Doel 3 – Tihange 2
Reactor Pressure Vessel Assessment

WGIAGE
April 2013

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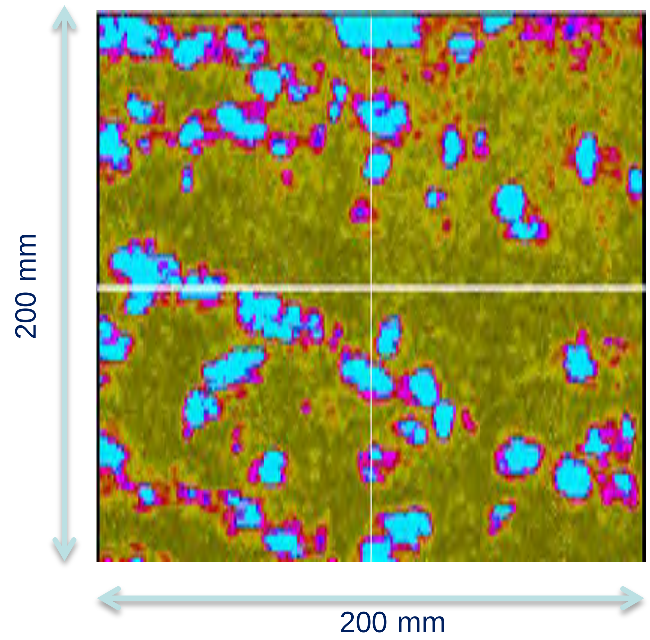
Context

- Beltline examination
  - Following the French Operating Experience (Tricastin)
  - Searching for underclad cracks up to 30 mm depth
  - Height facing fuel assemblies +/- 200 mm
  - Ultrasonic Testing based on flaw tip diffraction, qualified in 2011-2012. This ISI is much more sensitive than the manufacturing examinations of 1975
  - The intervention involved
    - A detection phase aiming at identifying indications that may correspond to underclad cracks
    - A characterization phase that confirms or denies the underclad crack diagnostic, and in case of confirmation, sizes the crack depth & length

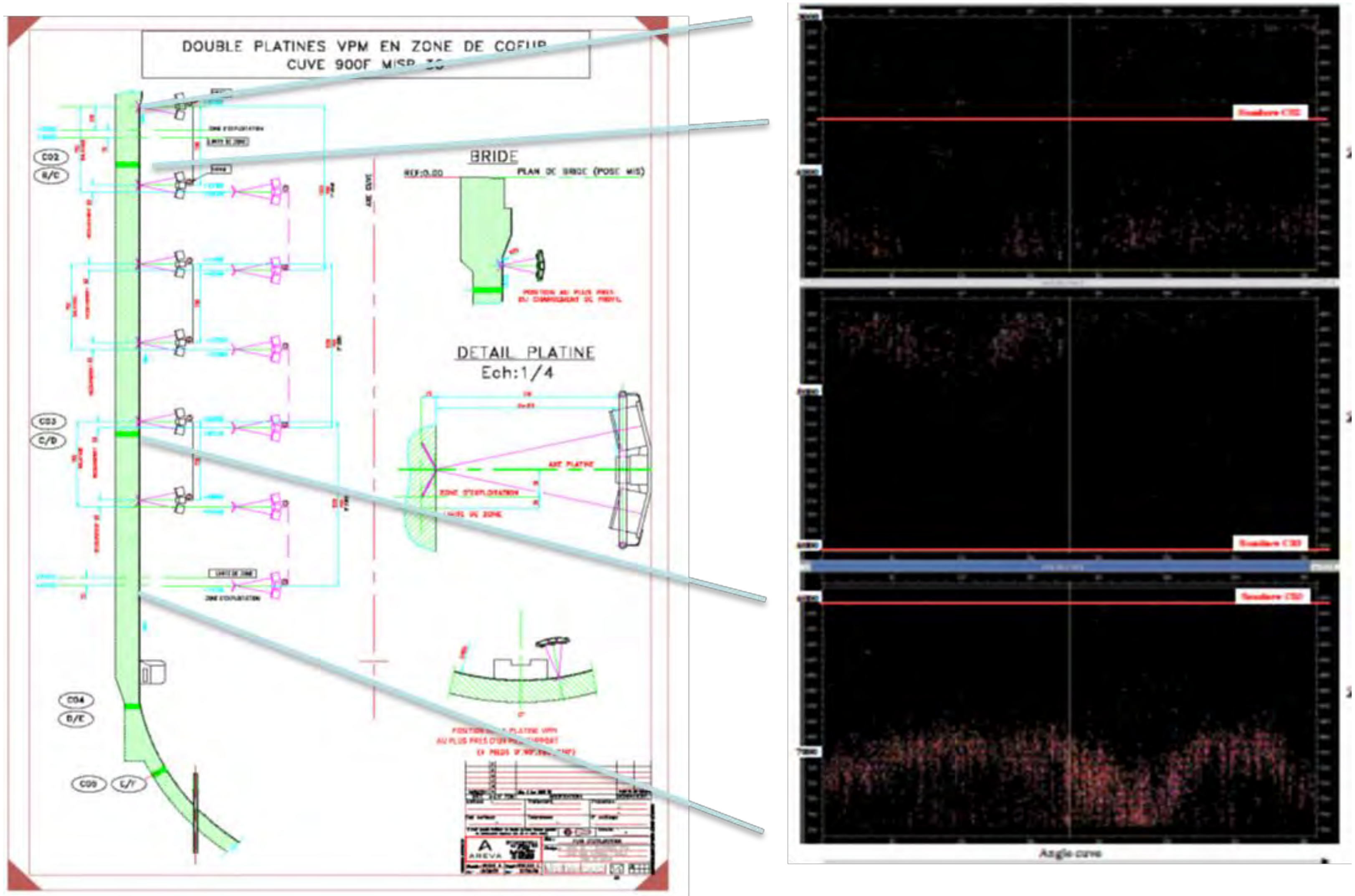
Inspections Results– Doel 3 (1st inspection)

- June 2012: start outage (scheduled until 12/7)
- No underclad cracking, but suspicion of other indications
  - Detection method not calibrated for such kind of indications nor for these locations

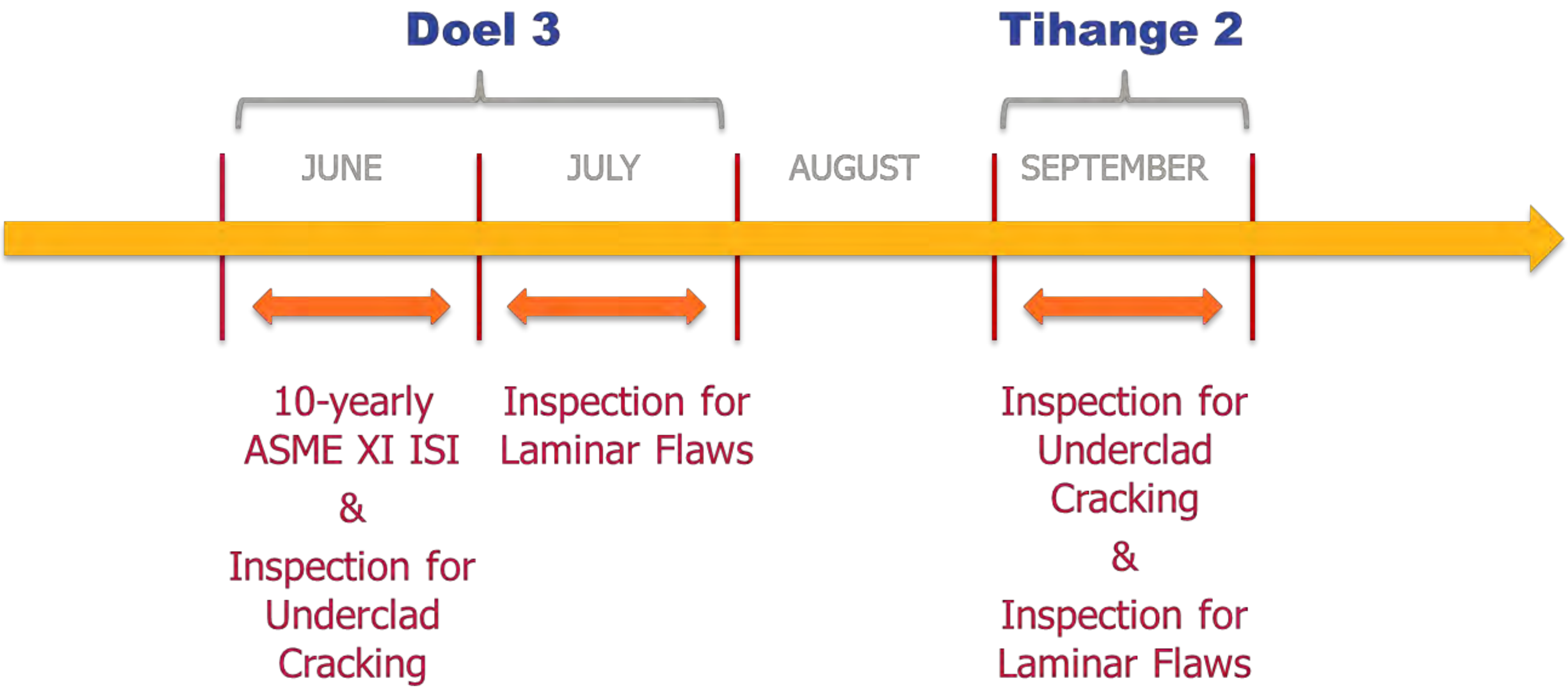


- ✓ In most characterized areas (200 x 200 x 30 mm), many unanticipated indications appeared.
- ✓ As they reflect essentially the L0 beam, they appear (nearly-) laminar and concentrate in the central part of each beltline shell.

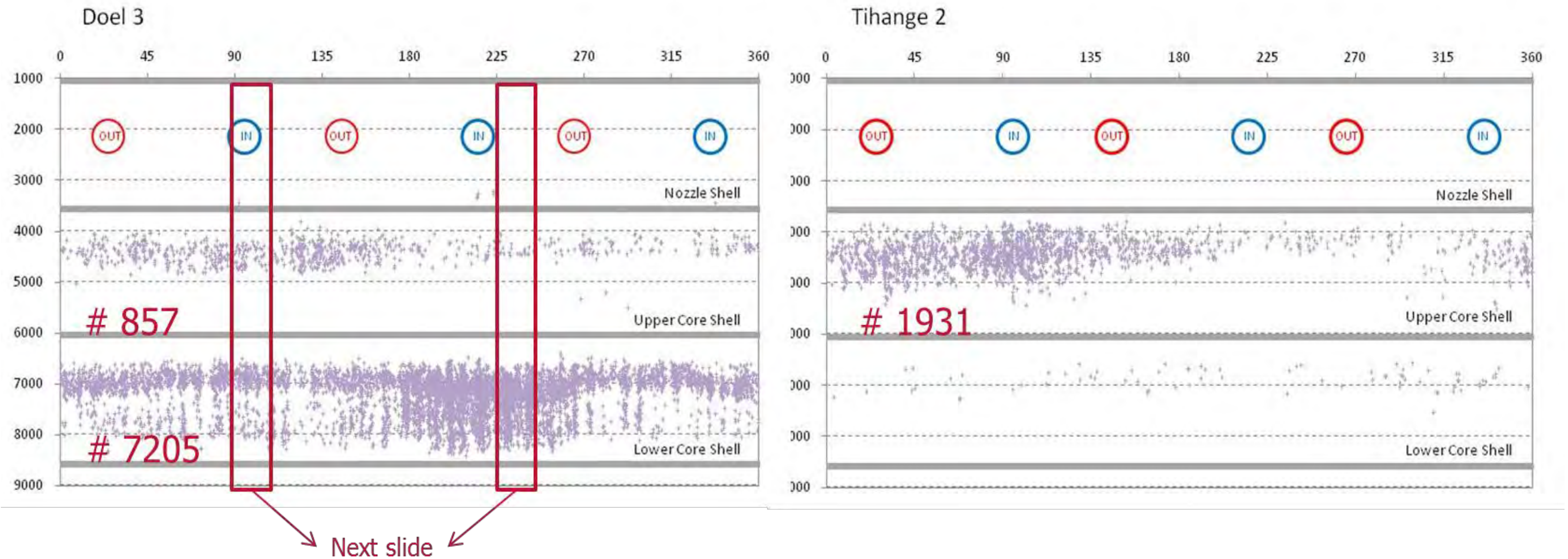
Inspections Results– Doel 3 (1^{st} inspection)



Inspections

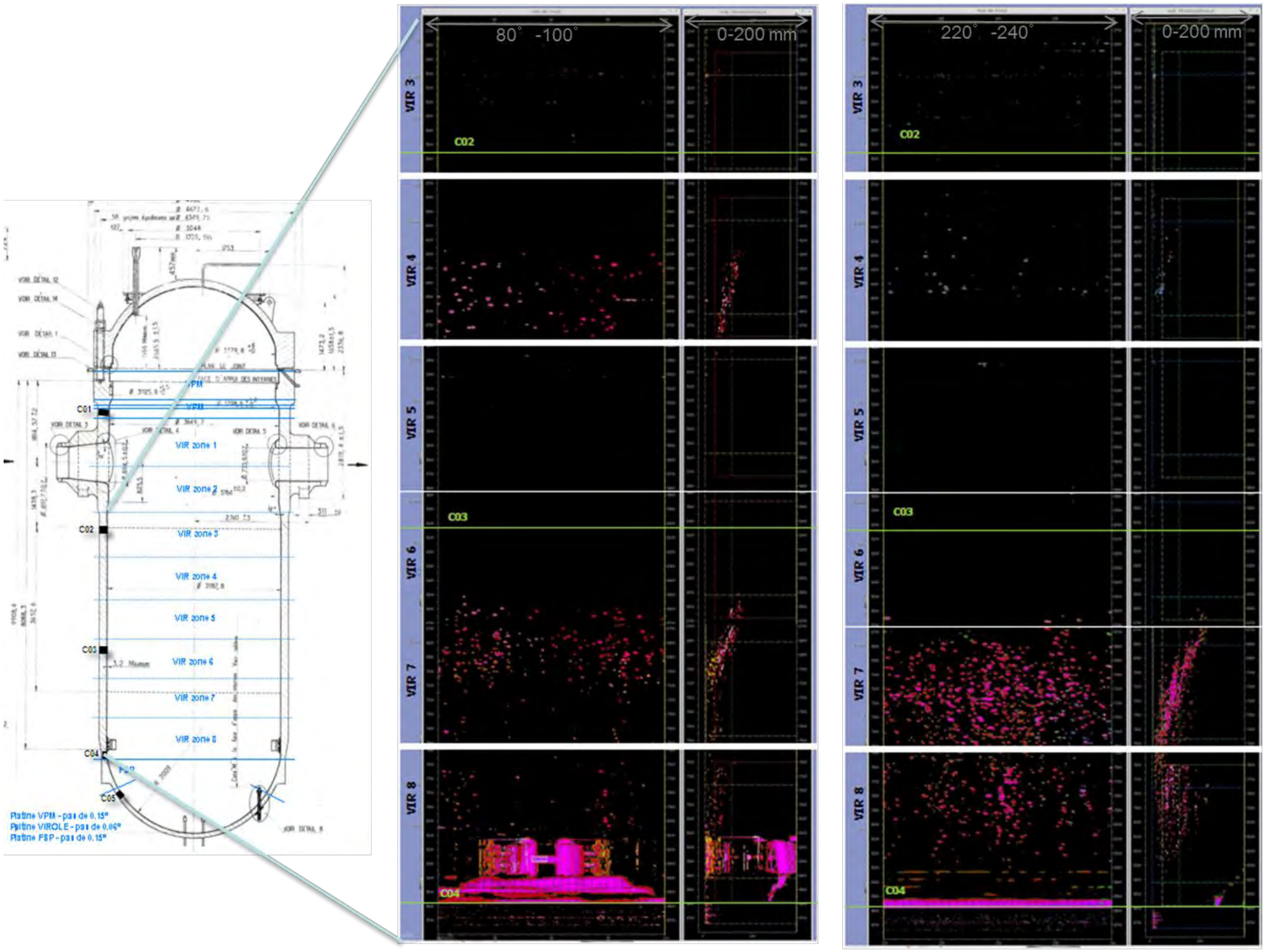


Inspections Results

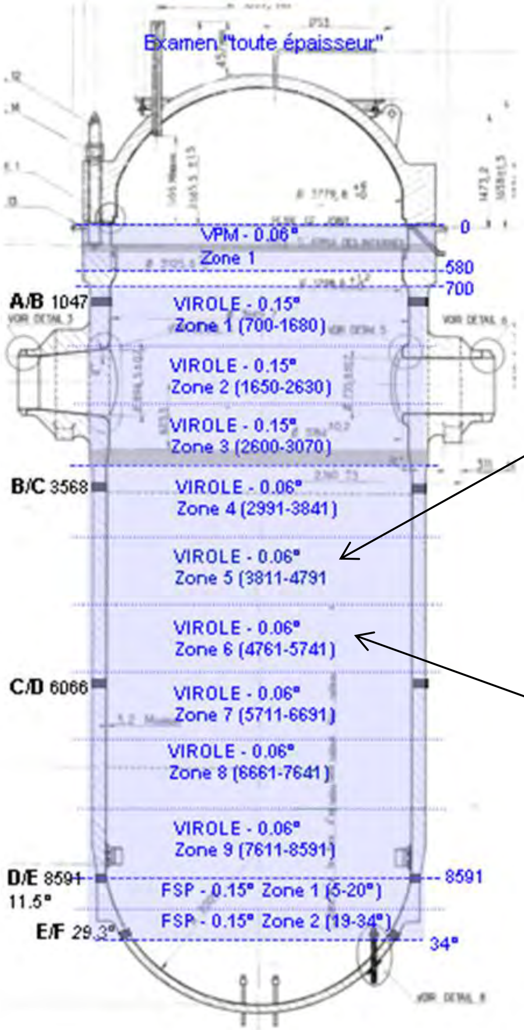


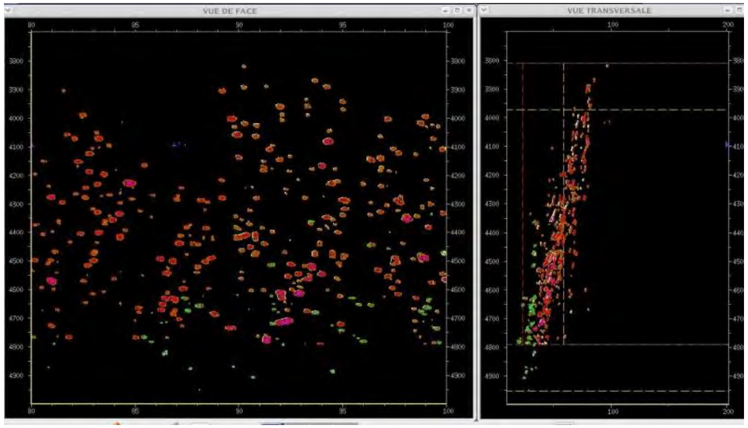
Density in 1st 100mm of thickness:
- Maximum: 25.8 flaws/litre
- Average: 2.1 flaws/litre

Inspections Results– Doel 3 (2^{nd} inspection)

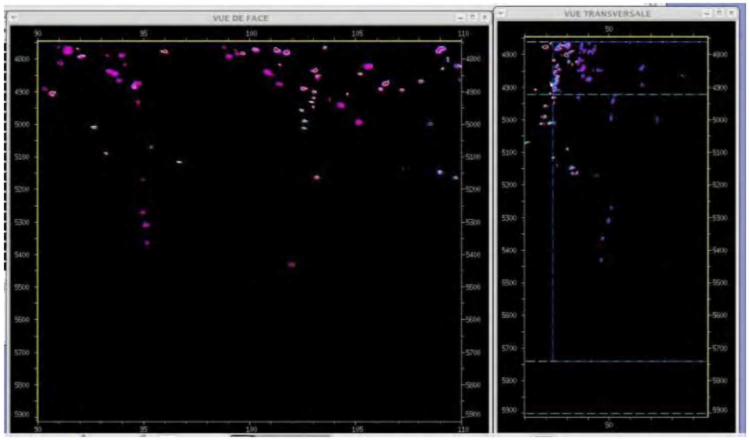


Inspections Results – Tihange 2





zone 5 80-100°



zone 6 90-110°

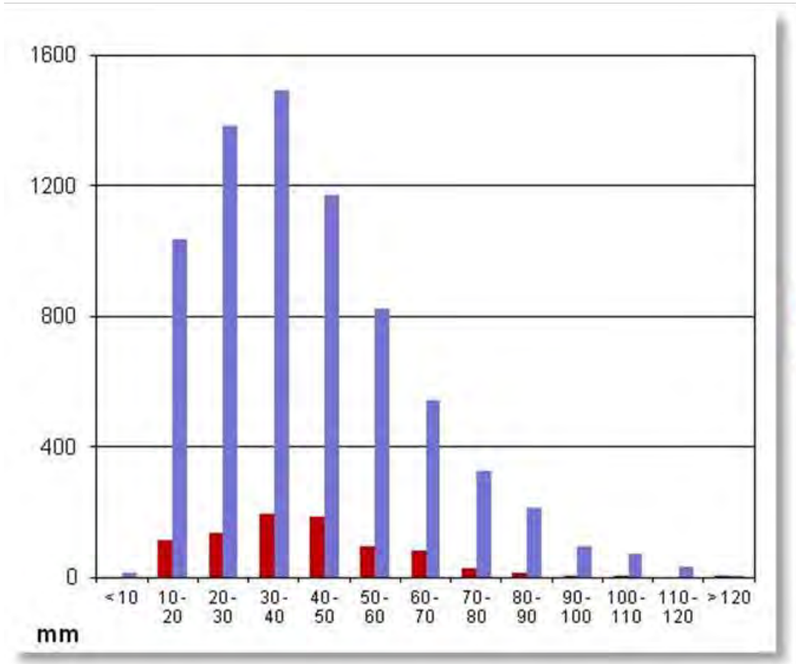
Inspection Results

Doel 3			Tihange 2	
Manufact.	2012	Component	2012	Manufact.
-	3	RVH flange	5	-
-	2	RPV flange	19	-
-	11	Nozzle shell	0	-
Indications	857	Upper core shell	1931	-
Indications	7205	Lower core shell	80	-
Indications	71	Transition ring	0	-
	8149	Total	2035	

Inspection Results

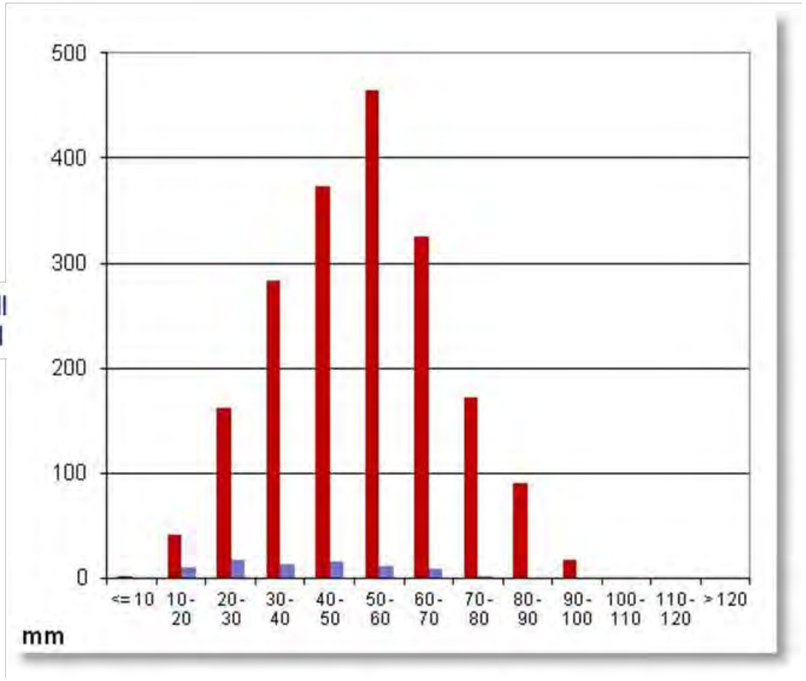
- Flaw position

Doel 3



upper core shell
lower core shell

Tihange 2



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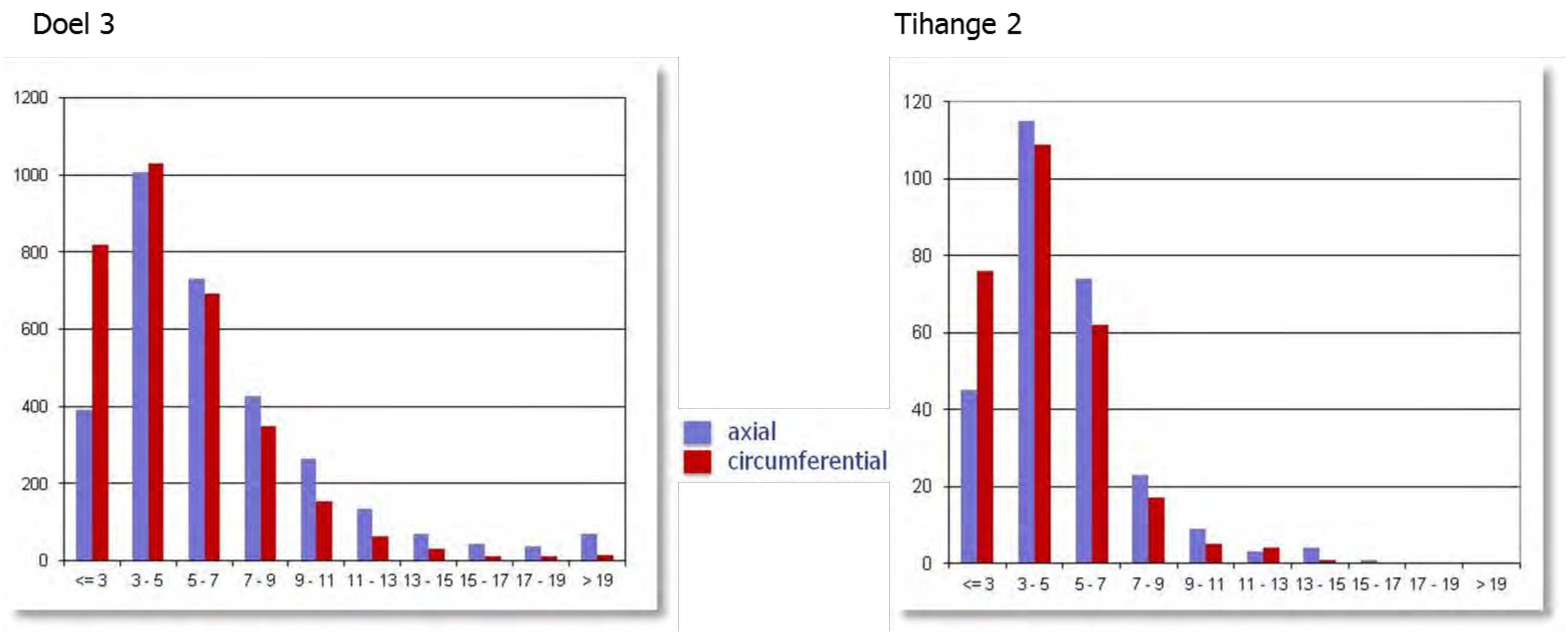
Inspection Results

