



D2.1 - Specifications for the online damage monitoring systems

WP2 - Task 2.1 October 2025

Document information

Grant Agreement Number	n° 101164810
Project Acronym	FIND
Project Name	Future Instrumentation and coNtrol based on innovative methods and Disruptive technologies for higher safety level
COO	Autorité de Sécurité Nucléaire et de Radioprotection (ASNR)
Project Coordinator	Bastien Poubeau
Project duration	1 October 2024 – 30 September 2028 (48 months)
Website	www.find-project.eu

Deliverable No./Milestone No	D2.1. Specifications for the online damage monitoring systems
Dissemination Level	PU
Work Package	2
Lead beneficiary	Andreas Schumm EDF
File Name	FIND_D2.1_specifications_normal operating conditions

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History

Date	Submitted by	Reviewed by	Approved by	Version (Notes)
19/06/2025	Andreas Schumm (EDF)	Zaiqing Que (VTT)	Bastien Poubeau (ASNR)	V0, submitted to the End-User Group for review
24/10/2025	Andreas Schumm (EDF)	Zaiqing Que (VTT)	Bastien Poubeau (ASNR)	V1, submitted to the European Commission



Summary

D2.1 defines the specifications for the use-cases related to defect detection during the normal operation of NPPs. The specifications present conditions to satisfy detect defects before they lead to “abnormal situations”. In this way, the associated technologies will strengthen the first level of defense-in-depth, related to the prevention of incidents and accidents. Another deliverable of FIND, D2.2, will present the specifications of the accidental instrumentation systems developed in the project.

Five use-cases were addressed, including the raw service water pipes, pre-heater siphons, steam extraction line of the high-pressure turbine, stress corrosion cracking on the primary circuit, thermal and vibrational fatigue on the pressurizer expansion line and double contained wall penetration of the sump recirculation line. For the different use-cases, specifications, nuclear codes, literature and operational experience of nuclear power plants were reviewed to identify the characteristics of the degradation mechanisms and defects to be detected. This deliverable defines precise specifications for the use-cases selected for the project and provides orientations for the technical developments performed in WP3 and the tests carried out in WP4.

This deliverable is made publicly available to inform other technology providers beyond FIND partners about the needs of nuclear operators, for them to propose relevant solutions.

Keywords

Specifications; Use-cases; Degradation; NDE; Prevention of abnormal situations



Abbreviations and acronyms

Acronym	Description
EC	European Commission
ATWS	Anticipated Transient Without Scram
EU	European Union
ENSREG	European Nuclear Safety Regulators Group
ESWS	Essential Service Water System
EUG	End-User Group
FAC	Flow-accelerated corrosion
LTO	Long-term operation
MCL	Main Coolant Line
NDT	Non-destructive testing
NPP	Nuclear Power Plant
PWR	Pressurized water reactor
SCC	Stress corrosion cracking
SCSS	Safety Containment Spray System
SG	Steam generator
SHM	Structural health monitoring
SIS	Safety Injection System
TPR	Topical Peer Review
UT	Ultrasonic testing
WP	Work Package
WWER (or VVER)	Water-Water Energetic Reactor

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Introduction

Six use-cases were selected and discussed, including the raw service water pipes, pre-heater siphons, steam extraction line of the high-pressure turbine, stress corrosion cracking on the primary circuit, thermal and vibrational fatigue on the pressurizer expansion line and double contained wall penetration of the sump recirculation line. For the different use-cases, detailed specifications, nuclear codes, literature and operational experience of operating plants were reviewed to identify the characteristics of the degradation mechanisms and defects to be detected.

Current safety approaches rely on periodic inspections to detect and characterise defects in metallic components. For each component and type of defect, a safety criterium is defined. It can correspond to the maximal wall thinning for local and global defects, or to a maximal crack size. These values are given in the report.

The objective of monitoring technologies is to better anticipate the degradation of components. This should avoid unplanned reactor outages and streamline maintenance processes, giving time to supply spare parts and prepare operations. So, monitoring technologies must detect defects well before they reach their maximal acceptable size. They should also be able to determine the kinetics of the degradation, which can sometimes accelerate or decelerate. Giving a precise objective for the detection limit of the systems is premature, because it depends on the maintenance strategy of each operator. In any case, the better the precision and the detection limits are, the higher the added-value of the system will be. The same applies for stress monitoring systems.

The OPEX elements presented below have been identified from significant events reported by French operators to the nuclear safety authorities. For confidentiality reasons, only a summary of these events is provided, and only the aspects relevant to the FIND project are mentioned. These significant events – which correspond in whole or in part to the “use cases” for which technical solutions are being considered – were selected for various reasons:

- To justify the relevance of the selected use-cases.
- To illustrate and clarify the specifications.
- To identify new use-cases to be addressed in continuation of FIND (WP 2.3).

These events were collected using the PIREX application (Integrated Platform for Operational Feedback), developed and maintained by the ASNR. PIREX contains all the significant events reported across all French facilities since their commissioning (over 50,000 events to date). Equipped with advanced text mining features, PIREX enabled the identification of relevant operational scenarios to be explored within the FIND project.

This deliverable defines precise specifications for the use-cases selected for the project and provides orientations for the technical developments performed in WP3 and the tests carried out in WP4. Precise requirements like measurement sensitivity, response time, and resistance to environmental conditions were defined for the use-cases selected.

For clarity, all bibliographic references for a specific use-case are grouped in a table at the end of each section.

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This document includes implications of partners from all WPs and has been reviewed by the EUG. Partners acknowledge the substantial contribution of the results coming from the Euratom-funded INCEFA SCALE project.



1. Raw service water pipes

1.1. Presentation of the systems of interest

Concealed pipes may belong to safety-related systems. A typical example in WWER and Western PWR designs is the Essential Service Water System (ESWS), which supplies cooling water to the spent fuel pool and safety-related equipment (Figure 1-1). Such pipes operate at low or even atmospheric pressure. They may only be partially filled by water in some sections, the rest of the volume being occupied by air. In this case, condensation can form on the upper inner wall of the pipe.

The ESWS includes multiple concealed pipe sections: in/out pipes between the heat sink (sea or river) and the reactor building, spray pond in/out pipes, and distribution lines. These pipes are buried underground at depths of up to 6 meters. The bedding and backfill consist of loose, damp sand with varying density. Figure 1-1 shows pipe examples after excavation. This system must adapt to the local topology of the site, so it is less standardized than other NPP systems. They have a long service lifetime, typically 30 years.

Inspection of buried pipes is considered internationally as critical for ageing monitoring in the context of long-term operation (LTO). The action plan of ENSREG (European Nuclear Safety Regulators Group) resulting from the 1st topical peer review (TPR) on ageing management states: “EC2.2. ENSREG will recommend the EC to promote research on the application of NDT techniques to identify defects in long lengths of concealed pipework to address the topic in EU Framework Programme for Research and Innovation”.



**Figure 1-1: ESWS pipes (after excavation) (ref. to National Report: First Topical Peer Review Ageing Management. State Nuclear Regulatory Inspectorate of Ukraine. Kyiv 2017
www.ensreg.eu/sites/default/files/attachments/ukraine.pdf**

1.2. Measuring large deformations

1.2.1. Rationale

Among the degradation mechanisms, the settlement, soil displacements and other external loads must be considered for LTO [1]. Soft soil conditions or environmental condition changes, such as groundwater elevation, may lead to pipe deformation or even failure (Figure 1-2).

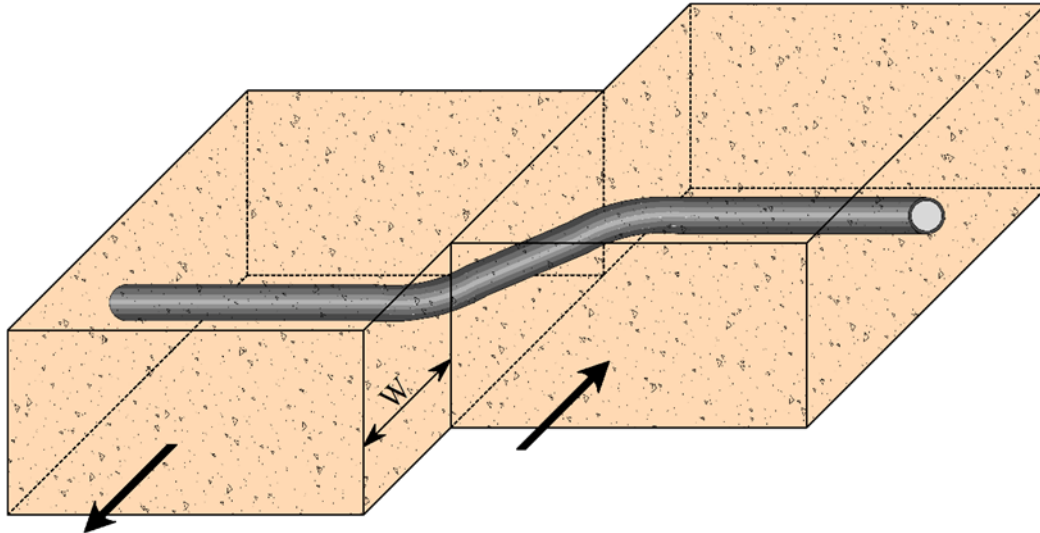


Figure 1-2: Schematic representation of a pipeline crossing a strike slip fault at a crossing angle [1].

The main mechanism for determining the parameters related to settlement and soil displacements is direct or indirect measurements of vertical distances to determine structure elevations or soil surface in the case of underground utilities [1]. An optical or laser device is used to measure the elevations of object points (points to be monitored) and reference points. Indications of surface soil irregularities, depressions, or sinkholes also give indications of potential settlement and should be investigated further. Some care must, however, be exerted when considering only surface soil monitoring. Depending on soil layers, subsidence of a layer below the surface grade may occur without or with reduced visual impact on the surface grade (Figure 1-3) [1].

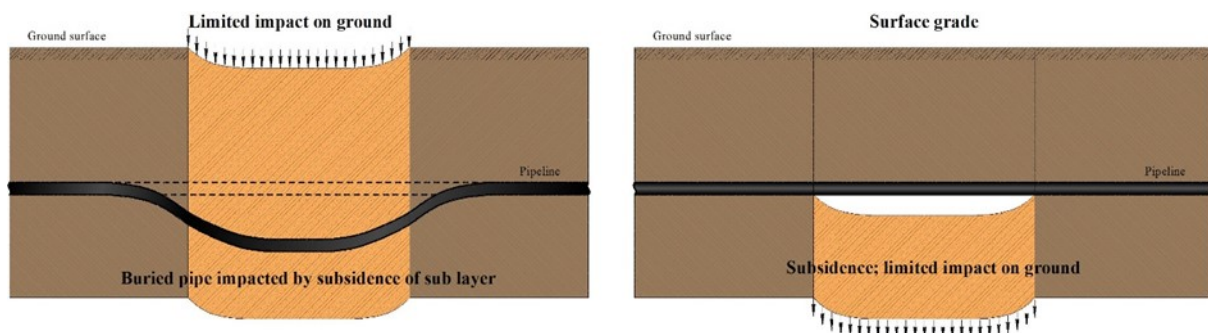


Figure 1-3: Surface soil monitoring, subsidence on sublayer soil and limited impact on surface grade [1].

One way to increase the reliability of concealed (underground) pipes is through direct instrumental examination of pipelines. But classical Non-destructive testing (NDT) requires digging the pipe section out, which is so costly and only a few measurement points are investigated at a specific time. The presence of coating (Neoprene as usual) and support systems make measurements more difficult. Therefore, it is relevant to develop approaches for permanent and long-term monitoring of stresses during pipeline operation.

For this purpose, the system should be able to provide online monitoring results for chosen pipeline cross-sections. The positions (cross-sections) of measurement are selected based on the preliminary analysis and predictive calculations, potential failure locations of pipes and piping configuration. The data gathered from each measured cross-sections are transferred to the operator in any convenient form: communication cable, radio signal, etc. depending on plant piping system configuration.

Considering the absence of utilities like electricity and communication tools available in the vicinity, the monitoring system must be equipped with an autonomous power module. Given the low frequency of measurements (usually ground displacement processes occur gradually), the available power module can ensure the system's operation for a long time and, if necessary, be charged externally.

The objective of the system is to reconstruct the stress field in the component. The lower detection limit and the measurement uncertainty should be between 5% and 10% of the yield stress of the material of interest. The upper limit of the measurement range should be higher than the yield stress. For typical steels, it corresponds to a strain of 2% maximum.

1.2.2. Specifications: large deformations of raw service water pipes

General information			
Component(s) concerned	Concealed (underground) pipes	Degradation mechanism	• Freeze-thaw cycling; • Settlement and soil displacement.
Safety function	Heat sink		
Type of measure to be developed	Digital twin: reconstruction of the stress field in the pipe.	Reactor designs concerned	PWR, WWER
Lead contributor	IPP	Other contributors	VTT
Current methods applied	An optical or laser device	Limitations of current methods	Is not always representative. Depending on soil layers, subsidence of a layer below the surface grade may occur without or with reduced visual impact on the surface grade.
Component characteristics			
Diameter	630 – 2040 mm	Thickness	8 – 14 mm
Length	Up to several kilometres	Geometry	Straight, sectional elbows
Material type	17GS, Steel 20, VSt3pt6, 11G2S1-12		

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Surface characteristics	Outer coatings: mastic bitumen very reinforced (reinforced fiberglass in two layers), epoxy, cementitious		
Transported fluid			
Type of fluid	Untreated river or sea water	Temperature	5 – 65 °C
Velocity	0.3 – 3 m/s	Pressure	0.6 – 6.0 kg/cm ²
Chemical composition	pH of 6.5 – 8.5	Physical state	liquid
Stresses to be monitored – one table for each stress monitored			
Type of stress	Mechanical stresses	Detection range	Lower bound: <10% of the yield stress Upper bound: >yield stress
Frequency	1 measurement per hour		
Sources of signal interference			
Material inhomogeneity	Same as above	Vibration	Same as above
Electromagnetic	Same as above	Others	-
Sensor environment			
Temperature	5 – 65 °C	Humidity	Natural soil humidity 100 – 15 %
Pressure	Atmospheric	Irradiation	No
Data transfer	Pipes are usually located far from industrial building, so medium range (< 1 km) wireless communication is preferred.	Electricity supply	No source of electricity is available in most cases. Self-powered devices are preferred.
References			
Ref: Scientific publications	Yaskovets Z., Orynyak I. Monitoring and calculation of the stressed state of underground main gas pipelines in the areas of mine productions (in Ukraine). Tekhnicheskaya diagnostika i nerazrushayushchiy kontrol. 2018. №2. P. 45–52. I Orynyak, Z Yaskovets, R Mazuryk Novel numerical approach to analysis of axial stress accumulation in pipelines subjected to mine subsidence. Journal of Pipeline Systems Engineering and Practice, 2019 https://doi.org/10.1061/(ASCE)PS.1949-1204.0000405		
Ref: National and international standards	[1] IAEA Nuclear Energy Series No. NP-T-3.20. Buried and underground piping and Tank Ageing Management for Nuclear Power Plants		



1.3. Detecting local defects

1.3.1. Rationale

Only about 2% of the total ESWS underground pipe length is accessible for inspection. High-moisture environments with elevated electrical conductivity, alkalinity, or acidity increase the risk of corrosion. Over time, soil movement and settlement can impose additional external stress on buried pipes, leading to cracks and structural damage.

