.

The inspection programmes have been improved over the years taking into account lessons learned from major events in foreign NPPs for example Surry-2 (1986) and Loviisa-1 (1990). An assessment of operator’s inspection programmes in the context of the Mihama-3 event (2004) indicated no deficiencies.

4.7 Japan

The Japanese "Act on the Regulation of Nuclear Source material, Nuclear Fuel Material and Reactors" [31] requires that the nuclear power reactor licensees maintain their facilities, including components such as piping, to conform to the technical standards. FAC is managed as part of efforts for pipe wall thinning management according to stipulations of related legislations, regulations and regulatory guides and requirements of academic societies and associations' codes. It is also necessary to satisfy various requirements, including the measures against aging, for the piping and other components during the design, operation and periodic evaluation of each plant.

The structure and integrity of piping conforming to these provisions are specified in the JSME Code [32]. Specifically, for FAC, it is required that the thickness not be less than the minimum required thickness (T_{sr}), and that pipe wall thinning management is performed to ensure this. Licensees must perform periodical inspections and confirm that the facilities concerned conform to the stipulated technical standards. According to the Rules for Commercial Power Reactors [33], the methods used for the periodic inspections must be carried out by non-destructive examinations and visual inspections to check for damage, deformation, wear, or any abnormal condition. The methods, frequency of inspection must be submitted to, and approved by the Japanese Nuclear Regulation Authority prior to the commissioning of a commercial nuclear power plant. There are specific codes for BWR [34] and PWR [35] plants concerning the management of pipe wall thinning that cover the scope and appropriate methods. In addition, the following must be taken into account: 1) Pipe thinning phenomenon from the outside surface, 2) wall thickness management of branch connections, and 3) the method of wall thickness management from the first wll thickness measurement. Trend monitoring of pipe wall thinning is performed continuously from the beginning of operation and is to be based on the pipe wall thinning measurements.

Ageing management is based on a technical evaluation of nuclear power facilities and includes the following:

- A technical assessment of the ageing degradation of components and other elements which have aged 30 for years or more after commissioning for each subsequent 10 years (Ageing management technical evaluation), and

- Based on the technical assessment, the strategy for maintenance management to be performed during the next 10 years (long-term maintenance management policy) is developed and/or changed, if necessary.

The pipe wall thinning of ageing management technical evaluations is specified in the ageing-management implementation guide [36] as follows:

- Since pipe wall thinning is a degradation phenomenon that should be managed by performing an evaluation of its occurrence and thinning rate at all times, it is not necessary to select the pipe wall thinning for measures other than ageing management:

  - The ageing degradation phenomena that generate, or for which generation cannot be excluded, on the components and structures included in the ageing management technical evaluation, that are not covered by any of the above, and

  - The aging degradation phenomena for which the deterioration degradation management is performed appropriately, corresponding to the characteristic change over time with routine maintenance management.

- However, for the routine degradation management of pipe wall thinning the way of thinking, methods, plans, and records of deterioration for the surveillance of degradation tendencies must be described in the ageing management technical evaluation report.

With regard to the seismic safety evaluation on pipe wall thinning, the example of process of seismic-safety evaluation on the pipe wall thinning is shown in the Seismic Safety Evaluation of Review Manual for Technical Evaluation of Measures for Aging Management [36]. This process is an example of evaluation based on the technical standard on the pipe wall thinning of the JSME, assuming the necessary uniform wall thickness against the service pressure at elbows, reducers and the lower stream that are regarded as channeling portions.

Based on the aging management technical evaluation results, the strategy for maintenance management (long term maintenance management policy) is developed and if necessary revised. This is a part of the application document for the operational safety program which must be approved by the national government.

An example of the long-term maintenance management policy on the PWR pipe wall thinning is as follows. For the corrosion (erosion-corrosion and erosion) from the header inner surface of carbon steel piping, such as the Main Steam system piping, the following matters are performed. However, the policy is a short-term plan that should be started within five years and completed within 10 years after development of the plan:

- Based on the PWR Code [35] and the inspection results, the need for maintenance is determined, and the management guideline of the secondary-system pipe wall thickness is revised when necessary.

- For the piping other than the piping subject to the management guideline of the secondary-system pipe wall thickness, in order to obtain further knowledge of pipe wall thinning phenomena, the wall-thickness is measured and the data are accumulated.

- The wall-thinning trend is managed by the pipe wall thickness management system, the need for maintenance is determined based on the wall-thinning trend data, and the implementation plan is revised when necessary.

- For the carbon steel piping (drain line piping) for which the seismic safety evaluation was performed based on the actual wall-thickness measurements, the seismic safety is re-evaluated based on the future wall-thinning progress estimated by the extrapolation of actual measurements.

An example of the long-term maintenance management policy on the BWR pipe wall thinning is as follows:

- For the erosion corrosion and erosion of the inner surface of carbon steel piping and low-alloy-steel piping, based on the BWR Code [34], when the outcome of the safety infrastructure research is obtained, the necessity of feedback to maintenance is determined, and the in-company guideline is revised when necessary.

- For the carbon steel piping (extraction system of gland steam system) for which the seismic safety evaluation was performed based on the actual wall thickness measurements, the seismic safety is re-evaluated based on the future wall-thinning progress estimated by extrapolation of actual measurements.

- When the outcome of the safety infrastructure research on the seismic safety evaluation method supposing pipe wall thinning is obtained, the necessity of feedback to maintenance is determined, and the implementation plan is developed when necessary.

4.8 Korea (Republic of)

The FAC management programme of carbon steel piping system is one of the augmented in-service inspection programmes in Korea. It was initially started in 1987 following the Surry-2 event in the U.S.

Based on NSAC-202L [22], Korea Hydro & Nuclear Power (KHNP) developed the Corporate Standard Procedure No. 10 “Thinned Pipe Management Program” to provide an integrated guideline. It has been applied to the secondary side carbon steel piping system in all Korean NPPs (PWR and PHWR) since 2002. It provides detailed procedures concerning susceptibility analysis, wear rate prediction, component selection to be inspected, wall thickness measurement, remaining life evaluation, follow-up actions including analytical integrity evaluation, repair/replacement, and documentation. The utility utilizes the CHECWORKS computer code for predicting wear rate. Wall-thinning mechanisms (Cavitation, flashing, liquid drop impingement etc.) other than FAC can be also considered in the component selection process for ISI based on operating experience. Effectiveness of the secondary side water chemistry programme in the connection with the FAC management programme is also reviewed in accordance with the Steam Generator Management Programme.

For each plant, a FAC inspection plan is incorporated into the Long-Term ISI Plan. For carbon steel piping systems in a newly constructed NPP, an initial thickness is measured and the FAC database is established during the pre-service inspection period.

4.9 Slovak Republic

In response to recurring observations of wall thinning of the 5th to 8th stages of the steam extraction piping at Bohunice Nuclear Power Plant (EBO), Slovenské elektrárne, a.s. (SEAS) began to address FAC systematically in 1992. Building on the experience with the EBO FAC programme, an equivalent FAC programme was implemented in 1996 for the Mochovce Nuclear Power Plant (EMO) prior to Unit 1 entering commercial operation in 1996 and Unit 2 in 2000. Listed in Table 1 are the systems monitored for FAC according to the FAC Programme Guidelines of SEAS.

Table 1: Scope of SEAS FAC Programme

List of Pipe Lines Monitored for FAC	
Steam from SG to TG	Heating condensate from separator to feedwater tank
Inlet of feedwater pump	Heating condensate from HPH
Outlet of feedwater pump	Heating condensate from LPH
Condensate from LPH to feedwater	Steam to deaerators
5th -8th Extraction steam system of TG	HP, LP gland steam
Main condensate from 1st level	Drainage of steam pipelines
Steam collector 0,7 MPa	

Between 1992 and 1995, a mathematical model of the secondary circuit pipeline for EBO was developed and it was modified for the EMO in 1996. Initial analysis of the technical state of selected pipelines systems was carried out and critical components with the highest corrosion rate were identified. The first measurements were carried out on the critical components, and components selected on the basis of experience from other plants. SEAS began performing measurements of critical components in sufficient time which has resulted in there being enough time to plan manufacturing of new components and the replacement of critical components. Currently, SEAS selects components for wall thickness measurements on the basis of the following criteria:

- Minimum measured thickness < minimum allowable thickness + 2 mm (2 mm is the margin for early detection in critical components, including the uncertainty of measurements)
- Calculated lifetime of components is sufficient for safe operation for at least 1 campaign
- Engineering consideration and experience
- Component was evaluated as critical at the other unit or turbine
- The component has not been measured for over 3 years

All the necessary information about the components monitored is stored in an aging management database. Evaluation and selection of components is carried out using the software which is part of the aging management database.

Secondary-side piping at EBO and EMO is made of carbon steel with low concentrations (from 0.00 to 0.03 %) of chromium, molybdenum, and copper. As a result, wall thinning problems were noted initially in EBO in 1992. Sections of the carbon steel extraction steam piping were replaced with piping of new material. The new material contained higher concentrations of chromium and molybdenum. In the EMO the first leaks appeared on the extraction steam pipelines after 6 years operation. In 2010, a project was initiated at EMO to replace components in the extraction steam pipelines. New components were made from material with higher levels of chromium and molybdenum (1.4 %). The replacement will be completed in 2014.

Also evaluated is the impact of chemical parameters on FAC. The main parameters monitored are pH, concentration of ammonia, hydrazine and oxygen. The corrosion rate is to a certain extent proportional to the concentration of iron in the feed water. Introduction of an ethanolamine regime at pH of 9.2 instead of the ammonia-hydrazine regime has helped us decrease the concentration of iron.

Before power uprate at both EBO and EMO to 107 % the impact of a power increase on the loss of wall thickness pipelines secondary circuit by flow accelerated corrosion was evaluated. The calculations show that after a thermal power increase to 107 % the rate of damage due to FAC will also increase in most parts of the secondary circuit. This increase is expected to be negligible, however.

In 2005, an effort began to assess the monitoring FAC by acoustic emission. This was done within the research programme entitled “Development & Implementation of a System for the Detection of Pipeline Damage Caused by Flowing Medium under Operational Conditions of the EBO NPP.”

4.10 Spain

Spanish NPPs perform periodic surveillance of piping systems and components susceptible to the degradation mechanism of flow accelerated corrosion (FAC) from the moment they come into operation. Systems or parts thereof considered eligible for this degradation mechanism are those made of carbon steel or low alloy steels, with circulating water or wet steam.

Surveillance programmes are developed following Generic Letter 89-08 of the USNRC [8] and EPRI report EPRI NSAC-202L-R3 [7]. Some plants also make use of the EdF guide *“Corrosion des aciers dans la vapeur humide circulant à grande vitesse”* (Corrosion of Steels Due to Wet Steam at High Speed) [37].

These surveillance programmes are included in some cases in the In-service Inspection Programme of the NPP, and in other cases they are developed as separate surveillance programmes specific for this degradation mechanism, following the requirements of the CSN Guide IS-23 about In-service Inspection [38]. These programmes specify the criteria for eligibility of systems and components, the scope, the criteria for selecting the areas to be inspected at each refuelling outage, the methodology to follow during the performance of the inspections, assessment of the inspection results, the acceptance criteria, as well as the steps to be followed in case the acceptance criteria are exceeded, that is to say, repair or replacement of such areas.

Recently, as a consequence of the development of the Ageing Management Plan (AMP) of the Spanish NPPs, following the requirements of the CSN Guide IS-22 [39], all activities performed by the plants related to prevention, mitigation, inspection or monitoring and control of this degradation mechanism, have been assessed by CSN according to the attributes considered in chapter XI of NUREG-1801 “Generic Aging Lessons Learned (GALL) Report” Rev. 2 [40]. This assessment introduced improvements in the FAC programmes in order to adapt them to the GALL report, affecting the scope of the systems, the use of predictive codes, such as CHECKWORKS or other similar codes. These improvements are being implemented by all NPPs following the AMP.

4.11 Sweden

The Swedish Radiation Safety Authority (SSM requires that the licensees should have a program for management of ageing degradation and damage. With regard to reported FAC related events: in Sweden, between 1970 and 2010, FAC has been the most frequent degradation mechanism. Power upgrades and the fact that the plants are planning for long term operation will lead to these regulations being tightened and the frequency of regulatory inspections with regard to ageing management will increase. A combination of operational experience feedback and predictive FAC computer codes are examples of important tools to manage FAC.

A qualitative approach to risk-informed in-service inspection (RI-ISI) has been applied for many years in Sweden. It is based upon control grouping using a matrix combining potential risk of damage occurring in a given system or component with the consequences of failure, and the anticipated severity of fuel damage and a subsequent radioactive release. The qualitatively based approach to control groups for mechanical components is based on the assignment to an inspection group in accordance with the magnitude of risk assessed for the specific component. Structural parts for which the risks are assessed to be the highest are assigned to control group A, Those with lesser risk to control group B, and those with the lowest risk to control group C. The percentage of a given control group which must be inspected decreases with the risk. In principle 100% of control group A must be examined, with a frequency not exceeding ten years. To determine the division into the control groups the following risk matrix has been developed enabling the assignment of a damage index and a consequence index to each component and parts thereof, see Table 2.