- Flaw dimensions (depth 0-25 mm)

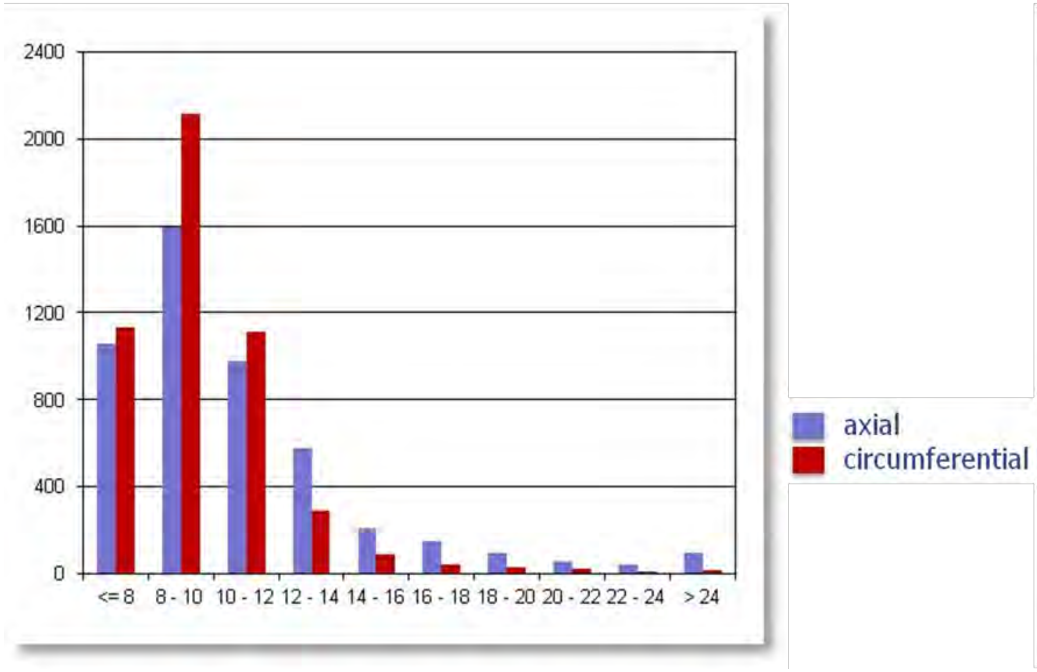


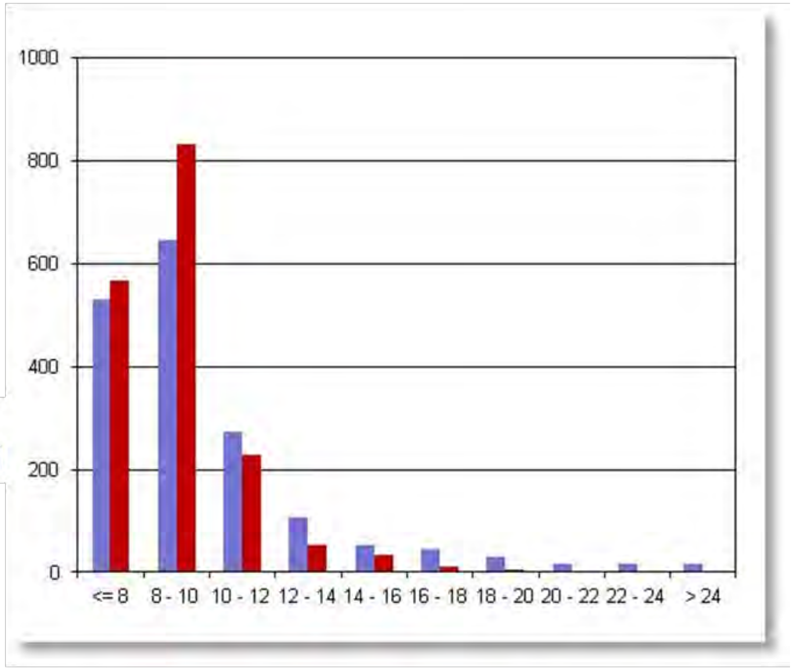
Inspection Results

- Flaw dimensions (depth 25-120 mm)

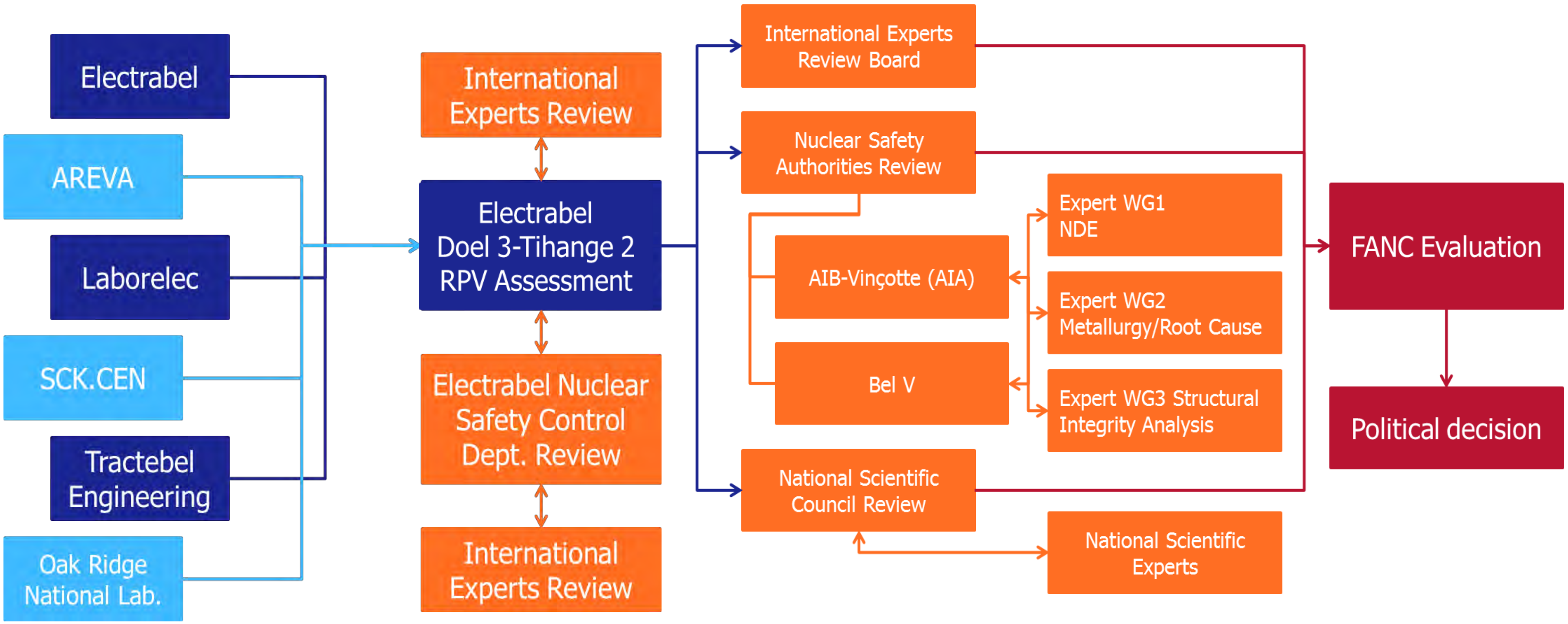
Doel 3

Tihange 2





Safety Case Editors and Reviewers

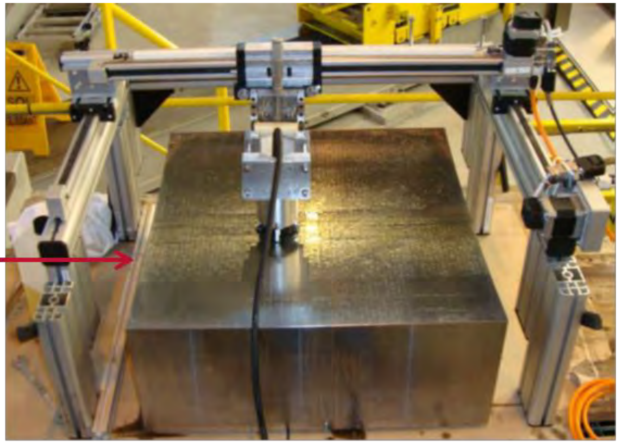


Inspection Validation & Qualification

- Sample 500x500 mm taken from rejected AREVA SG shell VB395 containing hydrogen flakes
- Sample machined to 200 mm thickness (as RPV); no cladding
- UT flaw detection and sizing capability validated through destructive examination of 18 representative flaws in terms of size, location and slope

AREVA
SG shell VB395

AREVA
SG shell VB395 and sample VB395/1



LABORELEC
UT inspection AREVA sample VB395/1

Inspection Validation & Qualification

- Conclusions of UT validation on the VB395/1 block:
  - No flaw missed by UT inspection
  - UT slightly overestimates the size of flaws and hence slightly underestimates the ligaments between neighbouring flaws
  - Between neighbouring flaws there is a ligament composed of sound material
  - Use of angle beams up to 20° does not improve the UT flaw detection and sizing capability
- MIS-B qualification to be done:
  - Based on 2nd sample of VB395 shell (1800x1400x200mm)
  - Quality heat treatment and stainless steel cladding to be applied
  - Non-destructive examination by MIS-B
  - Destructive examination

Manufacturing vs Inservice Inspection

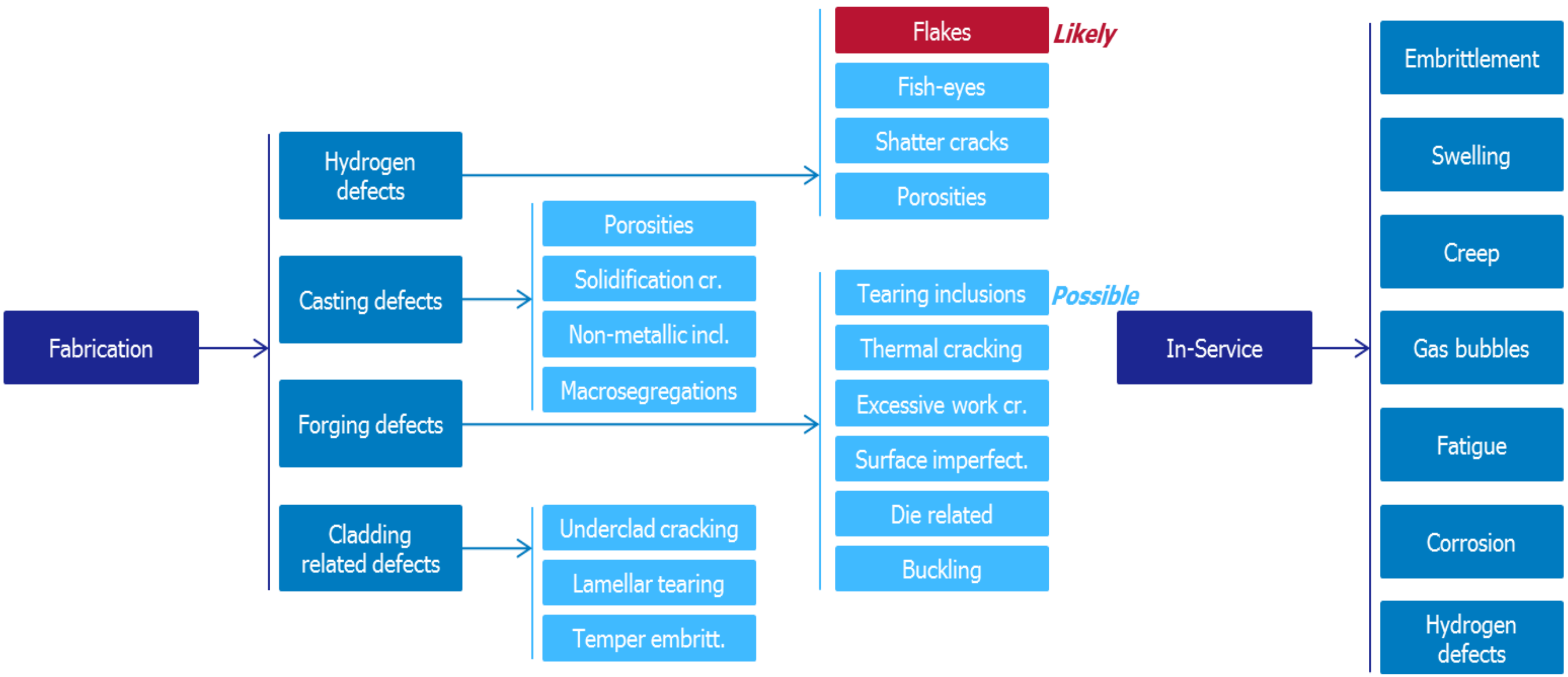
- Technique:
  - Manufacturing: manual inspection with standard probe from unclad outer surface
  - ISI 2012: automated inspection with focused probes through clad inner surface
- Sensitivity:
  - Manufacturing: rejection based on loss of back-wall signal
  - ISI 2012: recording from 18 dB below signal reflected by a 2 mm diameter hole side-drilled in an unclad calibration specimen -> by far more sensitive than manufacturing examination
- Acoustic modelling:
  - Simulation with CIVA software (CEA) of manufacturing and in-service inspection of the RPV shells
  - Conclusions:
    - At least part of the indications observed in 2012 should have been reported at the end of manufacturing
    - Probably, the indications did not exceed the acceptance criteria of the applicable procedures (ASME III is quite tolerant as regards rejection)

Manufacturing vs Inservice Inspection

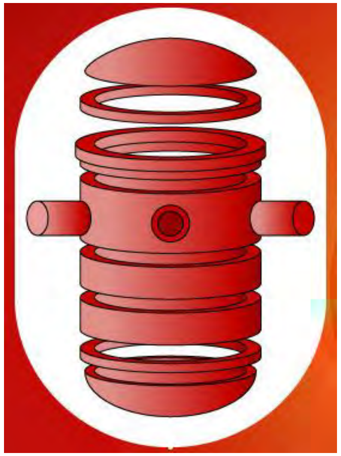
- AREVA Benchmark: Comparison of UT inspection procedures applied to shell VB395 with hydrogen flakes
    - Rotterdam Nuclear procedure (1974, applied to Doel 3 – Tihange 2 RPV shells)
    - EDF specification 900 MWe fleet (1976)
    - Creusot Forge procedure (2012)

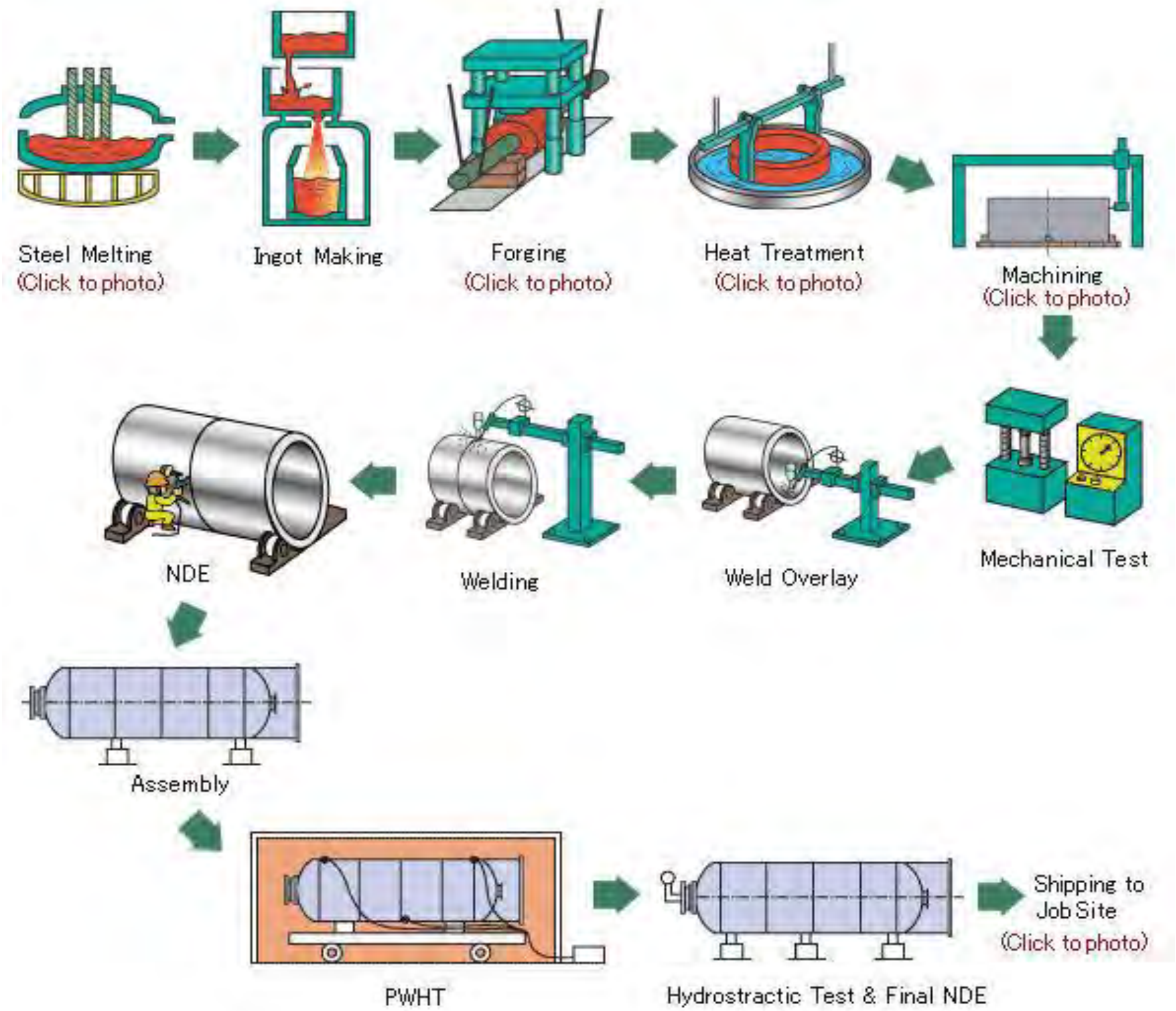
Criterion	RN	EDF	Creusot
Non reportable indications	0	7	3
Reportable indications	50	43	47
Acceptable reportable indications	49	17	0
Non acceptable reportable indications	1	26	47

Material Flaw Formation Mechanisms

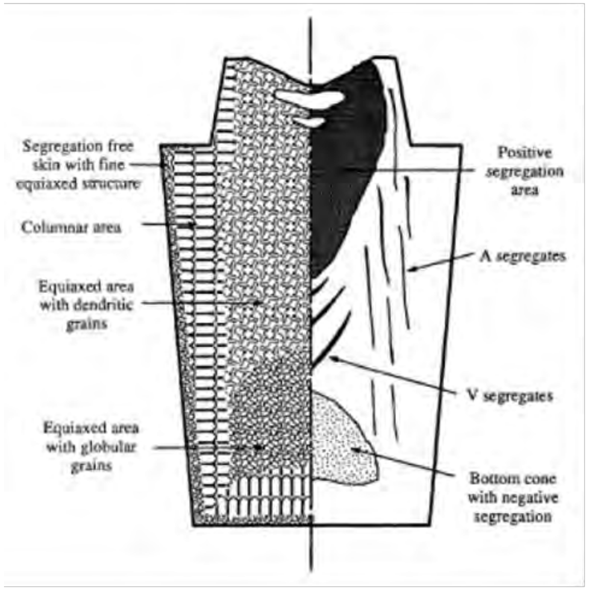


Material Flaw Formation Mechanisms

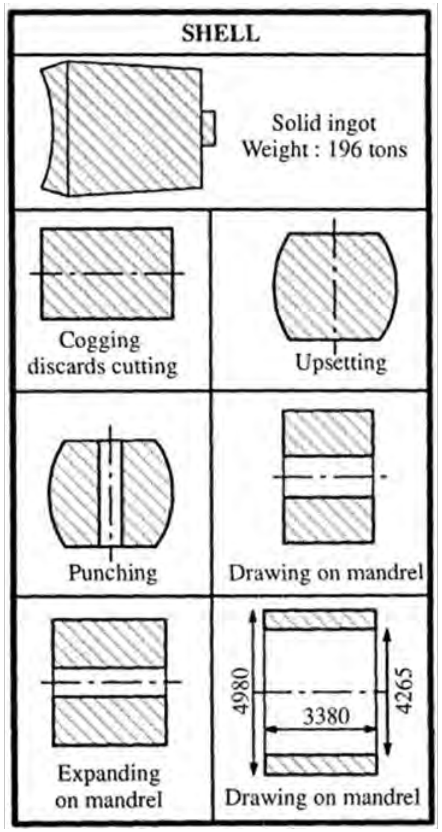


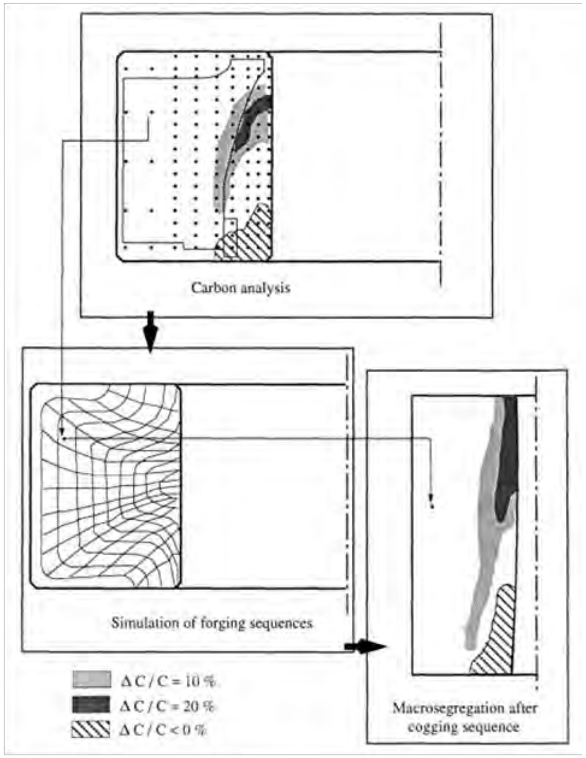


Hydrogen Flakes



[Cl. Benhamou, 1977]

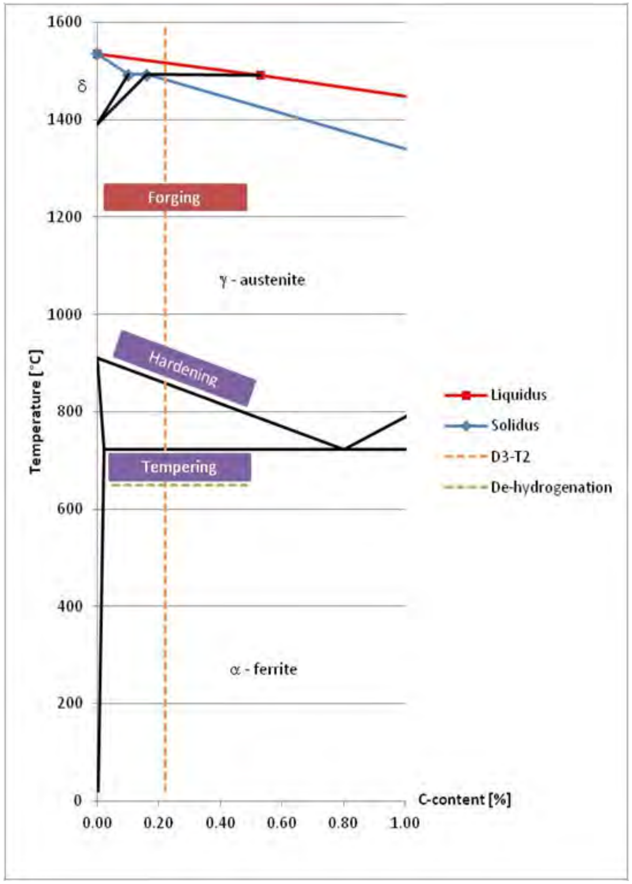




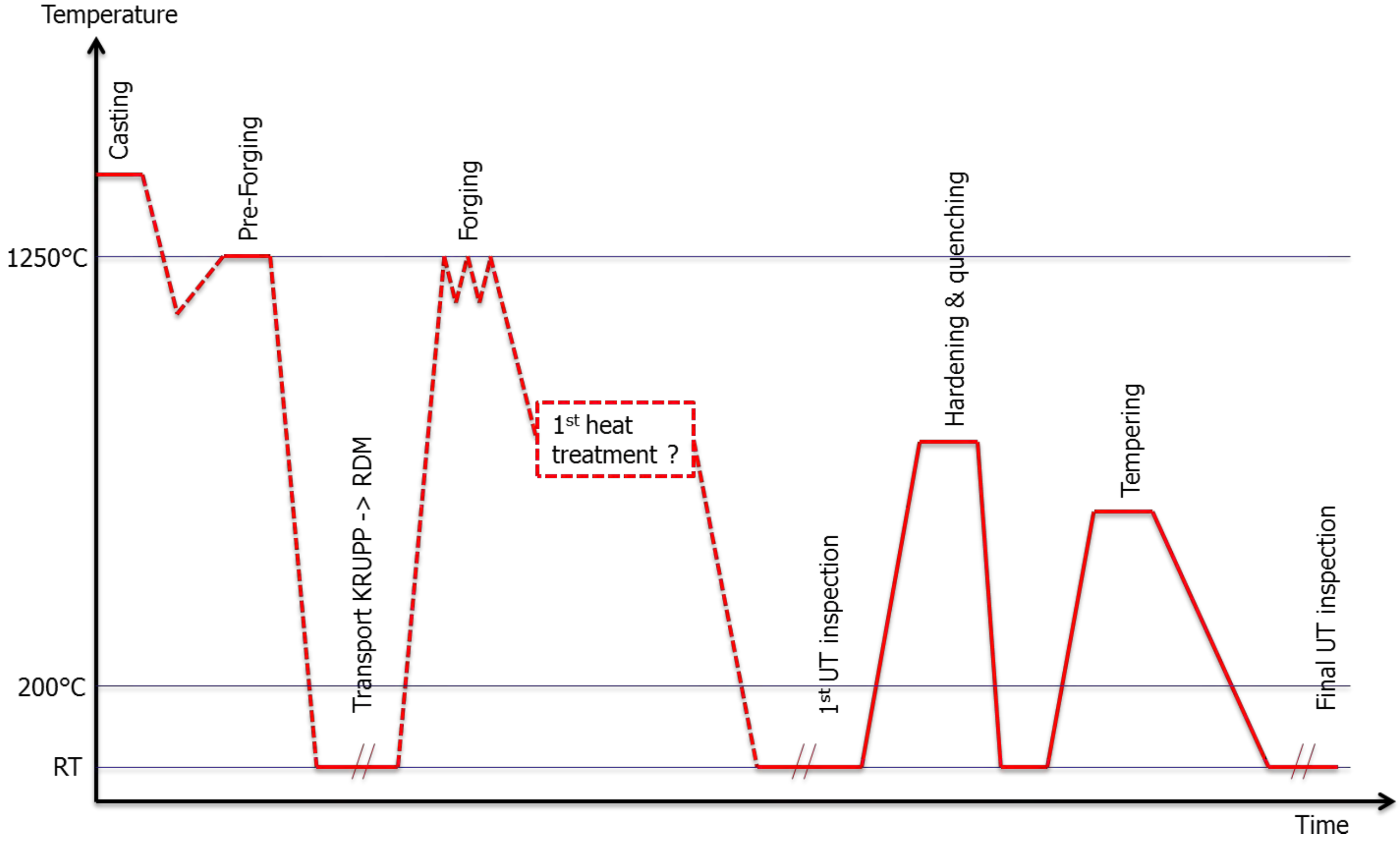
Piercing ϕ 475 – H 3422 ~ 4.7 t
(AREVA procedure: ϕ 820 mm)