Figure 1-4 shows examples of a buried service water pipe and service water pipes in a gallery, both on French sites.



Figure 1-4: Service water pipes, buried or in galleries.

Within the reactor building, the raw service feedwater pipes are not buried. Figure 1-5 shows a raw feedwater pipe in a French nuclear plant. Figure 1-6 shows examples of support structures, which render sections of the pipe inaccessible for UT wall thickness measurements.

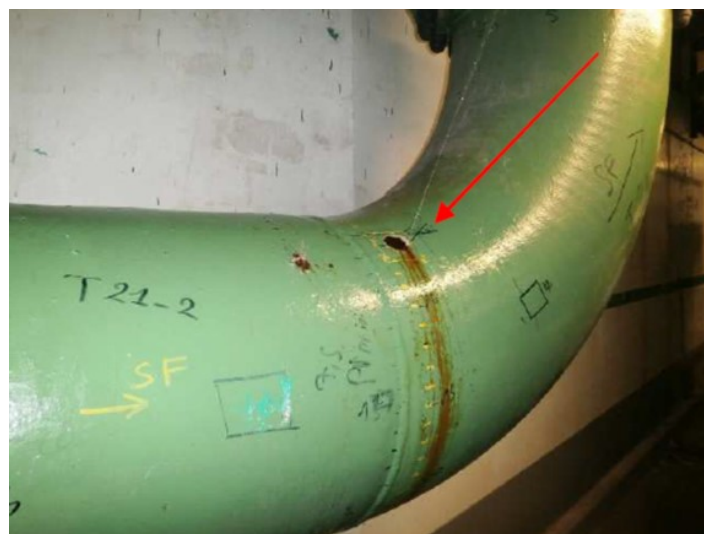


Figure 1-5: Raw feedwater line of Gravelines plant with leak due to cavernous corrosion.

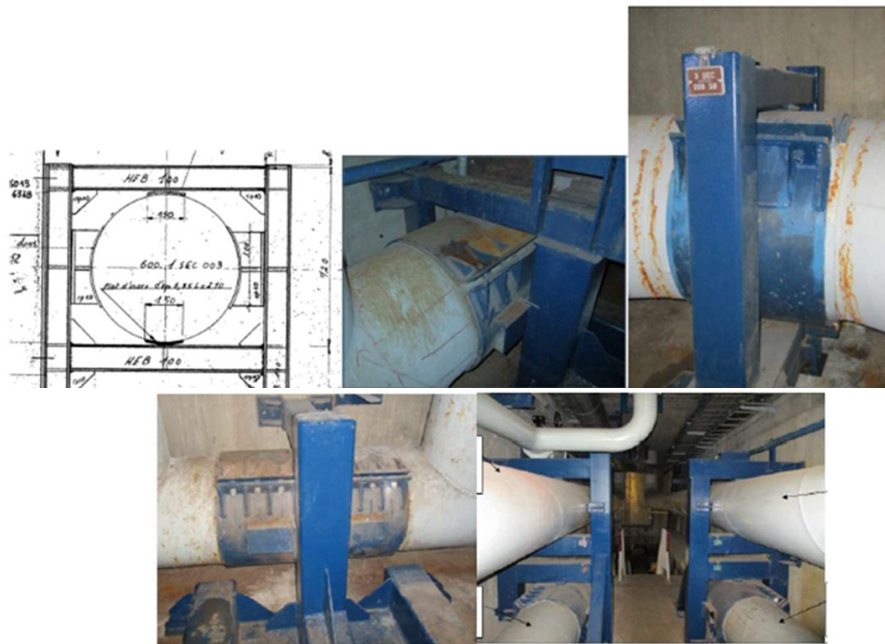


Figure 1-6: Raw feedwater lines with typical examples of support structures.

The ageing mechanisms of piping metal and their effects on metal are presented in Table 1-1, basic parameters that define the metal conditions are in Table 1-2.

Table 1-1: Ageing mechanisms and effects.

Effects on metal	Metal ageing mechanisms			
	Low- and high-cycle fatigue	General corrosion	Local corrosion	Flow accelerated corrosion
Thinning	-	×	-	×
Change of mechanical properties	×	-	-	-
Pitting corrosion	-	-	×	-
Erosion (local thinning)	-	×	-	-
Electrochemical corrosion	-	-	-	×

Table 1-2: Basic parameters defining the metal conditions.

Effect of metal ageing	Parameters defining the metal conditions
Thinning	Wall thickness
Change of mechanical properties	Safety margin, yield point, relative elongation, relative reduction

Effect of metal ageing	Parameters defining the metal conditions
Pitting corrosion	Number of defects per area unit, depth of defects
Erosion (local thinning)	Damage area, wall thickness
Electrochemical corrosion	Cracks

Two remote technologies are currently being applied for monitoring of the underground ESWS pipes at operating NPPs in Ukraine:

- magnetic tomography method;
- acoustic emission monitoring.

These methods have some constraints as noted in Table 1-3.

Table 1-3: Currently applying technologies (methods) and their limitations.

Method	Constraints
Magnetic tomography method	Cannot detect defects which do not change stress field within material (i.e. blow holes, pitting) Hindered by interference from AC power lines Requires baseline readings when pipeline is free from significant degradation to obtain magnetic signature Not reliable for cased piping, anodic zones on bare pipe and shielded corrosion activity
Acoustic emission monitoring	Can be subject to background noise Sensitivity decreases with pipe size and distance Limited distance between acoustic meters, requiring pits to install sensors on the surface of the pipe (if/where needed)

Parameters of importance in selecting a monitoring technology are as follows (the list is not exhaustive):

- Sensitivity: detecting the minimal leaks, ensuring maximum distance for leak detection and for assessing the overall corrosion thinning of pipe walls;
- Accuracy: detecting the leak coordinates with acceptable accuracy and determination of the average metal thickness with the highest possible precision;
- Reliability: ensuring the reliability of measurements under conditions of multi-wave propagation, acoustic wave interference and other characteristics of large-diameter pipelines;
- Robustness: eliminating the influence of acoustic and electromagnetic interference on measurement results, simplicity, operational safety, ease of maintenance as well as the ability to use operational experience for improving monitoring.

Considering the specific features of the ESWS underground pipes and constraints of the currently applied monitoring methods, it is expected that technology to be implemented will have the following basic characteristics:

1. Remote monitoring of the ESWS underground pipes (diameters range Ø 820 - Ø 2240, max water pressure - 6.0 kg/cm²) without any intrusion to the system operation (not changing pressure and flowrate)
2. Using the existing access points to the pipes (no excavation for pits)

3. Precise identification of the small leakages (less than defined in OLCs)
4. Precise assessment of the average metal thickness of pipeline sections between access points
5. Operational capability in conditions of intense electromagnetic interference
6. Ability to adapt to conditions of pipe access, laying and types of insulation
7. Data transfer that does not interfere with the current I&C at plant site

The existing acoustic parametric correlation method, currently used for monitoring heat and water urban pipes, will be adapted for application to the ESWS underground pipes in WP3.

1.3.2. Specifications for local defects

General information			
Component(s) concerned	Partially concealed (underground) pipes of the Essential service water system (ESWS)	Degradation mechanism	Crevice corrosion, thinning, change of mechanical properties, pitting corrosion, erosion, electrochemical corrosion
Safety function	Supplying water for safety system equipment cooling, heat removal from spent fuel pools, etc. System important to safety that uses chemically treated water in a closed loop, isolated from raw water sources (VVER design). In French PWR designs, underground pipes can transport raw water (like sea water).		
Type of measure to be developed	Periodic monitoring	Reactor designs concerned	PWR, WWER designs
Lead contributor	EDF	Other contributors	IPP, SSTC NRS
Current methods applied	Magnetic tomography Acoustic emission	Limitations of current methods	<u>Magnetic tomography</u> : interference from AC power lines; limitation for cased piping, anodic zones on bare pipe and shielded corrosion activity. <u>Acoustic emission</u> : background noise; sensitivity decreases with pipe size and distance; limited distance between acoustic meters, requiring pits to install sensors
Component characteristics			
Diameter	800 – 2250 mm	Thickness	9 – 14 mm
Length	280 – 1300 m	Geometry	Straight sections and pipe bends, including 90° bends, underground at a depth of up to 6 m

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Material type	Carbon steel	Specific properties (if relevant)	
Surface characteristics	The inner surface is in general not coated. However, raw feedwater pipes at EDF in contact with salt water are fitted with a neoprene coating inside. The outer surface – waterproofing coating: bituminous primer (3 mm), reinforcing wrapping of glass fleece (3 mm), bituminous-polymer mastic (3 mm). Total thickness 9±1 mm.		
Transported fluid			
Type of fluid	Chemically treated water	Temperature	5.0 – 65.0 °C
Velocity	0.3 – 3.0 m/s (estimated)	Pressure	0.6 – 6.0 kg/cm ²
Chemical composition	pH of 6.5 – 8.5 total hardness < 7 mg·eq/dm ³ carbonate hardness < 2.5 mg·eq/dm ³ total salt content <800 mg/dm ³	Physical state	Liquid
Degradation mechanism and characteristics of the defect to be detected			
Geometry of the defects	Pitting or crevice corrosion cavities, uniform decrease in metal thickness across the surface	Maximal acceptable size	Minimum acceptable pipe thickness established for each specific pipe group, typically 60–70% of nominal value (e.g. for pipes 2040×10 – minimum wall thickness is 6 mm).
Expected localisation	Outer surface		
Kinetics	Usually slow, except for crevice corrosion which can develop in less than one year.		
Specifications for the detection of leakages			
Magnitude	Leakage	Detection range	Minor leakage (reaching up to tens of liters per hour). A reduction in discharge pressure at the pump outlet could be a signal that leakage occurs.
Precision	0.5 m precision for leak location detection (proposed by the partners)	Response time	<10 min
Sources of signal interference			
Material inhomogeneity	Acoustic features of large diameter pipelines including multi-wave	Vibration	Sources of external acoustic interference: noise from water



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	propagation and interference		consumers, pumps, valves
Electromagnetic	Radio interference, near high-voltage power lines and transformer stations	Others	Interference from underground high voltage cables above the pipeline
Sensor environment			
Temperature	5.0 – 65.0 °C	Humidity	Less than 80%
Pressure	Normal atmospheric pressure	Irradiation	No
Main safety constraints on hardware	Need to be checked	Electricity supply	Not available
Data transfer	Analogical/numerical, middle range (< 1 km) wireless (no local access to physical data network)		
References			
Ref: Mechanical codes	Former PNAE G-7-008-89 “Rules for design and safe operation of equipment and piping of nuclear power facilities” replaced by NP 306.2.227-2020 and SOU NAEK 158:2020, SOU NAEK 159:2020		
Ref: Scientific publications	Vladimirsky, A., Vladimirsky, I., Dybach, O., 2022. Parametric Analysis of Correlation Functions for Acoustic Monitoring and Assessment of Underground Piping at NPPs. ISSN 2073-6321. Nuclear and radiation safety 3(95).2022		
Ref: National and international standards	NP 306.2.227-2020. General safety requirements for design and operation of equipment and pipelines of nuclear plants NP 306.2.210-2017. General requirements for the management of aging of elements and structures and long-term operation of nuclear power units. SOU NAEK 158:2020. Ensuring technical safety. Technical requirements for the design and safe operation of equipment and pipelines of nuclear power plants with VVER reactors SOU NAEK 159:2020. Ensuring technical safety. Welding and surfacing of equipment and pipelines of nuclear power plants with VVER reactors. Technical requirements SOU NAEK 051:2015. Maintenance and repair. Application of the modernized acoustic-emission diagnostic method for assessing the technical condition of pipelines and equipment of NPP power units.		
Operating experience	National Report: First Topical Peer Review Ageing Management, State Nuclear Regulatory Inspectorate of Ukraine, Kyiv 2017. Cavernous corrosion observed in Gravelines plant in 2019 with a full thickness loss over 16 months, leak rate 12 L/h.		

1.3.3. Additional operating experience feedback regarding local defects

The following events happened in French NPPs illustrate local defect use-case:

Date: 2017-10-15
NPP: GRAVELINES
Title: Reactor shutdown in accordance with operating procedure in response to a leak on the ESWS



What happened: On 15/10/2017, a leak was detected on the ESWS downstream of the SEC/RRI heat exchangers. Train A of the SEC/RRI systems was shut down, and thickness measurements were carried out around the orifice, which was measured to be 13 mm in diameter. The area found to have reduced wall thickness was estimated at 50 cm². The leak rate was assessed at 4.5 m³/h. The degradation originated from a phenomenon of crevice corrosion under neoprene.

Consequences: In a LOCA scenario, the unavailability of one RRI train directly results in the loss of one EAS train and, eventually, in the loss of the overall effectiveness of the RIS residual heat removal function (as the overall capacity of the EAS system is reduced).

Interest: Event selected by EDF as use case for FIND project.

Date: 2020-05-26

NPP: GRAVELINES

Title: Shutdown of Unit 6 following detection of a leak on a section of ESWS pipes

What happened: During a routine inspection of the Unit 6, the operators detected a leak on the ESWS pipes. A team was sent to locate and assess the leak. They observed a leak estimated at 7.2 m³/h. As repair could not be completed within the allotted time (24 hours), the shutdown of Unit 6 was initiated.