Table 2: Swedish Control Group Matrix

Damage Index	**Consequence Index**		
	High	**Medium**	**Low**
High (I)	A	A	B
Medium (II)	A	B	C
Low (III)	B	C	C

The consequence index expresses in a qualitative manner the likelihood that a crack or other degradation process will result in fuel damage, discharge of large amounts of radioactive substances, or other forms of damage which could lead to health problems or an accident. The consequence index is determined mainly by the margin to such consequences as the result of a break or malfunction of the specific component or part of a system. Two aspects are important when determining the assignment of the consequence index:

- System margins – how many systems or system circuits are essential in relation to the number available,
- Thermal margins – how much the fuel can be heated up in relation to acceptable margins.

The damage index is determined by the loading conditions, environment and material in relation to the dimensions of the component. Components or parts which may be exposed to loads or other conditions which experience has shown can result in damage or degradation should be assigned the highest damage index. Components which experience has shown are not expected to be subjected to loads or other conditions which will result in damage are assigned damage index II, and components exposed to minimal loads or other benign operational conditions are assigned damage index III.

Structural elements that may be exposed to FAC are often assigned to control group C. This is mainly due to the severity of the consequences to reactor safety of FAC damage in the systems where this mechanism occurs. More detailed consequence analysis using PSA conducted by the Swedish licensees has shown, however, that the current basis for assignment of consequence index may need to be revised in particular for the plant cooling systems.

4.12 Switzerland

In Switzerland no major pipe failures due to FAC have been observed in recent years. For components classified in terms of safety relevance an ageing surveillance programme has been required by the regulatory authority since 1991. Within this programme a catalogue of ageing mechanisms has been developed that addresses FAC among other material ageing phenomena. As outlined below, for non-classified (non-Code) equipment all the utilities have developed their own inspection and maintenance programmes to address potential wall thinning.

Identification of locations for wall thickness measurements is supported by predictive programs such as WATHEC/COMSY. The following systems are typically evaluated for susceptibility to FAC: Feedwater, Main Steam, drain lines (e.g., off heaters, reheaters, moisture separators), extraction lines, auxiliary steam lines, sealing steam lines. Depending on the plant specific experience more or less systems are included in the FAC analysis. The FAC mitigation is supported by the control of water chemistry. The following strategies are used by the Swiss NPPs.

4.12.1 NPP Leibstadt (BWR)

Since 1989 Leibstadt NPP has used the program WATHEC/COMSY by AREVA NP to assess the relevance of FAC systematically. For the last two decades the annual in-service-inspection programme has been determined by the EROSKO program. For this approach the measured wall thickness results are used for the annual improvement of the computer-aided analysis.

The EROSKO analysis is conducted in two steps. In a first step (“screening analysis”) the systems of the water-steam-cycle are analysed. For this analysis the EROSKO-program uses the following parameters: operating time, temperature, pressure, piping material, isometrics, water chemistry, etc. The simulation allows the exclusion of systems without the potential for FAC.

In a second step a detailed analysis of areas is performed, showing where loss of material could occur theoretically in the system. The detailed analysis produces annual lists containing information of critical areas of the systems which may be affected by FAC. The list includes the theoretical loss of material per year, a classification of the components of a system based on the hazard potential and a recommended date for an in-service-inspection to measure the wall thickness of the critical areas.

The results of the wall thickness measurements enable the "calibration" of the predicted susceptibility for components. The "calibrated" calculation of the remaining service life of each component allows individual inspection intervals to be scheduled. This interactive approach improves the year to year ability to predict (potential) susceptibility of piping components and particularly the effectiveness of the inspection programme for the next refuelling outage.

The components under surveillance by the EROSKO-program have not yet shown significant degradation. Therefore no refurbishment of these components has been necessary. Notwithstanding that, KKL did replace small bore piping mainly degraded by droplet impingement. Those components are out of scope of the EROSKO-program.

4.12.2 NPP Mühleberg (BWR)

Since 1998, periodic wall thickness measurements have been carried out on the piping of turbine generator sets A and B. In 2006 a detailed study was carried out for KKM to classify all relevant systems of the secondary coolant system and their piping parts systematically into risk groups.

The aim was to evaluate the stress level in the pipe system under operating conditions, as well as during an earthquake considering Operating Basic Earthquake (OBE) and Safe Shutdown Earthquake (SSE). The results show that the system during normal and bypass operation is at an acceptable stress level. The steam hammer load during turbine trip is also under control. OBE and SSE are in the permissible range and no supplementary supports are required in the valve areas. This result was possible only because the pipe systems modelled end at fixed points such as the turbine, condenser and moisture separator reheater (MSR).

The evaluation of the pipe wall thickness is necessary to determine the most critical locations. In order to have an overview of the stress behaviour in the pipe system during normal operation and bypass blow out, a pipe model for the steam lines from the penetration wall to the turbine/condenser was generated and a structural analysis was carried out. The values of the minimum wall thicknesses were calculated. The calculated stress levels indicate that the system is operating at a medium stress level.

Henceforth the study shall serve as a guideline for future periodic wall thickness measurements. Locations and times of measurements will be documented and relevant conditions analysed and documented.

4.12.3 NPP Gösgen (PWR)

Since 2005, NPP Gösgen has used the program EROSKO (which is part of COMSY) by AREVA NP to assess the relevance of FAC systematically in the complete secondary coolant system. The purpose of the program is to locate areas with a high hazard potential due to an elevated loss of material. The in-service-inspection programme is adjusted to the results of the programme. Previously, a Gösgen internal surveillance programme was implemented. NPP Gösgen has the same methodology as NPP Beznau, which also uses the EROSKO program to identify the areas for thickness measurements, see above. The results of the wall thickness measurements are used for the revision of the in-detail analysis. The revision of the analysis causes an adjustment of the component classification and also recommends dates for the next wall thickness measurements.

In the refuelling-period 2007, based on the EROSKO-Analysis, an elevated loss of material was detected in specific locations of the steam generator blowdown (SGBD) system. The wall-thickness never fell below the minimum tolerable wall-thickness. Nevertheless, due to this elevated loss of material the affected piping sections were replaced.

4.12.4 NPP Beznau (PWR, 2 Units)

For all the piping of both units in which flow accelerated corrosion, erosion-corrosion or droplet impingement cannot be ruled out a concept for a surveillance of these pipes regarding FAC has been created. Wear in other components such as tanks, heat exchangers, fittings or pumps are not covered in this concept. The surveillance of these components is carried out by periodical inspections and maintenance. Wear by erosion-corrosion induced by water drops on carbon steel pipes occurs most often in saturated and wet steam areas at NPP Beznau.

In 1989 the feed water and condensate main pipes were evaluated by Siemens using the WATHEC program regarding the susceptibility to FAC-induced wall thinning. The wall thickness of the most susceptible areas according to the calculations has since been tested periodically and there has not been any further wear recorded. AREVA has also tested the main steam elbow outside the containment in line A unit 1 with the new program COMSY. In their report they indicate an estimated wear of a maximum 0.04mm per year. This is assessed as "very small". Basically it can be said, that the expected wear, and therefore the risk of damage by FAC, is very small.

Periodic wall thickness measurements are carried out of the susceptible areas to find out if there has been any wear, and if so to what degree. The monitoring concept of the FAC programme at NPP Beznau is extensive and covers areas for which exists plant-specific and industry wide experience with FAC.

Further mitigation of FAC is achieved by changing the relevant factors. This is achieved through the water chemistry (pH-value and Oxygen-content), the steam moisture content at the exit of the steam generator (improved moisture separators in the new steam generators) and the replacement of carbon steel pipes with stainless steel pipes. The quantities of sludge are recorded to assess the efficiency of the measures taken with the water chemistry and to have a better overview of the wear in the secondary circuit. This applies to the filter of the waste water cleaning system, the sedimentation of the waste water tank and the removed sludge of the steam generator blow-down system. Since the replacement of the steam generators and the introduction of the "high-AVT-chemistry" the recorded quantities of sludge have been small when compared internationally.

All periodic inspections are governed by the ISI programmes at NPP Beznau. In addition to the ISI programme a supplementary programme is scheduled every year at a team meeting. The responsible supervisor for piping calls this meeting at the beginning of the year. All inspection results are filed internally. The results of repeated measurements are recorded in a comparison table. Any changes of one or more measuring points and the average per measurement plane are calculated in this table. These tables serve to visualize the test results and the changes over a period of time, and serve as a base for the following programme. The tables are updated annually after the outages.

4.13 USA

Generic Letter 89-08 [8] requested that all licensees implement a long-term FAC detection programme to prevent pipe failures in high-energy (single- and two-phase) carbon steel piping systems. The programmes are developed by each utility using plant specific conditions, industry-wide operating experience, engineering judgment, NDE techniques, and computer analysis of high energy carbon steel piping systems. As stated in the NRC Inspection Manual, Procedure 49001 [41], “the long term program must be well defined, with clearly documented results, and must include a complete analysis of the susceptible systems, inspection of the most susceptible piping components, repair or replacement of damaged piping components, trending of inspection data in order to determine FAC rates, and continued analysis based on inspection findings.”

Most if not all U.S. utilities utilize the CHECWORKS computer code for tracking and predicting wear. The EPRI Report NSAC-202L, "Recommendations for an Effective Flow Accelerated Corrosion Program" is used to select the most susceptible locations for inspection.

NUREG-1801, “Generic Aging Lessons Learned (GALL) Report" [40], is referenced as a technical basis document in NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants” (SRP-LR) [42]. The GALL Report identifies aging management programmes (AMP) that were determined to be acceptable to manage aging effects of systems, structures and components (SSC) in the scope of license renewal, as required by 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.” Section XI.M17, "Flow-Accelerated Corrosion" of NUREG-1801, contains guidance relative to an acceptable aging management programme. The GALL report indicates that an applicant's FAC programme can be based on the EPRI Guidelines in NSAC-202L. Furthermore, the GALL report indicates that an acceptable FAC program "includes the use of a predictive code, such as CHECWORKS,"

5. FAC EVENT POPLATION DATA

The FAC event population in the CODAP Event Database is summarized in this section. This event population includes non-through wall defects and through-wall defects. The former group represents NDE results where the measured wall thickness is below the minimum wall thickness allowed by the acceptance criteria as formulated in FAC programmes.

5.1 Susceptible Piping Systems

Available service experience data shows that piping components with complex geometries are frequently susceptible to FAC. Typical components which have been most susceptible include the following:

- Tees and branch connections
- Expanders[3] and reducers
- Long- and short-radius elbows
- Steam traps[4]
- Exit nozzles
- Orifices
- Valve bodies with flow changes
- Any significant inner surface discontinuity

Piping systems that are susceptible include safety related as well as balance-of-plant (BOP) systems. BOP piping systems are not included in the nuclear steam supply system (NSSS). That is, BOP piping is outside the Reactor Coolant Pressure Boundary, RCPB. Examples of carbon steel piping systems which are typically monitored for FAC in nuclear power plants include the following:

- Feedwater; in BWR plants it consists of piping outside containment, in PWR plants it consists of piping inside and outside containment.
- Condensate
- Feedwater heater drains and vents
- Moisture separator drains
- Moisture separator reheater drains
- Extraction steam
- Steam generator blowdown (in PHWR and PWR plants)

[3] A reducer with the flow from the small end to the large end.

[4] A steam trap is used to drain condensate from a steam line. A drain pocket is welded to the bottom of the pipe to be drained.

- Feeder lines that are integral part of the Primary Heat Transport System (PHTS) in Pressurized Heavy Water Reactors (PHWRs)[43][44].
- High Pressure Coolant Injection (HPCI) pump steam supply & drain (BWR)
- Main Steam; it consists of piping outside containment including turbine bypass piping
- Auxiliary Steam
- Auxiliary Feedwater
- Auxiliary Feedwater pump steam supply
- Cross around (large-diameter wet steam piping between HP turbine and moisture separator reheater and (relatively) dry steam between moisture separator reheater and LP turbine).

5.2 Reporting of FAC Events

The FAC operating experience data is recorded in different types of information systems such as Condition Reports, Work Orders, and databases for FAC inspection results. Reportable occurrence reports or licensee event reports capture significant events. In addition, information notices (or equivalent documents) are issued by regulators to inform licensees about generic issues. Members of the CHECWORKS® Users Group (CHUG) exchange operating experience data on a regular basis (twice annually). In the U.S., the Institute of Nuclear Power Operators (INPO) operates the Equipment Performance and Information Exchange System (EPIX) which is used by plant operators to exchange information on FAC related issues.