Hydrogen Flakes

- Phases:
  - γ : austenite (face-centered cubic)
  - α : ferrite (body-centered cubic)
- Transformation starts earlier in matrix than in segregated areas
- Solubility of H:
  - higher in liquid phase than in solid phase
  - higher in γ than in α
- Hence, H diffuses from matrix to segregated areas
- Accumulation of H e.g. at MnS inclusions
- Mitigative action: de-hydrogenation heat treatment



Hydrogen Flakes



Hydrogen Flakes

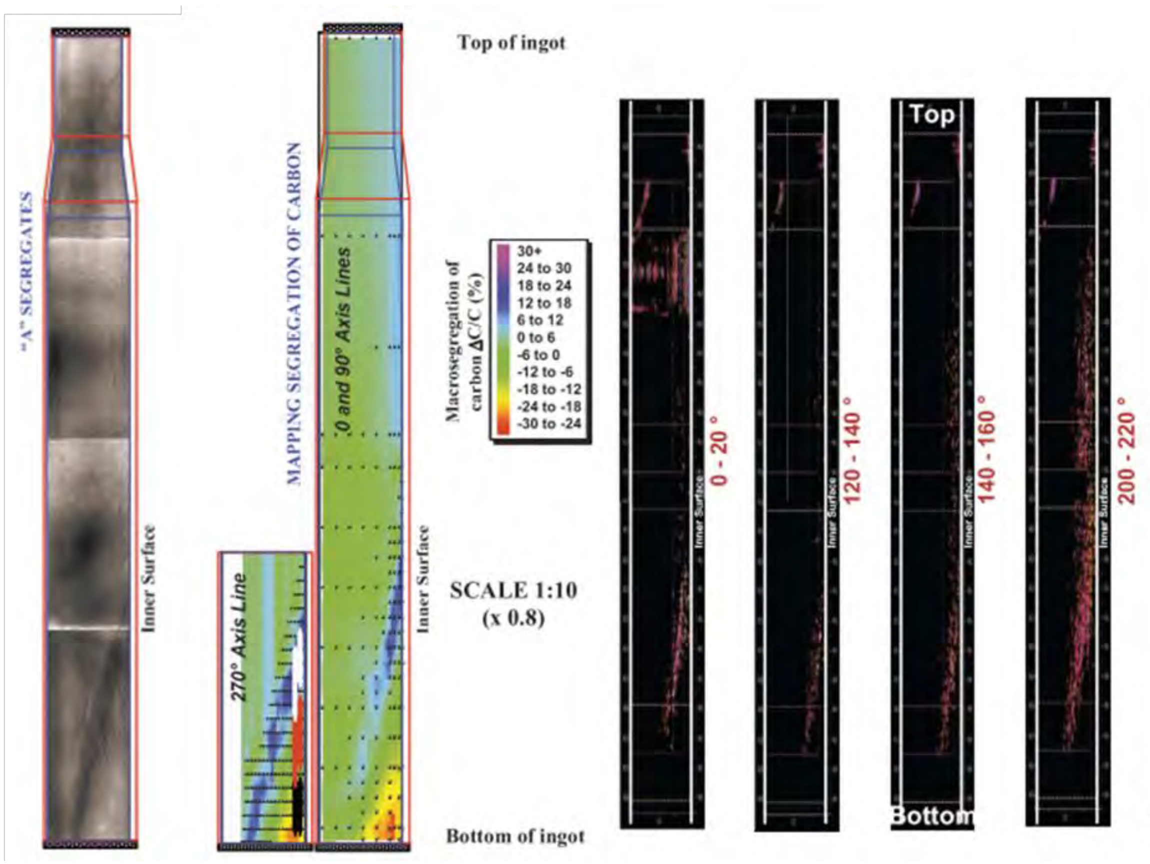
Doel 3					Tihange 2			
Ingot size [ton]	S_{laddle} [%]	H [ppm]	# ind.	Component	# ind.	H [ppm]	S_{laddle} [%]	Ingot size [ton]
92	0.011	1.170	3	RVH flange	5	1.260	0.010	92
84	0.010	1.450	2	RPV flange	19	1.170	0.010	84
110	0.008	1.000	11	Nozzle shell	0	0.900	0.010	110
110	0.011	1.400	857	Upper core shell	1931	1.530	0.010	110
110	0.007	1.500	7205	Lower core shell	80	1.440	0.009	110
93	0.009	1.500	71(2)	Transition ring	0(1)	1.170	0.010	55

Risk colour code: High – Medium - Low

(1) Transported in hot condition from KRUPP to RN
(2) # indications in Doel 3 part only; Tihange 2 part of same ingot rejected

Hydrogen Flakes

Areva nozzle shell



Doel 3 core shell

Hydrogen Flakes

- Piercing diameter of the ingot smaller than typical AREVA procedure (475 mm vs. 820 mm), eliminating less of the macrosegregation
- No evidence of de-hydrogenation heat treatment in RDM Files
- Location of indications in zones of positive segregation consistent with greater affinity of hydrogen for these zones
  - Last ones to transform from austenite to ferrite, greater solubility of hydrogen in austenite
  - Composition makes these zones more sensitive to cracking
- Quasi-laminar orientation consistent with the fact that hydrogen flakes are generated preferentially on MnS inclusions, oriented circumferentially and flattened by the forging
- Typical shape and dimension 4-14 mm consistent with ISI results
- When conditions for hydrogen flaking are met they appear in large numbers

Structural Integrity Analysis

- Deterministic assessment of the flaws in conformity with the principles of fracture mechanics according to Section XI of the ASME Code.
- Fracture mechanics: ensuring structural integrity requires:

Driving force for fracture<resistance to fracture

$$K_{Applied} < K_{Ic}$$

- Section XI (IWB-3612): requires application of a safety coefficient

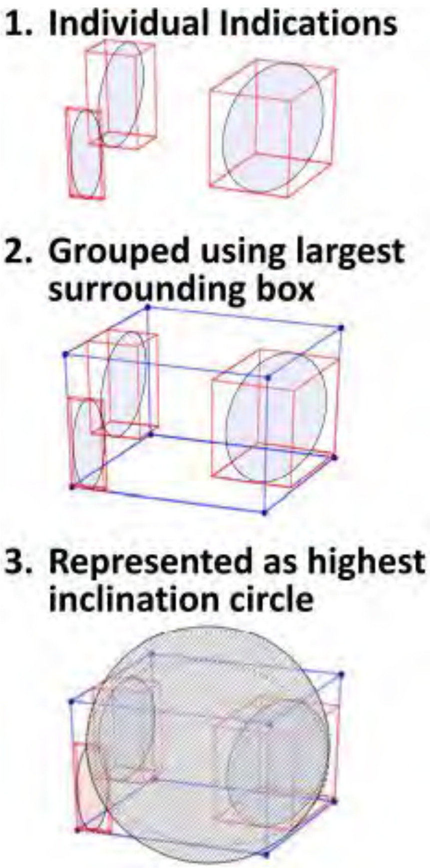
Under Level A and B loadings: $K_{Applied} < K_{Ic} /\sqrt{10}$

Under Level C and D loadings: $K_{Applied} < K_{Ic} /\sqrt{2}$

Structural Integrity Analysis

$$K_{Applied} = K_{Applied}(applied\ stress, flaw\ size, flaw\ shape, flaw\ orientation)$$

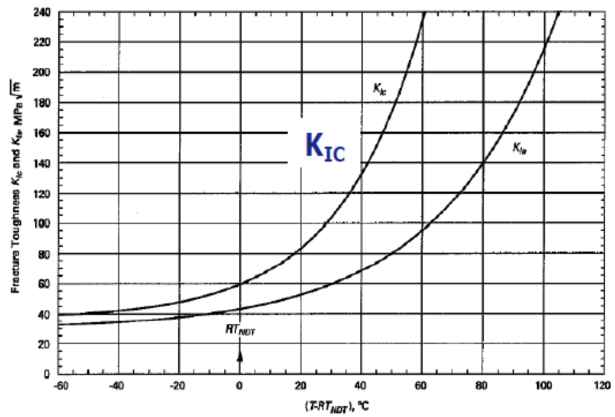
- Applied stress: stress field at the flaw location as obtained from stress analysis of the RPV under assumed loadings
- Flaw size: from UT measurement + grouping of closely-spaced flaws by application of specific proximity rules
- Flaw shape: flaws assumed to be circular
- Flaw orientation: from UT measurement and application of the proximity rules



Structural Integrity Analysis

$K_{Ic} = K_{Ic}(T, RT_{NDT})$

FIG. A-4200-1M LOWER BOUND K_{Ia} AND K_{Ic} TEST DATA FOR SA-533 GRADE B CLASS 1, SA-508 CLASS 2, AND SA-508 CLASS 3 STEELS



- Fracture Toughness K_{Ic}:
  obtained from the ASME Fracture Toughness curve
- Temperature T: temperature at the flaw location

from thermal analysis of the RPV under assumed loadings

- Reference temperature of nil-ductility transition RT_{NDT}: account have been taken for the effects of :
  - Irradiation (FIS formula)
  - Flaw orientation (testing program)
  - Macro-segregations (testing program)
  - Flakes (testing program)

Structural Integrity Analysis

- 400 samples material test program

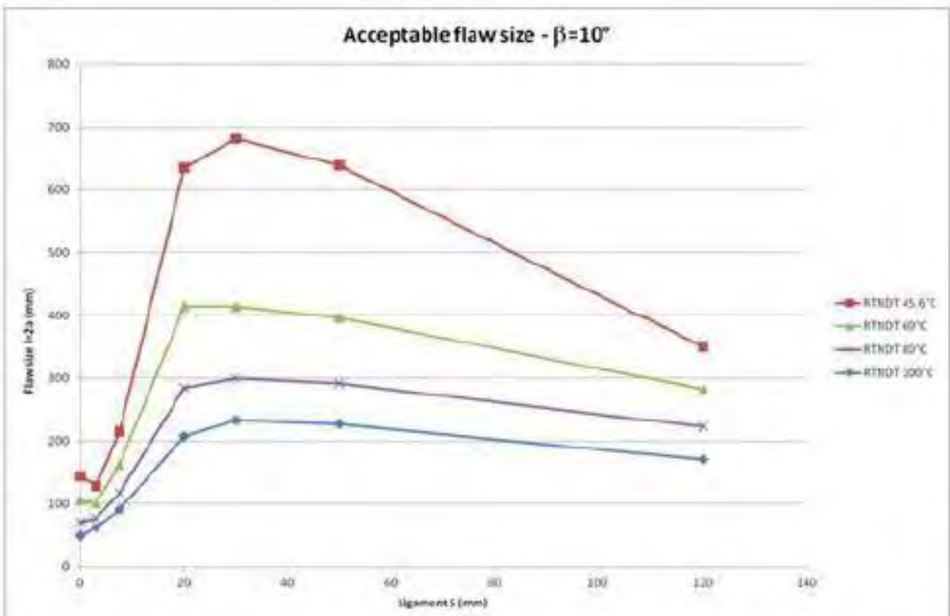
Source	Component type	Characteristic	Lab	Orient	Segreg	Irrad
Doel 3	Existing Surveillance program	Irradiated	SCK.CEN			X
	Spare surveillance program block	Un-irradiated	SCK.CEN	X		
	Nozzle shell cut-out H1	Macro-segregations	SCK.CEN AREVA	X	X	
AREVA	Nozzle shell cut-out H2BQ3	Macro-segregations	AREVA	X	X	
	SG shell VB395	Hydrogen flakes	AREVA LABORELEC	*Preliminary UT validation*		
Literature	German research program 1970-1980	Hydrogen flakes	-	X	X	X

- Maximum $\Delta RT_{NDT,\ FIS\ formula,\ segregated\ zone} = 17°C$
  - For the most unfavourable chemical composition
  - At the peak fluence location

→ ΔRT_{NDT} = 50°C considered in Structural Integrity Analyses

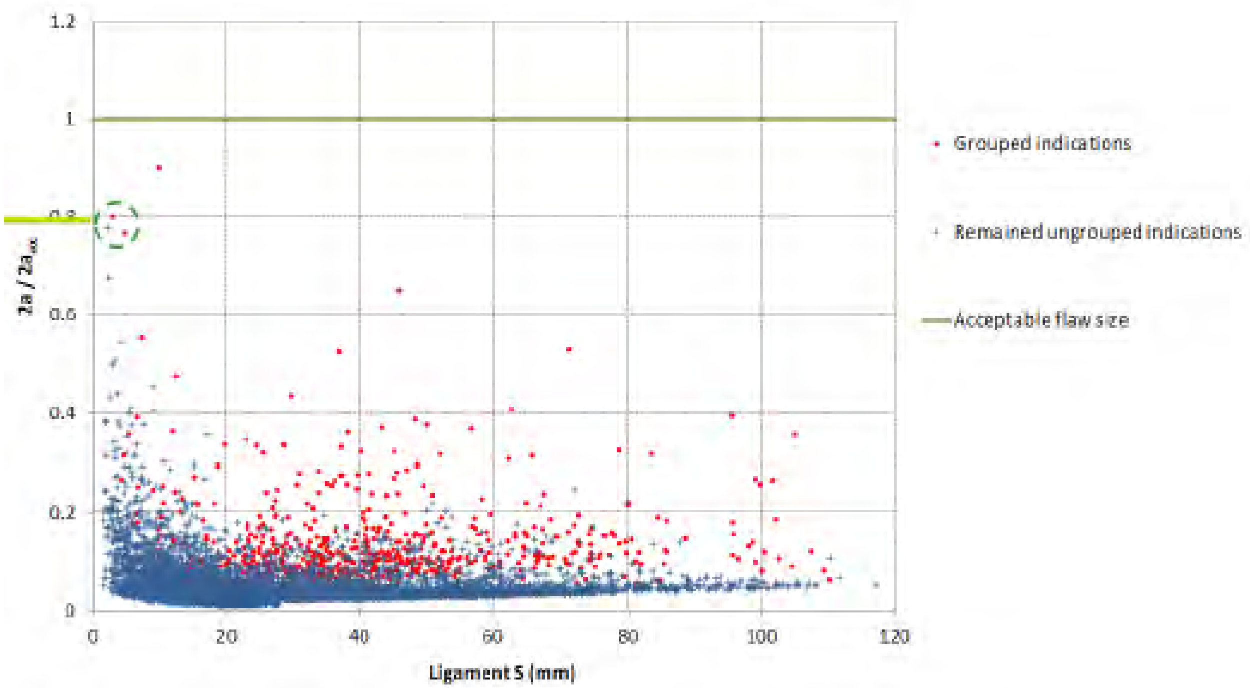
Structural Integrity Analysis

- Practically, the structural integrity assessment has not been performed from the evaluation of $K_{Applied}$ and K_{Ic} for each flaw but from the comparison of the actual flaw size with the acceptable flaw size.
- For the determination of the acceptable flaw size, the effects of flaw inclination, RT_{NDT}, and ligament (proximity to inner surface) are accounted for.



Structural Integrity Analysis

- Flaw assessment is as follows: for each flaw, the ratio of the measured flaw size to the acceptable flaw size is determined. Flaw is acceptable is the ratio is lower than 1.0



Structural Integrity Analysis

- Flaw Acceptability Analyses – margin to acceptable flaw size:
  - Doel 3 RPV:
    - At least 22% for individual flaws
    - At least 10% for grouped flaws
  - Tihange 2 RPV:
    - At least 63% for individual flaws
    - At least 79% for grouped flaws
- ASME III Primary stress limits are met
- Fatigue Crack Growth analysis:
  - No significant crack growth

Structural Integrity Analysis

- Additionnal material testing

Source	Component type	Characteristic	Objective
Doel 3	Nozzle shell cut-out H1	Macro-segregations	Characterization of ghost line properties
AREVA	SG shell VB395	Hydrogen flakes	Small scale tensile and fracture toughness tests covering the ligaments between flakes
			Characterization of material in zones without flakes (for comparison with zones with flakes)
			Large scale tensile testing of specimen with flakes parallel with specimen axis (ASME III analysis)
			Direct measurement of fracture toughness at the tip of hydrogen flakes
			Large scale tensile testing of specimen with hydrogen flakes in an orientation comparable to the orientation of the flaws in Doel 3/Tihange 2. Structural Integrity Analysis of specimen through Finite Element analysis and comparison with test.

Structural Integrity Analysis

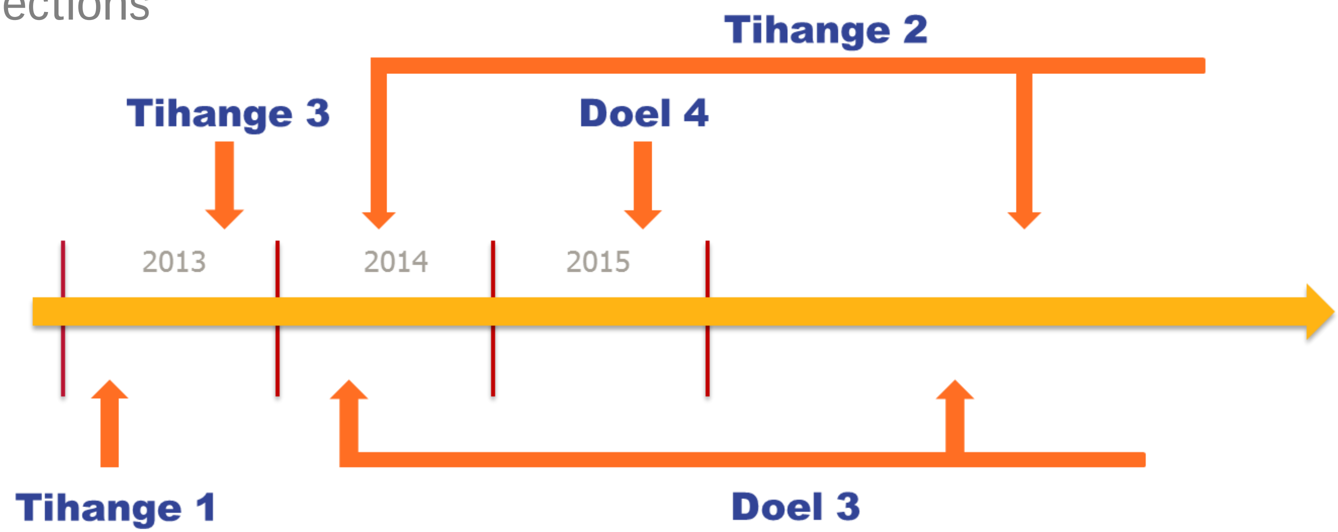
- ASME XI Appendix G analysis:
  - Update of p-T curves to include embrittlement of RPV at the end of its service lifetime and the additional shift in RT_{NDT} of 50°C
  - Re-evaluation of Low Temperature Overpressure Protection leads to adaptation of some operating limits
- 10 CFR 50.61 Deterministic PTS analysis:
  - RTNDT at the end of service lifetime, including additional shift in RT_{NDT} of 50°C, below 132°C for base metal and 149°C for circumferential welds
- 10 CFR 50.61a Probabilistic PTS analysis:
  - Large margin of FCI-Frequency of Crack Initiation with respect to acceptance criterion
  - FCI one decade lower than US Westinghouse reference plant

Operational Measures

- Adaptation of Technical Specifications (p-T limits)
- Doel 3

  Increase of Refueling Water Storage Tanks minimum water temperature up to 30°C (currently 7°C)

  → Additional 19% margin w.r.t. acceptable flaw size for flaws close to the inner surface
- Future inspections



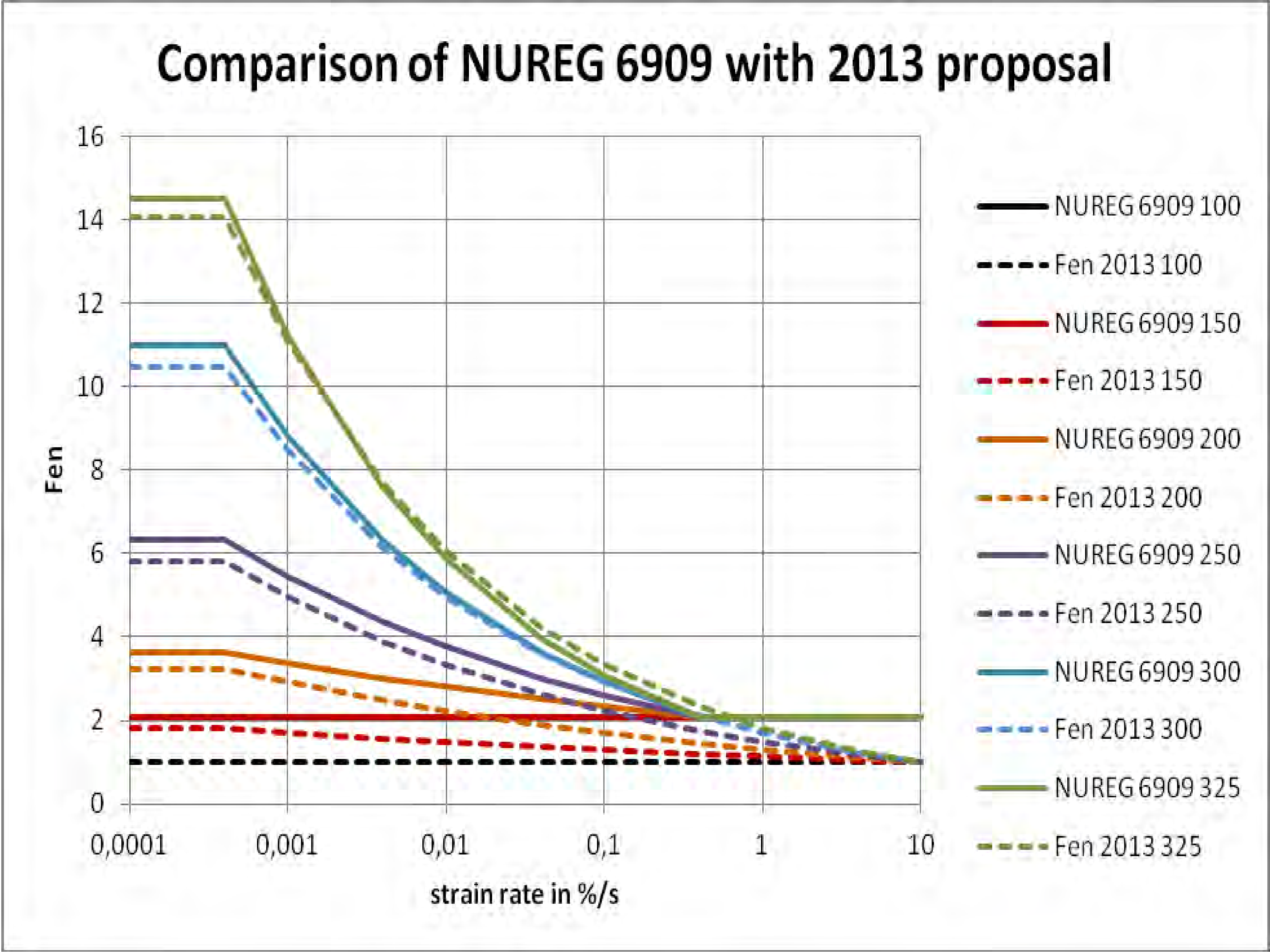
Conclusion

- 05/12/2012 Electrabel Safety case reports
- 11/01/2013 Report of the National Scientific Expert Group on the RPVD3T2
- 15/01/2013 Doel 3 - Tihange 2 RPV issue: International Expert Review Board Final Report
- 30/01/2013 Doel 3 and Tihange 2 reactor pressure vessels: Provisional evaluation report (FANC)

http://www.fanc.fgov.be/GED/00000000/3300/3391.pdf

Conclusion

- In the current state of knowledge and given the available data, the open issues identified along the assessment process and described in the current evaluation report do not represent conditions that require a definitive shutdown of the Doel 3 and Tihange 2 reactor units.
- However, these open issues lead to some uncertainties that might reduce the conservatism of the licensee's safety demonstration and hence impair the level of confidence in the safe operability of the reactor units in question.
- As a consequence, the Federal Agency for Nuclear Control considers that, in the current state, the Doel 3 and Tihange 2 reactor units may only restart after the requirements listed under §12.2 hereafter are fulfilled by the licensee.
- Actions regarding the inservice inspection, the metallurgical origin and evolution of the indications, the material properties, the structural integrity of the reactor pressure vessels, and the action plan proposed by the licensee.