Consequences: In a LOCA scenario, the unavailability of one RRI train directly results in the loss of one EAS train and, eventually, in the loss of the overall effectiveness of the RIS residual heat removal function (as the overall capacity of the EAS system is reduced).

Interest: The leak was caused by the appearance of crevice corrosion under the neoprene coating, resulting from contact between seawater and steel.

2. Low pressure Preheater siphon

2.1. Rationale

The low-pressure preheater siphon is a 6200 mm long component with only the top section accessible for instrumentation, the bulk of the length (more than 5 m) being surrounded by concrete, with a thermal insulation between the siphon and the concrete structure. Siphons extracted and replaced in the Tricastin plant showed superficial corrosion in the lower section. A guided wave screening technique is sought to scan the inaccessible section for wall thickness loss due to corrosion. The picture below in Figure 2-1 shows the accessible section with the thermal insulation still in place; it would be removed for the installation of the instrumentation. The second figure in Figure 2-2 shows the actual geometry, including the inaccessible section surrounded by concrete. Figure 2-3 shows the schematic of EDF preheater siphon in place.



Figure 2-1: Preheater siphon in place with heat insulation.



Figure 2-2: Extracted Preheater siphon (inner and outer tubing).

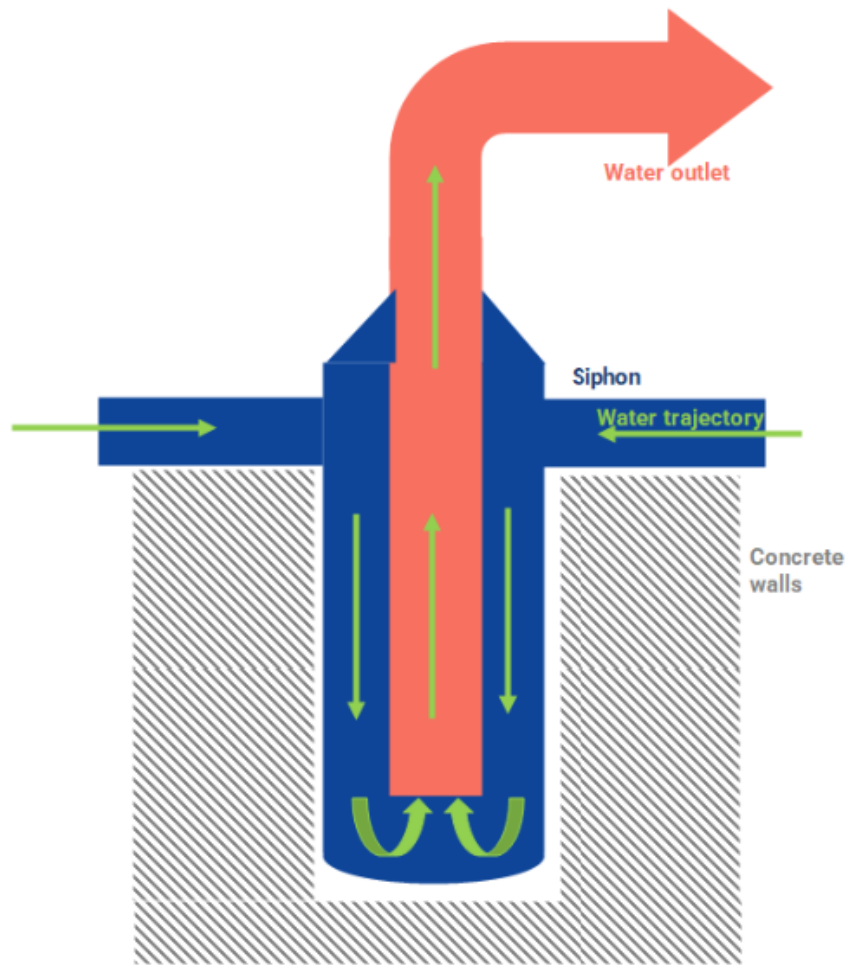


Figure 2-3: Schematic representation of preheater siphon in place.

2.2. Specifications

General information			
Component(s) concerned	Preheater siphon	Degradation mechanism	Flow accelerated corrosion
Security requirement	Pressurised equipment integrity		
Type of measure	Remaining wall thickness	Reactor designs concerned	PWR 900 MWe series
Lead contributor	EDF	Other contributors	KTU, CEA
Current methods applied	None		
Component characteristics			
Diameter	508 mm	Thickness	8 mm
Length	6200 mm	Geometry	Straight
Material type	Tu42B	Specific properties (if relevant)	Weld close to the siphon bottom

Surface characteristics	No coating		
Transported fluid			
Type of fluid	Water	Temperature	184 °C
Velocity	0.95 m/s	Pressure	11 bar
Chemical composition	pH at 25 °C: between 9.15 and 9.65	Physical state	Liquid
Degradation mechanism (for SHM)			
Geometry of the defects	FAC	Maximal acceptable size	30% of wall thickness loss
Expected localisation	Outer surface, lower end		
Kinetics	Slow (several years)		
Sources of signal interference			
Material inhomogeneity	Circumferential weld close to the bottom	Vibration	Unknown but possible
Sensor environment			
Temperature	10 – 40 °C	Humidity	Atmospheric
Pressure	Atmospheric	Irradiation	None
Main safety constraints on hardware	Must not interfere with thermal shielding		
Electricity supply and data transfer	Access to electricity and physical data network can be provided but with additional costs. Self-powered device with short range wireless data transmission is preferred.		
References			
Operating experience	4 siphons replaced in Tricastin plant, only superficial corrosion observed.		

3. Dryer superheater extraction line

3.1. Rationale

3.1.1. Introduction about Flow Accelerated Corrosion

Flow-accelerated corrosion (FAC) is a complex ageing phenomenon. It is sometimes also referred to as flow-assisted corrosion or incorrectly as erosion-corrosion.

FAC leads to wall thinning by metal loss of carbon steel piping and equipment exposed to flowing water, wet steam or a combination of both. FAC is the result of dissolution of the surface film of the steel, which is transported away from the site of dissolution by the movement of water [1][2].

The rate of flow accelerated corrosion depends on the combination of different parameters such as materials composition, geometry, water chemistry, temperature, and hydrodynamics.



This degradation phenomenon may reduce a component wall thickness over a large area. If the wall thickness becomes insufficient to withstand internal pressure, it may suddenly explode or break, causing the release of an important quantity of energy and be a threat for the safety of workers and affect the plant reliability.

Pipe wall thinning rates as high as 3 mm/year have been observed. Pipe ruptures and leaks caused by FAC have occurred at fossil plants, nuclear plants, and industrial processing plants [2].

3.1.2. User case of Tihange 3

In Tihange 3, the high-pressure steam turbine extraction line, or more precisely, the line coming from the dryer/superheater to the superheater purge tank (see Figure 3-2 and Figure 3-3), has been identified as highly susceptible to FAC thanks to a predictive tool and inspection campaigns in 2015 and 2018 (UT) that revealed loss of thickness (up to 1.3 mm in elbow E6 and 2.19 mm in elbow E8 in 3 years). It is therefore proposed as a use-case for testing new sensors technologies developed in FIND.

At Tihange NPP, FAC is currently managed as follows:

1. First, critical locations are determined with the EPRI predictive software Checkworks and thanks to the return of experience of other NPP.
2. Then, these areas are inspected thanks to classic ultrasonic means. The wall thickness is measured in a grid of points uniformly spread over the component (see Figure 3-1 for example).
3. If necessary, follow-up inspections are planned and/or temporary repairing and replacement of components during the next outage when needed.

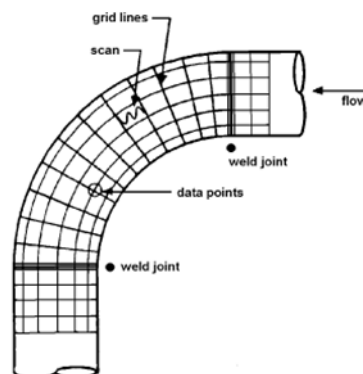


Figure 3-1: Example of measurement grid (EPRI) [2].

There is also the possibility to mitigate this ageing phenomenon by playing on the steam/water chemistry (increasing the pH) and selecting FAC resistant materials (higher chromium content) in case of replacements.

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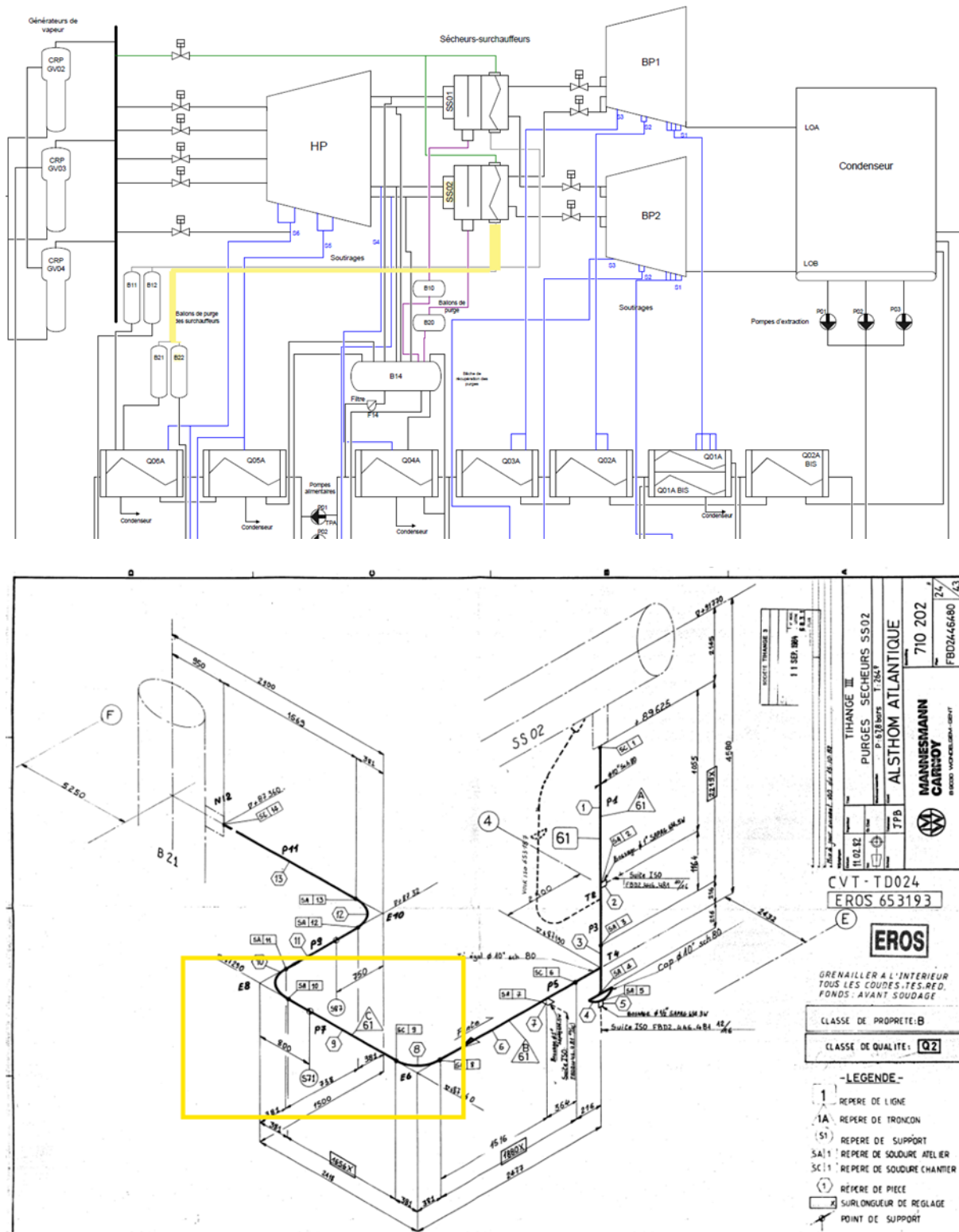


Figure 3-2: Technical drawings of the high-pressure steam turbine extraction line.



Figure 3-3: Images of the high-pressure steam turbine extraction line.

3.2. Specifications

General information			
Component(s) concerned	High pressure steam turbine extraction line. More precisely, the line coming from the dryer-superheater to superheater purge tank. See Figure 3-2 and Figure 3-3.	Degradation mechanism	Flow accelerated corrosion
Safety function	No safety function However, flow accelerated corrosion issues could impact the safety of workers (in case of pipe or equipment rupture for example) as well as the normal operation of the plant.		
Type of measure	SHM (ultrasonic guided waves)	Reactor designs concerned	PWR
Lead contributor	Tractebel	Other contributors	CEA, Framatome, IPP
Current methods applied	- Determination of the critical locations with predictive software and ultrasonic inspections of the susceptible areas. The wall thickness is measured in a grid of points uniformly spread over the component. When needed, follow-up inspections and/or temporary repairing/replacement of components during the next outage.		
Limitations of current methods	- Spot inspections No continuous monitoring; - Time consuming: for the measurements, the preparation of the measurements (location identification, assembly of scaffolding, removal of the insulation, cleaning of the piping if necessary, etc.) and the processing of the measurements - Difficult to monitor large areas. - Some zones could be inaccessible for UT examination.		