Data records on wall thinning are obtained from non-destructive examinations (NDEs). For safety related piping included in ASME Section XI In-Service Inspection Program, items with flaws and repair/replacement activities are recorded in the "Owners Activity Reports" (Form OAR-1, or equivalent) and submitted to a regulatory agency.

5.3 High-Level FAC Data Summary

The total FAC event population in the CODAP Event Database consists of ca. 1,990 records involving pipe wall thinning below minimum allowable wall thickness, through-wall leaks and ruptures for the period 1970-2012; Table 3. The data records on "wall thinning" have resulted in corrective actions, including temporary repairs[5] (e.g., weld build-up, welded patch, and engineered clamp), in-kind pipe replacement or replacement with FAC-resistant material. Of the total event population, ca. 80 % represents U.S. operating experience. The non-US data is limited to selected representative events. Hence, the FAC event population is inhomogeneous, reflecting different raw data screening criteria.

[5] The term "temporary repair" (or non-Code repair) implies that a permanent repair is made within some pre-defined period. Different regulatory positions apply to temporary repairs of pipe flaws. As an example, the NRC position (c.f. Generic Letter 90-05) is that such repairs are applicable until the next scheduled outage exceeding 30 days, but no later than the next scheduled refueling outage. For safety-related piping a non-Code repair relief request must be submitted for review and approval.

Table 3: FAC Event Population

Region	Plant Type	Failure Mode	CODAP Event Database FAC Event Records					
			Total No. of Records	1970-79	1980-89	1990-99	2000-09	2010-12
North America	BWR	Wall Thinning	138	3	72	17	44	2
		Through-Wall Leak	183	30	43	45	52	12
		Rupture	10	1	5	2	2	0
	PHWR	Wall Thinning	57	0	24	5	27	1
		Through-Wall Leak	11	0	2	6	2	1
		Rupture	1	1	0	0	0	0
	PWR	Wall Thinning	1060	0	783	78	192	7
		Through-Wall Leak	160	10	52	49	39	10
		Rupture	35	0	19	12	4	0
Asia	BWR	Wall Thinning	18	0	0	7	10	1
		Through-Wall Leak	11	1	2	2	6	0
		Rupture	1	0	0	1	0	0
	PHWR	Wall Thinning	6	0	0	0	6	0
		Through-Wall Leak	3	0	0	0	3	0
		Rupture	1	0	0	0	1	0
	PWR	Wall Thinning	32	0	0	15	15	2
		Through-Wall Leak	7	0	3	1	3	0
		Rupture	2	0	0	1	1	0

Region	Plant Type	Failure Mode	CODAP Event Database FAC Event Records					
			Total No. of Records	1970-79	1980-89	1990-99	2000-09	2010-12
Europe	BWR	Wall Thinning	62	2	10	38	11	1
		Through-Wall Leak	48	1	22	7	17	1
		Rupture	3	1	2	0	0	0
	PWR	Wall Thinning	30	0	8	7	14	1
		Through-Wall Leak	86	0	24	42	19	1
		Rupture	22	2	2	10	8	0
		Totals:	1987	52	1073	345	476	40

In Figure 2, the total event population is organized according to mode of degradation/failure and the affected plant system. The largest contributors to this event population are FAC in Extraction Steam piping (two-phase flow) and FAC in Feedwater piping (single-phase flow). The PHTS event population consists of small bore (≤ DN100) reactor outlet feed pipes of cold drawn carbon steel. This event population is unique to PHWR plants.

In Figure 3 the total FAC event population is organized by pipe size ("small-bore" vs. "large-bore" piping) and failure mode (wall thinning, through-wall leak and rupture). FAC programmes typically define "small-bore" as piping of nominal size 100 mm (DN100) or less. In Figure 4, the total FAC event population is organized by safety class (ASME III Code Class). The Code Class 1 event population consists of wall thinning and minor through-wall defects in PHWR feeder outlet piping. The Code Class 2 event population consists of wall thinning in PWR Feedwater piping inside containment.

In Figure 5 the FAC event population for the period 1990-2012 is organized by plant type and calendar year. Figures 6 and 7 summarize the FAC event population data for the period 2000-2012. In Figure 6 the event population data is organized by time period and failure mode, whereas in Figure 7 the event population data is organized by time period, plant type and failure mode. Finally, in Figure 8 the event population is organized by component age at the time of failure.

Figure 2.: FAC Event Records by Plant System

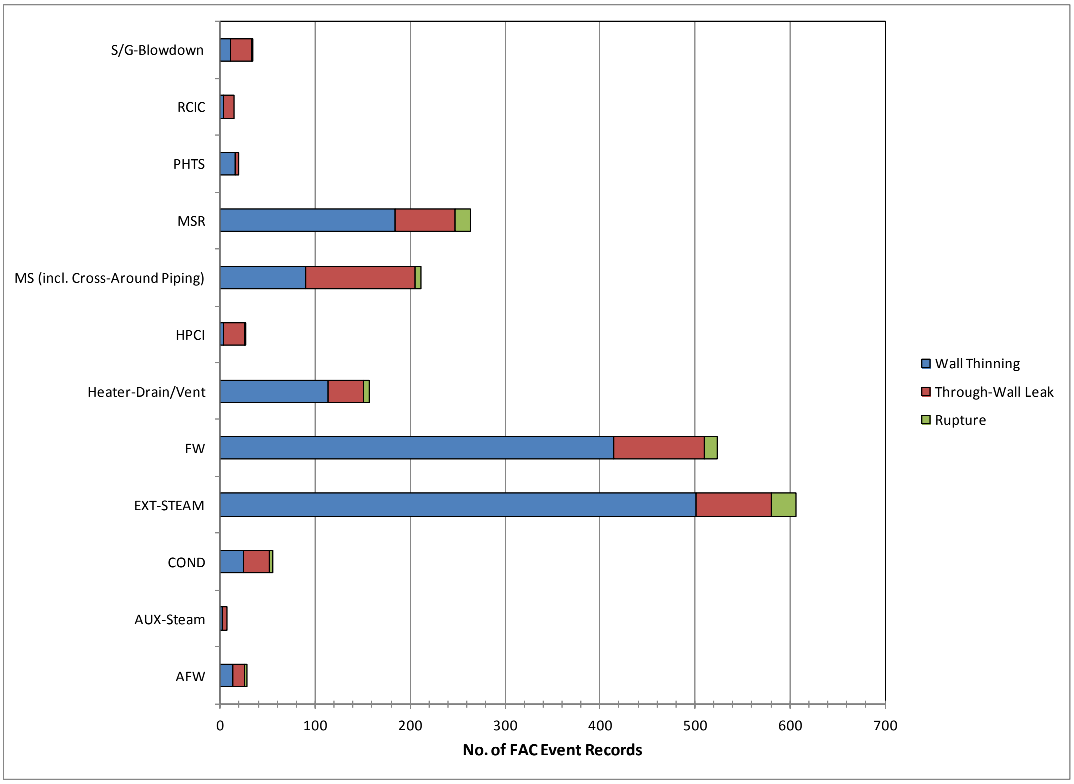
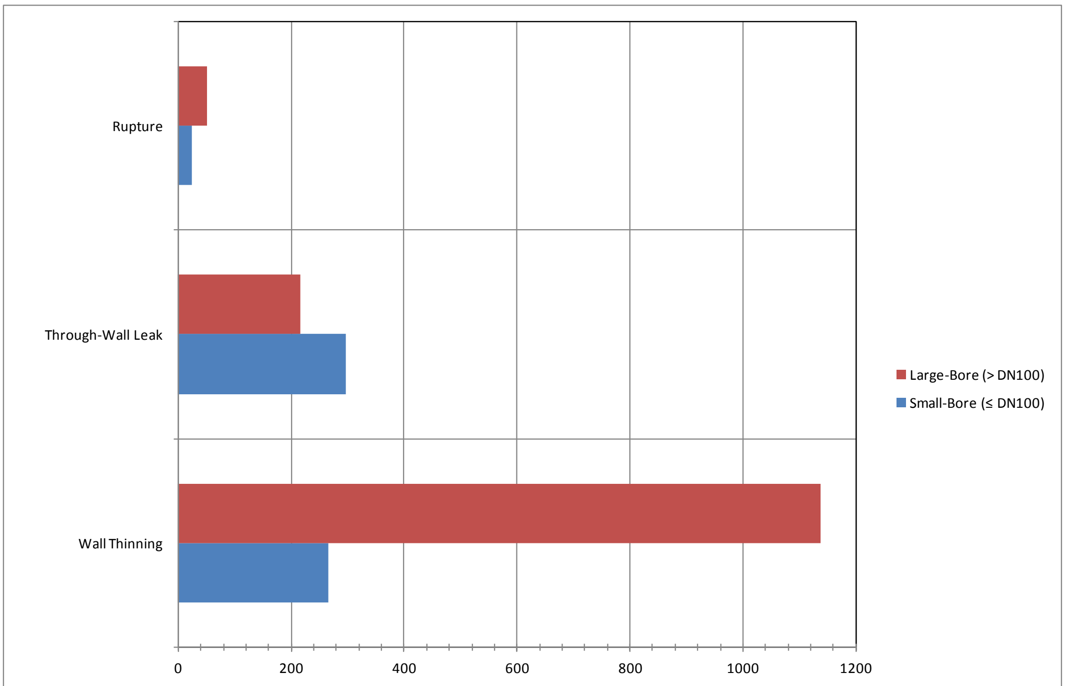
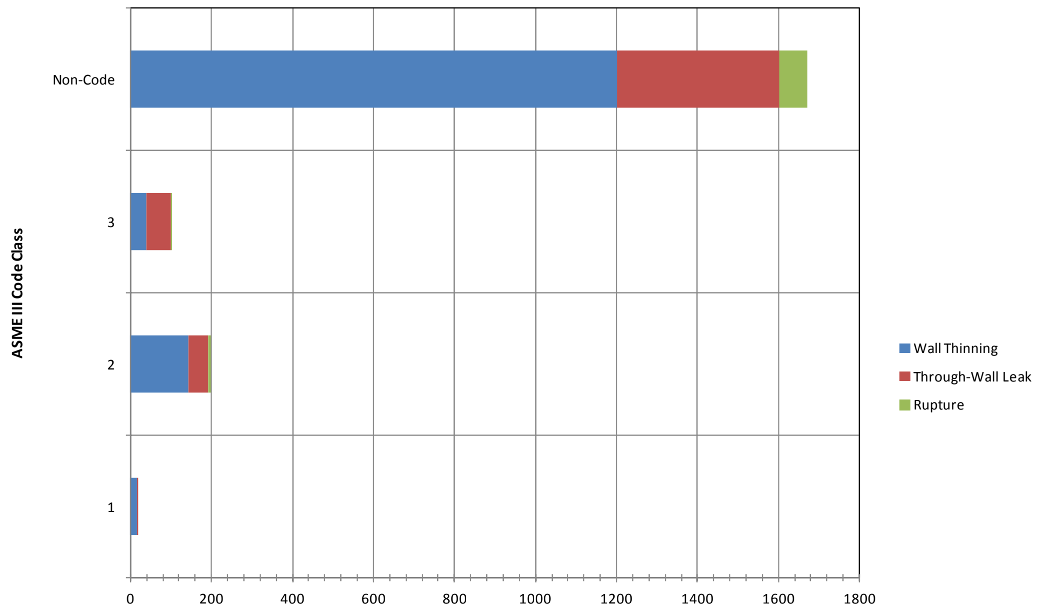


Figure 3: FAC Event Population by Pipe Size



	Wall Thinning	Through-Wall Leak	Rupture
Large-Bore (> DN100)	1138	217	50
Small-Bore (≤ DN100)	265	296	25