Comparison of NUREG 6909 with 2013 proposal
Fen
16
14
12
10
8
6
4
2
0
0,0001
0,001
0,01
0,1
1
10
strain rate in %/s
NUREG 6909 100
Fen 2013 100
NUREG 6909 150
Fen 2013 150
NUREG 6909 200
Fen 2013 200
NUREG 6909 250
Fen 2013 250
NUREG 6909 300
Fen 2013 300
NUREG 6909 325
Fen 2013 325

d) F_{en} evaluation (last Argone-USNRC draft proposal – 2013)

- $F_{en} = \exp(-T' \varepsilon' O')$

with:

- T'
  - $T' = 0$ for $T < 100°C$
  - $T' = (T - 100)/250$
    for $100 < T < 325°C$
  - $T' = 0.9$ for $T > 325°C$
- ε'
  - $\varepsilon' = 0$ for $\dot{\varepsilon} > 10\%/s$
  - $\varepsilon' = \ln(\dot{\varepsilon}/10)$
    for $0,0004 < \dot{\varepsilon} < 10\%/s$
  - $\varepsilon' = \ln(0,0004 / 10)$
    for $\dot{\varepsilon} < 0,0004\%/s$
- $O' = 0,29$ for DO<0.1 ppm
- The comparison of this new draft proposal of F_{en} with previous proposal of NUREG 6909 [3] is done in figure 8.

Universität Stuttgart

MPA STUTTGART
Otto-Graf-Institut
Materialprüfungsanstalt Universität Stuttgart

Fatigue behavior of ferritic and austenitic steel under air and environmental conditions

Karl-Heinz Herter, MPA Universität Stuttgart

18th WGIAGE METAL SUB-GROUP MEETING
April 9 and 10, 2013,
OECD, Paris

Content

- Introduction
- Experimental investigations
  - LCF tests in air environment (austenitic and ferritic material, austenitic cladding)
  - LCF tests in BWR environment (austenitic and ferritic material)
- Fatigue best-fit and design curves
- Summary & Outlook

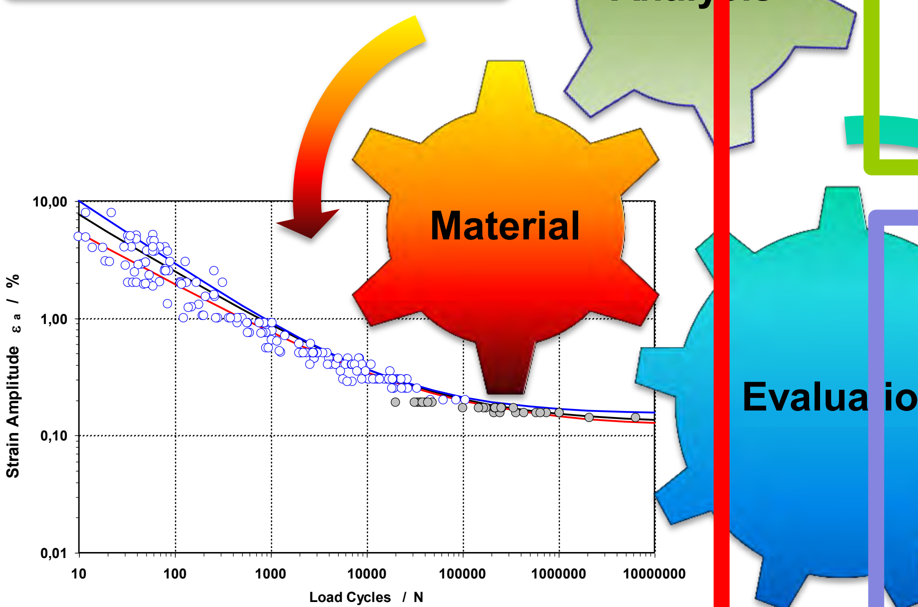
Introduction

- **Relevance of fatigue design due to severe damage effects**

  - Fatigue failure of a railway wheel (1875) due to the fact, that cyclic loaded materials are less resistant than under static loads

  - August Wöhler (1819-1914), first systematic investigation of S-N Curves, also known as Wöhler curves, to characterize the fatigue behavior of materials (presentation of his work at the Paris Exposition in 1867 brought it to a wide international audience)

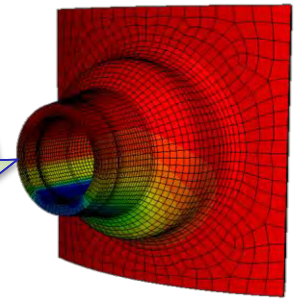
Introduction

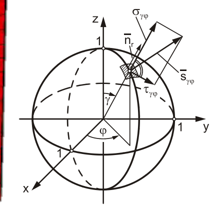


Fatigue Strength

- Stress-/Strain amplitude
- Temperature
- Surface
- Size effect
- Multiaxiality
- Time-function / Phase shift
- Environment
- Strain rate
- …

Stress- / Strain - Analysis





- Load-time-function
- Load amplitude
- geometry
- Boundary conditions
- Multiaxial stress state
- …

Fatigue Evaluation Concept

Fatigue Strength Assessment

…mental …ssment

Full…cale …omponent Test
…ests up to failure of t… component under t… loading conditio… N_F=f(loading), eg. acc. to ASME-VIII, Div.2 or up to the technical crack initiation N_i=f(loading)

Analytical / Numerical Assessment

Global Concept
Based on external forces or based on nominal stresses within the component sections in question

Local Concept
Based on local stresses and related to critical values of local phenomena like crack initiation or crack propagation

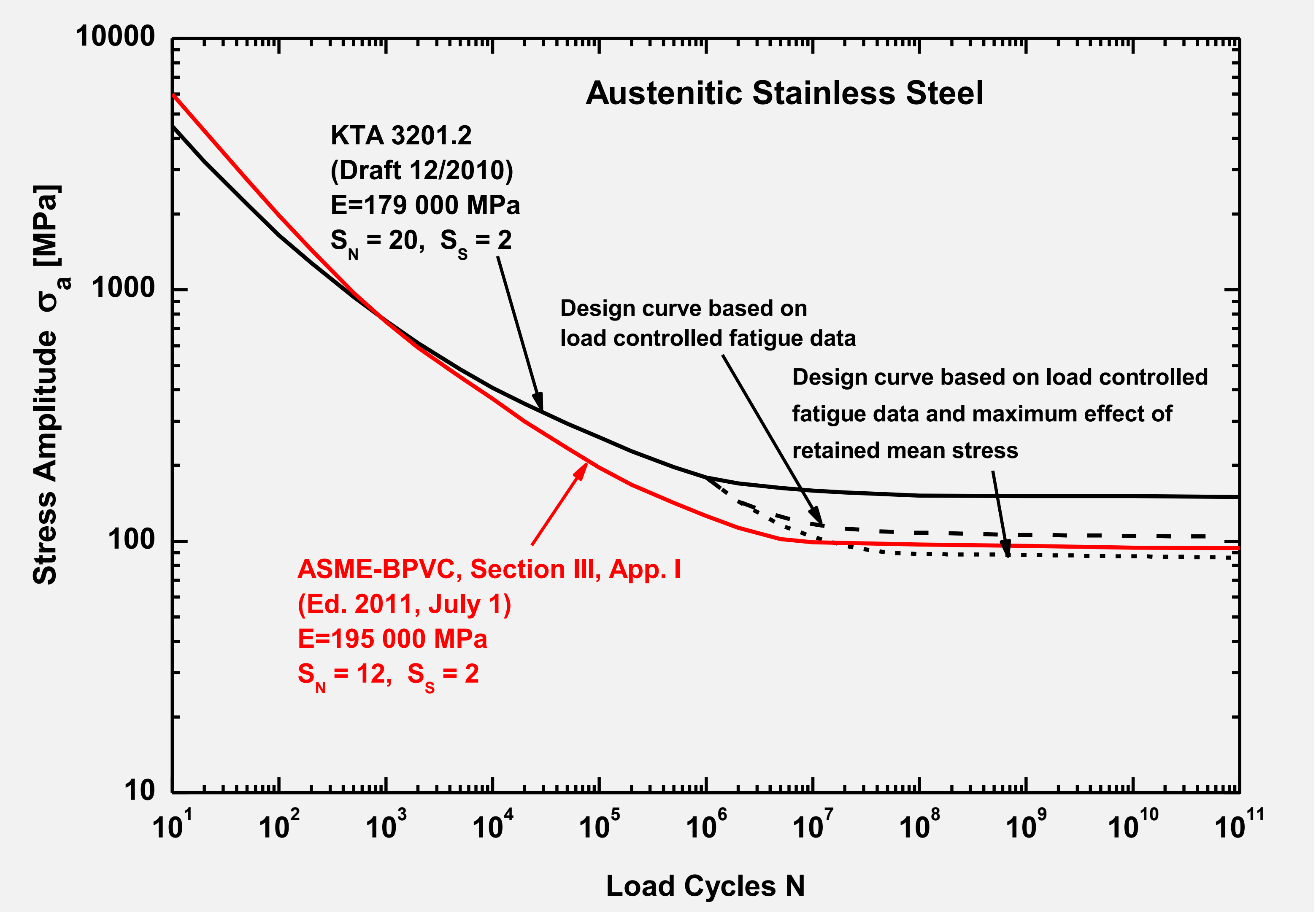
Nominal Stress Concept
- *Nominal Stress-S-N Curve*
  - Material
  - Geometry
  - Surface finish / Weld quality
- *Nominal Stress Collektive*
  - Loading sequence
  - Geometry
  - Load amplitude
- *Accumulation of damage* N_F=f(σ_{Nom})

Structural Stress Concept
- *Structural Stress (S) resp. Strain (ε)-N Curve*
  - Material
  - Geometry
  - Surface finish / Weld quality
- *Structural Stress resp. Strain Collektive*
  - Loading sequence
  - Geometry
  - Load amplitude
- *Akkumulation of damage* N_F=f(σ_{Struc})

Local Notch stress Concept
- *Strain (ε)-N Curve / cyclic Stress-Strain Curve*
  - Material
  - Weld quality
- *Local Notch Stress*
  - Geometry
  - Load amplitude
- *Stress-Strain History*
  - Load-time function
- *Accumulation of damage* N_i=f(σ_{Notch})

Crack Propagation Concept
- *Crack Growth Curve (cyclic)*
  - Material
  - Load amplitude
- *Load-Time Function* ΔK_{eff}
  - Geometry
  - Crack geometry
- *Crack Growth / Lifetime*
- *Woehler-Line / Service-Life Curves (crack init. Up to failure,* $\Delta\sigma_{Nom}$*)*

- **Fatigue design curves for austenitic stainless steels**



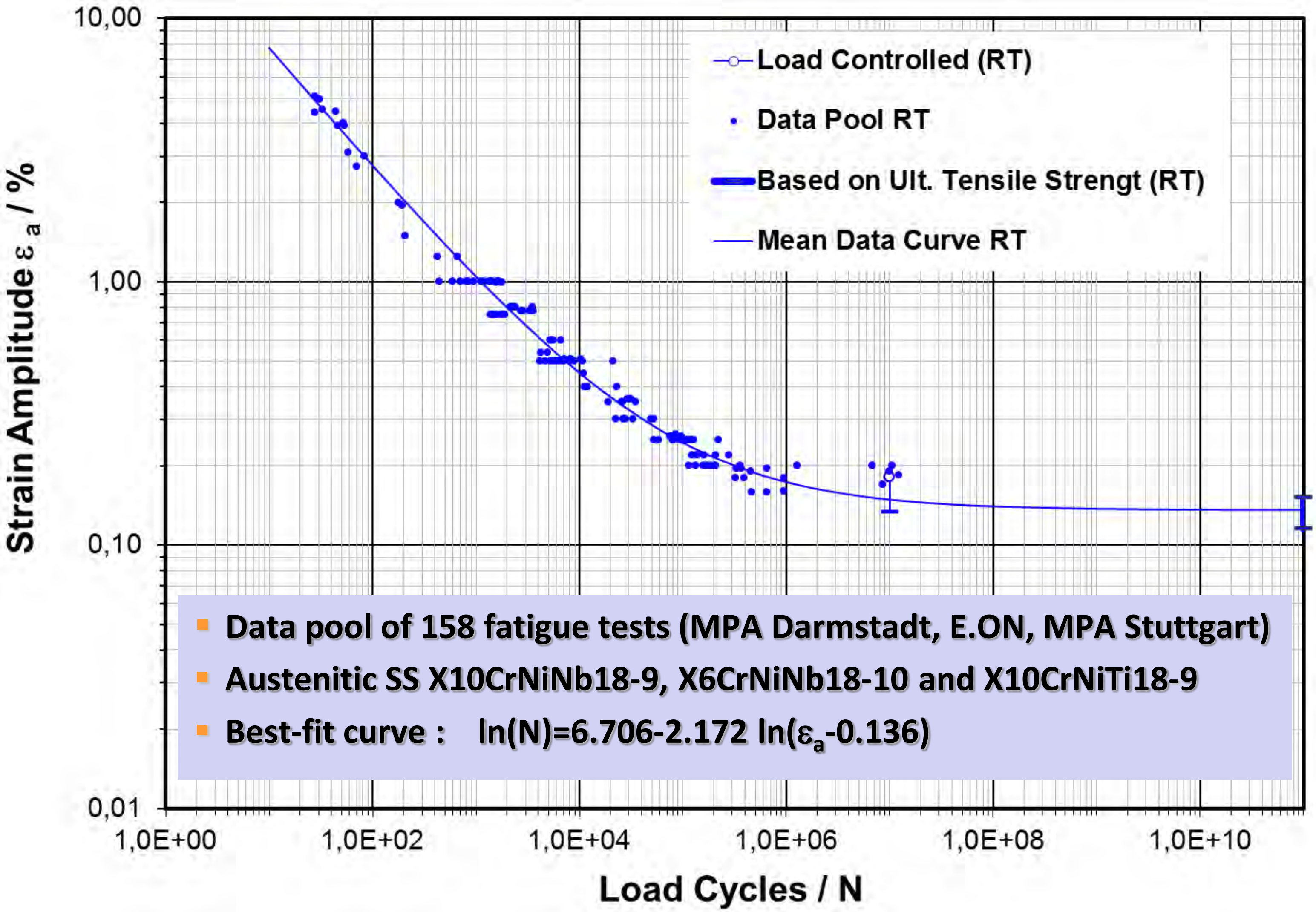
Effects influencing the fatigue life

Parameter	KTA 3201.2, chap. 7.8 Fatigue design curves		Sub-factors acc. to NUREG/CR-6909			
			Load Cycles			Stress
	Cycles	**Load**	**Low Alloy Steels**	**Carbon Steels**	**Austenitic Stainless Steels**	
Material Variability and Scatter of Data						
Temperature			1.5 (covered by factor 2.0 up to 300°C)	1.5 (covered by factor 2.0 up to 300°C)	No influence < 350°C	
Strain Rate			covered by factor 2.0	covered by factor 2.0		
Cyclic Strain Behavior			covered by factor 2.0	covered by factor 2.0		
Heat-to-Heat Variability			2.1 – 2.8 (95% Percentile, 95% confidence level)	2.1 – 2.8 (95% Percentile, 95% confidence level)	2.3 (95% Percentile, 95% confidence level)	
Factor	**2.0**					
Size Effect			1.2 – 1.4	1.2 – 1.4	1.2 – 1.4	
Weldment			(covered by factor 2.5)	(covered by factor 2.5)	(covered by factor 2.5)	
Factor	**2,5**					
Surface Finish, Environment						
Surface Finish			2 – 3.5 (covered by factor 4.0)	2 – 3.5 (covered by factor 4.0)	2 – 3 (covered by factor 4.0)	
Factor	**4.0**					
Load Sequence			1.2 – 2	1.2 – 2	1.2 – 2	2
Total Faktor	**20**	**2**	**6 - 27**			**2**

Experimental investigations

- **LCF / HCF fatigue tests**
  - **Strain and stress controlled in air environment at RT, 240°C, 288°C and 350°C**
  - **Strain controlled in BWR environment at 240°C**
    - **Oxygen content of the water is adjusted to 400 ppb, conductivity 0.05 µS/cm**
    - **Few test by dosing sulphate (90 ppb SO_4 by adding highly diluted sulfuric acid H_2SO_4), conductivity 0.8 µS/cm**
    - **Flow velocity 0.004 m/s**
  - **Materials used in German nuclear power plants**
    - **Low alloy ferritic steel 20MnMoNi5-5 (similar to ASTM A533, Gr. B, Cl. 2)**
    - **Low alloy ferritic steel 22NiMoCr3-7 (similar to ASTM A508, Cl. 2)**
    - **Austenitic SS X6CrNiNb18-10 and X10CrNiNb18-9 (similar to ASTM TP347)**
    - **Austenitic SS X10CrNiTi18-9 (similar to ASTM TP321)**
  - **Test data from smooth cylindrical specimens with fully reversed (R=-1) loading condition, diameter 10 mm, surface roughness R_z < 2µm**

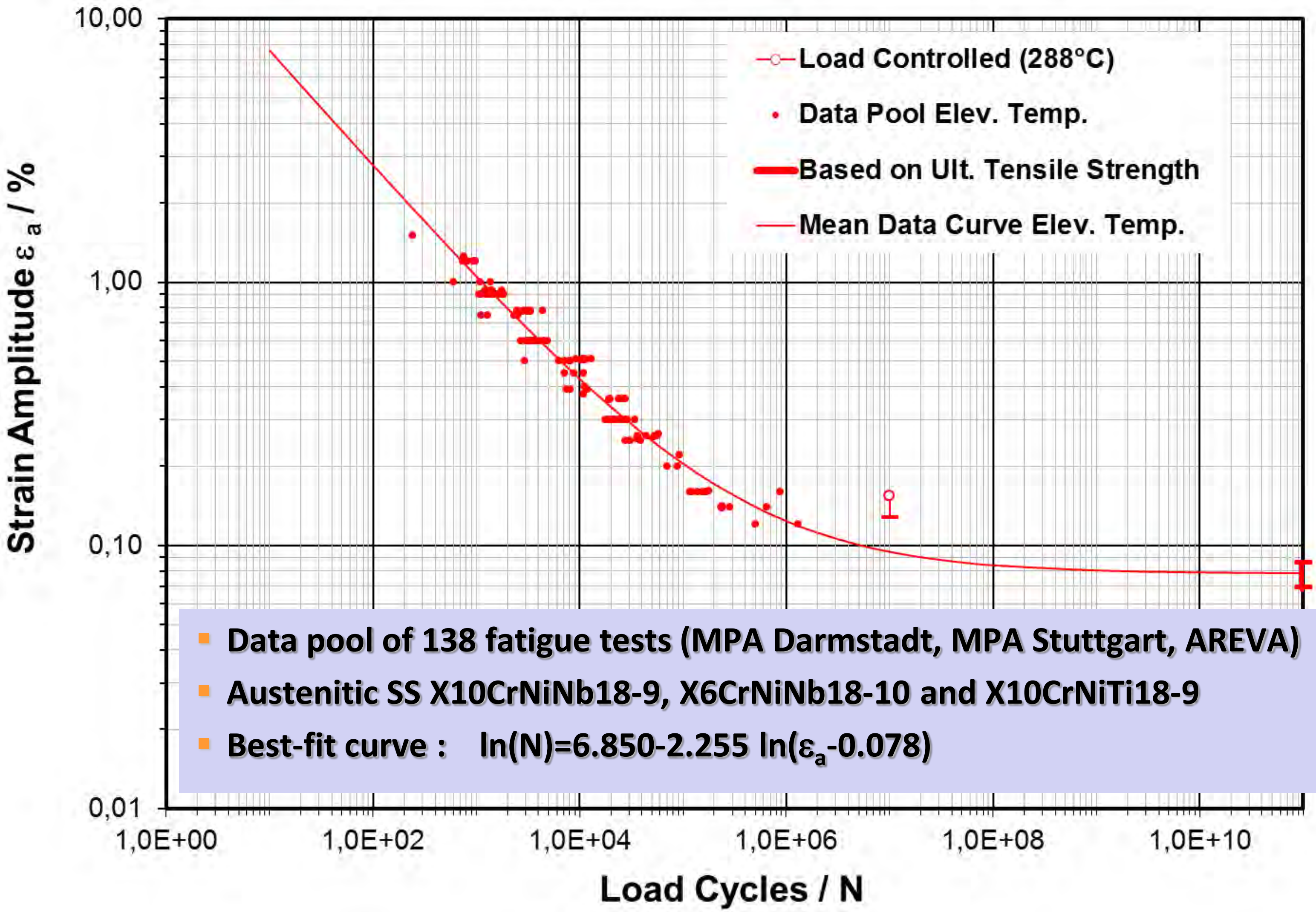
- **LCF tests in air environment at room temperature (RT) - austenitic stainless steel**



- **Data pool of 158 fatigue tests (MPA Darmstadt, E.ON, MPA Stuttgart)**
- **Austenitic SS X10CrNiNb18-9, X6CrNiNb18-10 and X10CrNiTi18-9**
- **Best-fit curve :** $\ln(N)=6.706-2.172\ \ln(\varepsilon_a-0.136)$

Experimental investigations

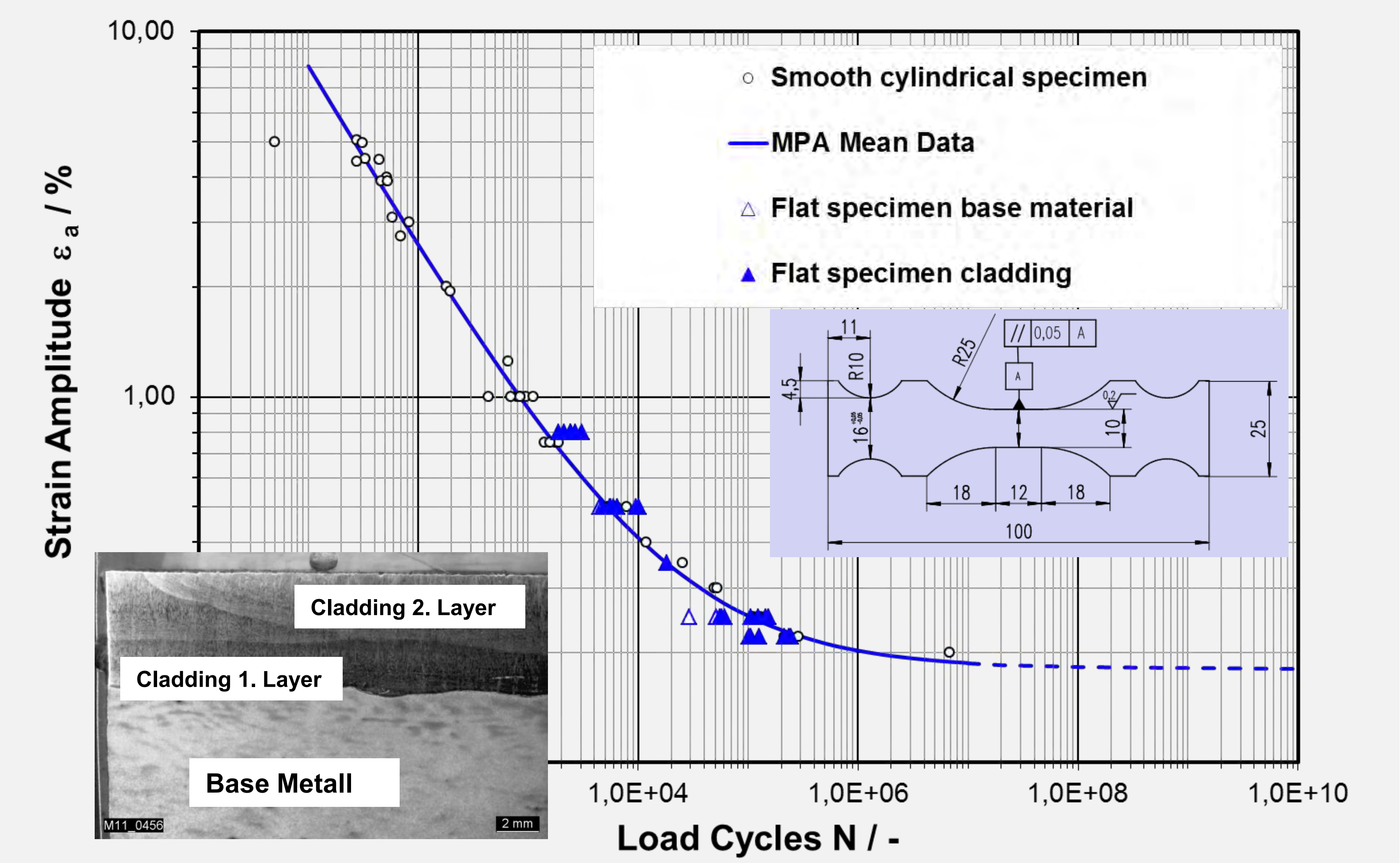
- LCF tests in air environment at elevated temperatures
  - austenitic stainless steel



- Data pool of 138 fatigue tests (MPA Darmstadt, MPA Stuttgart, AREVA)
- Austenitic SS X10CrNiNb18-9, X6CrNiNb18-10 and X10CrNiTi18-9
- Best-fit curve : $\ln(N)=6.850-2.255 \ln(\varepsilon_a-0.078)$

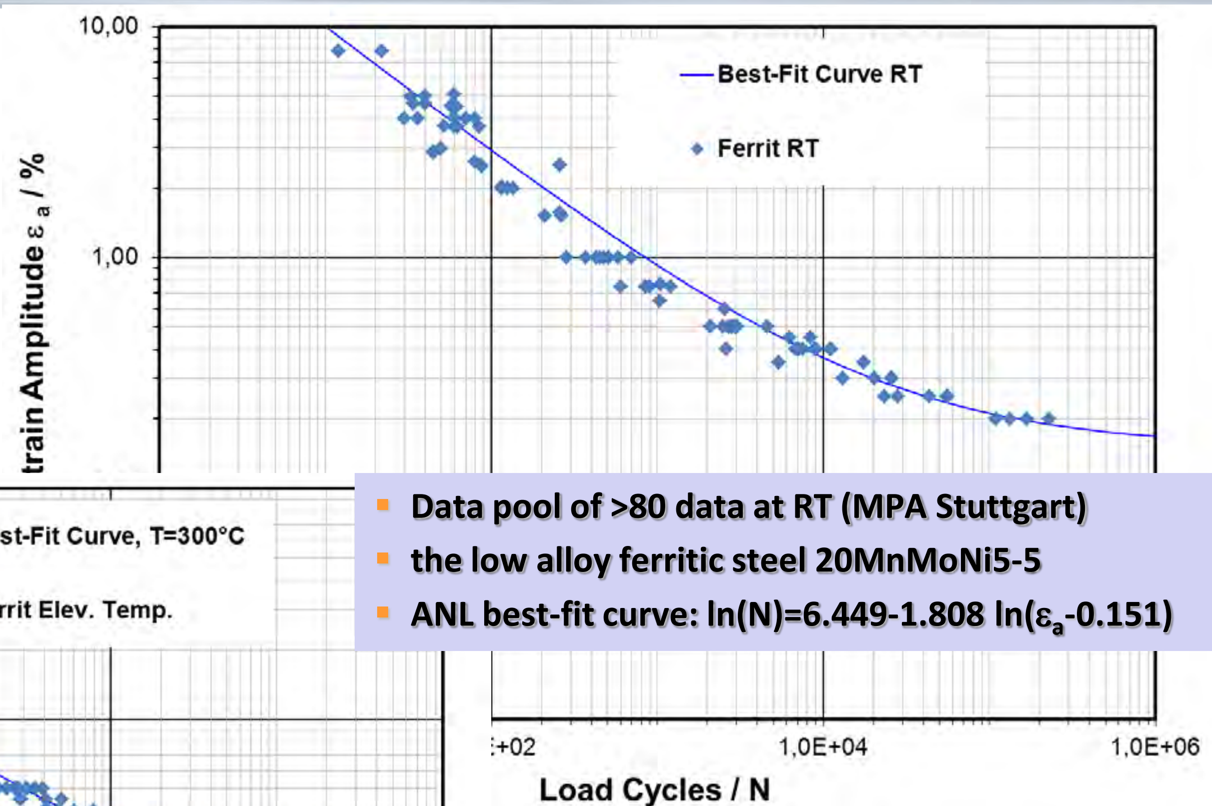
LCF tests in air environment at RT - austenitic cladding