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Component characteristics			
Diameter	10" (273.05 mm)	Thickness	15 mm (nominal)
Length	2418 mm between E6 and E8 (see isometric in Figure 3.2)	Geometry	Elbow + straight pipe
Material type	Carbon Steel, A106b (approx. 0.03-0.11% Chromium)		
Surface characteristics	Inside: No coating, no cladding. Potentially some rugosity due to FAC degradation mechanism. Outside: No coating, no cladding or paint. Insulation to be removed for the installation of the sensors.		
Transported fluid			
Type of fluid	Mixture of steam and heated water	Temperature	Around 270 °C Service T: 264 °C.
Chemical composition	No detailed data available. Mix of steam/water from the secondary system. Water treated with ammoniac.	Pressure	Service pressure: 67.8 bars.
Degradation mechanism			
Geometry of the defects	Reduction of the wall thickness	Expected localisation	Inner surface
Maximal acceptable size	Minimal critical thickness: 11.35 mm * Calculated with the design pressure, which is above the service pressure		
Kinetics	Complex kinetics, see description of FAC phenomenon in the text above.		
Sources of signal interference			
Vibration	Yes (located in the power plant engine room). No measurements or spectra available. Measures on-site could be planned before installation.		
Electromagnetic	Possibly. Measures on-site could be planned before installation.		
Sensor environment			
Temperature	Temperature of the fluid inside the pipe is around 270 °C. No measure on the outside of the pipe is available.	Humidity	Yes, air indoor uncontrolled.
Pressure	Atmospheric	Irradiation	No
Main safety constraints on hardware	No halogens.	Electricity supply	Classic electric plugs are usually available in these areas of the NPP.
Data transfer	Data could be stored locally and then transferred manually (but it should be avoided as much as possible to send workers in the vicinity of high energy systems for safety reasons) or using a short-range transmission device.		
References			
Operating experience	[1] Electric Power Research Institute (EPRI), Recommendations for an Effective Flow-Accelerated Corrosion Program (NSAC-202L-R4), 2013. [2] IAEA, Safety Reports Series No.82, Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL). [3] Tihange Operating Experience. [4] Mihama NPP accident in Japan (2002). [5] Surry NPP accident in the USA (December 1986).		

4. Stress-corrosion cracking on the primary circuit

4.1. Rationale about Stress Corrosion Cracking

Stress corrosion cracking (SCC) is the growth of crack formation in a corrosive environment. SCC is highly chemically specific in that certain alloys are likely to undergo SCC only when exposed to certain chemical environments.

SCC mechanism implies a complex interaction of mechanical conditions, type of alloy, and characteristics of the environment. In literature, there are often 3 phases described [1], see Figure 4-1:

1. Phase 1: Usually, SCC starts with the early localized corrosion or mechanical defects, such as pitting, local inter-granular attack, scratches or other pre-existing surface defects.
2. Phase 2: Subsequently, these defects evolve, pit growth and localized concentration of stress, which leads to short cracks with a slow propagation rate.
3. Phase 3: Finally, when reaching the long crack regime, the component may fail.

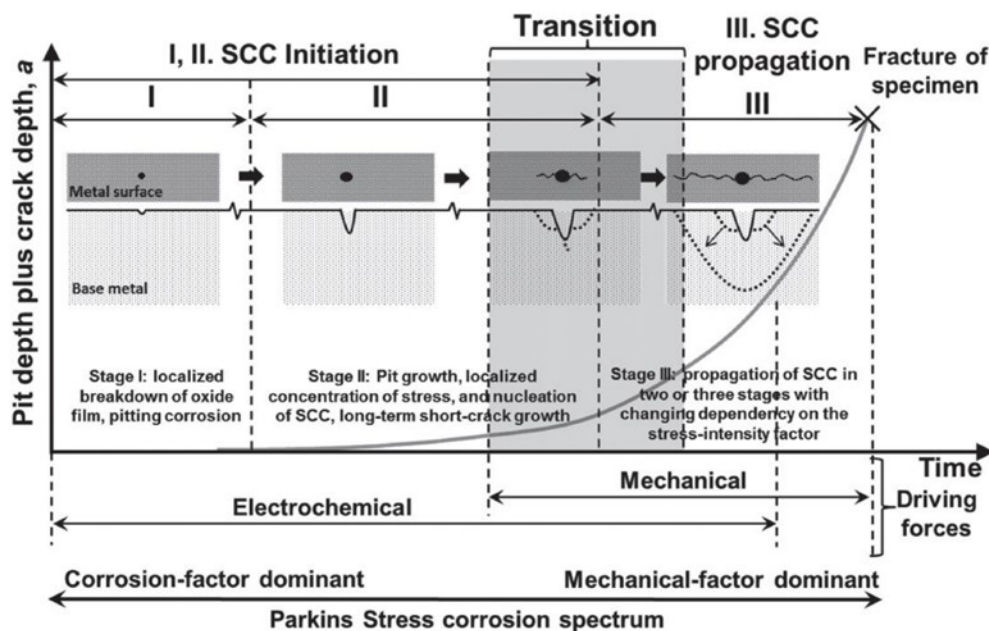


Figure 4-1: Schematic diagram of a three-stage model for SCC progression [1].

The evolution of the whole damage path is generally slow and time consuming. Consequently, in-situ monitoring of the transition from initiation to propagation can be very useful to estimate the SCC risks. The system will measure the degradation kinetics and not the defect size directly.

NPPs are mostly affected by intergranular SCC, for which crack growth events are less energetic than transgranular SCC. Fatigue crack growth can generate acoustic events similar to transgranular SCC. In some cases, the propagation phase can also stop before the failure of the component, thanks to the release of the stresses that initiated the phenomenon.

4.2. Specifications

A large variety of components of the primary circuit may be confronted to SCC. The table is filled for the safety injection line of the Emergency Core Cooling System, which is one of the most safety significant components exposed to SCC.

It is worth noting that in most of the French SCC cases spotted in 2021, the SCC cracks are relatively shallow in respect to the wall thickness. Only in two cases with local repair welding, the SCC cracks extended over 50% of the wall thickness (in one case even over 90%). The repair welding significantly influences the weld residual stress distribution and enhances the SCC propagation. Therefore, the instrumentations on welds particularly with repair welding for detecting the external mechanical stresses need additional attention.

General information			
Component(s) concerned	Weldings, especially dissimilar metal welds in the primary circuit	Degradation mechanism	SCC
Safety function	Enclosure of primary circuit		
Type of measure	SHM, e.g., Acoustic Emission	Reactor designs concerned	Most designs
Lead contributor	Framatome	Other contributors	CEA, ASNR
Current methods applied	Ultra-sonic testing during outages	Limitations of current methods	Only applicable during outages; considerable effort and time consuming; irradiation of workers
Component characteristics (Safety injection line)			
Diameter	8 to 14 inches	Thickness	25 to 35mm
Length	Various	Geometry	Elbows (45 or 90°), presence of counterbores without post-welding grindings
Material type	316L or 304L austenitic stainless steel	Specific properties (if relevant)	Dissimilar metal welds or other stress sources
Surface characteristics	Rugosity < 6.5 µm No grinding, no coating, no cladding	Free comments	Presence of thermal insulation
Transported fluid			
Type of fluid	Details depend on reactor type	Temperature	Ca. 300 °C
Velocity	To be defined	Pressure	150 – 160 bar
Chemical composition	Primary coolant	Physical state	Fluid
Degradation mechanism			
Geometry of the defects	Pits, cracks	Maximal acceptable size	2 mm high, 15 mm long (approx.)

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Expected localisation	Inner surface, base metal close to the weld	Orientation	Circumferential
Kinetics	Slow (several years)		
Sources of signal interference			
Material inhomogeneity	SCC occurs in the HAZ of welds	Vibration	Events shall be classified: the vibration or structure-born noise frequency range can be suppressed
Electromagnetic	EMC requirements acc. to IEC 62003 shall be fulfilled by the equipment		
Sensor environment			
Temperature	Ca. 300 °C (plus gamma heating)	Humidity	90% non-condensing
Pressure	Normal pressure	Irradiation	Various, but significant (typically, 100 mGy/h)
Main safety constraints on hardware	No aluminium, use of fire-resistant material, fire-retardent cables	Electricity supply	Pre-Amplifiers in containment shall be supplied by a central cabinet
Data transfer	Analog		
References			
Ref: Mechanical codes	[1] Luigi Calabrese, Edoardo Proverbio, “A Review on the Applications of Acoustic Emission Technique in the Study of Stress Corrosion Cracking”, Corrosion and Materials Degradation 2021, 2, pp. 1-30.		
Ref: Scientific publications	[2] Stress Corrosion Cracking in Light Water Reactors: Good Practices and Lessons Learned, IAEA Nuclear Energy Series No. NP-T-3.13. [3] Hugh 1. Logan, Studies of the Stress-Corrosion Cracking of Low-Carbon Steels, JOURNAL OF RESEARCH of the National Bureau of Standards-C. Engineering and Instrumentation, Vol. 66C, No. 4, October- December 1962. [4] Ulla Ehrnstén, Peter L. Andresen, Zaiqing Que, “A review of stress corrosion cracking of austenitic stainless steels in PWR primary water”, Journal of Nuclear Materials, Volume 588, 2024, 154815, ISSN 0022-3115, https://doi.org/10.1016/j.jnucmat.2023.154815 .		
Operating experience	In 2007, Mihama-2 SG DMW, 1.6 mm deep crack after 11 years of operation. Very high surface hardness resulting from grinding. In 2010, Daya Bay-1 RHR, leak after 15 y operation in non-solution annealed elbow outside HAZ. CW of 30–35%. The material/elbow was manufactured in France/Germany and was not solution annealed after cold forming. Cracks were found also in Daya Bay-2. In 2020, Ohi-3 crack in PRZ spray line after 19 y of operation. Unusual (high) heat input during welding, coupled with high (residual) stresses and high surface hardness exceeding HV 300.		



	Over 200 IGSCC cracks have been confirmed by EDF since end of 2021 in the French NPP fleets.
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4.3. Additional operating experience feedback regarding stress corrosion cracking

It is worth noting that approximately 140 events mentioning SCC phenomena are recorded in the PIREX database. Not all these events necessarily led to significant leaks, but some are quite old. An extraction of PIREX (Figure 4-2) shows the number of occurrences of SCC events identified in the database each year.

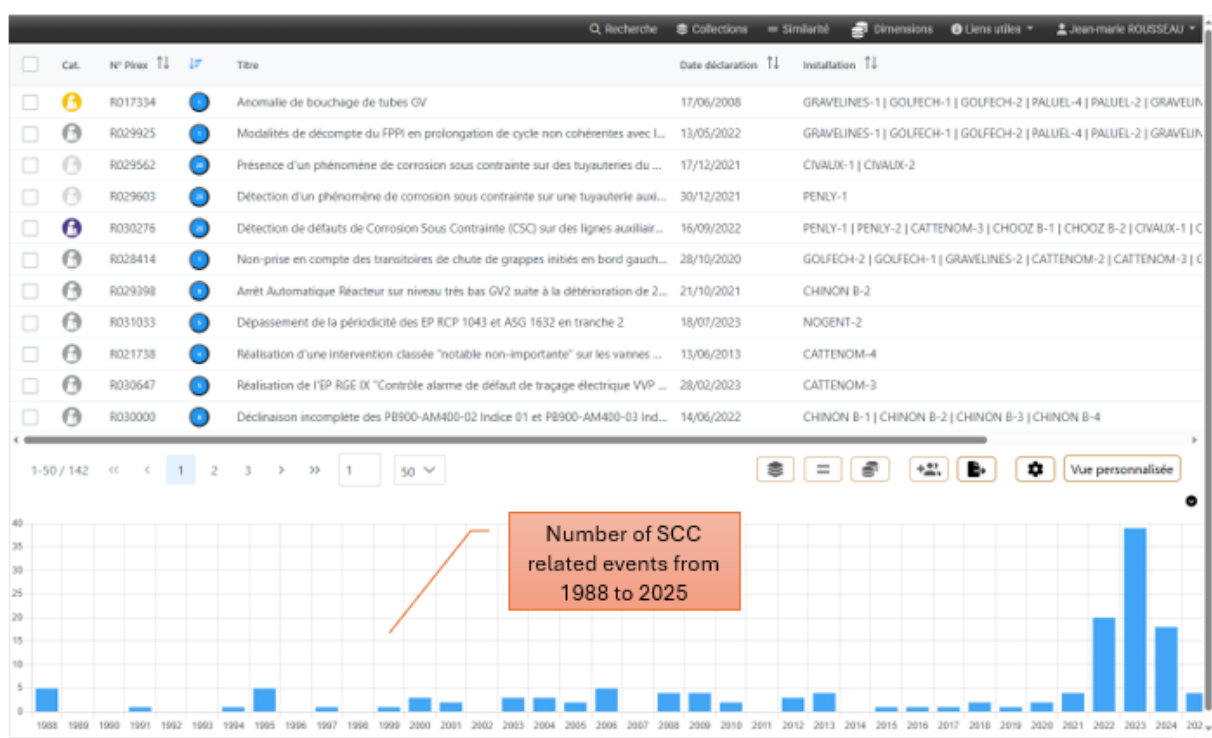


Figure 4-2: Extraction of safety events in French NPPs corresponding to the keyword "SCC".

Some of them are summarized hereafter:

Date: 2022-09-16

NPPs: CIVAUX – CHOOZ – CATTENOM – PENLY

Title: Detection of Stress Corrosion Cracking (SCC) defects on auxiliary lines and thermal fatigue defects on cold leg RIS lines

What happened: During preventive maintenance inspections on the auxiliary lines of the primary circuit in October 2021, indications were detected on welds of piping in the Safety Injection System on the cold leg. Laboratory analyses conducted after cutting out and sending the welds for examination confirmed the presence of intergranular stress corrosion cracking. This type of degradation had not been anticipated on these auxiliary lines connected to the main primary circuit.

The same type of defect was subsequently identified on the reactors at Penly, Cattenom, and Chooz.

The unexpected nature of these defects led EDF to reconsider the historical ultrasonic inspections previously performed and to define a comprehensive new investigation program. This program enabled the mapping of areas potentially affected by stress corrosion and, among other things, revealed that the only non-destructive testing techniques available in early 2022 was insufficiently reliable to detect and characterize such indications. EDF therefore initiated an additional program of destructive examinations to define the extent of the phenomenon, characterize any observed defects, and address them.

By the end of April 2023, over 200 samples—comprising more than 200 welds and several geometric discontinuities (e.g., radiographic plugs)—had been analyzed in laboratories. Some defects found in welds were the result of prior repairs, sometimes dating back to the initial construction phase. Thermal fatigue phenomena are suspected to be contributing factors.

Consequences: Despite the size of some defects detected in the welds, the rupture assumption for piping, previously considered as an initiating event, remains covered by the safety demonstration studies for LOCA.

Interest: Demonstrated limitations of non-destructive testing techniques (ultrasonics, etc.) in detecting stress corrosion cracking phenomena that are “masked” by historical weld repairs.

Date: 1988-02-18

NPP: DAMPIERRE

Title: Through-wall crack on ECCS piping

What happened: The unit was operating at 100% nominal power. A leak from a circular crack approximately 1 cm in size was discovered on a pipe located between the ECCS tank and a drain valve.