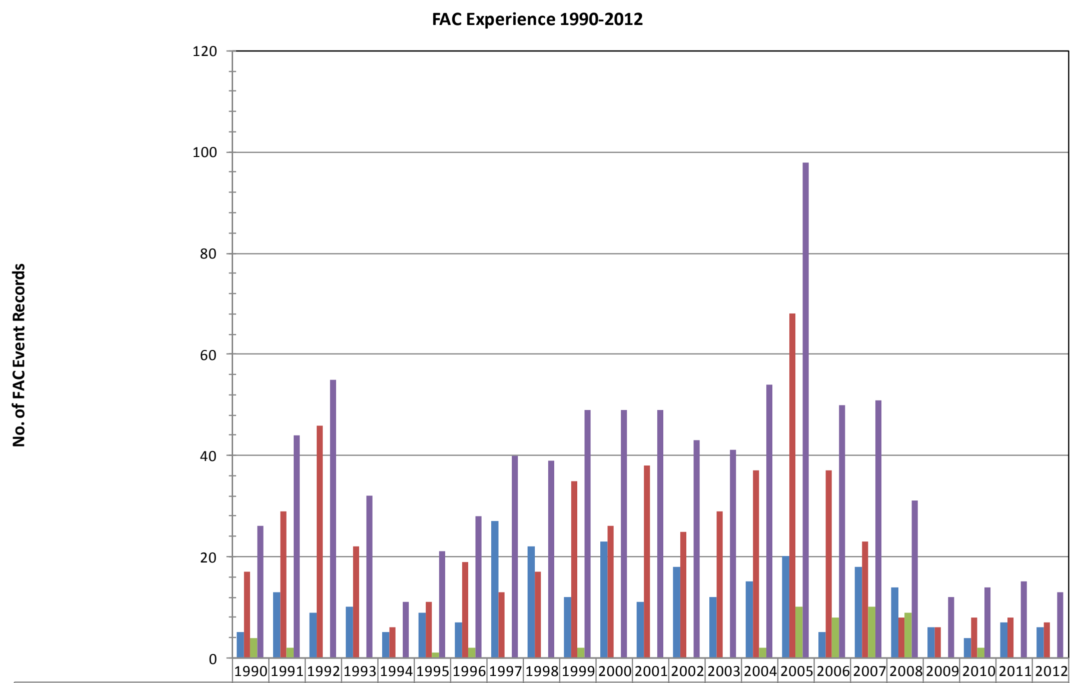
Figure 4: FAC Event Population by ASME III Code Class



	1	2	3	Non-Code
Wall Thinning	17	144	40	1202
Through-Wall Leak	3	49	61	400
Rupture		4	2	69

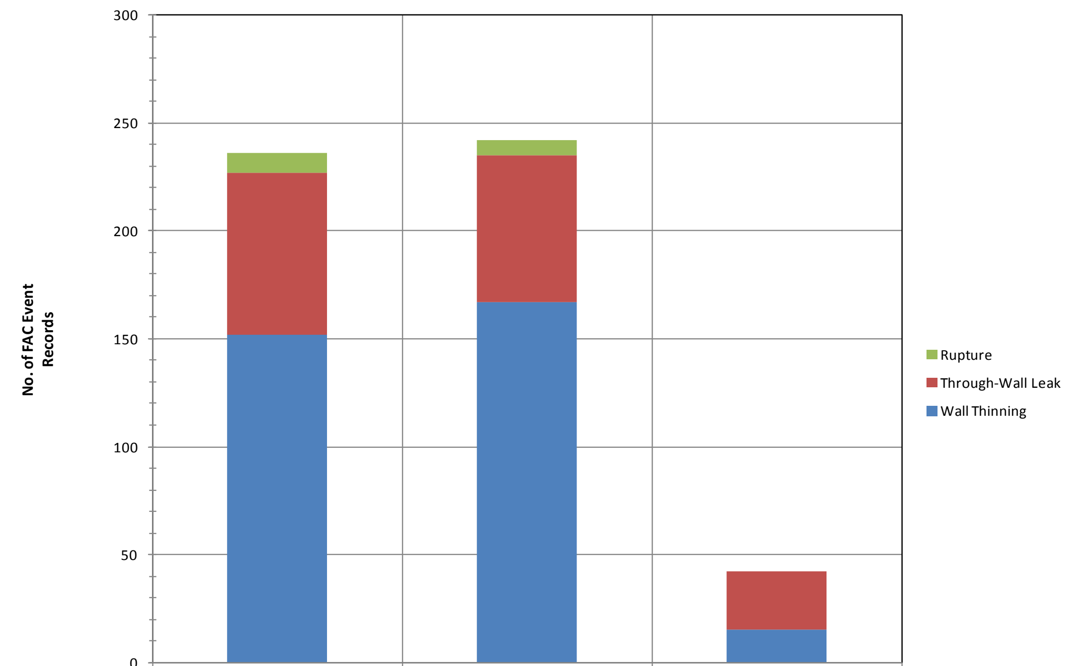
No. of FAC Event Records

Figure 5: FAC Event Records by Plant Type & Calendar Year



	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012
BWR (278 Records)	5	13	9	10	5	9	7	27	22	12	23	11	18	12	15	20	5	18	14	6	4	7	6
PWR (535 Records)	17	29	46	22	6	11	19	13	17	35	26	38	25	29	37	68	37	23	8	6	8	8	7
PHWR/CANDU (52 Records)	4	2				1	2			2					2	10	8	10	9		2		
All Plant Types (865) Records)	26	44	55	32	11	21	28	40	39	49	49	49	43	41	54	98	50	51	31	12	14	15	13

Figure 6: FAC Event Population (i) for 2000-2012



	2000-04	2005-09	2010-12
Rupture	9	7	
Through-Wall Leak	75	68	27
Wall Thinning	152	167	15

Figure 7: FAC Event Population (ii) for 2000-2012

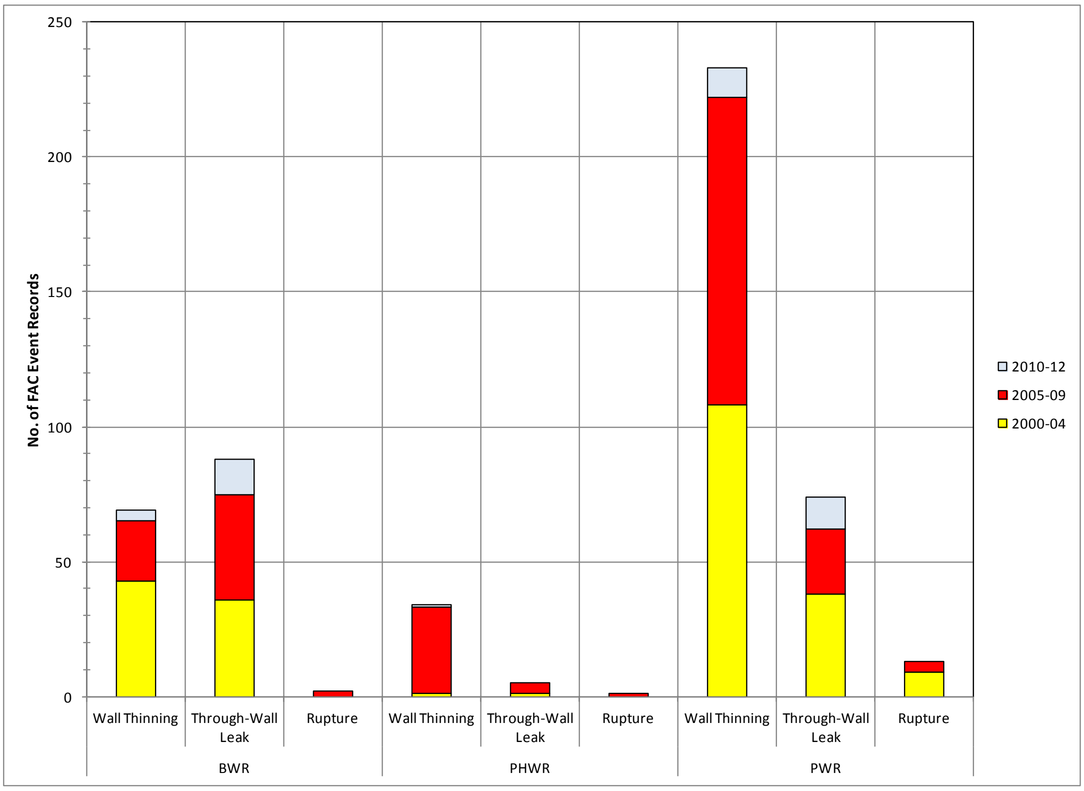
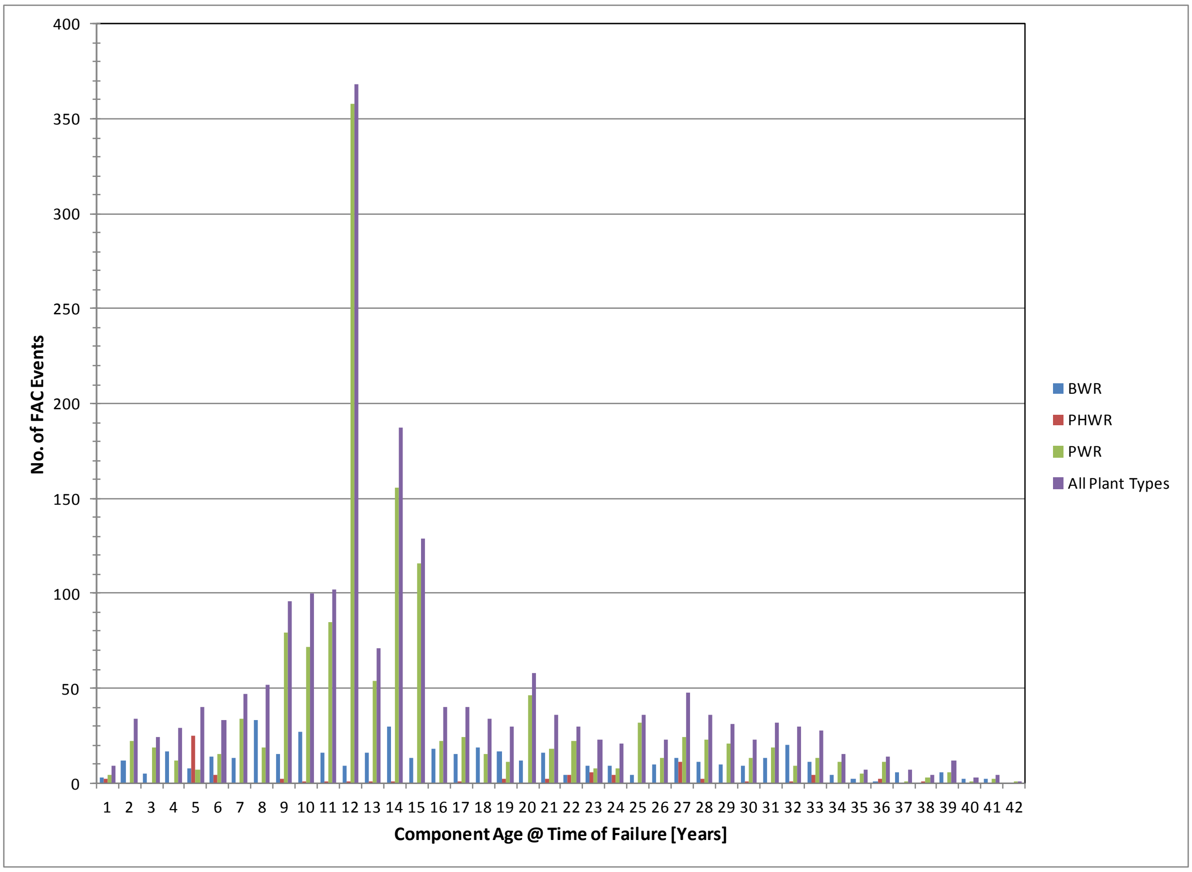


Figure 8: FAC Events as a Function of Component Age at the Time of Failure



In reviewing the FAC event population, the difference between the BWR- and PWR-specific service experience is noteworthy. As described in Section 3, this difference is attributed to chemical effects

5.4 Completeness & Comprehensiveness of CODAP

The CODAP Event Database is a relational database consisting of approximately 100 uniquely defined database fields that capture various piping reliability attributes and influencing factors. CODAP supports a wide range of applications, from high-level data analyses to identify global failure trends and patterns, to advanced applications in support of structural reliability modelling and PSA. The completeness and comprehensiveness of a database like CODAP are important factors in determining the 'fitness-for-use.' Completeness and comprehensiveness of a service experience database should be ensured through a sustained and systematic maintenance and updating process. Completeness is an indication of whether or not all the data necessary to meet current and future analysis demands are available within a database. The comprehensiveness of an event database is concerned with how well its structure and content correctly capture key piping reliability attributes and influence factors.

The data records on wall thinning due to FAC are obtained from non-destructive examinations. According to Figure 3, for the small-bore (or small-diameter) piping, the population of wall thinning data is less than that for through-wall leaks; i.e., 265:296 (ratio of non-through-wall to through-wall data records) vs. 1238:217 for large-bore piping. This observation points to potential data incompleteness that is explained by the evolving nature of the FAC programmes. In general, pre-1995 there has not been a systematic effort to track data on high-wear locations for small-diameter piping. Often, the FAC-inspections for this category of piping have been based on engineering judgment and inspection-for-cause whenever a through-wall leak has occurred. The event population for the two categories of piping reflect different inspection practices and procedures, and an evolving FAC management strategy.

For the small-bore piping, the event population in Figure 3 may not reflect the actual FAC-susceptibility at any given piping location. In general, a point estimate of the frequency of pipe failure (where "failure" includes both small and large through-wall leaks but not wall thinning ($t_{Meas} < t_{Min}$)), λ, is given by the following expression:

$$\lambda = \frac{n_F}{NT} \tag{4}$$

Where:

n_F = Number of failure events including both small and large leaks in the service data;

T = Total time over which FAC event data was collected;

N = Number of components that provided the observed FAC event data.