Taken from RPV wall of NPP Biblis Block C (1. layer: Thermanit 23/11 Enb (X2CrNiNb24-12, material-no. 1.4556); 2. layer: Thermanit 22/11 Enb (X2CrNiNb21-10, material-no. 1.4555)), thickness of cladding ca. 7 mm

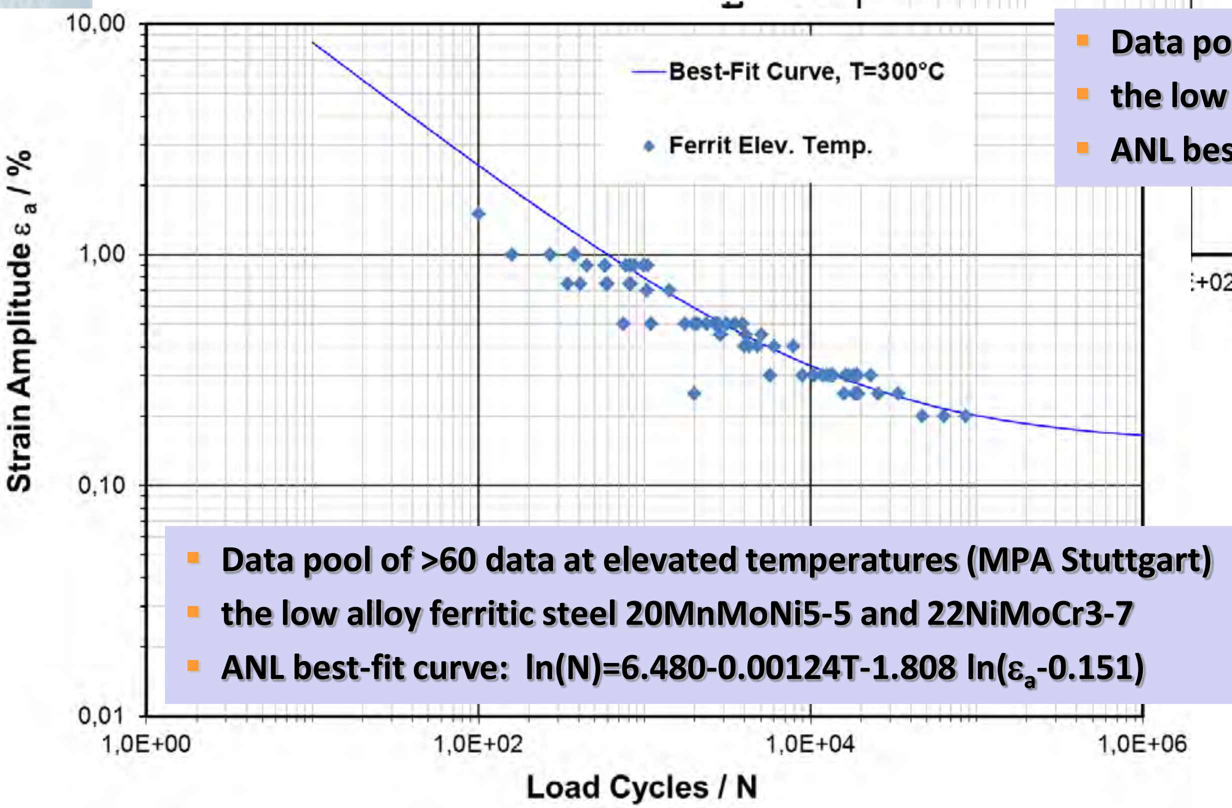


Experimental investigations

- **LCF tests in air environment - low alloy steels**



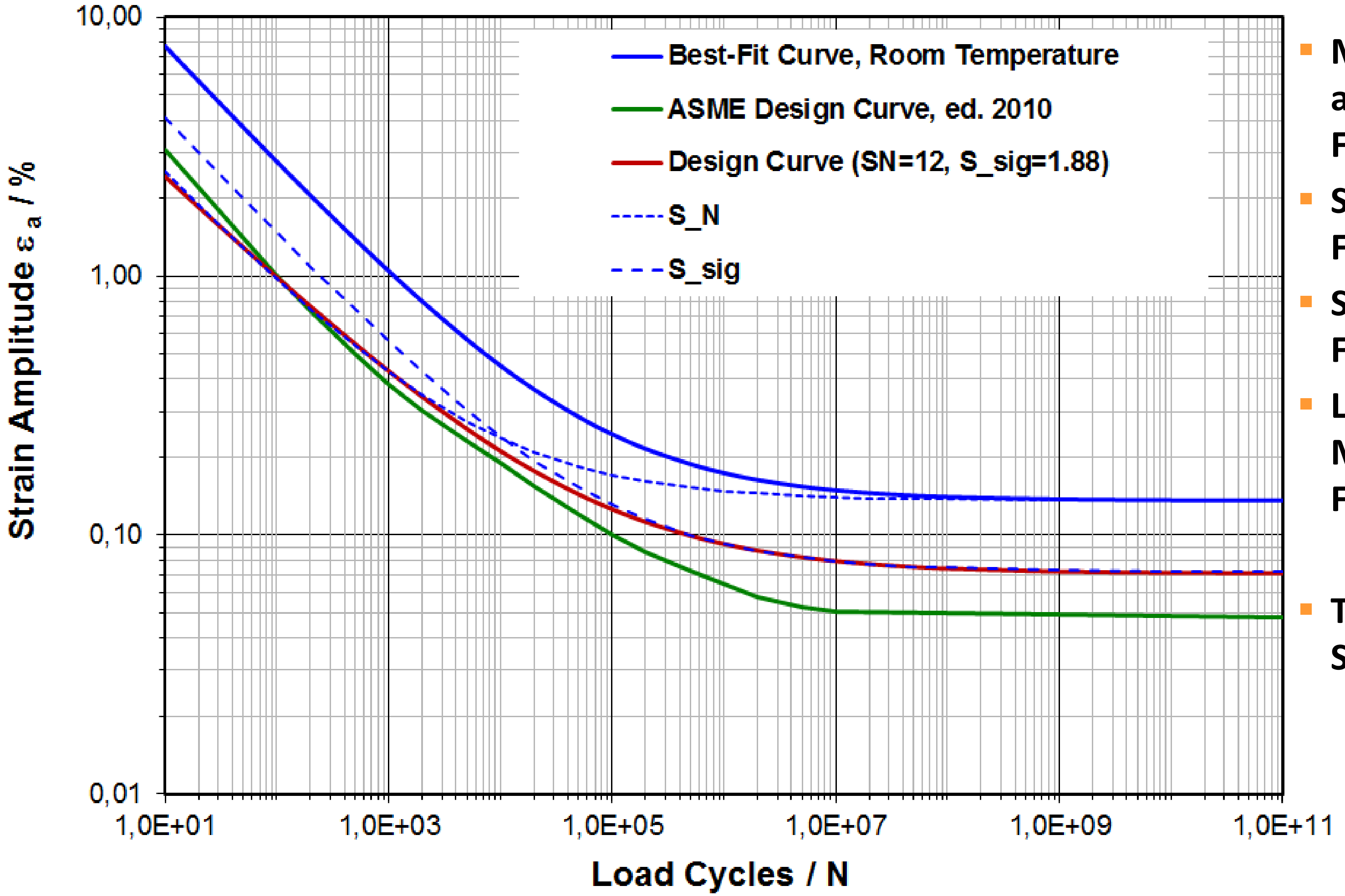
- Data pool of >80 data at RT (MPA Stuttgart)
- the low alloy ferritic steel 20MnMoNi5-5
- ANL best-fit curve: $\ln(N)=6.449-1.808\ \ln(\varepsilon_a-0.151)$



- Data pool of >60 data at elevated temperatures (MPA Stuttgart)
- the low alloy ferritic steel 20MnMoNi5-5 and 22NiMoCr3-7
- ANL best-fit curve: $\ln(N)=6.480-0.00124T-1.808\ \ln(\varepsilon_a-0.151)$

Fatigue best-fit and design curves

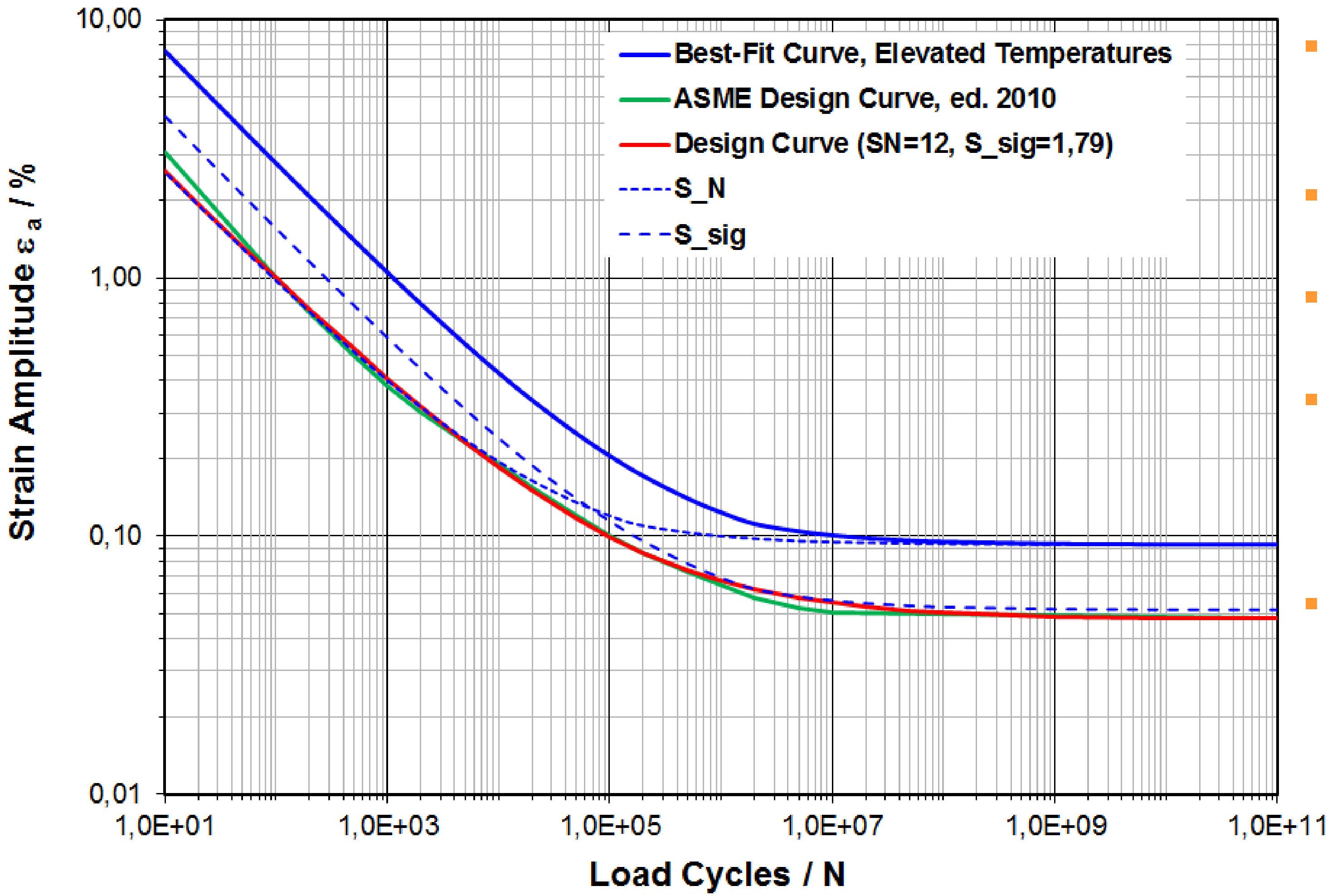
- **Best-fit and design curves in air environment at room temperature (RT) - austenitic stainless steel** (X10CrNiNb18-9, X6CrNiNb18-10 and X10CrNiTi18-9)



- **Material variability and scatter of data F(-2s) = 1.27**
- **Size effect F_e = 1. 09**
- **Surface finish F_s= 1.27**
- **Loading history Mean stress F_m = 1.07**
- **Total reduction S_σ = 1.88**

Fatigue best-fit and design curves

- **Best-fit and design curves in air environment at elevated temperatures - austenitic stainless steel**
  (X10CrNiNb18-9, X6CrNiNb18-10 and X10CrNiTi18-9)



- Material variability and scatter of data
  F(-2s) = 1.27
- Size effect
  F_e = 1. 09
- Surface finish
  F_s= 1.23
- Loading history
  Mean stress
  F_m = 1.05
- Total reduction
  S_σ = 1.79

Fatigue best-fit and design curves

- **Detrimental & Transferability Factor S_N for Cycles N**
  - **Comparison with NUREG/CR-6909**

	ASME ed.<2007	NUREG/CR -6909	ASME ed.>2007, Reg. Guide 1.207	KTA proposal		EDF proposal (PVP2012)
				Room temp.	Elev. Temp.	
Material variability and scatter of data (min. to mean)	2	2.1 – 2.8				2.5
Size effect	2.5	1.2 – 1.4				1.6
Surface finish, atmosphere, …	4	2.0 – 3.5				2*1.5
Loading history	1	1.2 – 2.0				1
Total	**20**	**6.0 – 27.4**	**12**	**12**	**12**	**12**

Reduction on $\Delta\varepsilon_{tot}$ / 2	2		2			1.4

Fatigue best-fit and design curves

- **Detrimental & Transferability Factor S_σ for Load**
  - **Comparison with NUREG/CR-6909**

	ASME ed.<2007	*NUREG/CR -6909*	*ASME ed.>2007, Reg. Guide 1.207*	*KTA proposal*		*EDF proposal (PVP2012)*
				Room temp.	*Elev. Temp.*	
Material variability and scatter of data (min. to mean)				1.27	1.27	
Size effect				1.09	1.09	
Surface finish, atmosphere, …				1.27	1.23	
Loading history, mean stress				1.07	1.05	
Total Reduction on $\Delta\varepsilon_{tot}$ / 2	**2**		**2**	**1.88**	**1.79**	**1.4**
Reduction on N	20	6.0 – 27.4	12	12	12	12

Fatigue best-fit and design curves

- **Fatigue analysis**
  KTA Safety Standard 3201.1 Draft for Revision 11/2012
  - For materials X10CrNiNb18-9, X6CrNiNb18-10 and X10CrNiTi18-9 new design curves can be approximated at elevated temperatures in terms of strain amplitude by

$$\ln(N) = 4.500 - 2.3650 \ln(\varepsilon_a - 0.0478)$$

  - If environmental effects cannot be excluded, actions shall be taken at the time when the cumulative usage factor reaches the fixed threshold value of CUF = 0.4. For operation beyond these threshold values one of the following measures shall be taken:
    - **Integration of the parts/areas concerned into the inspection/ monitoring program or**
    - **Performance of service relevant laboratory tests or**
    - **Fatigue analyses considering environmental reduction factors (F_{en}) and realistic boundary conditions.**

Summary & Outlook

Summary

- Based on a substantial data pool of austenitic stainless steels and low-alloy ferritic steels best-fit curves are developed
- The best-fit curves are similar to those included in ASME-Code
- The influencing factors to develope the fatigue design curves are verified
- The environmental effects can be followed by the ANL F_{en} factor

Outlook

- General accepted best-fit curves for materials used in German Nuclear power plants
- General accepted design curves based on the best-fit curves and general accepted factors S_N and S_σ
- Observe international developments (ASME-Code, RCC-M)

End of presentation

Thank you for your attention

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DE LA RECHERCHE À L'INDUSTRIE

BENCH-KJ

BENCHMARK ON THE ANALYTICAL EVALUATION OF THE FRACTURE MECHANICS PARAMETERS K AND J FOR DIFFERENT COMPONENTS AND LOADS

S. MARIE, CEA

C. FAIDY, AFCEN

www.cea.fr

GOALS

TASK 1 RESULTS – KI CALCULATION

TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

NEXT STEPS

GOALS

TASK 1 RESULTS – KI CALCULATION

TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

NEXT STEPS

GOALS

Important work in the world on the development of analytical tools for defect assessment (in-service defect assessment, LBB,..) has been made and continues:

- K_I compendia development
- J & reference stress compendia
- Different loading conditions: mechanical loading, thermal loadings, residual stresses,….
- Various component and defect geometries (pipes, elbows,….)

Interest for a comparison of the available solutions (A16, RSE-M, R6, API,…)

- On representative situations
  - Typical fatigue thermal stress distribution for K_I
  - Combination of mechanical & thermal loadings
  - Treatment of cracked weld joints

These analytical tools are more or less complex

- Training activities for young engineers are very important

GOALS

Draft description of the proposed benchmark prepared in 2010

- First official version validated by the OECD/IAGE Metal group : 07/03/2011
- proposed on a period of 24 months (for technical work) + 12 months for conclusions
- Focus on Analytical solutions (some partners provided additional F.E. results) – reference solutions from CEA F.E. calculations

Technical work in 7 tasks

1. Elastic KI evaluation (completed)
   - Simple cases on cracked pipes under mechanical loading
   - ‘exponential’ stress distribution relevant of thermal loading

2. J calculation for surface cracks in pipes (completed)
   - Single mechanical load
   - Combined mechanical load
   - Thermal (+ mechanical) load

3. J calculation for through wall cracks in pipes (completed)

GOALS

Technical work in 7 tasks

4. J calculation for surface cracks in elbows (completed – results not analyzed yet)
    - Single mechanical load
    - Combined mechanical load
    - Thermal (+ mechanical) load

5. Particular cases
    - Imposed displacement loading condition
    - Embedded cracks
    - Underclad cracks
    - Through clad cracks

6. J calculation in weld joints
    - Cracked pipes
    - Mechanical loading conditions only

7. Recommendations
    - Synthesis of the comparison
    - Identification of possible improvement of the different procedures

GOALS

30 interested partners

UK (5)
TWI
EDF ENERGY
Frazer-Nash
AMEC
Zentech Inter. Ltd

Hungary (3)
VEIKI Energia
KFKI AEKI
BAY-LOGI

Belgium (1)
Tractebel

Sweden (2)
SSM
Inspecta

France (3)
CEA
EDF
AREVA

Czech republic (1)
IAM Brno

Netherlands (1)
JRC Petten

Finland (1)
VTT

Spain(1)
CSN

Italy (1)
ENEA

Germany (1)
GRS

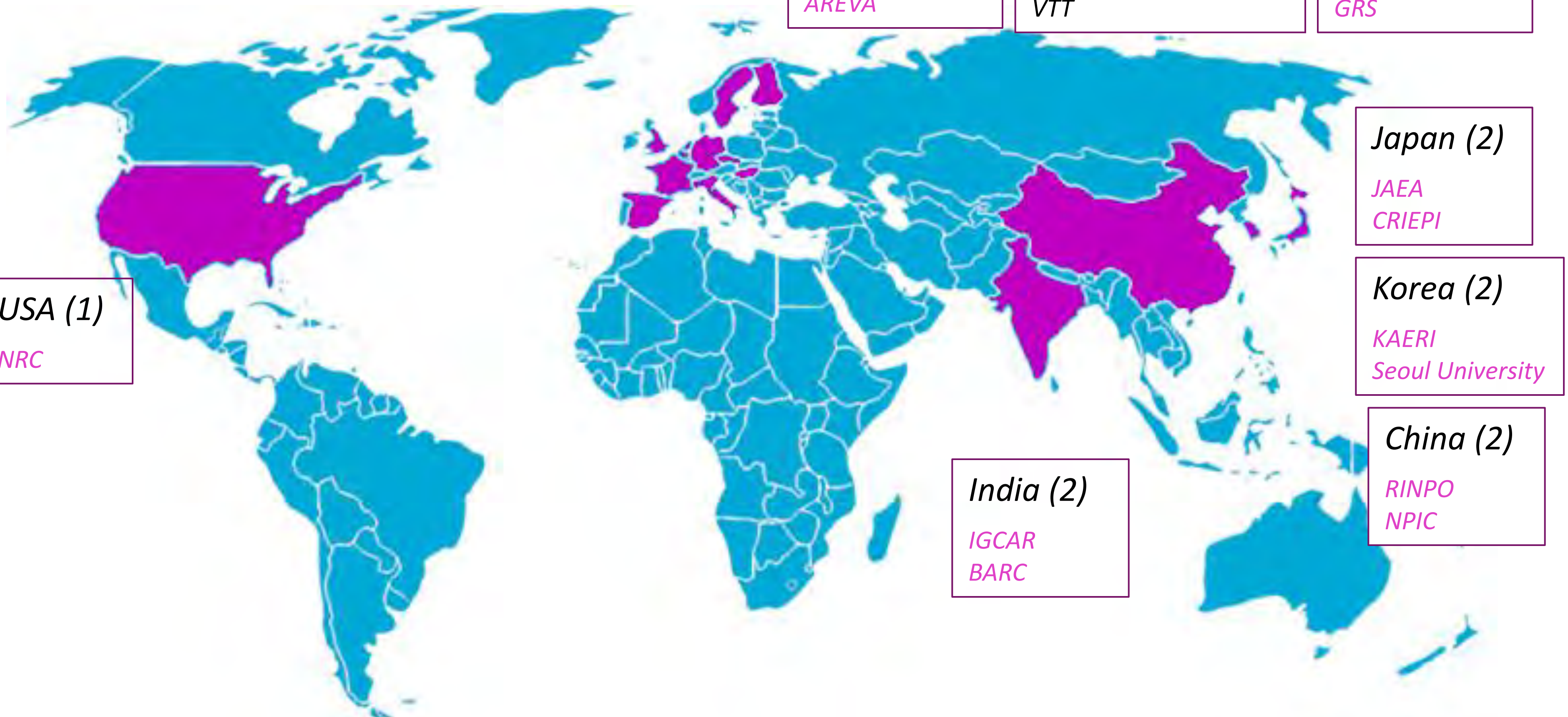
Japan (2)
JAEA
CRIEPI

Korea (2)
KAERI
Seoul University

China (2)
RINPO
NPIC

India (2)
IGCAR
BARC

USA (1)
NRC



GOALS

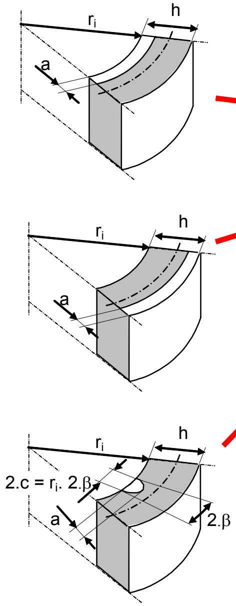
TASK 1 RESULTS – KI CALCULATION

TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

NEXT STEPS

PRESENTATION OF THE TASK 1 CASES

- 3 sub-tasks:
  - Circumferential surface crack in cylinder (3 geometries, 2 loading conditions)



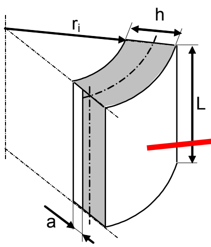
GEOMETRY						
Case #	Geometry #	Defect	a/h	c/a	h (mm)	De (mm)
K1	PIPE 1	CDAI – circumferetial internal axysimetric	0.1 – 0.25 – 0.5 – 0.75	-	60	660
K2	PIPE 2	CDAE – circumferetial external axysimetric	0.1 – 0.25 – 0.5 – 0.75	-	60	660
K3	PIPE 3	CDSI – circumferetial internal semi-elliptical	0.1 – 0.25 – 0.5 – 0.75	3	60	660

Loading condition #	P (MPa)	M1 (N.mm)	M2 (N.mm)
1	25	-	3.50E+09
2	-	1.70E+09	5.20E+09

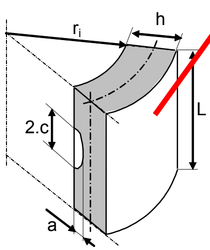
TASK 1 RESULTS – KI CALCULATION

PRESENTATION OF THE TASK 1 CASES

- 3 sub-tasks:
  - Circumferential surface crack in cylinder (3 geometries, 2 loading conditions)
  - Longitudinal surface crack in cylinder (2 geometries, 2 loading conditions)



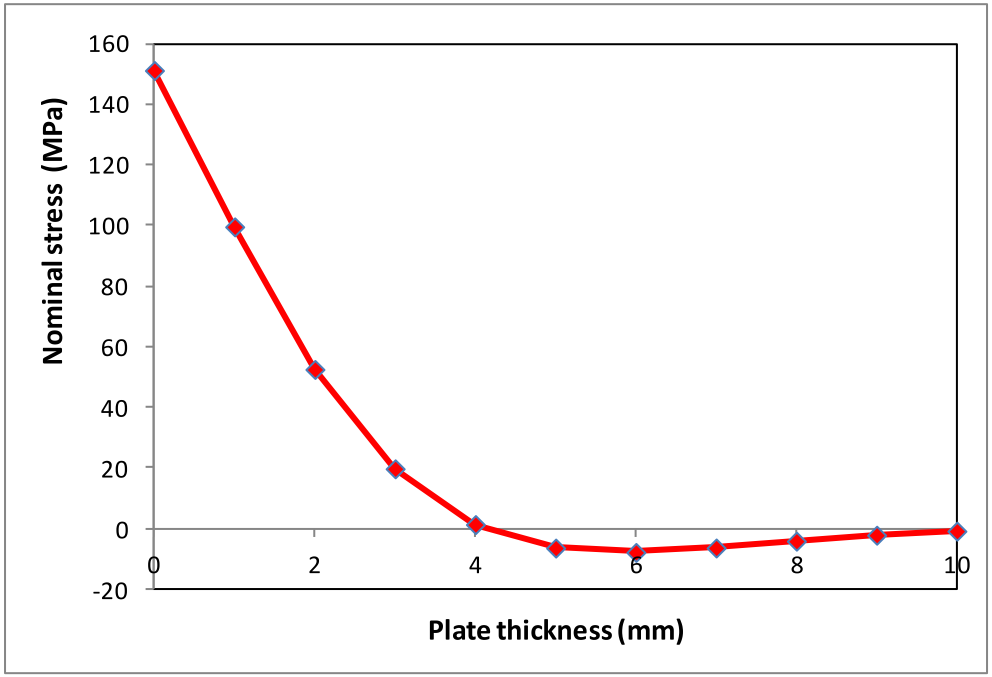
GEOMETRY						
Case #	Geometry #	Defect	a/h	c/a	h (mm)	De (mm)
K4	PIPE 1	LDII – longitudinal internal infinite	0.1 – 0.25 – 0.5 – 0.75	-	60	660
K5	PIPE 2	LDSI – longitudinal internal semi-elliptical	0.1 – 0.25 – 0.5 – 0.75	3	60	660