An expert analysis of the cracked pipe section concluded that the cause was external surface stress corrosion cracking. Contamination by sulfur and chlorine was observed. A risk analysis of corrosion under fibrous insulation materials containing sulfur was carried out, concluding that the sulfur present in the insulation was not harmful during normal unit operation.

Consequences: Risk of piping rupture and loss of the boron injection function.

Interest: The relevance of this event lies in its date: 1988! Stress corrosion issues are not new. Even at that time, corrosion was only detected once a leak occurred. In this regard, the FIND project focuses on two types of sensors designed to meet the needs of corrosion detection and leak localization.

5. Thermal and vibrational fatigue on the pressurizer surge line

5.1. Rationale

5.1.1. Thermal fatigue

General overview. According to [1] the reactor coolant components are subjected to several different ageing mechanisms. The surge and spray lines and several branch lines and nozzles

(charging line, safety injection lines ...) experience significant low cycle ($<10^4$ cycles) thermal fatigue caused by thermal stratification and cycling, and high cycle fatigue caused by striping.

Thermal fatigue is the major ageing mechanism for surge, spray and branch lines and their nozzles that are subject to thermal transients during plant startup/shutdown, thermal stratification, thermal shock, turbulent penetration, and thermal cycling. Fatigue design analyses of these piping components included the design basis transients. However, these initial analyses did not include thermal stratification phenomena present in the surge and spray lines, and thermal cycling phenomena present in the branch lines; these phenomena were discovered after the plants were placed in operation.

Thermal stratification phenomena. Thermal stratification in the surge line of pressurized water reactors (PWRs) is a significant factor influencing the structural integrity of the piping system. It occurs when cooler, denser water from the hot leg flows beneath the warmer, lighter coolant from the pressurizer during insurge, and vice versa during outsurge (see Figure 5-1). The likelihood of stratification depends on the Froude number, which measures the balance between inertial and buoyancy forces. When flow velocities are low and temperature differences are high, stratification is more pronounced, particularly during reactor heat-up and cooldown phases. This phenomenon can lead to uneven temperature distributions along the surge line, affecting straight pipes, nozzle ends, and elbows.

The temperature differences caused by stratification result in non-uniform thermal expansion, generating significant axial bending and hoop stresses within the piping system. These stresses, particularly at the interface between hot and cold fluids, fluctuate with flow rate changes, leading to fatigue damage over time. In cases where stratification extends over long horizontal sections, it can cause large cyclic stresses, potentially leading to significant structural deformation known as global thermal stratification.

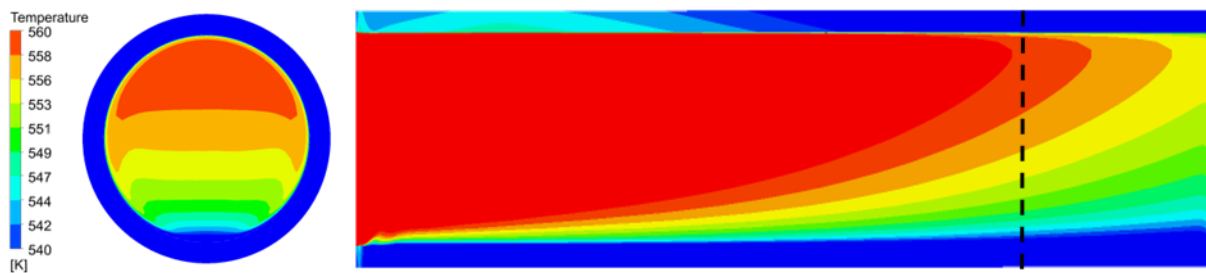


Figure 5-1: Thermal stratification phenomena.

When stratification occurs in shorter sections, cyclic thermal stratification can still cause fatigue damage without visible displacement. Even during normal full-power operation, some degree of stratification persists due to low surge line flow rates, though the resulting stresses are lower than those during heat-up and cooldown. The risks associated with thermal stratification prompted the U.S. Nuclear Regulatory Commission (NRC) to issue Bulletin 88-11, recommending inspections of PWR surge lines for fatigue damage and deformation [2] [3]. Utilities were advised to update their stress and fatigue analyses to ensure compliance with safety regulations, highlighting the critical need for monitoring and mitigating thermal stratification effects in nuclear plant operations. Currently, for the U.S. PWR fleet, the cyclic stratification is managed by guidance provided in MRP-146 [7]. The requirement to evaluate global and local stresses is presented in European documents [4] too.

For former USSR reactor design, the pressurizer surge line is susceptible to significant additional stresses due to the thermal stratification for WWER-440 especially. The available monitoring data from different Units confirms a significant temperature gradient along the pressurizer surge line cross-section under certain operations modes. The pressurizer surge line of WWER-1000 although to a lesser extent than in the case of WWER-440, notwithstanding is susceptible to additional stress due to the thermal stratification. The available monitoring data confirms the temperature gradient along the pressurizer surge line cross-section under certain operations modes.

For Ukrainian NPPs the additional stresses in nuclear power plants piping from thermal stratification must be considered for static strength assessment and low-cycle fatigue. Fatigue assessment is performed according to [6]. To monitor thermal stratification and then low-cycle fatigue assessment during LTO, the Conceptual Technical Solution [5] has been developed in Ukraine. It includes requirement for monitoring system and numerical analyses of stresses depending on measured values of temperature.

Generally, based on the worldwide LWR NPP units' operation experience, stratification is mainly observed in sections of pipelines not only for pressurizer surge line but for the following systems:

- Emergency core cooling (ECCS);
- Separate sections on hot and cold lines of MCL;
- SG normal feed water.

Currently, state-of-the art online fatigue monitoring systems are available to be used for real time counting of transients. Most well-known existing industry implemented solutions are FAMOS (FRAMATOME), SYSFAC (EDF), FATIGUEPRO (EPRI, SIA Inc.) and OPEDAS.

The developed systems are based on measurement temperature in specific cross sections of pipeline and further fatigue calculation (Figure 5-2). So, measuring the temperature of the outer surface of the pipeline itself is of no significant interest today, but a comprehensive, unified solution that allows to estimate (measure directly) the target parameters such as stresses and vibrations significantly increases the importance of such a system. At the same time, taking into account the specifics of the NPP regulatory framework, the system shouldn't damage the surface of pipelines or thermal insulation.

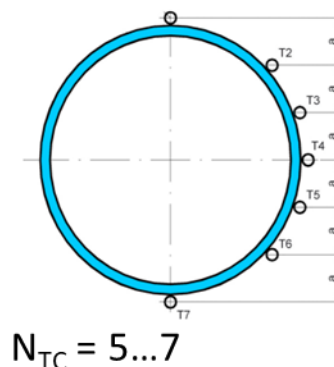


Figure 5-2: Thermocouples for thermal stratification monitoring.

Other issues that need to be resolved within Digital Twin for high energy pipes development include:

- Method of installing thermal sensors (surface, volumetric, cascade).
- Reliability of deformation readings under significant vibration accelerations.
- Method for absolutizing signals from strain sensors.
- Method for determining the flow rate of two-phase medium.
- Design features (materials, degree of system serviceability).
- Optimization of the sensors number for reliable construction of smoothed functions of the azimuthal temperature and deformation distributions.
- Stratification of a two-phase medium (method of calculating the main parameters from the medium side) requires validation.

5.1.2. Vibrational fatigue:

Vibratory fatigue is a high cycle mechanical fatigue mechanism which may affect any PWR primary piping system material; however, the degradation sites have been generally limited to small-diameter piping and socket weld locations in particular [1]. In these cases, vibratory fatigue cracking is caused by pump- or cavitation-induced pressure pulsations if an excitation frequency coincides with the structural (natural) frequency of the piping, as determined by the piping dimensions and layout. Pump-induced pressure pulsations occur at distinct frequencies (multiples of the pump speed) and are associated with positive displacement pumps such as those which may be found in charging systems. Cavitation induced pulsations consist of broadband frequencies (a band with a wide range of frequencies) and are associated with cavitating components such as an orifice in the letdown system.

Field experience has shown a variety of failure behaviour depending on the location of the vibratory fatigue cracking. The typical failure mode is pipe rupture for cracking in socket welds. In the absence of socket welds, the base metal near the small-diameter piping attachment becomes the most susceptible to such cracking and the typical failure mode becomes leakage.

Digital Twin for high energy pipes has two aims regarding vibrations measurement. First one is direct and provides traditional spectral analysis with further vibrational fatigue assessment. The second one is indirect: the sensors of vibration are used only as one of indicator for embedded software to be sure that obtained stresses and temperature distributions are processed correctly.

Fatigue monitoring program overview. To more precisely define the thermal stresses leading to fatigue usage and to provide an automated method to track the fatigue usage being accumulated at critical locations, several firms have developed online fatigue monitoring systems that directly query the plant process information system. For each critical location chosen to be monitored, the vendor has developed a model to compute the stress and fatigue usage.

5.1.3. Performance of the monitoring system

The objective of the monitoring system is to control the ageing of primary circuit components (especially the pressurizer surge line) due to low-cycle fatigue (approximately 10^4 - 10^5 cycles during their useful lifetime). Metallic materials employed in NPPs can suffer from low-cycle fatigues if they are confronted with stresses as low as 10% of their yield stress. So, the uncertainty of the monitoring system should not be higher than this value (10% of the yield stress). A lower measurement uncertainty (typical 1 to 5% of the yield stress) is desirable to increase the accuracy of models, especially if the monitoring system is used for high-cycle fatigue.

The detection range of the monitoring system does not need to be higher than 3 times the yield stress. Such levels of stresses are not expected during the normal operation of NPPs.

Mechanical stress cannot be measured directly: it can only be reconstructed indirectly from other physical measures (strain, temperature, acceleration...), based on models and associated assumptions. The monitoring system should cover the different sources of mechanical stress identified above.

The uncertainty reached in laboratories for primary conditions (VTT and ASNR in the EURATOM INCEFA SCALE project) is approximately 10% of the yield stress. For industrial pipes at ambient temperature, measurement uncertainties could be lowered to 1–2% of the yield stress. These results were obtained thanks to extensometers satisfying the 0.5 or 0.2 standard of the ISO 9513 norm. Similar uncertainties must now be reached for industrial components in primary conditions (350 °C and 16 MPa).

The critical location in surge line was identified as the safe end, where the nozzle and piping connect. According to the sensors installed on the safe end, the feature stress during normal operation can be –10 to 20 MPa and the typical strain range is -0.03 to 0.35%. Furthermore, the strain rate at the critical location, during the same recording period can be in the range of $\pm 0.0004\%/s$. The INCEFA SCALE project highlighted the large influence of the strain rate on material ageing: monitoring this magnitude would therefore be of high interest.

Component characteristics for different types of PWR

NPP type	Diameter, mm	Thickness, mm	Material type	Temperature, °C
Combustion Engineering	273 – 406	28 – 40.5	SS CF-8M replaced by SS 347	343
Babcock & Wilcox	273 – 406	29 – 40.5	SS 316	
Westinghouse plant AP1000	273 – 406	30 – 40.5	SS 316	360
Japanese design MHI	406	40.5	SS 316	360
German design (Areva NP - Siemens)	355	36	SS 347 mod	362
Former USSR design WWER-440	245	19	08Kh18N10T	325
Former USSR design WWER-1000	426	40	10GN2MFA	325

Configuration of piping for different types of PWR can be found in the Appendix of this document.

5.2. Specifications

General information			
Component(s) concerned	• Pressurizer Surge Line; • Emergency core cooling (ECCS); • Separate sections on hot and cold lines of MCL (Main Coolant Line); • SG normal feed water.	Degradation mechanism	Low-cycle fatigue In some cases, high-cycle fatigue
Type of measure	Reconstruction of the time-dependent mechanical stress field.	Reactor designs concerned	PWR
Lead contributor	IPP	Other contributors	Framatome, VTT
Current methods applied	FAMOS (Framatome) SYSFAC (EDF) FATIGUEPRO (EPRI, SIA Inc.) OPEDAS	Limitations of current methods	The developed systems are based on measurement temperature in specific cross sections of pipeline and further fatigue calculation. They don't measure target parameter (stresses) directly
Component characteristics			
Diameter	245 – 406 mm	Thickness	19 – 45 mm
Length	See configuration of pipeline on Fig.	Geometry	Straight, elbows
Material type	See Table below		
Surface characteristics	Outer coatings		
Transported fluid			
Type of fluid	Water	Temperature	Up to 400 °C
Velocity	<10 m/s	Pressure	15.5 – 15.69 MPa
Chemical composition	Primary coolant (borated water with LiOH additive)	Physical state	Liquid
Stresses to be monitored – one table for each stress monitored			
Type of stress	Thermal cycling, pressure variations, flow-induced vibrations		
Detection range	Between 10% and 300% of the material yield stress, with a resolution of 10% of the yield stress. Lower detection limit and resolution (1 – 5% of the yield stress) would be an advantage. Strain rates in the range +/-4.10 ⁻⁶ s ⁻¹ , with a resolution of 1.10 ⁻⁵ s ⁻¹		
Frequency	0 – 60 Hz for vibration monitoring ≈1 measure per second for deformation monitoring		

Sources of signal interference			
Vibration	Significant		
Sensor environment			
Temperature	15 – 60 °C (air) 350 – 400 °C (surface)	Humidity	<90%
Pressure	0.08 – 0.1 MPa	Irradiation	≈28 mGy/h
Data transfer	Wireless, analogical, numerical	Electricity supply	Limited availability of classical plugs. Self-powered sensors would have a key advantage.
References			
Ref: Scientific publications	J. Rudolph, S. Bergholz, W. Kleinöder, N. Wirtz / AREVA s fatigue concept (AFC) - an integrated and multidisciplinary approach to the fatigue assessment of NPP components / SMiRT 20 - Espoo (Helsinki), Finland. August 9-14, 2009 https://repository.lib.ncsu.edu/handle/1840.20/23638?show=full Boros, A. Aszódi / Analysis of thermal stratification in the primary circuit of a VVER-440 reactor with the CFX code / Nuclear Engineering and Design, Volume 238, Issue 3, 2008, P. 453-459, https://www.sciencedirect.com/science/article/pii/S0029549307003329		
Ref: National and international standards	[1] IAEA-TECDOC-1361. ASSESSMENT AND MANAGEMENT OF AGEING OF MAJOR NUCLEAR POWER PLANT COMPONENTS IMPORTANT TO SAFETY IAEA, VIENNA, 2003 [2] NRC, 1988a. NRC Bulletin No. 88-08 (Supplement 1, 2, 3): Thermal Stresses in Piping Connected to Reactor Coolant Systems, http://www.nrc.gov. [3] NRC, 1988b. NRC Bulletin No. 88-11: Pressurizer Surge Line Thermal Stratification, http://www.nrc.gov. [4] NEA, 2005. NEA CSNI, CSNI Integrity and Ageing Working Group. Thermal Cycling in LWR Components in OECD-NEA Member Countries, NEA/CSNI/R(2005)8. [5] KTR-S.1234.03-247.15 Conceptual Technical Solution for Monitoring and Accounting of Temperature Stratification in the Surge Line of the Pressurizer in VVER-1000 Nuclear Power Plant Units [6] PNAE G-7-002-86. Strength Calculation Standards for Equipment and Pipelines of Nuclear Power Installations. Moscow: Energoatomizdat, 1989. – 524 p. [7] ELECTRIC POWER RESEARCH INSTITUTE, "Materials Reliability Program: MRP-146/MRP-146S Implementation Survey Summary Report (MRP-275)," EPRI, Palo Alto, 2010.		