A point estimate of the total frequency of flaws, ϕ, is given by the following expression:

$$\phi = \frac{n_C}{N \cdot T \cdot f \cdot POD} + \frac{n_F}{NT} = \frac{n_C}{N \cdot T \cdot f \cdot POD} + \lambda \tag{5}$$

Where:

n_C = Number of non-through-wall flaw events

f = Fraction of FAC-susceptible locations inspected for pipe wall thinning

POD = Probability of detecting a flaw

Equation (5) accounts for the observed flaws as recorded in the database and the fact that only a fraction of the small-bore FAC-susceptible piping is inspected. The equation also reflects the fact that each failure in the database due to FAC has an additional flaw that eventually grew to produce the failure, the exposure parameter of which is the entire small-bore FAC-susceptible piping population at risk for failure. This is based on the insight that nearly all failures (i.e., through-wall leaks) are found not through NDE but fortuitously during routine leak inspections. This is an important observation because the entire piping population in the surveyed data is at risk for failure observation, but only a small fraction is at risk for the observation of non-through flaws which can only be found from NDE inspections. The ratio ϕ / λ is the factor by which to multiply the pipe failure rate to obtain the flaw (non-through wall crack) rate:

$$R_{C/F} = \frac{\phi}{\lambda} = \frac{n_C}{n_F \cdot f \cdot POD} + 1 \tag{6}$$

Where:

$R_{C/F}$ = Number of non-through wall flaws per leak event:

Equation (4) together with an assumption about the inspection scope (*f*) makes it possible to estimate $R_{C/F}$. Estimates of $R_{C/F}$ for the data set in Figure 3 are presented in Table 4 for different assumptions about the fraction of small-bore FAC-susceptible pipe locations that are inspected. An upper bound for *f* has been set to 25 % of small-bore FAC-susceptible pipe locations.

Table 4: Estimates of $R_{C/F}$ for Small-Bore FAC-Susceptible Piping

Number of Flaws (Wall Thinning)	**Number of Leaks**	***f* (Assumed Fraction of FAC-Susceptible Components Inspected)**	**POD (Probability of Detection)**	$R_{C\|F}$
265	296	0.05	0.50	36.8
		0.10	0.50	18.9
		0.25	0.50	8.2
		0.05	0.75	24.9
		0.10	0.75	12.9
		0.25	0.75	5.8
		0.05	0.90	20.9
		0.10	0.90	10.9
		0.25	0.90	5.0

The estimate for the ratio of cracks to leaks obtained in Table 3 reflects the degree to which certain small-bore piping is exposed to FAC. The evidence for the observed non-through-wall flaw frequency is based on an assumed exposed population that is only about 5 to 25 % of the exposed component population for failures as only the inspected pipe locations are available to produce this evidence. This fact combined with the additional implicit flaw that must have existed prior to each of the leak events, creates an underlying failure rate for non-through-wall flaws that is at least 5 times higher than the underlying failure rate for leaks. This is true even though the observed number of flaws is actually less than the observed number of failure events.

5.5 Effectiveness of Programs for FAC Wear Rate Prediction

As stated in NSAC-202 [22]. "…Accurate inspections are the foundation of an effective FAC program. Wall thickness measurements will establish the extent of wear in a given component, provide data to help evaluate FAC trends, and provide data to refine the predictive model. Thorough inspections are key to fulfilling these needs ….." The CODAP event database includes information on ISI histories and ISI programmatic errors or weaknesses applicable to specific events. The objective of any FAC program is to predict wear rates and prevent through-wall flaws or BBL-events from occurring. Despite the dedicated efforts to comprehensively manage FAC, pipe failures do occur due to less-than-adequate implementation of inspection plans. Included in the database are numerous examples of programmatic deficiencies. Listed below are examples of recent FAC events that are attributed programmatic deficiencies:

- <u>Through-Wall Leak Develops Prior to Next Scheduled Inspection</u>. This could be indicative of higher-than predicted wear rates. It also entails instances where pipe routing has been modified without corresponding change to FAC monitoring program. An example of the latter type of deficiency is the failed extraction steam drain line in a Japanese BWR plant in 2010. The turbine rotors had been replaced in 2009 to achieve higher thermal efficiency. The design change involved rerouting the drain piping, which in turn resulted in an increased drain flow and higher-than-anticipated wear rate.

- Failure Occurs in a Location Identified by Predictive Program as Susceptible. However, this information may not have been transferred to the inspection plan. An example of this type of deficiency is the ruptured first stage extraction steam line at a U.S. PWR plant. The event occurred during the first quarter of 2009. The failed line was listed in the FAC program documentation as susceptible, but was not monitored at the failed location.

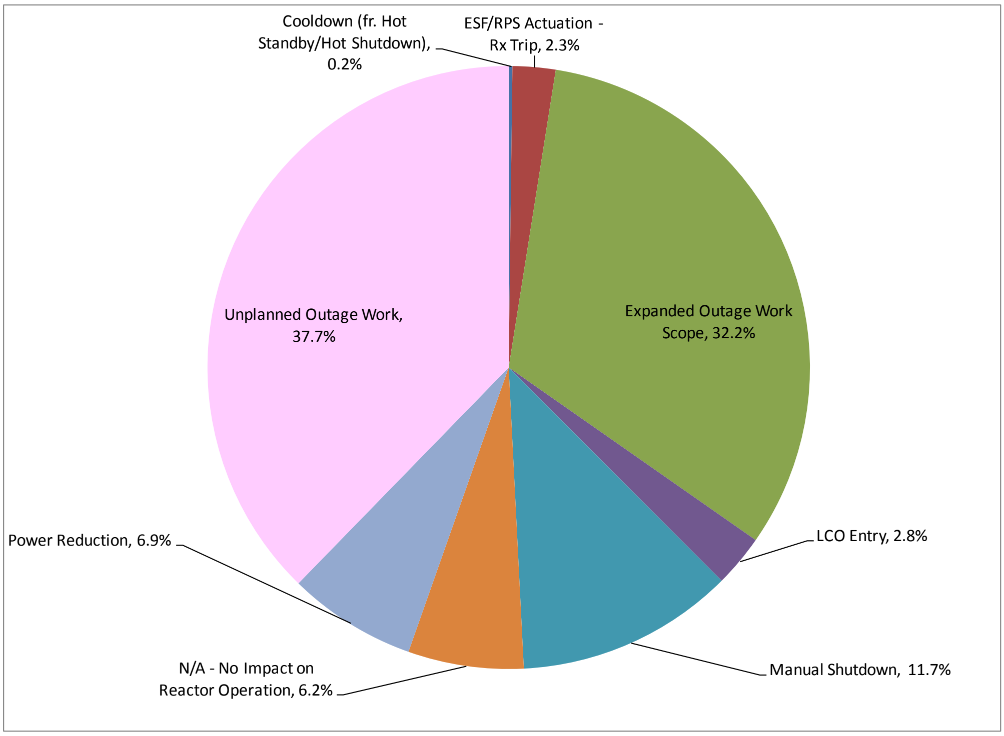
- Failure Occurs Due to FAC Software Model Input Errors. During the 3rd quarter 2006, a DN150 Moisture Separator Reheater drain line failed at a U.S PWR plant. The FAC monitoring program included inaccurate wear prediction due to a computer software modelling error that was due to "*failure of the organization to establish a proper level of verification*." It was also determined that the preparers, reviewers and approvers of the FAC procedure did not recognize that a formal second level of verification was needed to ensure a quality software model to ensure the safety of plant personnel and plant power generation reliability.

- Less-Than-Adequate FAC Program Implementation. In 1999, a U.S. PWR plant experienced a rupture of a DN150 MSR drain line. The failure occurred in a straight section of pipe immediately downstream a 45-degree elbow. A causal factor associated with this event is the manner in which the FAC inspection program provided guidance for inspecting downstream piping when performing fitting inspections. The program did not consider downstream piping inspections as mandatory, although it is a common practice. However, this downstream piping was not inspected when the upstream 45 degree elbow was inspected during previous refueling outages. Had this piping been previously inspected, it is probable an abnormal wear rate would have been detected. A second causal factor associated with this event was the perception that the upstream 45 degree elbow would be the most susceptible component for failure due to FAC. This causal factor contributed to the decisions made during previous refueling outages to not inspect the downstream piping associated with this elbow.

5.6 Operability & Safety Impacts of FAC Events

A high-energy line break can cause serious flooding and excessive steam release to the extent that safety related electrical and mechanical equipment functions are impaired; see for example Appendix A. Specific examples include failures of auxiliary feedwater pumps, emergency diesel generators, low- and high-voltage electrical safeguard buses, and safe shutdown panels. The vulnerability to these types of impacts varies significantly from plant-to-plant. Based on CODAP, Figure 9 is a high-level summary of the operational impact of FAC events. About 30% of the recorded events have occurred during routine power operation. Almost 40% of recorded events have resulted in unplanned outage work.

Figure 9: FAC Event Impact on Plant Operation



As stated in Section 1.3, FAC-induced major pipe failures tend to be sudden and energetic causing collateral damage. Also, FAC poses an occupational safety hazard. Furthermore, the current PSA practice includes detailed consideration of pipe failure due to FAC particularly in the context of internal flooding. As examples, the EPRI "Guidelines for Performing Internal Flooding Probabilistic Risk Assessment" [45] addresses high-energy line break (HELB) scenarios, which can produce flooding as well as other unique operability challenges to plant equipment. Similarly, the ASME/ANS "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" [46] states that the identification of potential flood sources shall include consideration of pressure boundary breaches in feedwater, condensate and steam systems. An event data collection such as CODAP supports such evaluations. The operating experience embedded in the CODAP database identifies BOP locations that are susceptible to FAC, and it supports the estimation of PSA model input parameters (see also Section 6).

6. FAC EVENT DATA ANALYSIS[6]

Included in this section is an example of FAC event data analysis. In CODAP, all database applications begin by querying the event database for specific information. It is the requirements of an application that provide for database query specifications [47]. A query is a request for certain event populations, for action on data, to perform calculations, to combine data from different tables, to add, change or delete data, or to create a new table that addresses a specific degradation mechanism such as FAC. Queries that are used to retrieve data from a table or to make calculations are referred to as "select queries." Queries that add, change, or delete data are called "action queries." Database queries facilitate the filtering and processing of event data according to user-defined application requirements.

6.1 Practical Data Analysis Guidelines

The ability of an event database like CODAP to support practical applications is closely linked to its completeness and comprehensiveness. Equally important is the knowledge and experience of an analyst in interpreting and applying a database given a certain analysis specification. In principle, CODAP supports three general categories of applications: 1) high-level, 2) risk-informed, and 3) advanced database applications. Examples of risk-informed applications include:

- Loss-of-coolant accident (LOCA) initiating event frequency estimation.

- Internal flooding PSA; e.g., derivation of internal flooding initiating event frequencies.

- HELB Analysis. Consideration of HELB in PSA includes estimation of Main Steam and Feedwater line break initiating event frequency. As stated in Section 5.5, HELB is also considered in internal flooding PSA.

- Significance determination process and accident precursor analysis to determine the risk-significance of pipe degradation or failure.

- Risk-informed in-service inspection (RI-ISI).

The quality of a data analysis task is a function of the analyst's knowledge and experience and how a parameter estimation task is structured to adequately address a specific application requirement. Guidelines and best practices for piping reliability analysis include the following elements:

- Knowledge: A fundamental basis for a qualified piping reliability analysis task rests on a deep understanding of how, the typically robust metallic piping systems degrade and fail or sustain damage to various off-normal operating environments. Also of importance is a deep understanding of piping system design principles, including the different piping construction/fabrication practices.

- Parameter Estimation Specifications: A chosen parameter estimation scheme must address the requirements of an application. For an example, an application may involve estimating a pipe rupture frequency as a function of equivalent break size or through-wall flow rate. Risk-informed applications often include consideration of different degradation mitigation strategies, in-service-inspection strategies, and leak inspection strategies. Hence, a large set

[6] Disclaimer: The CODAP PRG fully endorses the goals and objectives of the OECD/NEA database projects. Decisions about CODAP applications, including methodology and analysis techniques, are taken at the national level. Using open literature sources, the purpose of Section 6 is to provide an example of how CODAP can support the assessment of FAC-induced flooding scenarios. The analysis methodology used in this example is but one of several different technical approaches for calculating internal flooding initiating event frequencies.

of piping reliability parameter distributions may have to be generated to respond to the demands of a given application.

- Service Experience Data: Under what conditions can service experience data support quantitative piping reliability analysis? The completeness and comprehensiveness of a database are essential characteristics for it to support the derivation of "robust" reliability parameter estimates.

- Qualitative Analysis Requirements: Database query functions are defined to extract event populations and exposure term data from a comprehensive relational database. Often, a query definition must address a complex set of reliability attributes and influence factors. Furthermore, an iterative set of data processing steps may be required to obtain the input to a quantitative analysis.