Loading condition #	P (MPa)	M1 (N.mm)	M2 (N.mm)
1	50	-	-
2	50	-	6.0E+09

PRESENTATION OF THE TASK 1 CASES

- 3 sub-tasks:
  - Circumferential surface crack in cylinder (3 geometries, 2 loading conditions)
  - Longitudinal surface crack in cylinder (2 geometries, 2 loading conditions)
  - Plate under thermal loading (1 geometry, 1 loading condition)



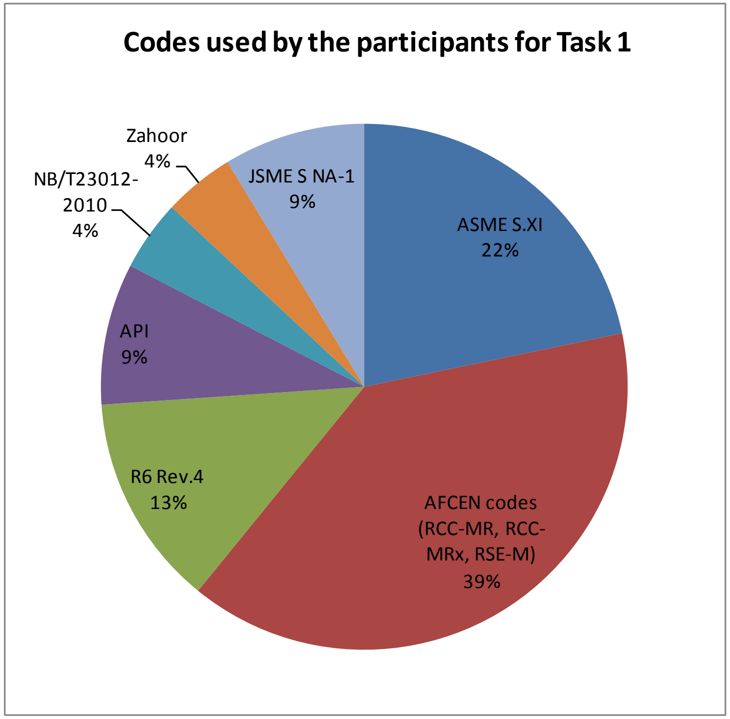
TASK 1 RESULTS – KI CALCULATION

Analysis of the results for Task1

- 21 answers received (11 fully complete – 15 complete on cracked pipes)

- Information on code(s) used
  - NRC, RINPO & Seoul Univ. provided results for two codes
  - AFCEN codes (RCC-MRx, RSE-M, RCC-MR) – 9
  - ASME Section XI – 5
  - R6 Rev.4 – 3
  - API 579 – 2
  - JSME – 2
  - NB/T23012-2010 – 1
  - Zahoor – 1

- 5 Partners provide F.E. results
  - Comparison with reference solutions



TASK 1 RESULTS – KI CALCULATION

Analysis of the results for Task1

- Technical work in two steps
  - First step : full blind test application
  - Second Step : relative errors are provided
    - Possibility for the partner to provide a new set of results
    - Understanding of the discrepancies

- For each step, analysis of :
  - The results homogeneity for a given code (when possible)
  - Comparison of the predictions obtained with the different codes

TASK 1 RESULTS – KI CALCULATION

Analysis of the results for Task1 : main conclusions

- F.E. reference calculation
  - CEA reference calculation are confirmed
  - Several errors have been identified in the F.E. models on :

    - The defect mesh (respect of the shape)

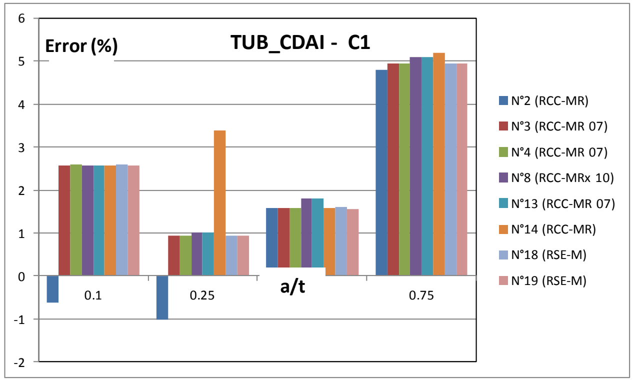
    - The pressure on crack lips for internal defects

    - Contact boundary condition to avoid any elements interpenetration when a part of the defect is submitted to compression

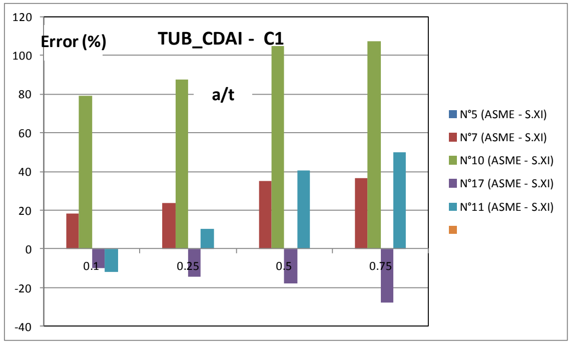
TASK 1 RESULTS – KI CALCULATION

Analysis of the results for Task1 : main conclusions

- Results homogeneity
  - Very good for AFCEN code (9 partners)
    - even for the fist step
  - Very good for R6 (3 partners)
    - But all partners using R6 forgot the pressure on the crack lip during the first step
  - Very good for JSME code (2 partners)
    - At the second step
  - Problem for ASME code users (5 partners)
    - No consistent results and often far from the F.E. solution
    - The possibility of different solutions in section XI does not explain the discrepancies
    - Some explanation in wrong nominal elastic stresses (F.E.), use of safety coefficient (1 partner)
    - But today, main part of these particular results is not understood yet



Example of the results consistency for AFCEN codes – case K1 – first step



Example of the results consistency for ASME code – case K1 – first step

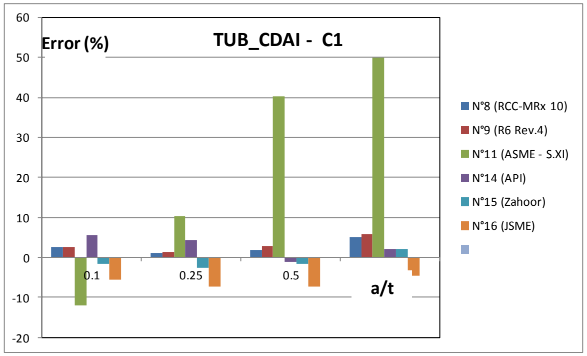
Analysis of the results for Task1 : main conclusions

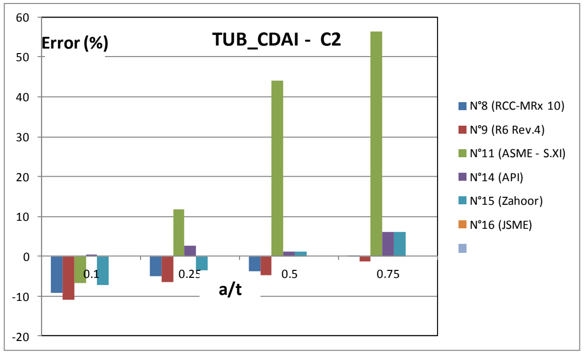
- Codes comparison
  - for each code, one representative partner is selected
  - This is made once the 2nd turn results have been provided to be sure that differences between the different codes is just due to the compendia accuracy
  - This still remains problematic for ASME code as none ASME partners sent revise results, and as it is not clear which is the correct result from the ASME Section XI application. We selected the partner who provided the closest results from the F.E. solution

TASK 1 RESULTS – KI CALCULATION

Analysis of the results for Task1 : main conclusions

- Codes comparison
  - For cracked pipes, AFCEN codes, R6 and API provide in general relatively correct estimation for the F.E. reference solution (less than max ± 10%).
  - JSME code provides also close results but systematically under predict the F.E. solution. The observed differences remains nevertheless less than -10%
  - Zahoor solution is in general relatively good, but the difference with F.E. calculation is often larger than the 4 first codes.
  - ASME results are really problematic. Surprisingly, whereas section XI, appendix C is based on Zahoor compendia, none ASME partner provided similar results than partner 15 who directly used Zahoor solution
  - For the plate case with an exponential nominal elastic stress distribution, all codes (bases on a polynomial representation of the nominal stress) provided very comparable results and all over predict F.E. reference solution.





Example of the code results comparison – case K1

TASK 1 RESULTS – KI CALCULATION

Analysis of the results for Task1 : main conclusions

- Codes comparison : Analysis of the errors

  - The more prescriptive is the code, the more homogeneous are the different contribution.
  - The most common error was the pressure on the crack lip which wasn't taken into account for internal defects.
  - Other particular errors have identified as no consideration for the end caps effect for circumferential defects.
  - Several mistakes have been reported on the nominal elastic stresses, in particular when using F.E. solution.

GOALS

TASK 1 RESULTS – KI CALCULATION

TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

NEXT STEPS

TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

Task 2: J calculation for cracked pipes with a surface crack

- 4 sub-tasks
  - Circumferential surface crack submitted to mechanical loadings (11 cases)
  - Longitudinal surface crack submitted to mechanical loadings (9 cases)
  - Elementary thermal loads (longitudinal defects: 7 cases, circumferential defects: 14 cases)
  - Combined mechanical + thermal load conditions (longitudinal defects: 5 cases, circumferential defects: 6 cases)

Task 3: J calculation for cracked pipes with a circumferential throughwall crack

- Mechanical loading conditions only – 4 cases

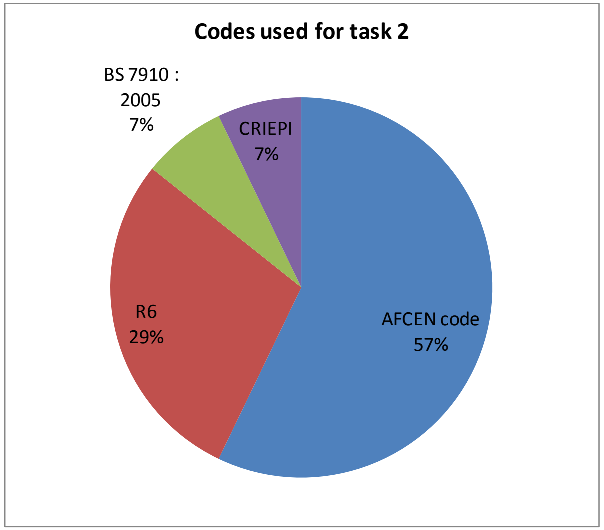
TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

Analysis of the results for Tasks 2-3

- 14 contributions with analytical solutions have been received: AREVA, GRS, INSPECTA, AMEC, JRC, CEA, KAERI, EDF, EDF Energy, GDF-SUEZ (only task 3), BARC, Seoul University, CRIEPI, TWI

- 3 contributions with F.E. results have been received: University of Seoul, RINPO, FORTUM (only Through-wall cracks)

- Information on code(s) used
  - AFCEN codes (mainly RCC-MRx A16 appendix) – 8 partners
  - R6 Rev.4 – 4 partners
  - CRIEPI – 1 Partner
  - BS 79910 : 2005 – 1 Partner



TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

Analysis of the results for Tasks 2 & 3

- Technical work in one step done today
  - First step : full blind test application
  - Some partners sent recently some additional information and/or new set of results

- For this first step, analysis of :
  - The results homogeneity fro a given code (when possible)
  - Comparison of the predictions obtained with the different codes

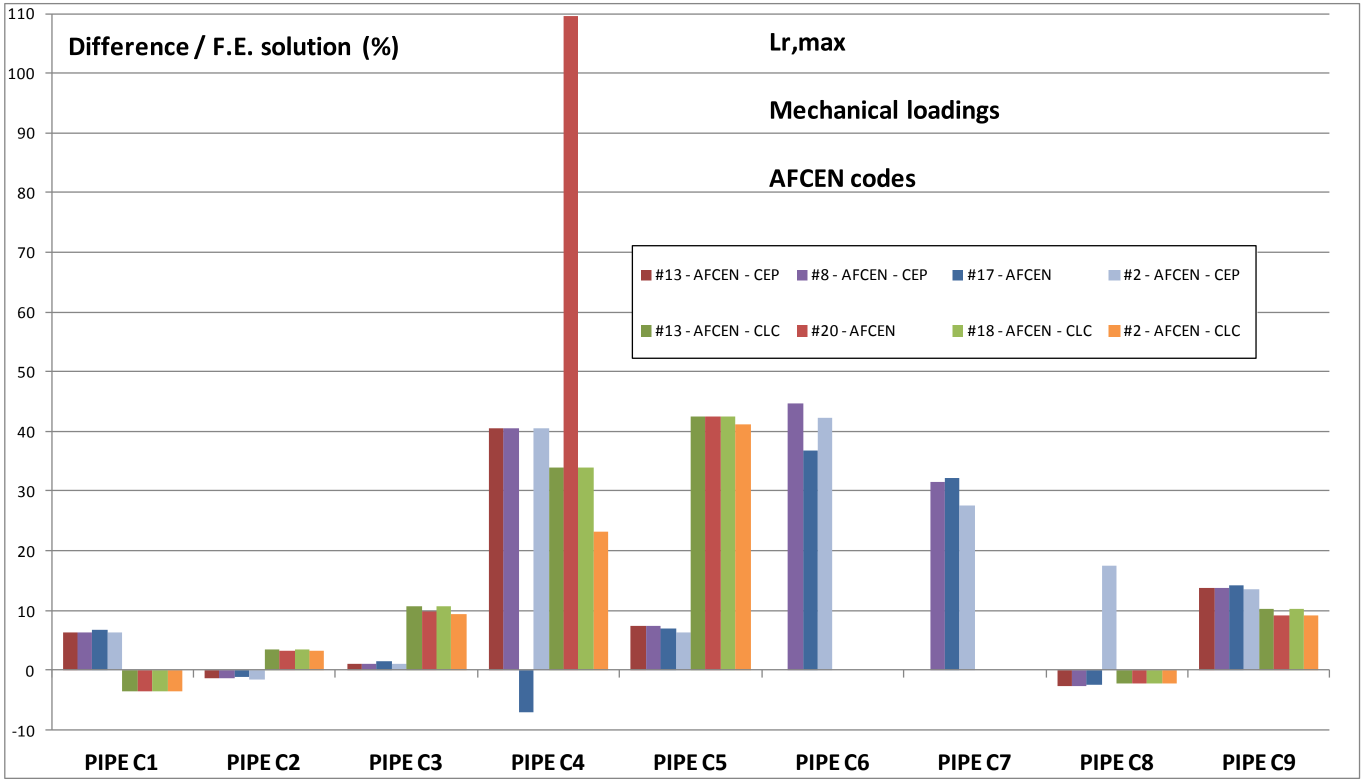
TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

Analysis of the results for Tasks 2-3 : main conclusions

- F.E. reference calculation
  - CEA reference calculation for surface cracks are confirmed
    - Some differences are obtained for longitudinal defects with M_2 BUT in CEA calculation, the defect is in the M_2 compressive part (which open the defect and then I the most conservative situation), whereas other partners put the defect in the tensile part
    - The result analysis have been performed only on cases for which at least two partners provided comparable F.E. results (within +-10%)
  - No agreement is obtained for throughwall cracks
    - One order of magnitude is found between CEA and one partner !
    - Analytical solutions appears to be close to the CEA reference solution
    - CEA F.E. procedure have been benchmarked with EDF and AREVA in the past

Analysis of the results for Tasks 2-3 : main conclusions

- Results homogeneity – task 2 – mechanical loadings
  - Very good for AFCEN code (8 partners)
    - Two options are available for the reference stress calculation – for each option, the homogeneity is ok



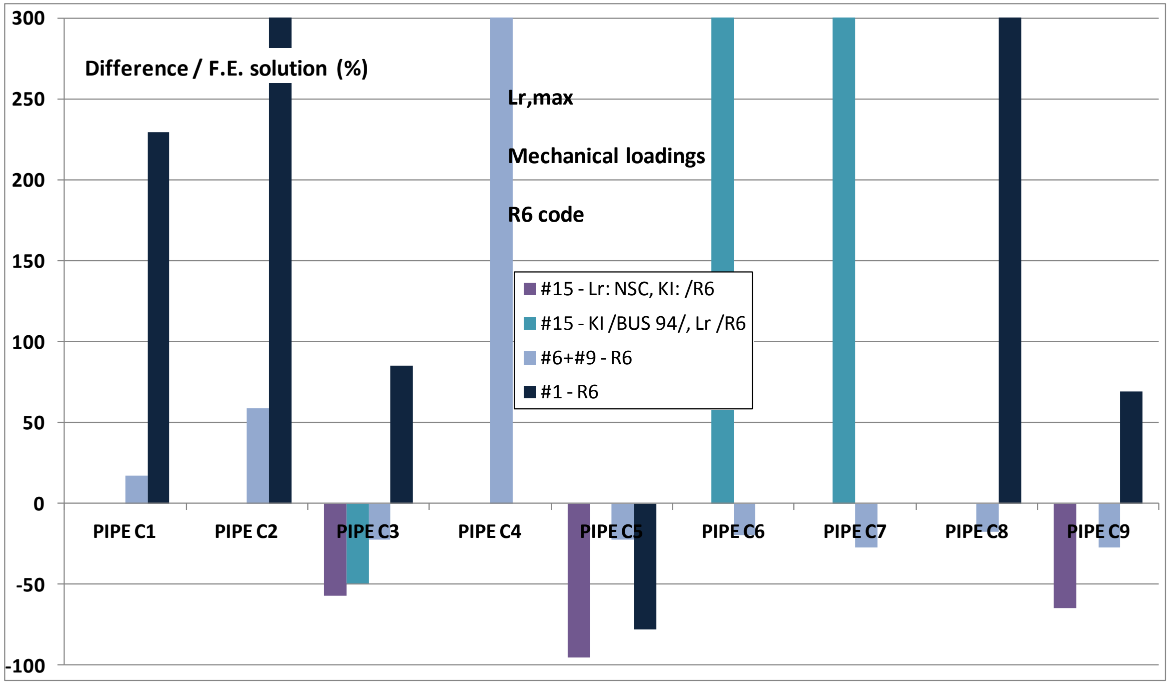
Task 2 - AFCEN codes – Circumferential defects – Mechanical loading Lrmax

TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

Analysis of the results for Tasks 2-3 : main conclusions

- Results homogeneity – task 2 – mechanical loadings
  - Very good for AFCEN code (8 partners)
    - Two options are available for the reference stress calculation – for each option, the homogeneity is ok
  - R6 (4 partners) : important differences – probably relevant of users' mistakes (partner #1 sent additional information)

Task 2 - AFCEN codes – Circumferential defects – Mechanical loading Lrmax



TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

Analysis of the results for Tasks 2-3 : main conclusions

- Results homogeneity – task 2 – mechanical loadings
  - Very good for AFCEN code (8 partners)
    - Two options are available for the reference stress calculation – for each option, the homogeneity is ok
  - R6 (4 partners) : important differences – probably relevant of users' mistakes (partner #1 sent additional information)

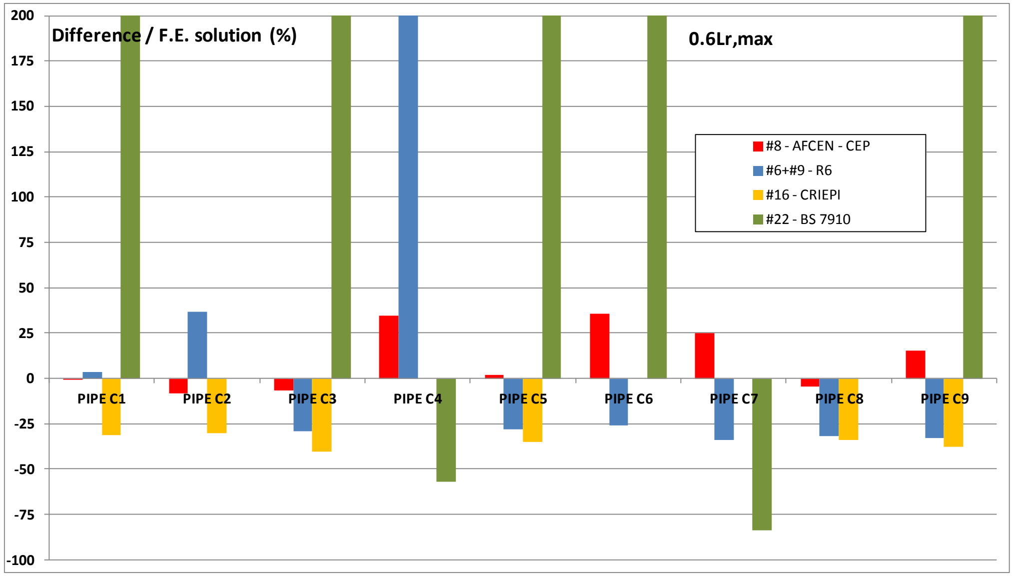
- Results homogeneity – task 2 – thermal (+mechanical) loadings
  - Very good for AFCEN code (6 partners) for pure ΔT_1 cases
    - For ΔT_2, major part of the partners did not considered ΔT_2 in K_I whereas solution is available (only 2 partners)

TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

Analysis of the results for Tasks 2-3 : main conclusions

- Codes comparison – task 2 - mechanical loadings
  - **BS 7910:2005** largely overestimates the J values. It wasn't even possible to produce values for mechanical loading intensity Lrmax.
  - **AFCEN codes** provide reasonable results, with prediction between -10% and 45% of F.E. results for circumferential defects and between 0% and 80% for longitudinal defects.
  - CRIEPI (Yellow) provide in general underestimates the reference value for circumferential defects (around -20%)
  - Situation is similar with **R6** code but with some positive cases with circumferential defects and, with a worst situation for longitudinal defects.

Task 2 – Codes comparison – Circumferential defects - mechanical loadings.

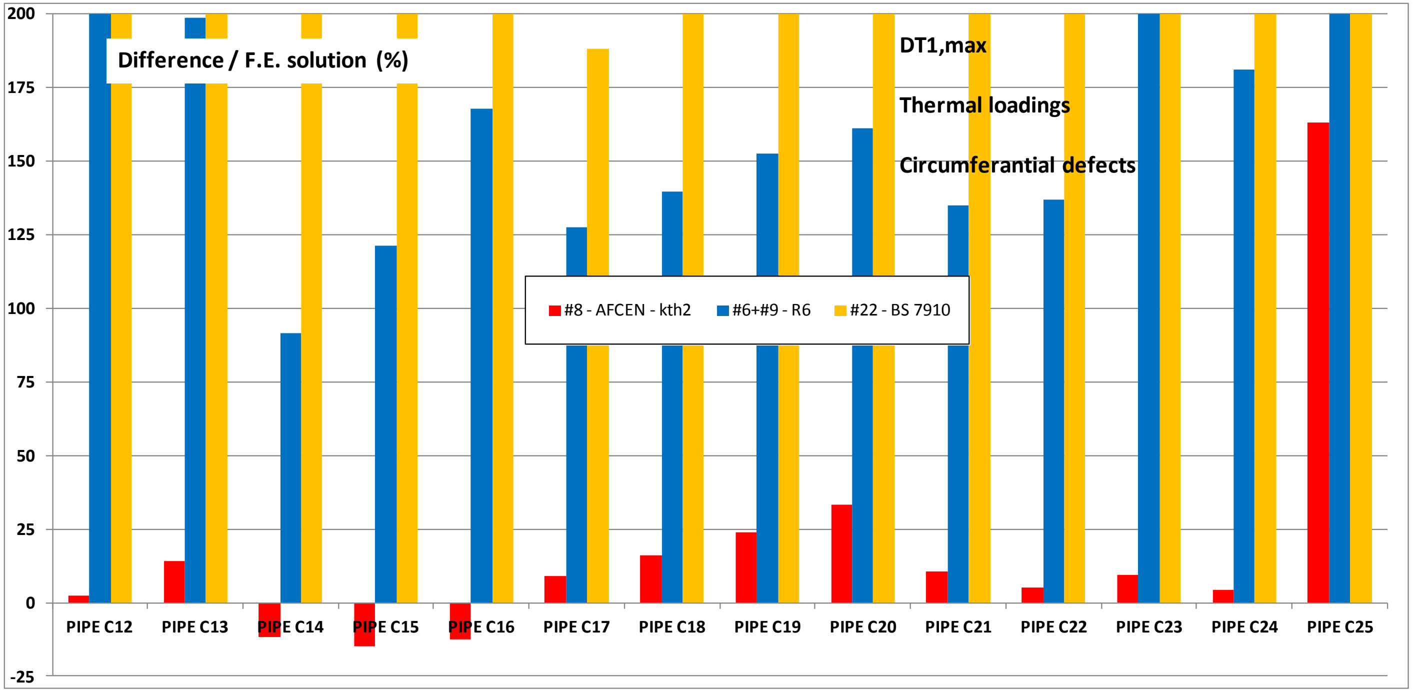


TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

Analysis of the results for Tasks 2-3 : main conclusions

- Codes comparison – task 2 – pure thermal loadings
  - **BS 7910:2005 (yellow)** results are over-conservative
  - **AFCEN codes** provide slight conservative prediction,
  - **R6** code appears more conservative (elastic solution)

Task 2 – Codes comparison – Circumferential defects - thermal loadings.