5.3. Additional operating experience feedback regarding fatigue

It is worth noting that approximately 520 events mentioning “vibrational or thermal fatigue” phenomena are recorded in the PIREX database. Not all these events necessarily led to significant leaks, but some are quite old (Figure 5-3).

Deliverable 2.1 – Specifications for the online damage monitoring systems Version 1 (19/06/2025)

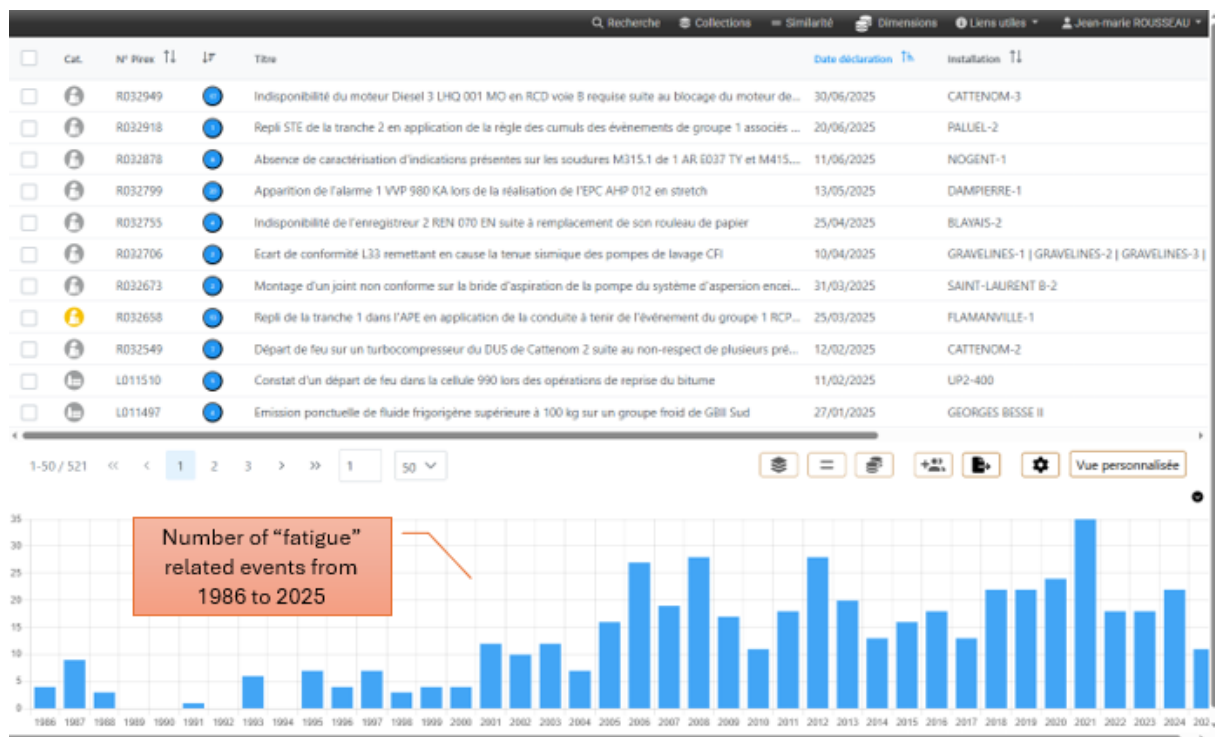


Figure 5-3: Safety events with the occurrence of the word “fatigue” (thermal, vibrational) encountered in French facilities.

Some of them are summarized hereafter:

Date: 1998-12-05

NPP: CIVAUX

Title: Breach on the RHRS at the bypass of the cooler

What happened: A leak of water (estimated at 30 m³/h) was discovered on a pipe of the reactor shutdown cooling system (Residual Heat Removal System, RHRS). The leak was caused by multiple cracks—displaying a “crazing” pattern—emerging on the outer surface of an elbow in the piping. After removal of the affected elbow, dye penetrant testing revealed cracks on the internal surface of the tee.

The RHRS piping, made of austenitic stainless steel, was subjected to significant temperature fluctuations downstream of a mixing point between two fluid streams (one at 180 °C, the other at 20 °C), under a pressure of 27 bar. The leak occurred after only 1,500 hours of operation.

Consequences: LOCA – The 2,800 m³ threshold of the Refuelling Water Storage tank was exceeded.

Interest: An unanticipated thermal fatigue phenomenon in a hot/cold fluid mixing zone (known as the “Farley-Tihange” phenomenon). Early detection of the cracks could have prevented the situation from evolving toward a leak.

Date: 2004-01-22

NPP: CATTENOM



Funded by
the European Union

Title: Transition to fallback state following an RCCS event

What happened: The reactor was in cycle extension, operating at 94% of nominal power. The transition to the fallback state was initiated due to the unavailability of the Reactor Component Cooling System (RCCS). This diagnosis was made following confirmation of a leak at the weld of a tapping point on the RCCS. The leak was caused by partial cracking at the weld bead of the connection between the RCCS tapping and the excess bleed supply line.

The pipe support had become detached from its anchor point. The leak was initially detected on 2003-12-07, but the risk of RCCS unavailability was only identified on 2004-01-20. Additionally, the absence of instrumentation on this pipework made it impossible to retrieve historical data that could have helped identify the stresses that led to the degradation of the pipe and its support.

Consequences: The leak could be compensated by water make-up. In the event of a full rupture at the weld, it would have resulted in the loss of RCCS water inventory.

Interest: The unavailability of the RCCS was identified with a 43-day delay. The crack remains unexplained due to the lack of instrumentation on this section of piping.

Date: 2006-06-21

NPP: FESSENHEIM

Title: Detection of a thermal fatigue cracking phenomenon on the steam supply line of the AFW pump and its inlet components

What happened: During an inspection of the Auxiliary Feedwater (AFW) valves, visual anomalies were observed. A dye penetrant test was then performed, revealing linear indications that were characterized by replication as thermal fatigue cracking. Investigations carried out on the entire AFW steam supply line of the turbo generator showed that a thermal fatigue cracking phenomenon had occurred upstream of the AFW turbine, mainly affecting the low points (lower manifold and drain ports of components located at low points). Generalized corrosion was also observed in areas where water stagnates.

The operation of the AFW turbine since the unit's startup—particularly during periodic testing with partial conditioning and the presence of water in the steam lines—caused severe thermal stresses on the most exposed components. The corrosion is due to the presence of water in the low points, caused by upstream/downstream valve leakage and the malfunctioning of the automatic drain system of the line.

Consequences: The loss of integrity of the AFW and the Emergency Supplied Distribution System (a common-mode failure) leading to the loss of both components has been analysed. This failure is covered by the design's functional redundancy, which ensures that the resulting situation does not compromise the safety demonstration for any design-basis accident.

Interest: Expert analysis of the AFW regulating valve body indicates a differential aeration corrosion phenomenon. Beyond the undetected corrosion, leakage from isolation valves remains a recurring issue.

Date: 2009-06-12

NPP: PALUEL

Title: Closure of the DELAS valve (Steam Isolation Valve of the Steam Generator) at 100% of nominal power leading to Reactor Automatic Shutdown.

What happened: The unit was in production when the DELAS valve unexpectedly closed. This transient led to the opening of the two safety valves of the Main Steam System, causing an automatic reactor trip due to very low steam generator level. The rupture and cracking observed on the valve stem were found to result from a slow-kinetics vibratory fatigue phenomenon.

Examination of the fillet radii in the groove area revealed manufacturing differences between valve stems: no specification of groove fillets on the manufacturer's design drawings; machining differences between original parts and subsequent supply batches (from 1990 onwards).

It is worth noting that EDF's analysis refers to a similar event that occurred in Slovenia.

Consequences: In case of failure of the Reactor Automatic Shutdown sequence, the ATWS (Anticipated Transient Without Scram) backup would have triggered the drop of the control rods and the start-up of the AFW system. The aggravating accident considered in combination with this transient (e.g., stuck-open steam generator safety valve) would be managed as an external Steam Line Break (SLB) outside the containment in the accidental procedures.

Interest: An example of vibratory fatigue-induced corrosion on a component other than piping, where a relatively fast unit shutdown is required, disrupting reactor operation.

Date: 2025-03-25:

NPP: FLAMANVILLE

Title: Reactor shutdown in accordance with operating procedures due to a primary circuit leak rate exceeding 2300 L/h

What happened: During reactor restart following a partial outage, a steam leak was detected in the reactor building. The initial estimated primary leak rate was 4979.9 L/h, triggering the application of accident operating procedures. Following a local inspection, the primary leak was located at the nozzle of a tapping point on the Reactor Coolant System (RCS) and re-estimated at 1200 to 1400 L/h.

Ongoing investigations suggest that a vibratory fatigue phenomenon may have caused weakening of the piping.

Consequences: In the event of a full rupture of the piping, the leak would have been classified as a small-break LOCA (Loss of Coolant Accident) with a diameter less than 25.4 mm.

Interest: The affected piping was not initially considered noteworthy, as it was not identified as a "sensitive tapping point" (due to the absence of known vibrations). However, vibrations have since been observed under specific configurations (on RHRS). This phenomenon may be generic.

Furthermore, other events highlight the diversity of systems affected by vibratory fatigue phenomena. For example: on the Emergency Core Cooling System (CHINON, 2000) and (SAINT-LAURENT, 2024); on the Safety Containment Spray Systems (GRAVELINES, 1995) and (NOGENT, 2021).



6. Double contained wall penetration of the sump recirculation line

6.1. Rationale

According to the general safety rules, the piping penetrations of the containment must be equipped with two isolation devices, one inside and other outside the containment.

However, for the French NPPs, the suction lines of the Safety Injection System (SIS) and Safety Containment Spray System (SCSS) are connected directly to the containment sump and are equipped with only one isolation valve outside the containment. So, a leak in the non-isolated part of piping (between the sump and the isolation valve) would present the containment bypass. To remedy this problem the piping between the sump and the first isolation valve is designed in double walls. The SIS (SCSS) piping of the system connected to the sump is surrounded by the sheath whose function is to confine the water in the event of a leak in the non-isolated part of piping. In this case the external wall plays the role of extension of the third barrier of confinement.

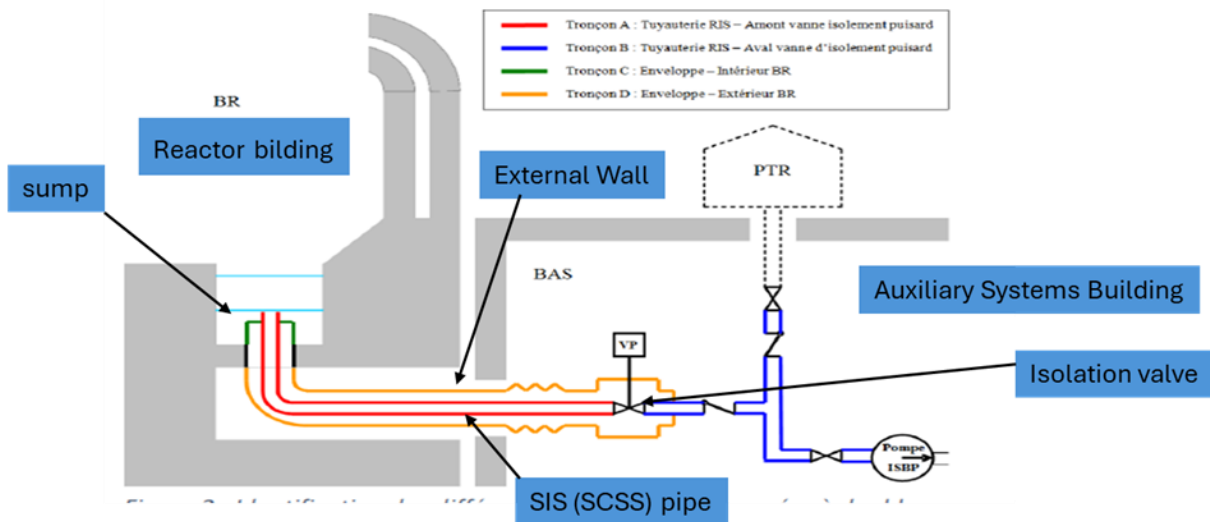


Figure 6-1: SIS piping of the system connected to the sump.

But, the main part of this double contained wall penetration, especially the space between the walls, is not accessible for inspection during the normal operation. One-off and non-exhaustive inspection is only possible when the reactor is in cold shutdown conditions.

So, it is impossible to verify and warranty the availability and integrity of this specific device in real time, which is a problem for the safety demonstration.

The safety interest is to detect in real time (asap) the loss of this specific containment penetration integrity. So, in the frame of the FIND project, the goal is to find ways allowing to check the integrity and, more precisely, to detect the leakage (into double-walled area) of this double-walled containment penetration during normal operation.

The system is located in the nuclear auxiliary building, out of the reactor building.