- Quantitative Analysis Requirements: Pipe failure rate calculation is based on event populations that reflect different piping designs. Therefore, an established practice is to apply Monte Carlo posterior weighting technique [48] to synthesize the variability in piping element (e.g., weld, susceptible area) counts and DM susceptibility. Pipe rupture frequencies are calculated for well-defined break sizes and resulting through-wall flow rates. Conditional rupture probability (CRP) models [49][50] are required for a pre-defined set of break size ranges according to parameter estimation specifications.

6.2 FAC Data Exploration

Before pursuing statistical parameter estimation it is essential to identify the different influencing factors that act on metallic piping components. In this section the U.S. service experience with FAC is explored as a precursor to estimation of HELB frequency. With CODAP, the following data processing steps are recommended:

- As a first step, CODAP is downloaded to a local computer and converted to a Microsoft® Access relational database. Next, develop a US-centred, FAC-specific sub-table by screening out all other database records. That is, non-US service experience data is screened out.

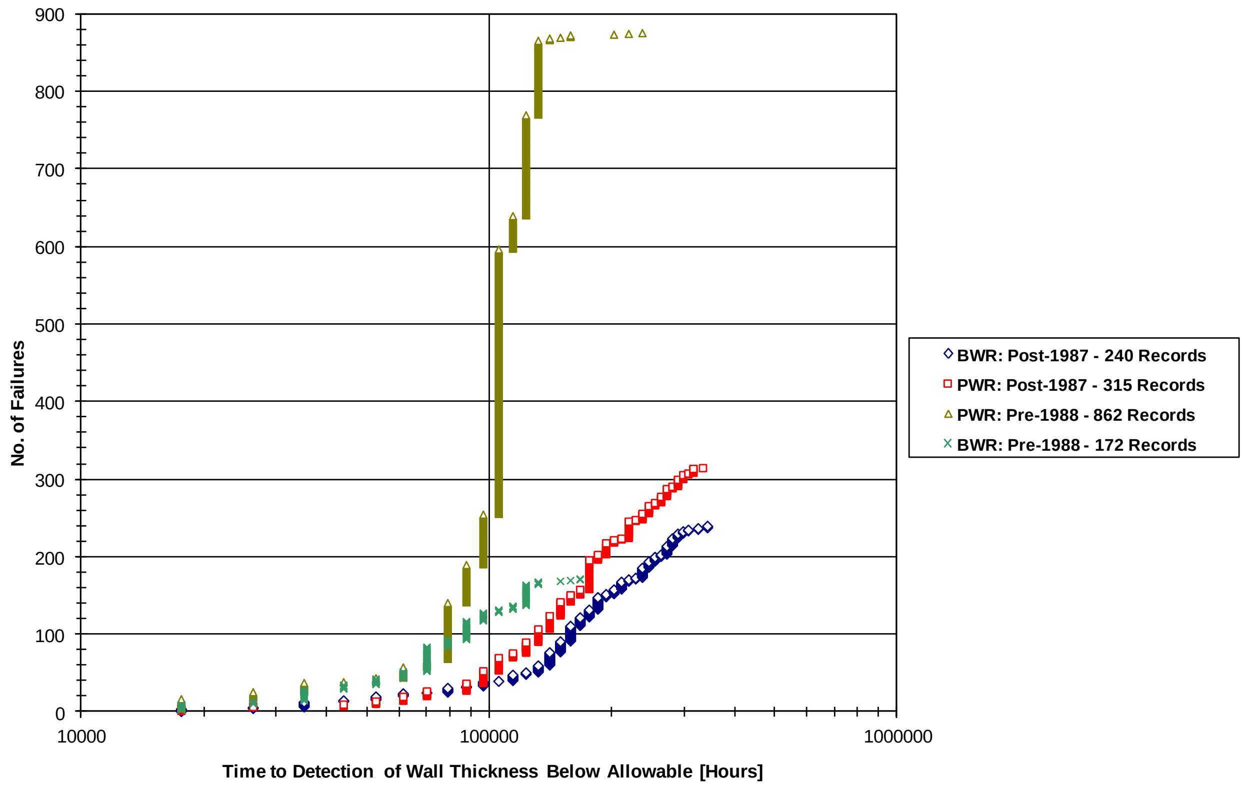
- Open the new FAC table and apply a data filter so that only the BWR-specific records are displayed. Again, apply a data filter to the 'Age' column and sort data in ascending order. Copy the two columns and paste onto a Microsoft® Excel worksheet. Add a 'rank order' column to the worksheet.

- Repeat the previous step for PWR plants.

- In the Excel worksheet, split the four columns (BWR/PWR and Rank/Age) in such a way that two pairs of columns are generated; one pair of columns includes failure date for period 1970 through 1987 and another pair includes failure data for period 1988 through 2012. Re-rank columns in ascending order. Next, for each record calculate the time to failure in hours. In this example, failure is interpreted to mean any degraded condition, including instances where t_{Min} (minimum pipe wall thickness) has been exceeded ($t_{Meas} > t_{Min}$) and pre-emptive pipe replacement is performed. This split of event data into two time period categories is warranted by the U.S. industry response to the resolution of FAC concerns. In July 1987 the U.S. NRC issued Bulletin 87-01 requesting all licensees to provide information to the NRC on their FAC experience and monitoring programmes for single-phase and two-phase high-energy carbon steel piping systems. In response to Bulletin 87-01, all licensees performed base-line inspections of all susceptible piping systems. The results of these inspections were reported to the NRC, and evaluations of inspection records were performed by EPRI. Based on the U.S. FAC experience, two event populations have been developed using the above data filtering process. One data pool represents FAC experience prior to implementing formalized FAC programmes (inspection requirements and enhanced secondary side water chemistry requirements). Another data pool is intended to depict the effect of improved FAC management on the piping performance. NUREG-1344 [4] is a good summary of the history of U.S. FAC experience.

- In Excel, prepare a chart showing the cumulative number of failures versus the time to detection of pipe wall thickness below T_{Min}. Figure 10 displays the total U.S. FAC event population as recorded in CODAP. The plots of FAC events are consistent with the FAC knowledge base. As an example, the pre-1988 BWR- and PWR-specific service experience is fully consistent with the predicted FAC susceptibility reflecting the differences in oxygen content and pH control. The post-1987 service experience shows the effects of improved FAC programme management and water chemistry control. From a statistical perspective a good estimation strategy would be to establish an empirical Bayes prior failure rate distribution using available service data and to update this prior using the post-1987 service experience data.

Figure 10: US FAC Experience



6.3 FAC-Specific Piping Reliability Parameter Estimation

The example in this section is concerned with the estimation of Extraction Steam piping rupture frequencies for application to a PWR HELB analysis as part of an internal flooding PSA study. To support the baseline calculations and some sensitivity calculations that were selected to develop risk management insights, a set of 6 analysis cases were devised as shown in Table 5. The variables used to define these cases include the break size, and data screening assumptions.

Table 5: Calculation Cases

Case	System	Pipe Size	Data Screening
1	Extraction Steam	≥ DN50	Post-1988 data only – “good” FAC inspection programme
2	Extraction Steam	> DN150	Post-1988 data only – “good” FAC inspection programme
3	Extraction Steam	≥ DN50	Data up to 1988 only – limited FAC inspection programme
4	Extraction Steam	> DN150	Data up to 1988 only – limited FAC inspection programme
5	Extraction Steam	≥ DN50 inch	FAC events removed to simulate the case of FAC mitigation by replacing carbon steel with stainless steel piping
6	Extraction Steam	> DN150	FAC events removed to simulate the case of FAC mitigation by replacing carbon steel with stainless steel piping

A failure rate and a rupture frequency had to be developed for each case and, hence, a total of 12 parameter distributions were developed. The dominant degradation mechanism in Extraction Steam piping is flow accelerated corrosion (FAC). Based on analysis specifications, rupture frequencies were developed for two rupture size cases: Ruptures with equivalent break sizes (EBS) between 50 and 150 mm diameter (EBS1), and ruptures with EBS > 150 mm in diameter (EBS2). The estimation of the rupture frequencies for each of these break size cases required the estimation of two parameters: 1) a failure rate, and 2) a conditional rupture probability that the break would be in the specified size range. The failure rate for each break size range is different because only pipes with a pipe diameter of at least 150 mm can produce a break size greater than 150 mm, whereas pipes as small as 50 mm in diameter can produce break sizes of 50 mm and greater. To support the estimation of these parameters, separate queries of the pipe failure database had to be made for pipe failures (cracks, leaks, wall-thinning, and ruptures) and ruptures in the prescribed break size ranges, and these queries had to be matched up against the appropriate estimate of the pipe component population exposure terms. A data-driven model of pipe rupture frequency is given by:

$$\rho_{i/x} = \frac{EventPopulation}{ExposureTerm} \times CRP \qquad (7)$$

Where

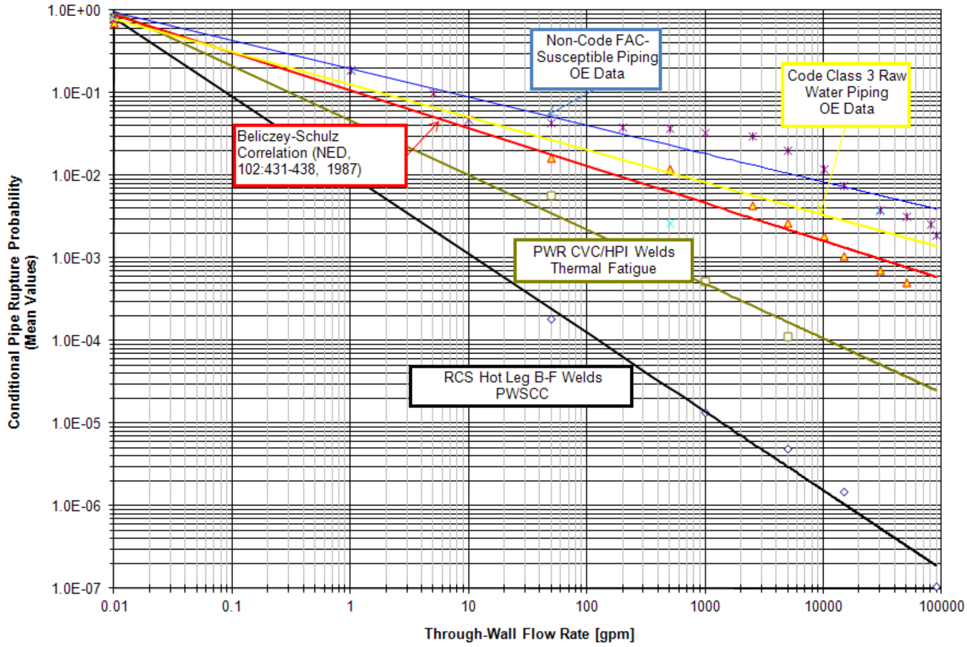
$\rho_{i|x}$ = Rupture frequency for pipe component i for rupture mode x

CRP = Conditional Rupture Probability

Based on success criteria as documented in the analysis specifications, for each set of failure rates, two rupture modes had to be distinguished: those with equivalent break sizes between 50 mm and 150", and those with break sizes in excess of 150 mm. Depending on the location of the pipe break either or both of these rupture modes may contribute to a specific HELB-initiated internal flooding initiating event. Separate conditional rupture probability models had to be developed to distinguish these cases.