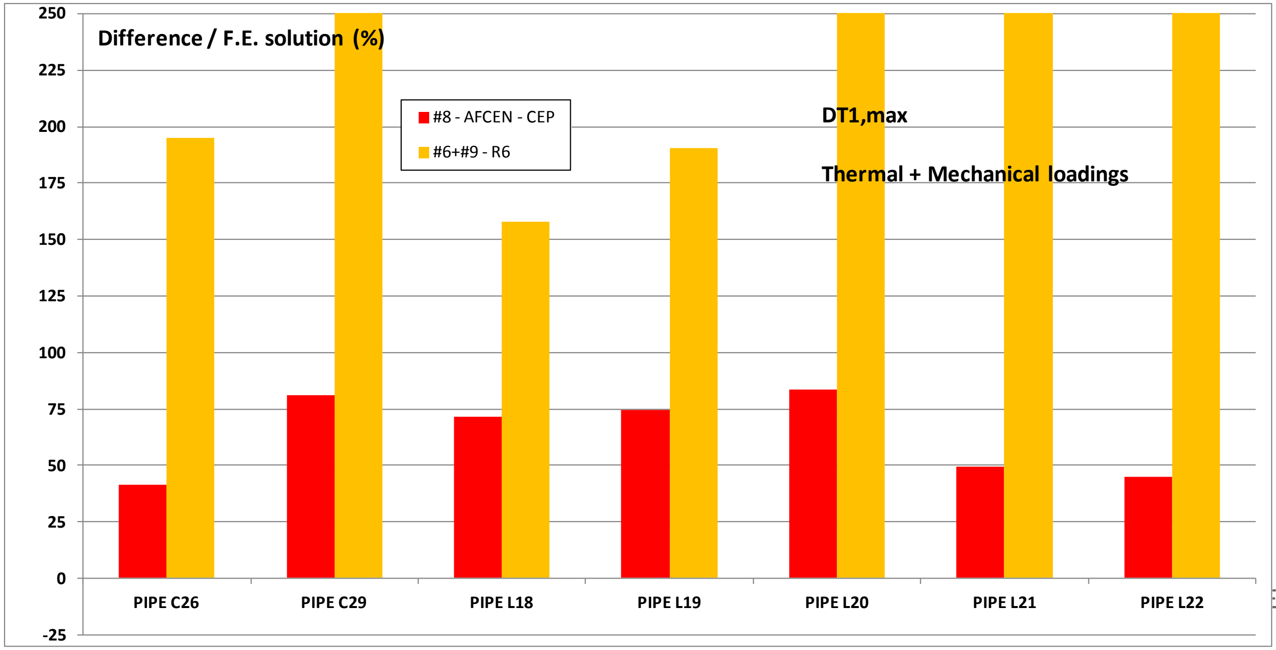


TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

Analysis of the results for Tasks 2-3 : main conclusions

- Codes comparison – task 2 – thermal +mechanical loadings
  - **AFCEN codes** provide slight conservative prediction,
  - **R6 (yellow)** code appears more conservative

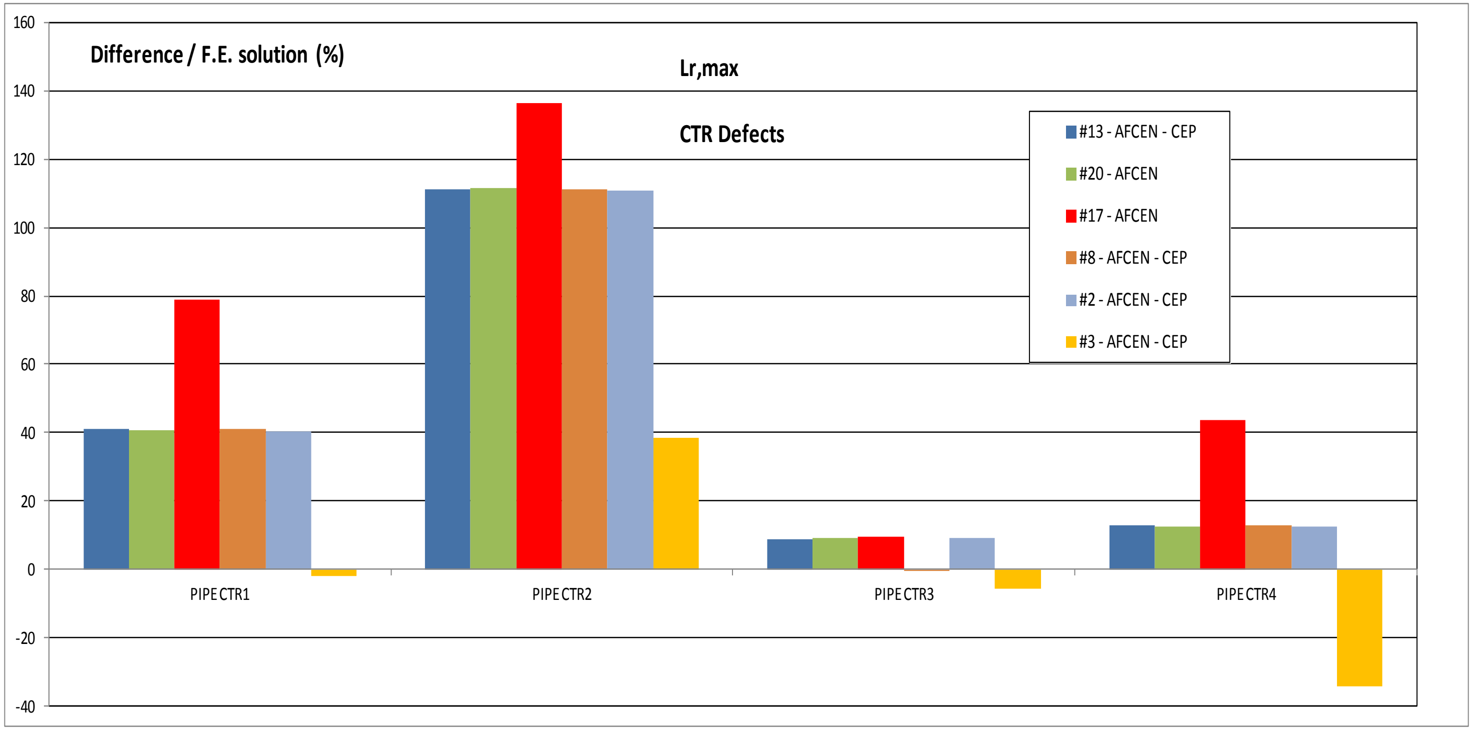
Task 2 – Codes comparison – Combined thermal + mechanical loadings – DT1,max.



TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

Analysis of the results for Tasks 2-3 : main conclusions

- Results homogeneity – task 3 – mechanical loadings
  - Very good for AFCEN code (6 partners)

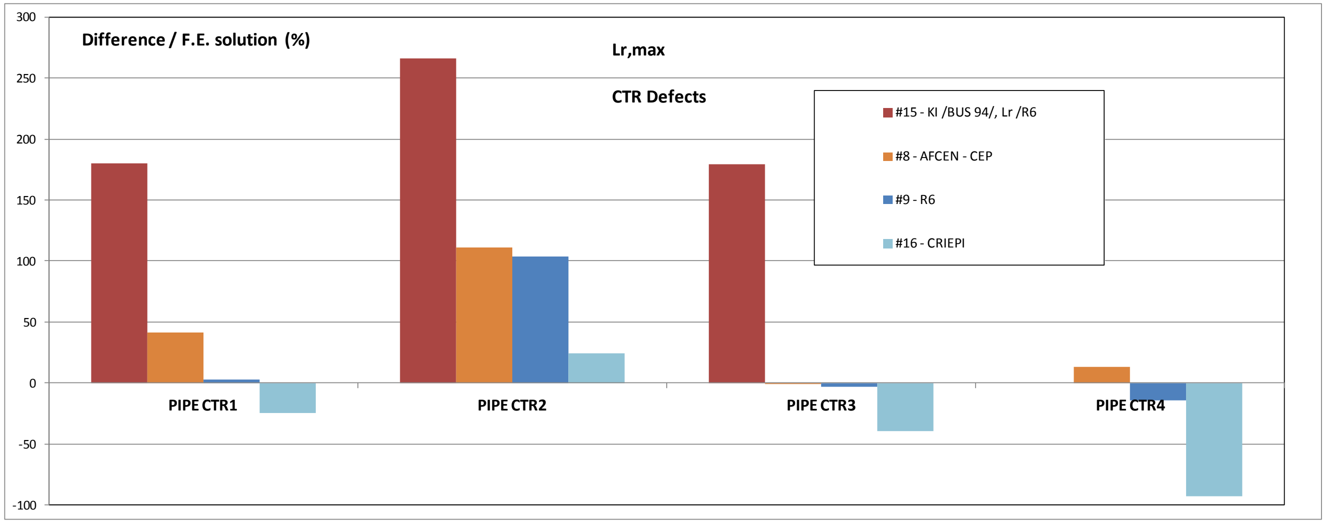


TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

Analysis of the results for Tasks 2-3 : main conclusions

- Results homogeneity – task 3 – mechanical loadings
  - Very good for AFCEN code (6 partners)

- Code comparison – task 3 – mechanical loadings
  - Comparable results for AFCEN codes and R6 procedure
  - Zahoor solution more conservative
  - CRIEPI not conservative (mistake ?)



TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

Analysis of the results for Tasks 2-3 : main conclusions

- Code comparison : Analysis of the discrepancies
  - Several partners did not considered ΔT_2 in K_I whereas solution is available (at least in AFCEN codes)
  - Error in the Elastic nominal stresses calculation when using F.E.
  - Thermal loadings : Elastic solution for R6 (more conservative) – Elastic-Plastic for AFCEN
  - When using the R6 code there is different R6 curves that can be used in estimating J. It should be stated which type of R6 curve that has been used (the accuracy at high load levels depends very much on which curve that is used)

GOALS

TASK 1 RESULTS – KI CALCULATION

TASKS 2-3 RESULTS – J CALCULATION IN CRACKED PIPES

NEXT STEPS

NEXT STEPS

Task 2-3 : Cracked pipes

- For F.E. Solution, a specific focus have to be made on through wall cracks
- Some particular results have been identified and must be investigated

Task 4 : Cracked elbows

- 8 Contributions received – not analyzed yet

Task 5 : particular cases elbows & Task 6 : Welds

- Contributions are awaited – still possible to participate

NEXT STEPS

Understanding of the discrepancies

- It is important to go as deep as possible in the results analysis
  - Main sources of errors made in the analyses or of differences between codes have been identified for task 1 and partially for task 2 and 3
  - But it is possible to go deeper :
    - Identification sources of differences for each code (different options ?,..)
    - Impact of the skill level of the user ?
- Once the 6 tasks performed, preparation of a questionnaire :
  - To have the maximum level of details on the compendia and options used
  - To evaluation the knowledge of the user (in fracture mechanics & in the code used)

Responses to Survey on Leak Before Break

Robert L. Tregoning *
Sr. Technical Advisor for Materials Engineering

Nuclear Regulatory Commission
Office of Nuclear Regulatory Research

CSNI WGIAGE Annual Meeting
April 9 – 12, 2013
Paris, Fr

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

** The views expressed herein are those of the author and do not represent official positions of the U.S. NRC.*

Introduction

- Eleven countries and JRC responded to the survey:
  - Belgium, Canada, Czech Republic, Finland, Germany, Japan, Netherlands, Slovakia, Sweden, Switzerland, USA

1. Regulations address possibility of pipe ruptures
   - All countries require consideration of pipe rupture as part of design basis
   - Canada referenced RD-337 for “new” power plants; not clear if regulations apply to existing power plants.

2. Credit for leak before break (LBB) in existing regulations

- Almost all countries directly credit LBB either within regulations or as a method to meet existing regulations.
  - Typically LBB is used to address local pipe rupture effects such as pipe whip or jet impingement
  - Some countries credit LBB for addressing asymmetric blow down loads
  - No country credit LBB for ECCS system design, containment design, equipment qualification, radiation exposure evaluation
- Germany does not credit LBB; requires specific leaks and breaks analyzed for various events.

3. Applicability of LBB to systems with active degradation

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- Almost all countries **do not** allow credit for LBB in systems susceptible to active degradation mechanisms such as stress corrosion cracking (SCC)
- Canada
  - No credit for new construction if SCC is possible
  - May allow use of LBB for managing degradation in existing plants if concrete evidence of LBB is provided
- US
  - SCC exists in some lines that were previously credited for LBB
  - Effectiveness of mitigation used to justify continued credit of LBB in the near-term

4. Piping systems applicable for LBB credit

- Most countries allow LBB credit for the following systems
  - PWR main coolant lines
  - BWR main steam and feedwater lines
- Several countries have also credit LBB in other systems
  - Surge line: Czech Republic, Netherlands, Slovakia, Sweden, US
  - Other branch piping systems: Czech Republic, Germany, Sweden, US
- Slovakia and US, in principal, could allow LBB credit for any high energy piping
- Canada is unique because of CANDU design
  - Allow for primary heat transport tubing
  - Consider as a defense in depth consideration for pressure tubes, feeder tubes, and steam generator tubes

5. Regulatory approval before crediting LBB

- All countries require regulatory approval in order to allow LBB to satisfy applicable regulations (i.e., in order to credit LBB)
- Exact requirements for approval vary slightly from country to country
- Japan
  - Cannot currently approve LBB in carbon steel piping
  - Relevant codes and standards have yet to be endorsed
- Slovakia's requirements
  - Operate three independent leakage detection systems
  - Perform NDT of LBB lines
  - Monitor the chemistry in these lines
  - Perform revisions, functional tests and maintenance of snubbers installed on primary piping and components.

6. Description of LBB evaluation procedure

- All countries typically only allow deterministic evaluations of LBB
- Technical basis often refer to the US Standard Review Plan (SRP) 3.6.3, "Leak-Before-Break Evaluation Procedures" or the German break preclusion (BP) concept
- Almost all evaluations contain similar requirements
  - Evaluation of piping line between anchor points
  - Evaluation of subcritical crack growth by fatigue
  - Evaluation of leak rate margins for calculated through-wall crack (TWC)
  - Evaluation of crack stability margin for calculated TWC
- Some countries identified additional considerations
  - Establish operating procedure to ensure timely response (Canada)
  - Require qualified inspections before and after LBB credited (Sweden)
  - Demonstration of stability after growing TWC in fatigue for an additional reactor operating life (Netherlands)
  - Several countries (e.g., Belgium Finland) identify additional loading sources to consider in the crack stability analysis

7. Comparison of LBB evaluations to SRP 3.6.3

- Some countries identified the German BP concept in RSK 79 as basis of their LBB evaluations: Germany, Netherlands
- Many countries referenced SRP 3.6.3 as the basis of their LBB evaluations
  - Belgium, Canada, Czech Republic, Sweden, Switzerland, Slovakia, US
  - Almost all countries have some slight differences with SRP 3.6.3 or specify additional requirements
- Finland identified their basis as a mixture of BP and SRP 3.6.3 concepts
- Japan indicated that SRP 3.6.3 is not the basis of their evaluation

8. Allowance of alternative LBB evaluations

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- No alternative evaluation procedures allowed
  - Czech Republic, Germany, Netherlands, Slovakia
- Up to now, no alternative procedures have been proposed by the operators or accepted by the regulator
  - Belgium, Switzerland
- Allow alternative approaches as long as adequacy is demonstrated, but have yet to approve such approaches
  - Finland, Japan*, US
- Have considered alternative approaches in LBB evaluations
  - Canada accepts probabilistic analyses and reviews their adequacy on a case-by-case basis
  - Sweden has used probabilistic analyses in concert with deterministic analyses to strengthen LBB case

* Japan answered no to this question, but in question 6 indicated that alternative approaches could be considered

9. Gaps in understanding of LBB

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- Assessing probabilistic evaluations and developing associated evaluation criteria (Switzerland, US)
- Understanding performance of dissimilar metal welds and components (Germany)
- Effect of degradation (e.g., SCC) and residual stress (JRC, US)
- Implementing near-field and far-field earthquake loading (Czech Republic)
- Developing requirements for leak rate systems (Sweden)
- Determining the proper crack morphology for evaluations (Sweden, US)
- Identifying appropriate load combinations for the stability analysis (Sweden)
- Quantitatively accounting for the effects of mitigation measures (US)

10. Current research activities related to LBB

- Canada
  - Evaluating consequences of high-energy piping failures
  - Assessing CANDU-specific probabilistic fracture mechanics (PFM) codes
- Czech Republic
  - Developing a probabilistic LBB computational code
- Finland
  - Evaluating deterministic and probabilistic aspects associated with LBB
- Germany
  - Conducting environmental fatigue research
  - Studying interaction among piping, supports, and structure during dynamic loading
- JRC (through NUGENIA)
  - Studying effect of degradation and residual stresses on LBB
- Switzerland
  - Conducting research on SCC and corrosion fatigue
- US
  - Developing a PFM code to evaluate LBB (xLPR)
  - Evaluating weld residual stress prediction methods
  - Participating in international PARTRIDGE program

11. Technical or regulatory concerns related to LBB

- Many similar responses as to question 9
- Crediting LBB for long-term operations (Netherlands, JRC)
- Crediting LBB in lines with active degradation mechanisms (Switzerland, US)
- Validating PFM codes (Canada, US)
- Effect of weld strength mismatch on crack stability (Finland)
- Effects of rupture on environmental qualification of equipment (Canada)
- Significance of differences among deterministic LBB analyses (Finland)
- Clarifying consequences that LBB is mitigating against (Sweden)
- Understanding failure probabilities associated with deterministic LBB analysis (Sweden)
- Assessing/crediting crack initiation times within LBB evaluations (US)

12. Interest in sharing information related to LBB

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- Highly interested
  - Canada, Czech Republic, Germany, Japan, Sweden, Switzerland, US
- Medium interest
  - Netherlands, JRC
- Low interest
  - Belgium
- No answer
  - Finland, Slovakia
- Some topics indicated for information sharing
  - More insight into approaches used elsewhere (Netherlands)
  - Experimental results related to performance of DMW components, fatigue, and dynamic loading effects (Germany)
  - Probabilistic LBB methods (Czech Republic, US)
  - Regulatory perspectives on LBB (Canada)
  - Review procedures for new LBB concepts (Switzerland, US)
  - Failure of bolted connections in manhole covers (Finland)

13. Interest in conducting collaborative research

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- Highly interested: Canada, Czech Republic, Japan, JRC, US
- Medium interest: Germany, Sweden
- Low interest: Belgium
- No interest: Netherlands,, Switzerland
- No answer: Finland, Slovakia
- Some topics indicated for collaborative research
  - Probabilistic LBB methods (Czech Republic, Sweden, US)
  - PWSCC in dissimilar metal welds (Belgium, US)
  - Regulatory perspectives on LBB (Canada)
  - Influence of near-field earthquake loading (Czech Republic)
  - Improvement of leak detection systems (Japan)
  - Consequences of asymmetric blowdown loads and how these depend on pipe break opening times and opening areas (Sweden)
  - Applying LBB using risk-informed methods (Sweden)
  - Pipe fracture by combined torsion plus bending (Japan)
  - Experimental results related to performance of DMW components, fatigue, and dynamic loading effects (Germany)

14. Interest in SOA report on LBB regulations, knowledge, and current activities

- Highly interested
  - Canada, Czech Republic, Germany, JRC
- Medium interest
  - Japan, Sweden, Switzerland, US
- Low interest
  - Belgium
- No interest
  - Netherlands
- No answer/unclear
  - Finland, Slovakia

15. Interest in collaboration in US xLPR program

- Interested
  - Canada, Czech Republic, Germany, Japan, Sweden
  - Japan is most interested in residual stresses in dissimilar metal welds and subcritical crack growth assessment
  - Sweden is already involved through PATRIDGE program
- Possible interest
  - Belgium, Finland
  - Belgium is most interested in following activities related to PWSCC in dissimilar metal welds
  - Finland is more concerned with thermal aging of dissimilar metal welds but could participate based on available funding
- No interest
  - Netherlands
- No answer/unclear
  - JRC, Switzerland, Slovakia

NUGENIA
NUclear GENeration II & III Association

www.nugenia.org

NUGENIA

International association dedicated to **safe, reliable and competitive nuclear** energy technology

NUGENIA is mandated by SNETP to coordinate nuclear Generation II & III R&D

SNETP
Sustainable Nuclear Energy Technology Platform

www.snetp.eu

What is NUGENIA?

- NUGENIA is an **international non-profit association** founded under Belgian legislation in November 2011, and launched in March 2012

- NUGENIA is dedicated to the research and development of nuclear fission technologies, with a focus on Generation II and III nuclear plants

Why NUGENIA?

- **Mission**
  - To be the integrated framework between industry, research and safety organisations for safe, reliable and competitive Gen II & III fission
- **Services**
  - To run an open innovation marketplace
  - To promote the emergence of joint research
  - To facilitate the implementation and dissemination of R&D results
- **Products**
  - R&D roadmap and coordinated project portfolio
  - Advanced scientific and technical base for Gen II & III technology
  - Support to harmonisation at European level, in particular for safety requirements

Technical scope: Generation II & III nuclear plants

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Nuclear Power Plants in Operation in Europe, July 2012

As of July 2012 there is a total of **185 nuclear power plant units** with an installed electric net capacity of 162 GWe in operation in Europe (five thereof in the Asian part of the Russian Federation) and **16 units** with an electric net capacity 14 GWe were **under construction in five countries.**

Country	in operation		under construction	
	number	net capacity MWe	number	net capacity MWe
Belgium	7	5,927	-	-
Bulgaria	2	1,906	-	-
Czech Repuplic	6	3,766	-	-
Finland	4	2,736	1	1,600
France	58	63,130	1	1,600
Germany	9	12,068	-	-
Hungary	4	1,889	-	-
Netherlands	1	482	-	-
Romania	2	1,300	-	-
Russian Federation	33	23,643	10	8,203
Slovakian Republic	4	1,816	2	782
Slovenia	1	668	-	-
Spain	8	7,567	-	-
Sweden	10	9,325	-	-
Switzerland	5	3,263	-	-
Ukraine	15	13,107	2	1,900
United Kingdom	16	9,246	-	-
total	**185**	**161,859**	**16**	**145,085**

Source: www.euronuclear.org

Preceeding Networks

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Merges Activities of 4 Networks

SARnet

Severe Accidents

SNETP
Sustainable Nuclear Energy Technology Platform

TWG Gen II & III

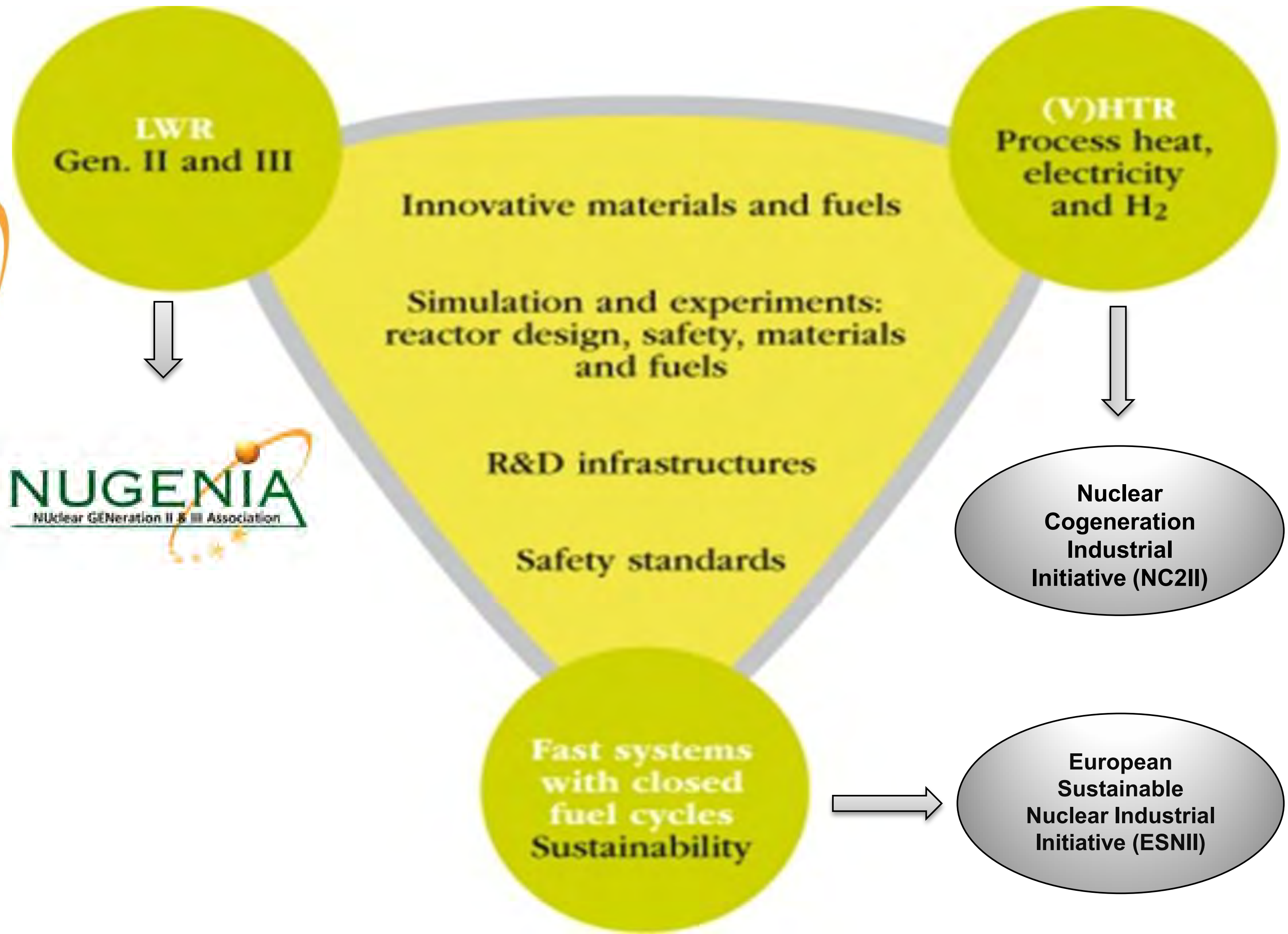
NULIFE

PLIM

ENIQ
EUROPEAN NETWORK FOR INSPECTION AND QUALIFICATION

ISI

NUGENIA in the SNETP frame

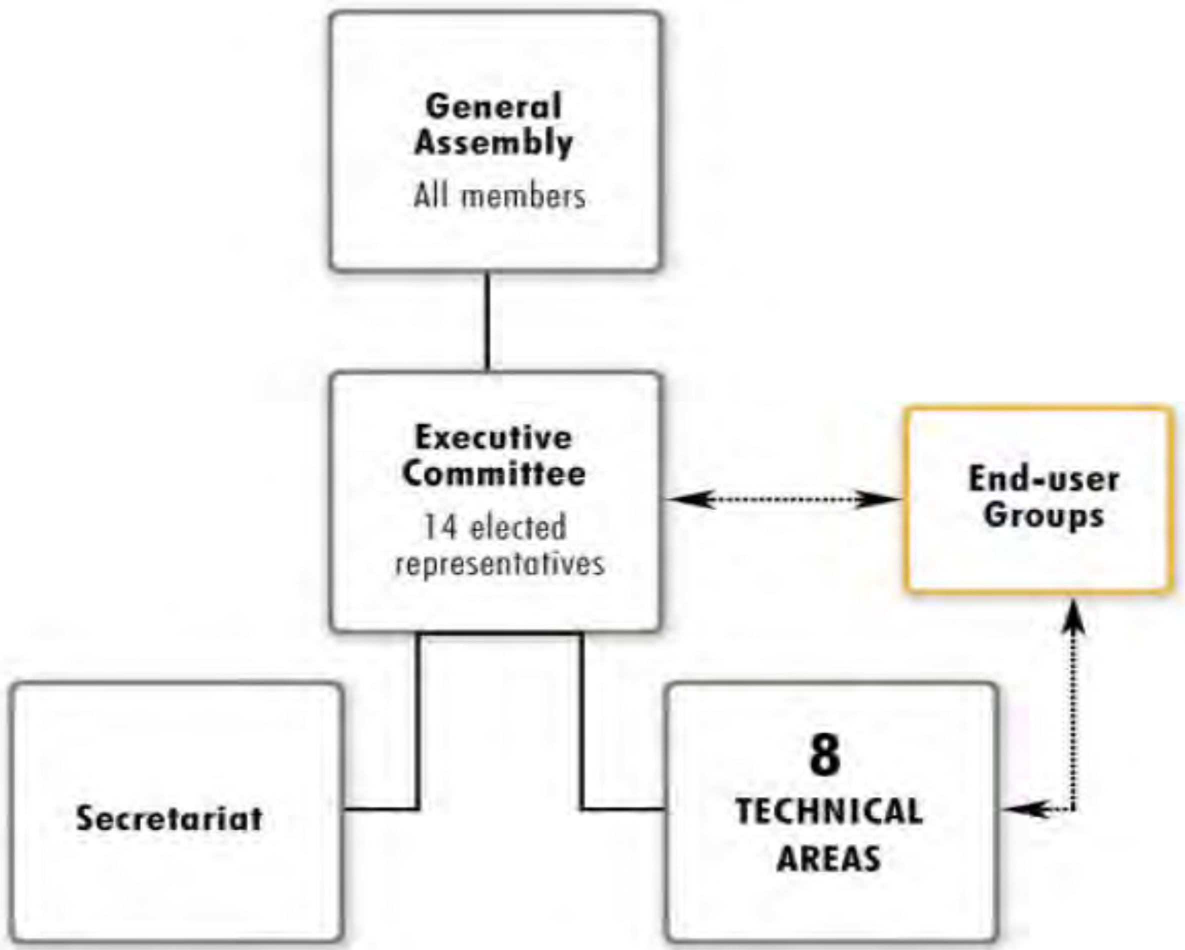


Who is NUGENIA?