6.2. Specifications

General information			
Component(s) concerned	SIS/SCSS Suction line from the reactor sumps: double-walled section	Degradation mechanism	Leakage of the inner pipe collected in the sheath.
Safety function	Recirculation function. Extension of the 3 rd confinement barrier.		
Type of measure	Continuous tightness control	Reactor designs concerned	Western-type PWRs (Westinghouse, Framatome)
Lead contributor	ASNR	Other contributors	Vuez
Current methods applied	Endoscopy and UT thickness measurement during service shutdown	Limitations of current methods	No real time assessment of the pipe tightness. Unclear origin of the water collected in the pipe (condensation or leakage)
Component characteristics			
Diameter	Pipe: 508 to 812 mm Sheath: 668 to 1007 mm	Thickness	Pipe: 7.92 to 10 mm Sheath: 10 to 20 mm
Length	5 to 10 m	Geometry	Elbow
Material type	Pipe: Z2 CN 18 10 (French designation of 304L) Sheath: A42 or Tu42 steel	Specific properties (if relevant)	No
Transported fluid			
Type of fluid	Borated water	Temperature	10 to 35 °C (during the normal operation)
Velocity	Static	Pressure	Pipe: 0.6 bar abs. Sheath: 1 bar abs.
Chemical composition	Primary water	Physical state	Liquid
Detection limits			
Amount of water to be detected	Minor leakages characteristic of a component degradation (drop by drop).	Response time	1 hour
Sources of signal interference			
Humidity sources	Condensation, possible leakage from the stuffing	Vibration	-

	box of the isolation valve		
Electromagnetic	-	Others	-
Sensor environment			
Temperature	10 to 35 °C	Humidity	≤35%
Pressure	Atmospheric	Irradiation	Negligeable
Main safety constraints on hardware	No degradation of the tightness of the system.	Electricity supply	Yes
Geometrical constraints	The pipe is 5 to 10m long and can only be accessed by a ½ inch fitting at its end.		
References			
Operating experience	The event occurred in Gravelines NPP in 2006, summarised within the section related to thermal fatigue could illustrate this use-case.		

7. Crosscutting issues

7.1. Indirect impact of unexpected material degradation on safety and radiation protection

Events involving Stress Corrosion Cracking (SCC) phenomena on primary circuit components reveal direct impacts on plant safety. Some of the events identified led to reactor trips. According to operating procedures, many other events required an unplanned plant shutdown if the failures could not be repaired within a specified delay, which was not always achieved.

Moreover, a significant number of events also show indirect impacts (collateral damage):

- Their in-situ detection and the resulting repairs often raise issues related to radiation protection for workers.
- Their late detection (often upon the appearance of a leak) disrupts operations (e.g., delays in restart schedules) or disorganizes maintenance activities, leading to other significant events or non-qualitites.

These direct and indirect consequences of the lack of effective detection methods justify innovation in this area, beyond the immediate safety concerns (such as component failure).

7.2. Resistance to radiation of instrumentation implanted in the reactor building

Operators like EDF define several thresholds to test the resistance to irradiation of components implanted in the reactor building, depending on their localisation and expected lifetime. These levels include a correction factor of 4 to account for the use of much higher dose rates in irradiators, as compared to real industrial conditions. In practice, a campaign of 1 month in an irradiator can simulate decades of ageing. This correction factor does not apply to evaluate the resistance in accidental conditions, for which the dose rate is similar in real and experimental conditions.

The following reference levels are recommended (more detailed information is confidential). They already include the correction factor:

- 5 kGy,
- 35 kGy,
- 70 kGy,
- 140 kGy,
- 250 kGy.
- 500 kGy,
- 1,000 kGy,
- 1,500 kGy.

The last value is defined only for components used in accidental conditions, which must resist to the cumulative effects of normal irradiation ageing and of accidental conditions. The effect of β -rays should be considered in some cases for accidental transients; nevertheless, it will not be possible in the frame of FIND due to the experimental capacities of consortium members.

These reference levels cover a large variety of localisations and lifetimes (the exact correspondence is confidential). They will be used during irradiation tests in IRMA (γ -ray irradiator): good operation tests will be performed when they are reached (possibly with additional intermediate tests).

Industrial good practices recommend testing the influence of the dose rate besides the total dose received, especially for electronic components. This can easily be investigated in IRMA by moving the samples closer to the sources.

7.3. Data transmission and electricity supply in the reactor building

One important safety function of the reactor building is to confine radioactive materials. In addition, main electric cabinets and electronic devices are situated out of the containment building, to protect sensitive materials from radiations. Therefore, all electricity supply and data collection wires must pass through complex wall penetrations, whose number is limited. Most of the penetrations are safety-classified devices that must respect some provisions (redundancy, no signal multiplexing...). Safety constraints are more stringent for accidental instrumentation (to be discussed in the deliverable D2.3). As a consequence, remaining wire sockets are extremely rare, and their use must be duly justified.

Wall penetrations are typically equipped with shielded cables, coaxial cables and optic fibres. The RS-485 standard can be used to transmit digital signals. Electricity supply usually includes 230V AC, 380V AC, 24V DC and 48V DC. Some secondary cabinets exist in the reactor building to connect low-energy systems.

The reactor designs allowing access to the reactor building during normal operation (like the EPR) offer more flexibility for the implantation of electronic components. Usually, they can withstand the radiation levels of the accessible zones of the reactor building. Manual data collection in the reactor building is theoretically feasible, eliminating the need to use wall penetrations.

7.4. Intrusiveness of monitoring technologies

Monitoring technologies must not affect the safety function of the materials on which they are installed, especially their mechanical integrity. So, they must not rely on destructive or highly intrusive coupling with the components. Rules may differ between countries and operators.

Some may accept qualified welding or gluing procedures, while others prohibit any modification of the pipe. In this case, only mechanical coupling like straps or clamps may be accepted.

8. Conclusion

Deliverable D2.1 gives precise specifications for FIND project partners to develop industry-relevant acquisition chains. Considering operational constraints from the start will shorten the time-to-market of our solutions. Moreover, this public document will help other technology providers beyond FIND to understand the constraints of the nuclear sector, which is so difficult to enter.

Industrial constraints have been discussed among partners to account for the different national contexts. They cannot be fully representative of all NPPs around the world.



Appendix

Configuration of piping for different types of PWR are shown in the following figures:

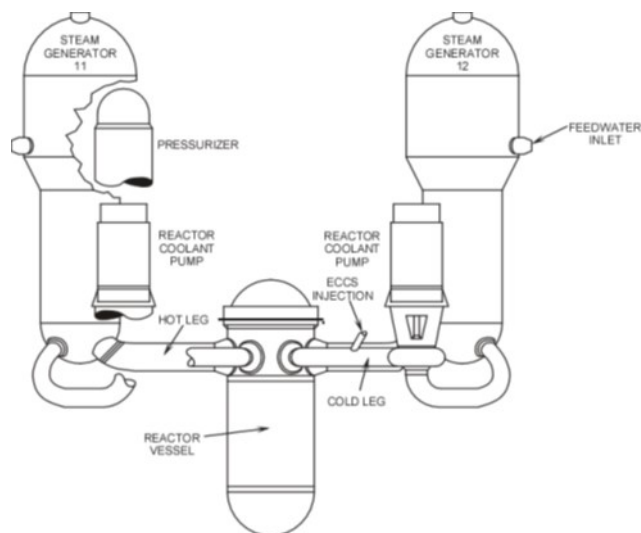


Figure 0-1: Reactor Coolant System of Combustion Engineering NPP.

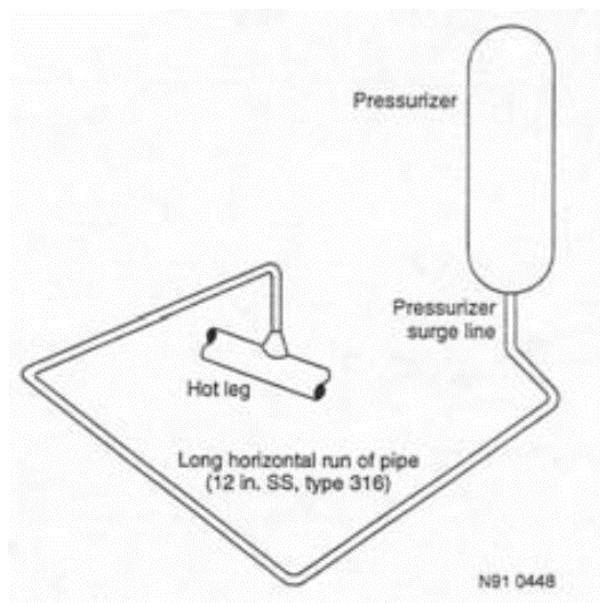


Figure 0-2: Typical combustion engineering pressurizer surge line layout Combustion Engineering NPP.

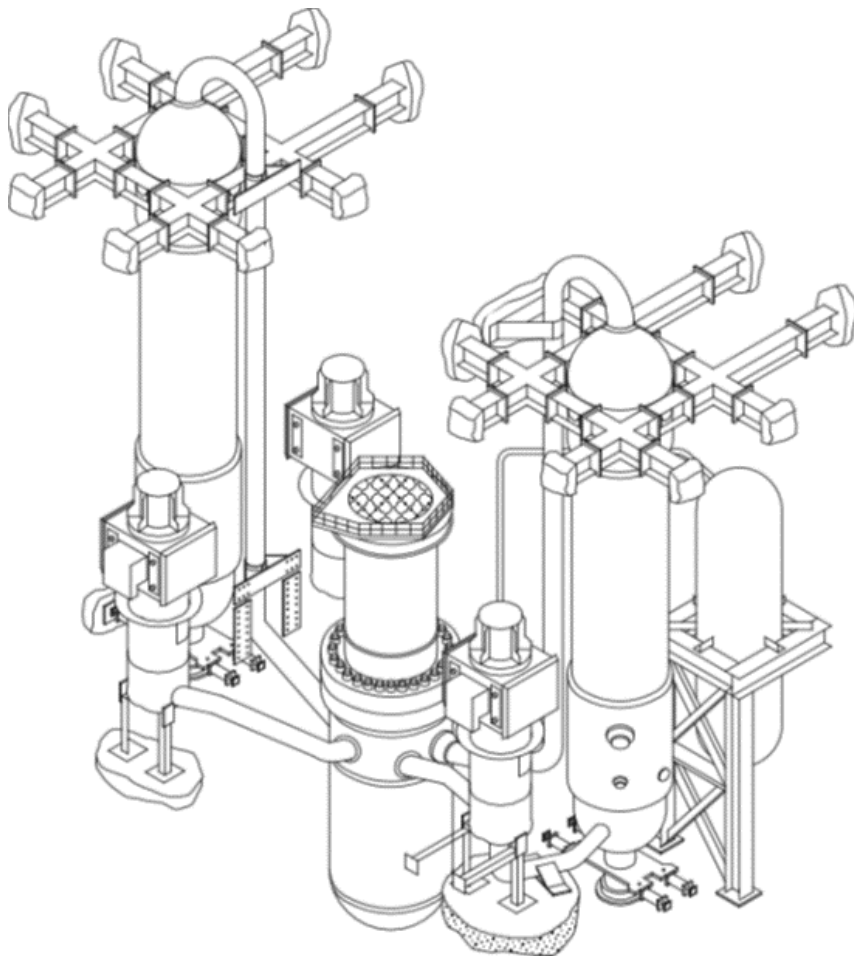


Figure 0-3: Reactor Coolant System of Babcock & Wilcox NPP.

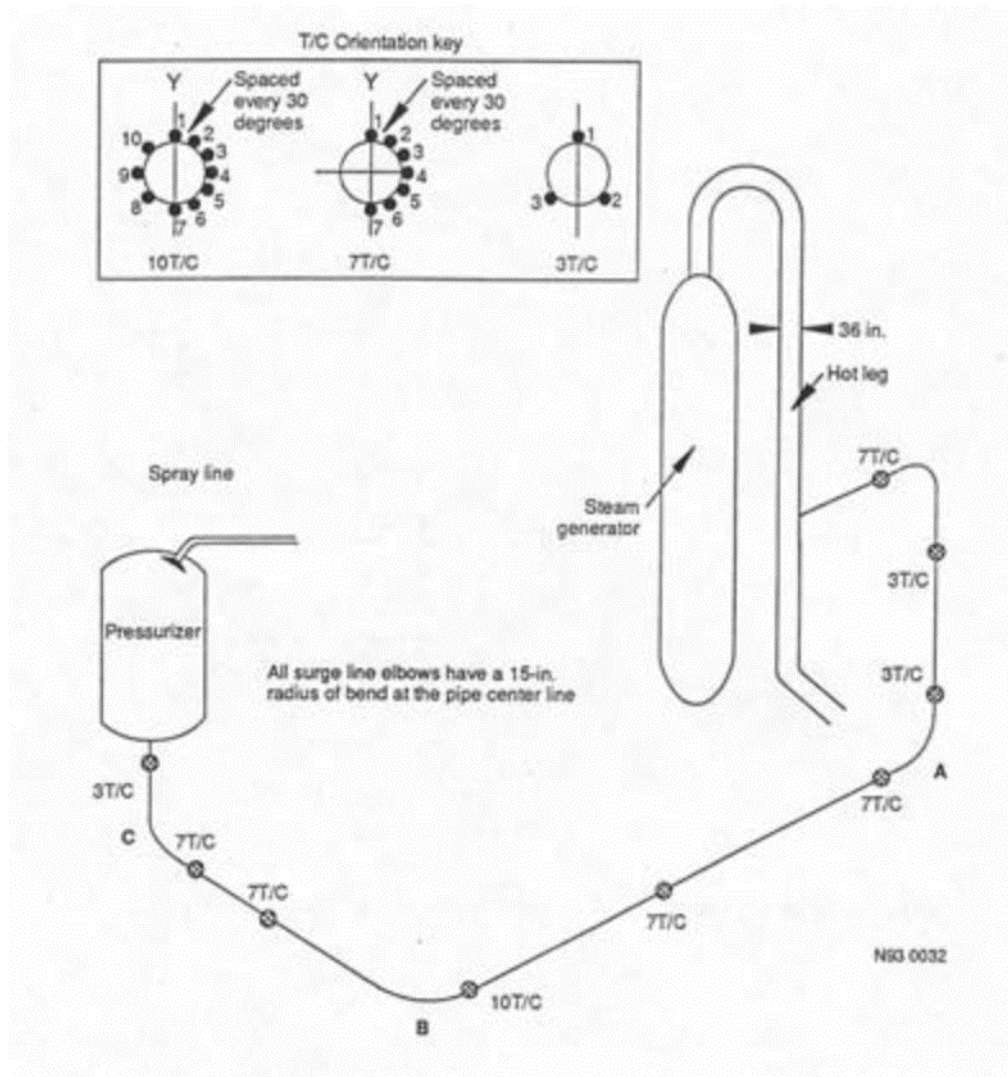


Figure 0-4: Typical Babcock & Wilcox pressurizer surge line layout. Locations of thermocouples for monitoring outside-surface temperatures are shown.