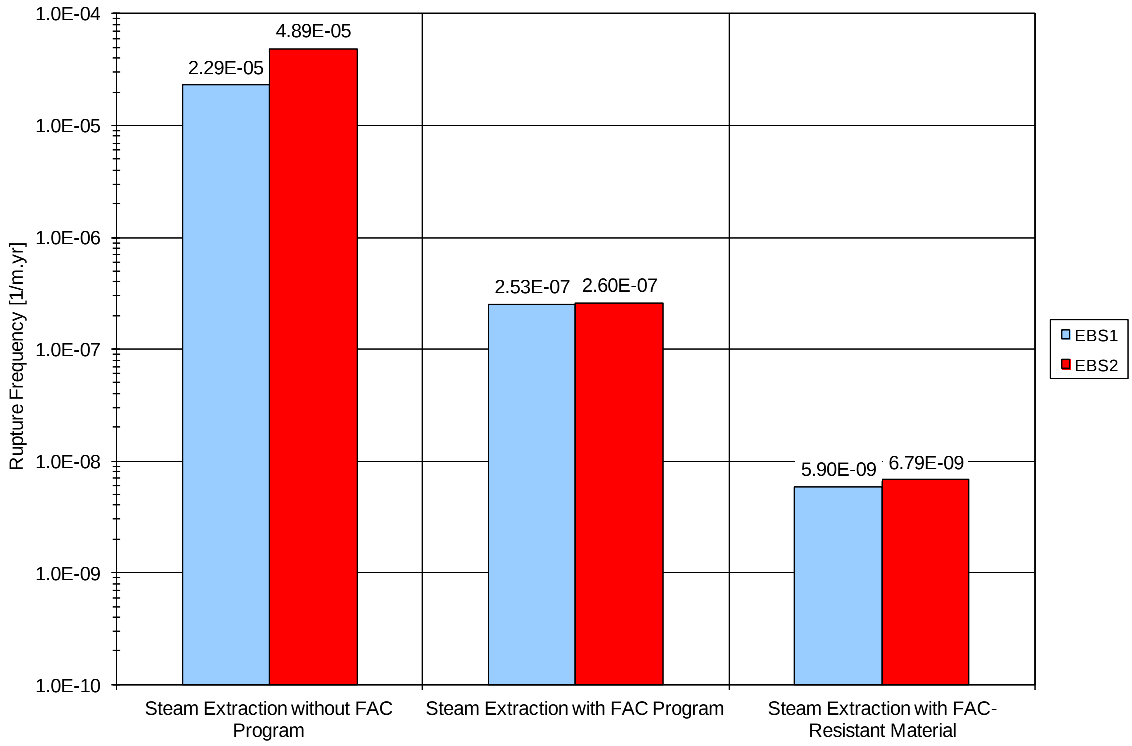
An event database like CODAP provides a valuable basis for developing conditional rupture probability models. Included in Figure 11 are examples of conditional rupture probabilities derived from operating experience (OE) data and theoretical studies. Results for the six calculation cases are summarized in Figure 12.[7] [49][50].

Figure 11: Conditional Rupture Probability According to Empirical & Theoretical Studies



[7] The technical details are available on the Internet at http://adams.nrc.gov/wba/; Accession Number ML053180483.

Figure 12: Results of Extraction Steam Piping Rupture Frequency Calculation



7. SUMMARY & CONCLUSIONS

7.1 Summary

The flow-accelerated corrosion (FAC) degradation mechanism was selected as the subject of the first CODAP Topical Report. All commercial nuclear power plants experience FAC of carbon steel and low-alloy steel piping. Significant progress has been made in managing and mitigating FAC. FAC management consists of a combination of analysis for susceptibility, non-destructive examination, repair and replacement. The CODAP Event Database includes a large volume of data records on significant pipe wall thinning, through-wall leakage and significant structural failures. The database includes detailed technical information on about 70 significant pipe failures that are attributed to FAC. All of these significant events have caused considerable collateral damage including flooding and spraying of equipment adjacent to a pipe break location.

Recent (2000-2012) operating experience demonstrates that FAC continues to cause damage to safety-related and non-safety-related piping. A primary cause of this recent experience is attributed to deficiencies in FAC programme implementation and management.

The CODAP Knowledge Base includes documentation on the national regulatory approaches to FAC management. Table 6 summarizes the different national FAC management approaches. Also included in the Knowledge Base are detailed descriptions of the different computer programs for the prediction of wear rates. Implementing plant-specific FAC monitoring strategies starts with development of a heat balance diagram (HBD that provides details on pressure, temperature, enthalpy and mass flow rate at every junction of the power generation cycle. Next, hydrodynamic, water chemistry, and materials data are used to predict FAC wear rates.

Table 6: Summary of the National Approaches to FAC Management

FAC Management Approach		
Country	**Current**	**Beyond 2013**
CA	Managed FAC programs that are based on plant-specific implementation of CHECKWORKS. Selective pipe replacement with improved materials	With focus on PHTS feeder piping, extensive R&D on FAC fundamentals. Evolving approach - accounts for new R&D results, enhanced NDE technology.
CH	Managed FAC programs that are based on plant-specific implementation of WATHEC/COMSY. Selective pipe replacement with improved materials.	No information available at time of writing
CZ	Managed FAC programs that are based on plant-specific implementation of CHECKWORKS. Selective pipe	No information available at time of writing

FAC Management Approach		
Country	**Current**	**Beyond 2013**
	replacement with improved materials	
DE	Inspection-for-cause, COMSY is also used to perform plant-wide screening for susceptible locations	No information available at time of writing
ES	Managed FAC programs that are based on plant-specific implementation of CHECKWORKS. Selective pipe replacement with improved materials	No information available at time of writing
FI	Managed FAC programs that are based on plant-specific implementation of COMSY. Selective pipe replacement with improved materials.	For Olkiluoto-3 EPR the COMSY code has been used to identify susceptible systems and locations as well as ascertain the material selections.
FR	Managed program based on plant-/design-specific applications of BRT-CICERO. Selective pipe replacement with improved materials	Evolving - accounts for new service experience, R&D results, progress with NDE technology. Development of requirements for "FAC-Free" BOP piping design.
JP	Inspection-for-cause, selective pipe replacement with improved materials	Extensive R&D on FAC fundamentals. Gradual implementation of "enhanced managed FAC programs"
KR	Managed FAC programs that are based on plant-specific implementation of CHECKWORKS. Selective pipe replacement with improved materials	Evolving - accounts for new service experience, R&D results, progress with NDE technology
SE	Inspection-for-cause; in the case of PWR units at Ringhals NPP, utilization of RI-ISI insights. Selective pipe replacement with improved materials	Gradual implementation of managed FAC programs. Full implementation of managed FAC program (based on COMSY) at Forsmark NPP mid-2013
SK	Inspection-for-cause, selective pipe	No information available at time of writing

FAC Management Approach		
Country	**Current**	**Beyond 2013**
	replacement with improved materials. Since 2005, monitoring of FAC by acoustic emissions.	
US	Managed FAC programs that are based on plant-specific implementation of CHECKWORKS. Selective pipe replacement with improved materials	Evolving - accounts for new service experience, R&D results, progress with NDE technology
TW	Managed FAC programs that are based on plant-specific implementation of CHECKWORKS. Selective pipe replacement with improved materials	Evolving - accounts for new service experience, R&D results, progress with NDE technology

7.2 Conclusions

The Topical Report on FAC explains the different parameters that affect the susceptibility of carbon and low-alloy steel piping to wall thinning. Parameters of interest include water chemistry, material composition and hydrodynamics. Next, the FAC management approaches of the thirteen CODAP member countries are summarized. As recorded in the CODAP Event Database, a high-level summary is provided of the international operating experience with FAC. Finally, an example is given of how the content of the CODAP Event Database may be processed in order to derive FAC-centric piping reliability parameters for use in probabilistic safety assessment (PSA) applications.

Insights from the several decades of FAC program implementation and monitoring and current R&D results are used in formulating FAC management strategies for new reactors. In theory, achievement of "FAC-free" piping systems is possible by taking into account the current state of FAC knowledge during the initial stages of new plant design.

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APPENDIX A

SELECTED SIGNIFICANT FAC EVENT DESCRIPTIONS[8][9]

[8] The CODAP Event Database includes extensive pipe failure event narratives. The selected, abbreviated event descriptions in Appendix A have been extracted from CODAP. Selection criteria include "safety significance of event," "resulting regulatory action," and/or "quality of root cause information."

[9] Photo credits: Figures A-1 & A-2 are reproduced from EPRI-1013249. Figure A-3 is reproduced courtesy of Japan Nuclear Energy Safety Organization (JNES; on March 1, 2014 the technical support organization became part of the Nuclear Regulatory Authority, NRA).

A.1 Feedwater Heater Drain Pipe Rupture at Trojan

CODAP EVENT ID 768: On March 9, 1985, a main feedwater isolation following a turbine trip at the Trojan plant (PWR) produced a pressure pulse that reached a maximum total pressure of approximately 6.0 MPa (875 psig) in the heater drain and feedwater system. The pressure surge ruptured a 368-mm (14.5-in.) diameter section of SA-106 Grade B carbon steel pipe in the feedwater heater drain pump discharge piping and released a steam-water mixture into the turbine building. The system flow velocity was 6.1 to 7.3 m/s (20 to 24 ft/s), and the normal operating pressure and temperature at the time of the break were about 3.1 MPa (450 psig) and 177°C (350°F), respectively. The ruptured portion of the piping section had been thinned from a nominal thickness of 9.5 to about 2.5 mm (0.375 to about 0.098 in.). Some of the thinning may have occurred during rupture. One worker received first and second-degree burns from the high-temperature fluid. Before this rupture, it was believed that only piping carrying two-phase fluid was susceptible to flow-accelerated corrosion and was, therefore, inspected in service. Because the ruptured drain pipe carried single-phase fluid, it was not inspected. As a result of this failure, the entire secondary system of the Trojan plant was evaluated to identify the sites susceptible to flow-accelerated corrosion, and then a sample of the sites was included in the inspection programme and subsequently inspected by ultrasonic examination. Repair and replacement of the damaged sections of piping were performed as necessary.

A.2 Feedwater Heater Extraction Steam Line Rupture at Hatch-2

CODAP EVENT ID 891: On April 24, 1986, Hatch Unit 2 (BWR) was in steady-state operation at approximately 85 % of rated thermal power, plant personnel were investigating the report of a large steam leak in the Condenser Bay Area. At that time, the "Generator or Exciter Field Ground Detection Relay" actuated tripping the main turbine. Consequently, the reactor scrammed and both recirculation pumps tripped due to a trip of the main turbine at greater than 30 % power. The relay, which is located in the generator exciter housing, actuated due to moisture build-up from steam condensing in the area around the main silicon control rectifier bridges which supply the main generator field. The steam was from the steam leak under investigation when the scram occurred. The leak was due to a through wall failure in the 6th stage feedwater heater extraction steam line at the downstream reducer leading to the 6th stage feedwater heater. The through wall failure was 18 inches long by up to 1 inch wide. The failure was caused by wet steam erosion.

The piping in the extraction steam lines is a carbon steel with a 0.3 to 0.6 % copper added ("Yoloy" material[16]). Ultrasonic test inspections were performed in order to locate any additional erosion in the extraction steam piping. The ultrasonic test inspections confirmed that no indications were noted in the 4th, 7th, or 8thstage lines. Also, on the 6th stage lines, no indications were found in the 10-inch or l4-inch lines. However, a 12 inch by 5 inch area that was slightly below ANSI B31.1 minimum wall tolerances in the area of a back to back elbow was found on a 20-inch line.

An ongoing ultrasonic test inspection programme was in place at Plant Hatch which examines approximately 30 points in the Extraction steam system during refuelling outages. This inspection had been completed during the unit 1 refuelling outage which began 27 November 1987. However, following this event, five additional points on the extraction steam line on unit 1 were tested per the ultrasonic test inspection programme with no indications found.

A new A-106B carbon steel reducer was installed to replace the failed "Yoloy" reducer on April 26, 1986. To correct the thin wall elbows, a weld overlay was performed on April 27, 1986 to bring the minimum wall thickness back above acceptable limits (0.370 inches).

[16] "Yoloy" is the U.S. trade name for this low-alloy steel, which has high resistance to general atmospheric corrosion.

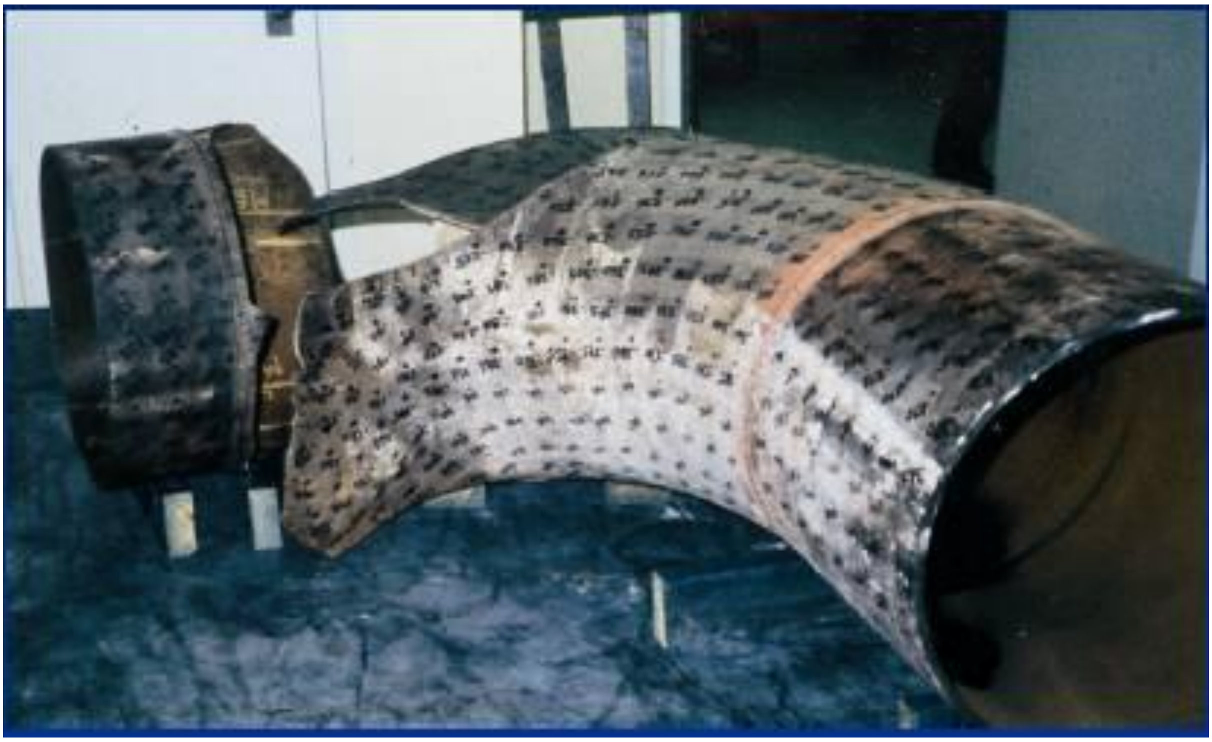
A.3 Main Feedwater Line Rupture at Surry Unit 2

CODAP EVENT ID 914: On December 9, 1986, a main steam isolation valve failed closed at Surry Unit 2 (PWR), and the resulting increased pressure in the steam generator collapsed the voids in the water. This caused the system pressure to surge beyond the normal operating pressure and led to a catastrophic failure of a 90-degree carbon .steel (SA-234 Grade WPB) elbow in the suction line to the main feed pump, as shown in Figure A-1. The diameter of the elbow was 460 mm (18 in.), and the design thickness was 13 mm (OS-in.). At the time of the event, the reactor was at full power and the feedwater was single phase, with a flow velocity of about 4.3 m/s (14 ft/s), a pH level in the range of 8.8 to 9.2, an oxygen content of about 4 ppb, and a coolant temperature and pressure of approximately 188°C (370°F) and 3.1 MPa (450 psig), respectively. Ammonia was used for the feedwater treatment. The examination of the ruptured elbow showed that the wall thinning was relatively uniform except in some local areas.

The wall thickness of the elbow was reduced from a nominal 13 mm (0.5 in.) to 0.38- to 1.22-mm (0.015- to 0.048-in.) in small local areas and to 2.3 mm (0.09 in.) in larger areas. Eight workers were burned by flashing feedwater, four of whom subsequently died. The flashing feedwater interacted with and disrupted the fire protection, security, and electrical distribution systems (USNRC 1988b). As a result of the Surry accident, the NRC staff asked that all utilities with operating nuclear power plants inspect their high-energy carbon steel piping.

New piping was installed at several locations in the Surry-2 feedwater system as a result of the pipe break. During the September 1988 outage, an elbow (installed in 1987) on the suction side of one of the main feedwater pumps was found to have lost 20 % of its 13-mm (0.5-in.) thick wall in 1.2 years. The NRC preliminarily concluded that this abnormally high rate of wall thinning may have coincided with a reduction in feedwater dissolved oxygen concentration. However, at the time the plant owner disagreed that the oxygen content was a major contributor. One explanation is that the accelerated thinning could have been aggravated by feedwater flowing into the steam generators that bypassed the feedwater heaters. This would have reduced the water temperature from its normal 190°C (370°F) to about 150°C (300°F). Later in the September 1988 outage, Virginia Power replaced a total of 125 piping segments with steel piping containing 2.5 wt % chromium.

Figure A-1: Ruptured Elbow in Feedwater Line at Surry 2



A.4 Failed High-Pressure Extraction Line at Fort Calhoun

CODAP EVENT ID 2506: On April 7, 1997, a DN300 extraction steam line to a feedwater heater ruptured in the turbine building. The rupture occurred in the 4th stage extraction steam piping, in a DN300 sweep elbow (radius equal to five times the pipe diameter). When operators heard a loud noise from the turbine building, the reactor was manually tripped. The rupture (estimated by the plant owner to be approximately 0.9 m long) occurred at the outer edge of a large radius bend in the extraction steam line; see Figure A-2.

Significant steam impingement damage to balance-of-plant Motor Control Centers 4C3 and 4C5 occurred. Additionally, collateral damage was experienced in several cable trays and pipe hangers, and insulation containing asbestos was blown throughout the turbine building. The fire suppression system actuated in the area and was subsequently isolated. Intermittent electrical system grounds occurred during the event. Insulation, containing asbestos, was blown throughout the turbine building.

No automatic safety-system actuations occurred during the event. However, portions of the fire protection system were actuated throughout the turbine building due to the heat and temperature rise associated with the steam rupture. The steam from the rupture caused seven wet pipe sprinkler heads to actuate in the basement level of the turbine building. These sprinkler heads were in the immediate vicinity of the steam leak and were designed to actuate at 160°F. The team noted that the steam in the vicinity of these sprinkler heads exceeded the actuation temperature of the sprinkler heads.

The deluge system for the turbine lube oil reservoir also actuated at the time of the event. This system contains 15 deluge nozzles that sprayed water in the area of the reservoir and on the main lube oil pumps. The system was actuated by a rate of temperature rise probe. (This device will cause a deluge actuation if a 15°F per minute temperature increase occurs.) The rate of temperature increase in the area of the steam rupture probably exceeded that required to actuate the detector. Due to the sprinkler and deluge system actuations, fire water system pressure decreased and caused the fire suppression water pumps to start automatically.

Figure A-2: Ruptured Long-Radius Elbow

Following activation of the fire suppression system, intermittent electrical grounds were received on Vital DC Bus 1 and 480V Bus 1B4C, both of which are safety related. These grounds on the safety-related buses were determined to be most likely due to grounds on the turbine building motor-control centers in the vicinity of the pipe rupture. These motor-control centers are fed by the safety-related buses and can be isolated by the tripping of the critical quality equipment feeder breakers which isolate the non-safety-related loads from their safety-related source.

The failure of the piping in the fourth stage extraction steam system was most likely due to flow accelerated corrosion. The design conditions of the fourth stage extraction system were 300 psig/425°F and the system was composed of primarily 12-inch diameter piping fabricated from A 106B carbon steel with a nominal wall thickness of .375 inches. The "fishmouth" break, which occurred, was approximately 4 feet long by 1 foot wide, and it was postulated that an approximately 2 - 4 inch wide by 4 foot long section of pipe was below minimum wall thickness before the rupture. The failure location occurred on what is known as a "large radius elbow." The as-found readings on the failed pipe revealed a minimum wall thickness of the rupture seam of .054 inches, whereas the minimum allowable pipe thickness was 0.126 inches. The failure location was modelled in the licensee's erosion/corrosion programme but the actual wall thickness had never been measured using non-destructive examination techniques. The licensee had relied on CHECWORKS to monitor the condition of the large radius elbows in the extraction steam system. The CHECWORKS methodology had predicted a lower wear rate on the large radius elbows relative to other potential wear locations within the fourth stage extraction steam system.

A.5 Condensate System Pipe Rupture at Mihama-3

CODAP EVENT ID 2960: While Mihama Power Station, Unit 3, was in operation at the rated thermal output, a fire alarm sounded in the central control room at 1522 hours on August 9, 2004. The control room operators determined that the alarm-generated area was on the second floor of the turbine building and checked the area to find that the building was filled with steam. Thus, it was judged that there was a high possibility of steam or high temperature water leakage from the secondary piping. The operator started emergency load reduction. While those operations took place, a "3A SG Feedwater < Steam Flow Inconsistency Trip1" alarm was generated and the reactor and then the turbine shut down automatically.

The operator made an inspection in the turbine building and confirmed a ruptured opening in an A-loop condensate pipe (Figure A-3) at 1730 hours on August 9, 2004, which was the feedwater line from the 4th feedwater heater to the deaerator running near the ceiling on the deaerator side at the 2nd floor of the turbine building. For the unit in question, the 21st periodical inspection was planned to start on August 14, 2004. In the turbine building, a total of 105 workers of KEPCO (the plant owner/operator) and affiliated companies were doing preparatory work for the periodical inspection at the time of occurrence of the accident. Of them, the affiliated company’s workers working near the ruptured A-loop condensate pipe were exposed to the steam and high temperature water flowed out from the ruptured opening, 5 were killed and 6 were injured.

According to the KEPCO Accident Report, two motor-driven auxiliary feed water pumps started automatically at 1528 hours on the day of the accident, followed by an automatic start of one turbine-driven auxiliary feedwater pump due to the abnormal low water level of the steam generator. After that, because the necessary flow rate of auxiliary feedwater was secured, so the auxiliary feedwater flow control valves A, B and C in the turbine-driven auxiliary feedwater lines were closed at 1532 hours to stop lowering the primary coolant temperature excessively.

After that, the water level of the steam generator was recovered and became stable, so the turbine-driven auxiliary feedwater pump was stopped at 1712 hours. To put this pump in an automatic standby condition, the operator tried to open the auxiliary flow control values A, B, and C at 1713 hours. However, the valves A and C stayed closed and no opening action took place. The operator tried to open the valves A and C again the next day, and both valves opened.

Figure A-3: Mihama-3 Condensate Pipe Failure

As a result of the root cause investigation, it was presumed that the backpressure of the valves in question exceeded the valve opening force while the pump was stopped, and this kind of system condition was not assumed in the design conditions for the valve, which was taken to be a cause. As a countermeasure, it was decided to replace the valve opening spring with one having a larger spring constant to provide the valve with a larger valve opening force than the maximum back pressure assumed in the design.

According to the KEPCO Accident Report, it was estimated as a result of the situation survey on the spot that high temperature water flowed out from the opening and then flowed down from the second floor to the first floor through the stairs and openings, and finally flowed into the turbine sump. With regard to steam blown out from the opening, it is estimated that the steam rapidly permeated almost the entire area of the turbine building immediately after the pipe rupture, and intruded into some portions of the control building, and the intermediate building adjacent to the turbine building.

In the region estimated to have touched high temperature water or steam that blew out from the pipe opening, there were the solenoid valves for main steam isolation valves, the control panels installed in the central control room, the instrument power facilities, the DC power facilities and the turbine-driven auxiliary feedwater pump as safety-related facilities. Of these, high temperature water intruded into the terminal box of one of the three solenoid valves for the main steam isolation valves, and one-sided grounding formed in the DC circuit; however, it operated normally at the accident. A trace of steam intrusion was found at the control panels installed in the central control room, the instrument power facilities and the DC power facilities; however, they operated normally during the accident.

According to the KEPCO Accident Report, Mihama-3 injects feedwater treatment chemicals from downstream of the condensate treatment equipment for corrosion inhibition of all the secondary piping and equipment. All volatile treatment (AVT) using ammonia (pH adjuster) and hydrazine (deoxidizer), as the feedwater treatment chemicals, has been used since the commissioning. As an anticorrosion measure for the steam generator tubes, boron injection had been performed from the 10th to the 15th operating cycles. From the 17th operating cycle, ethanolamine has been added as a pH adjuster.

KEPCO investigated the water quality control history since the commissioning of Mihama-3; and as a result, says that both the feed- and condensate water quality data have been maintained within the water quality control values, and that there was no variation in pH, dissolved oxygen, etc. Condenser tube leaks occurred twice in the past, and seawater flowed into the secondary cooling water. However, these events are considered to have no effect because the copper alloy does not corrode on the side in contact with the condensate water which was almost free of oxygen. The effect of boric acid on pipe wall thinning was investigated; however, no significant difference was observed in the thinning rate between with and without boron injection.

APPENDIX B

GLOSSARY OF TECHNICAL TERMS

Analysis Line. An "Analysis Line" is one or more physical lines of piping that have been analyzed together in the Predictive Plant Model of CHECWORKS™.

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.

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).

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.

Heat Balance Diagram (HBD). A schematic representation of the steam/condensate flow streams associated with the power generation cycle. The diagram contains information of steam properties (e.g., pressure, temperature, enthalpy and steam mass flow rate) at every junction of the cycle. The HBD provides input parameters to the development of a managed FAC program.

Line Correction Factor (LCF). According to the CHECKWORKS™ user's manual, the LCF is the median value of the ratios of measured wear for a given component divided by its predicted wear for a given "Analysis Line." A LCF of 1.0 is considered ideal as the measured wear equals the predicted wear (median value).

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).

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.

Time of Flight Diffraction (TOFD). The TOFD method of ultrasonic testing (UT) uses a pair of UT probes on opposite sides of a weld-joint or area of interest. A transmitter probe emits an ultrasonic pulse which is picked up by the receiver probe on the opposite side. In an undamaged part, the signals picked up by the receiver probe are from two waves: one that travels along the surface (lateral wave) and one that reflects off the far wall (back-wall reflection). When a discontinuity such as a crack is present, there is a diffraction of the ultrasonic sound wave from the top and bottom tips of the crack. Using the measured time of flight of the pulse, the depth of the crack tips can be calculated automatically by trigonometry application.