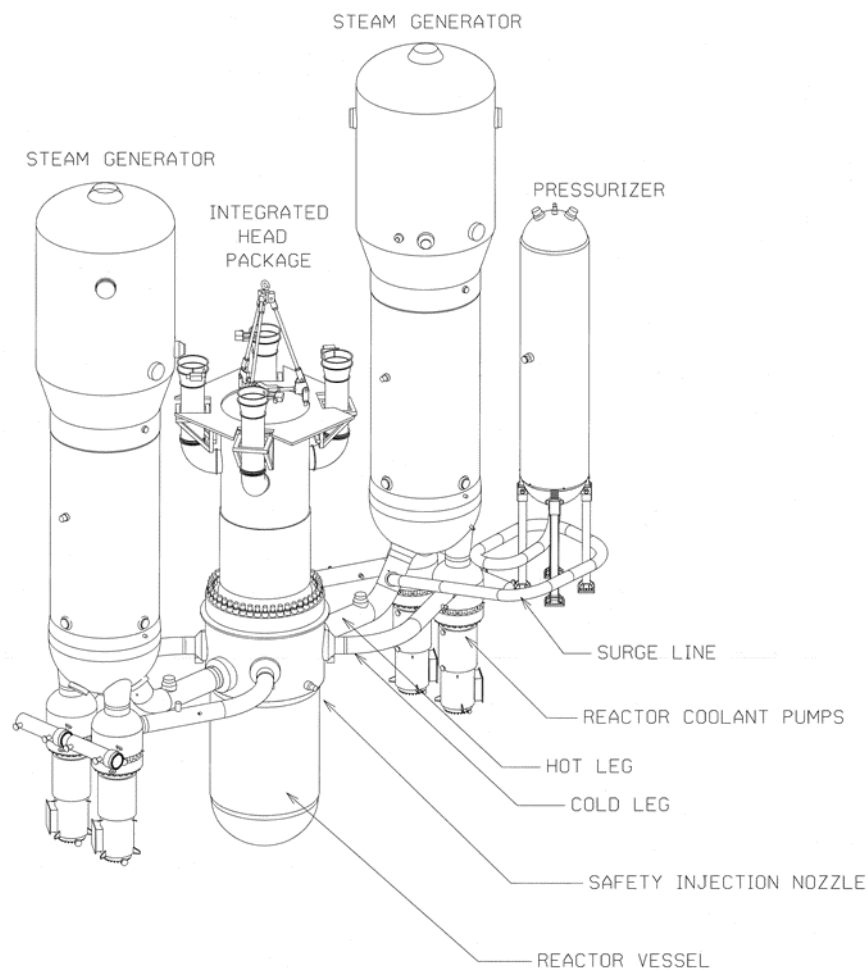


Figure 0-5: Reactor Coolant System of Westinghouse NPP.

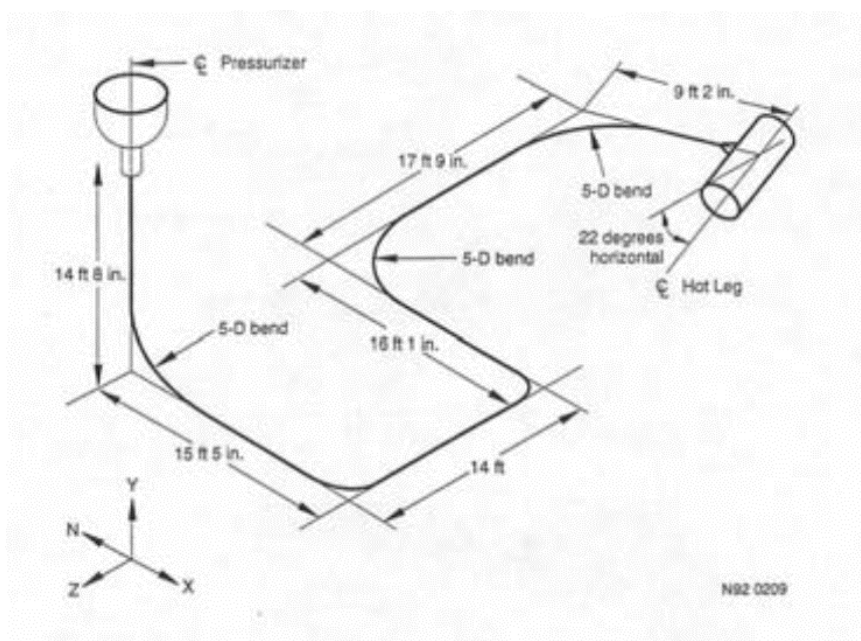


Figure 0-6: Westinghouse pressurizer surge line layout.

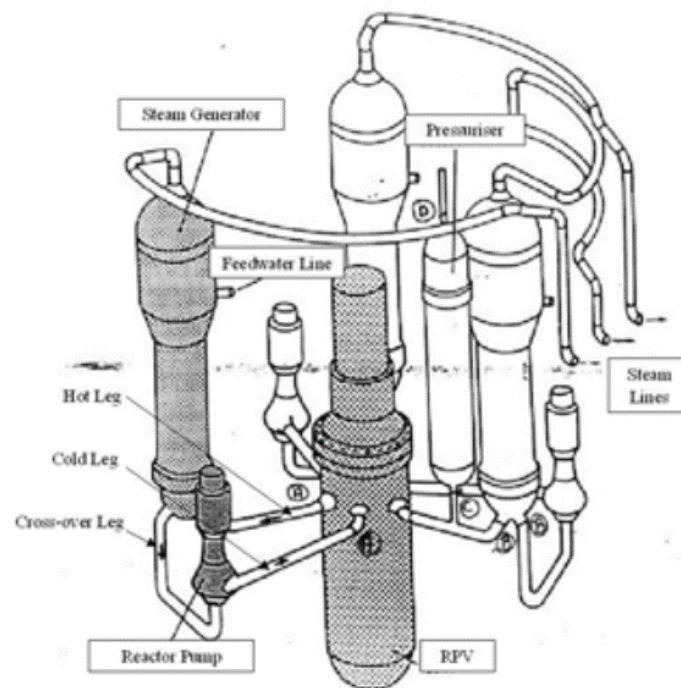


Figure 0-7: Sketch of a typical French PWR reactor coolant loop.

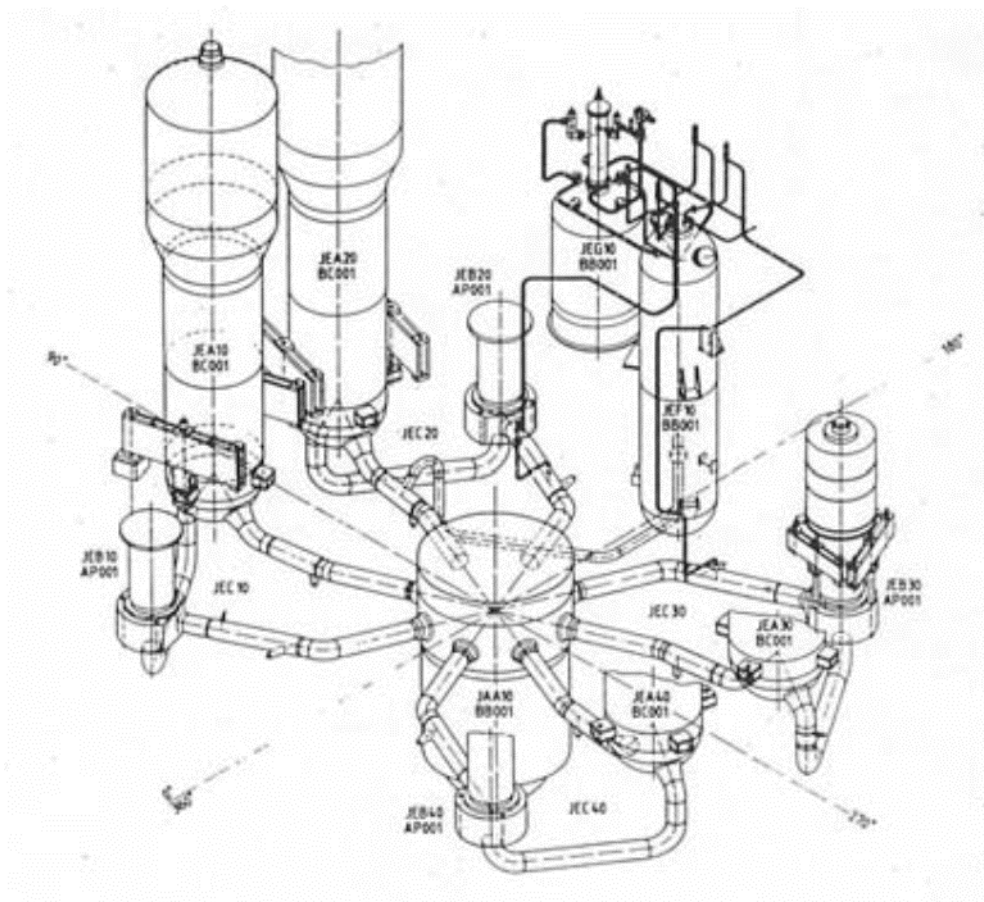


Figure 0-8: Schematic of a typical German PWR primary coolant system.

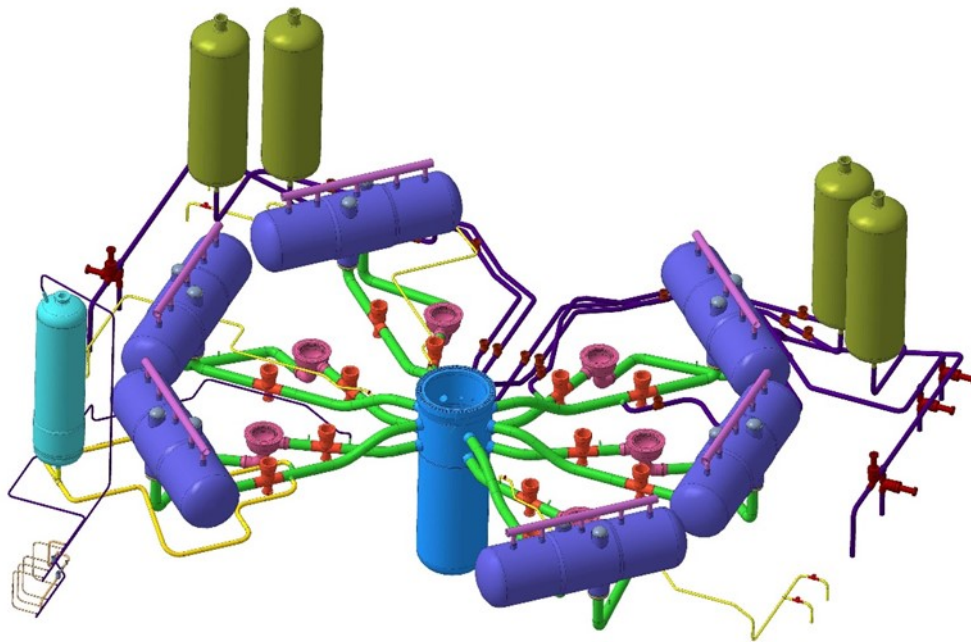


Figure 0-9: Schematic view on the primary system layout of a WWER 440, Model 213 reactor.

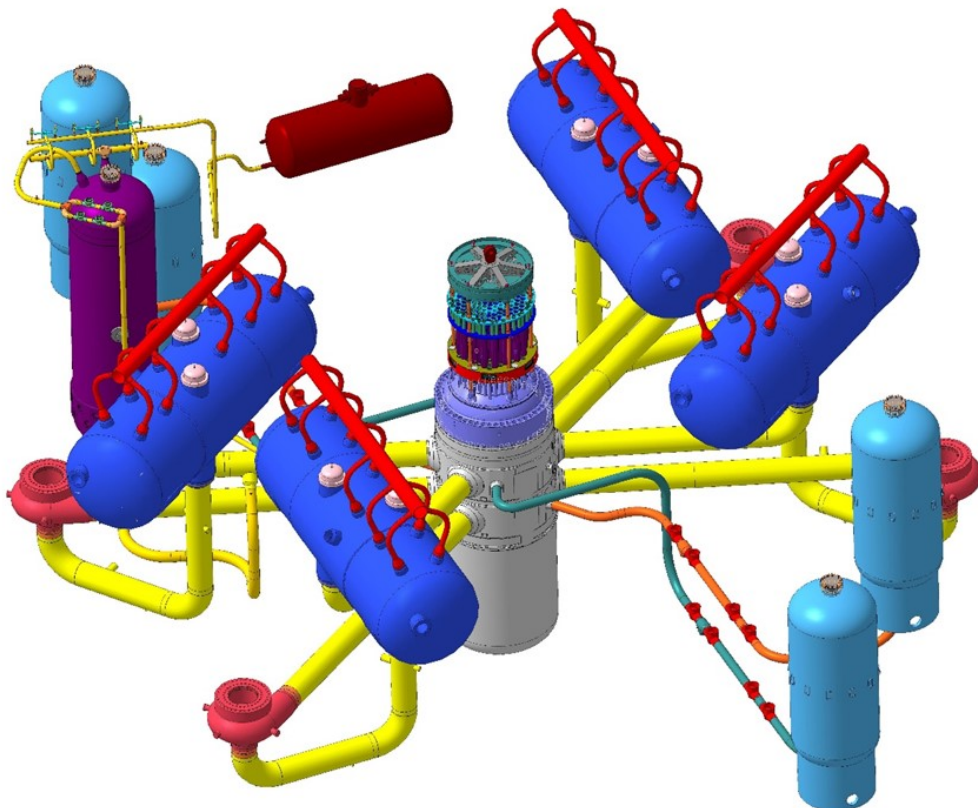


Figure 0-10: Schematic of a typical primary coolant loop for WWER 1000, Model 320 reactor.