- **Members: major nuclear stakeholders**
  - More than 60 members from 20 countries, from industry, utilities, research institutions and technical safety organisations

- **Honorary Members**

Governance structure



Technical scope: 8 areas

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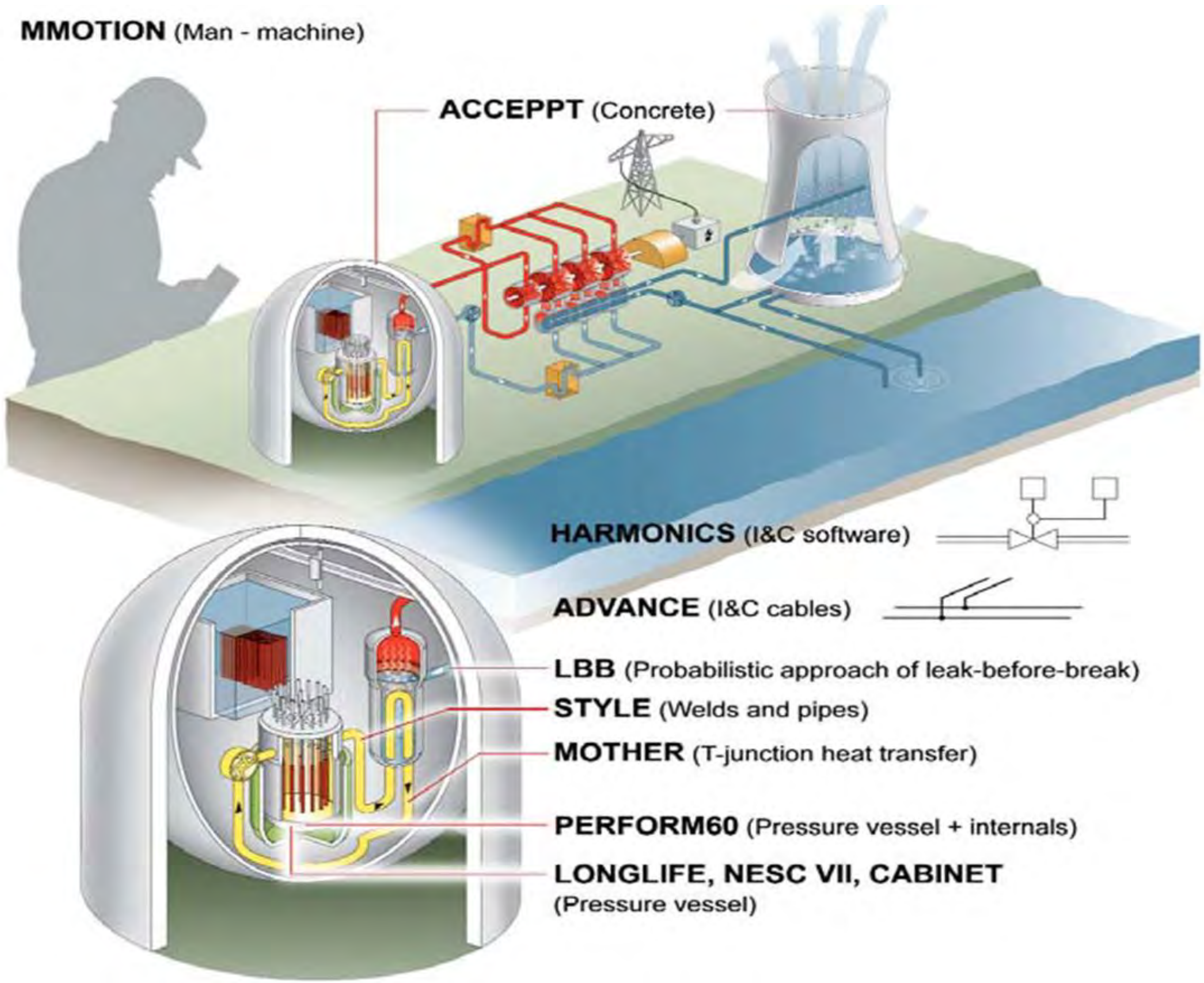
1. Plant safety and risk
2. Severe accidents
3. Core and reactor operation
4. System and component integrity
5. Fuel, waste and decommissioning
6. Innovative LWR design
7. Harmonisation
8. In-service inspection (ENIQ)

NUGENIA project portfolio

- Based on initial contributions from NULIFE, SARNET, ENIQ and SNETP Gen II/III working group



NUGENIA project portfolio

NUGENIA
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2006... | **2009** | **2010** | **2011** | **Year**

NESC VII - Application of WPS in RPV assessment including biaxial loading P

SCC - Stress corrosion cracking N

TF - Thermal fatigue N

I&C - Instrumentation & Control N

DMW - Dissimilar metal welds N

VERLIFE - Unified procedure for WWER components N

LBB - Probabilistic approach of leak-before-break N

MMOTION - Man-machine-organisation through innovative orientations for nuclear EC

PERFORM60 - Prediction of the effects of radiation for RPV and in-core materials using MSM-60 years foreseen plant life time EC

LONGLIFE - Treatment of long term irradiation embrittlement effects in RPV safety assessment EC

STYLE - Structural integrity for lifetime management - non-RPV components EC

CABINET - Constraint and biaxial loading effects and their interaction considering thermal transients P

ADVANCE - Ageing diagnostics and prognostics of low-voltage I&C cables EC

HARMONICS - Harmonised assessment of reliability of modern nuclear I&C software EC

ACCEPPT - Ageing of concrete and civil structures in nuclear power plants P

MOTHER - Modelling of T-junction heat transfer P

P Funded by the partners
N NULIFE pilot projects
EC Funded in part by EC

www.nugenia.org

Thank you for your attention

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NUGENIA is mandated by SNETP to coordinate nuclear Generation II&III R&D

www.snetp.eu

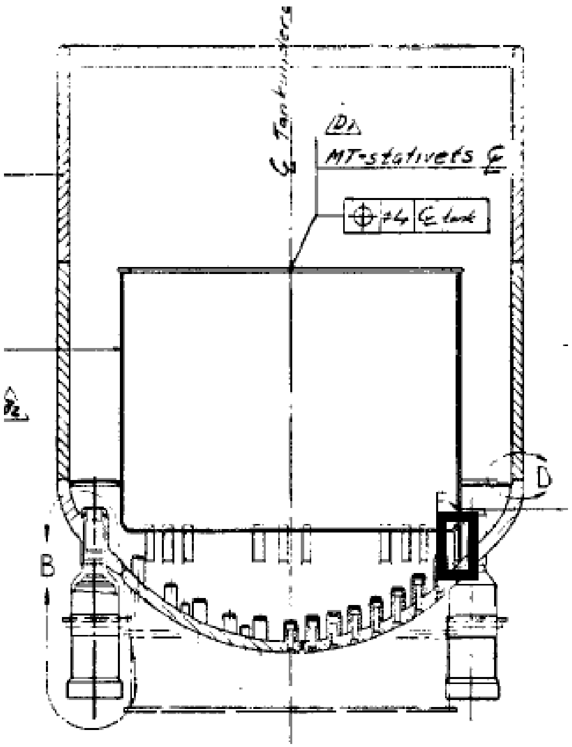
Detected cracks in core shroud welds in a Swedish BWR

Björn Brickstad
Swedish Radiation Safety Authority
Department for Nuclear Power Plant Safety,
Structural Integrity and Event Analysis

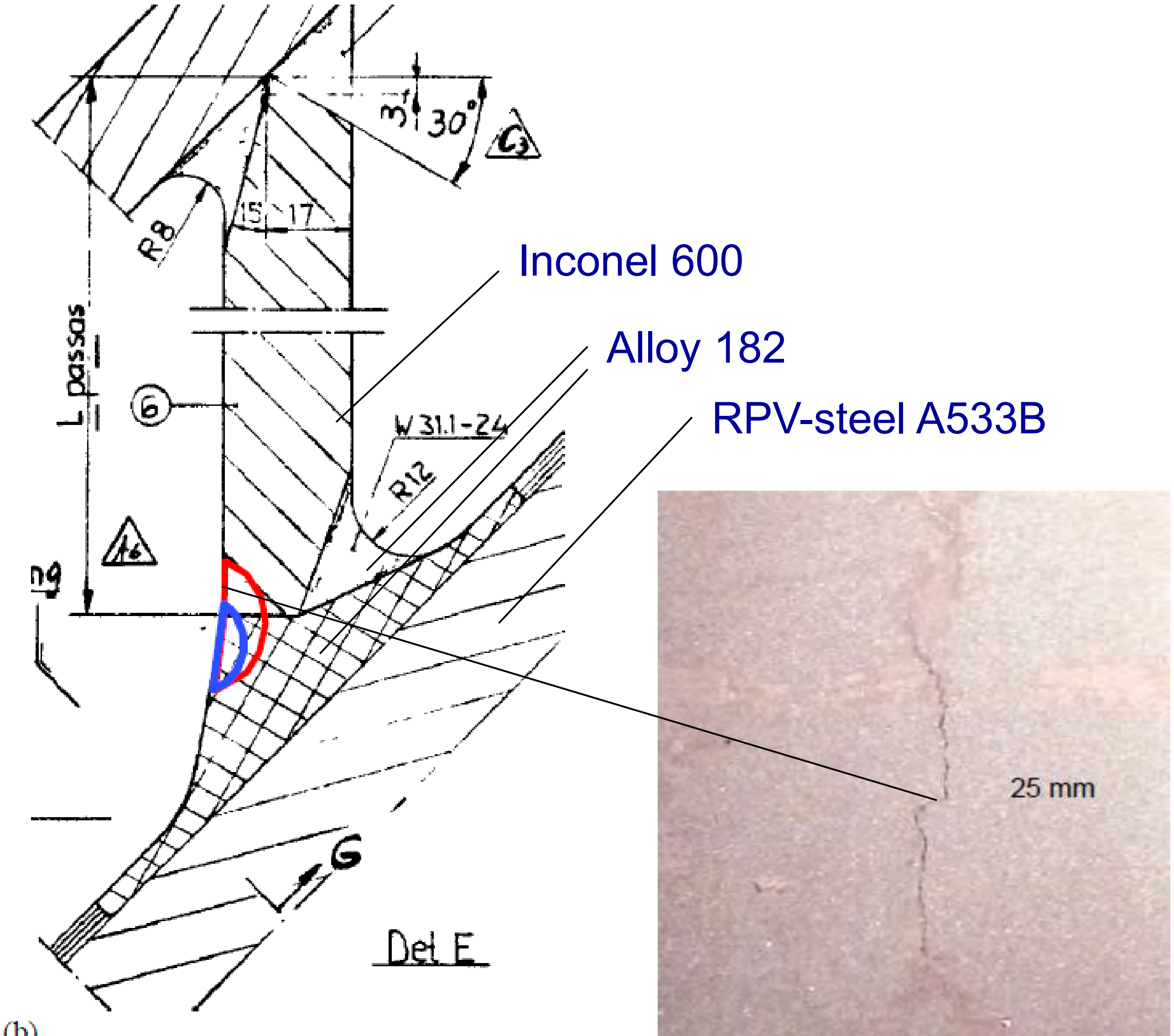
IAGE meeting, April 2013

Cracks were detected in the core shroud welds in a Swedish BWR

- Crack length measurements were performed 2011 by VT.
- Crack depth measurements were performed 2012 by UT.

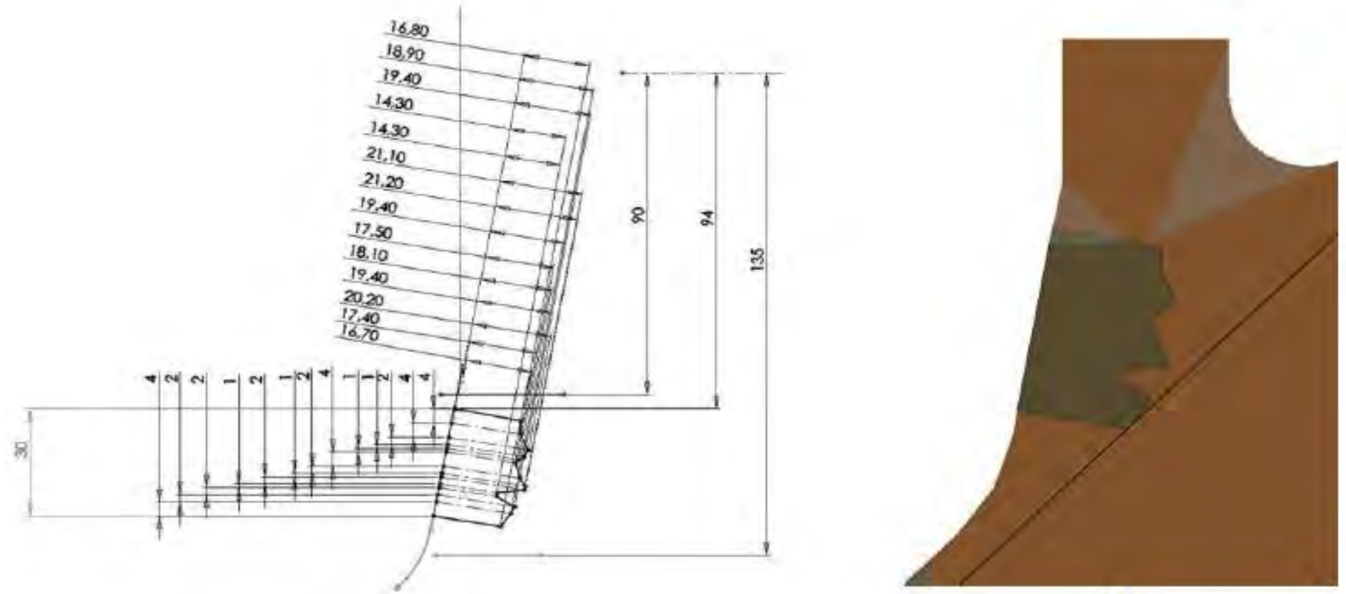


(a)



(b)

Detailed UT-measurements of the crack depth

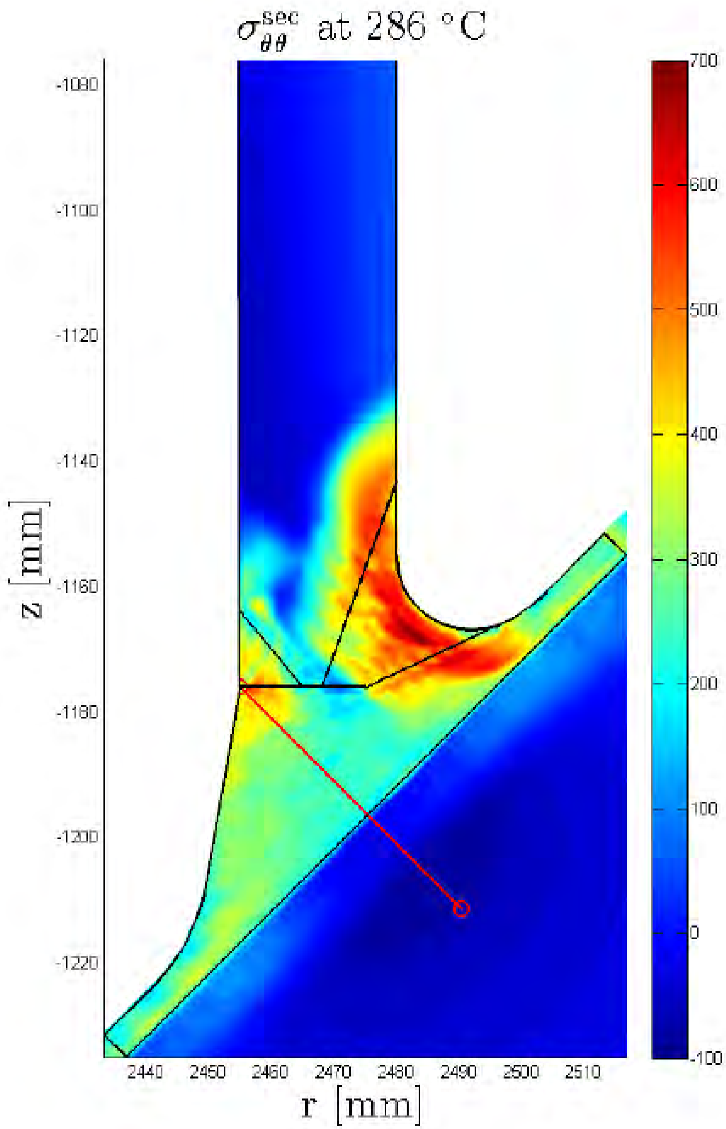


(a) (b)

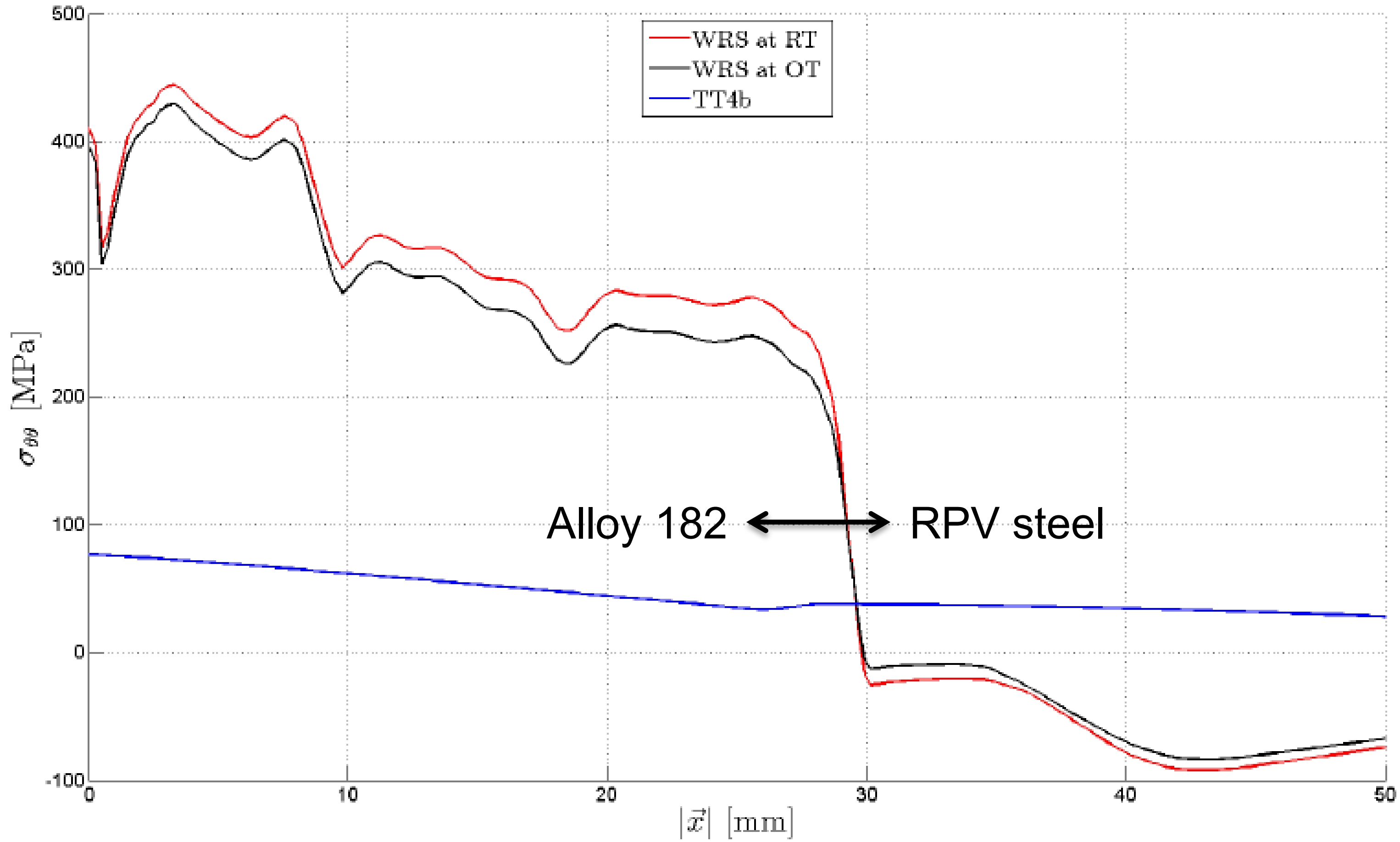
The cracks were judged to be caused by stress corrosion

Maximum crack length 25 mm
Maximum crack depth 21 mm

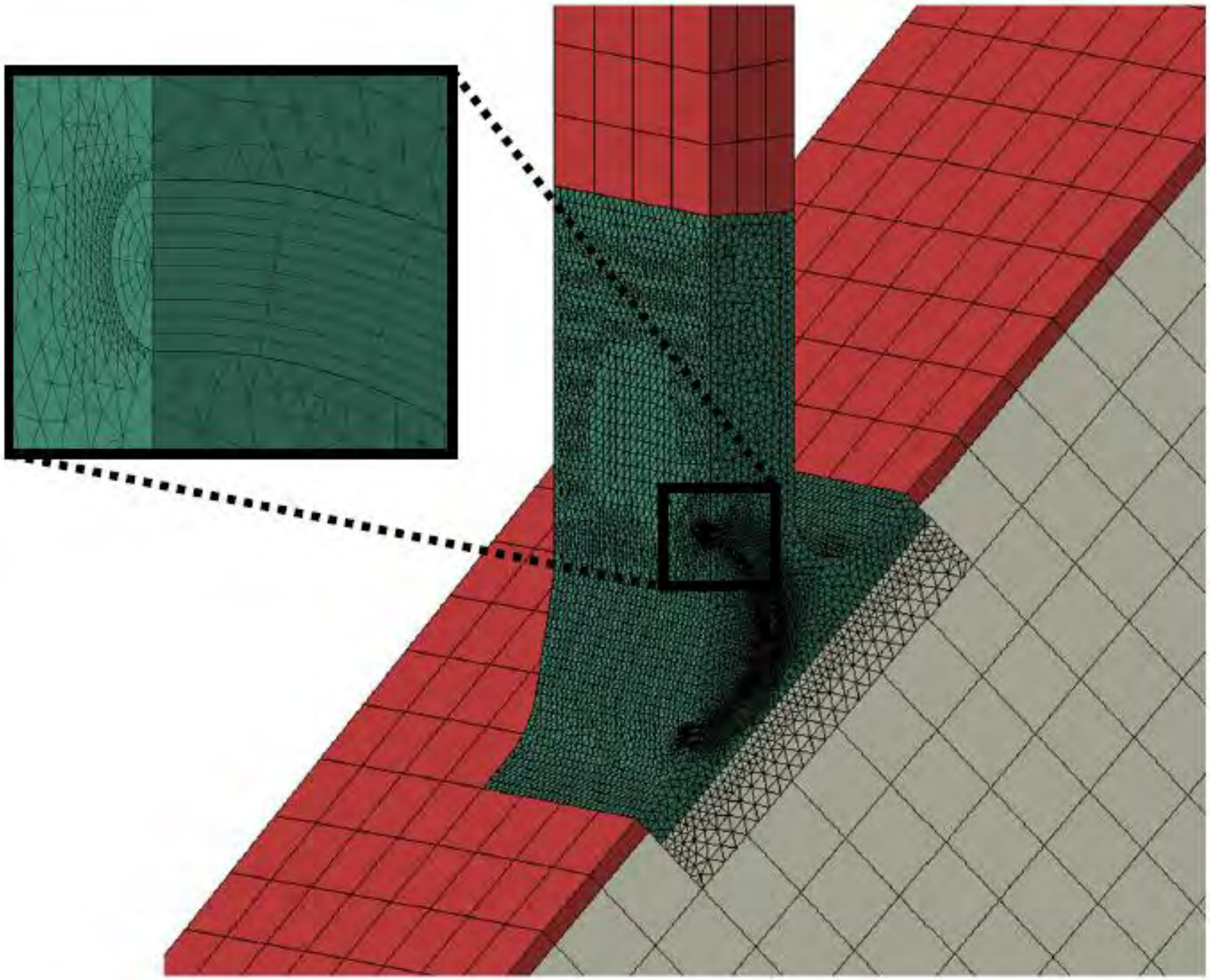
Hoop weld residual stress at operating temperature



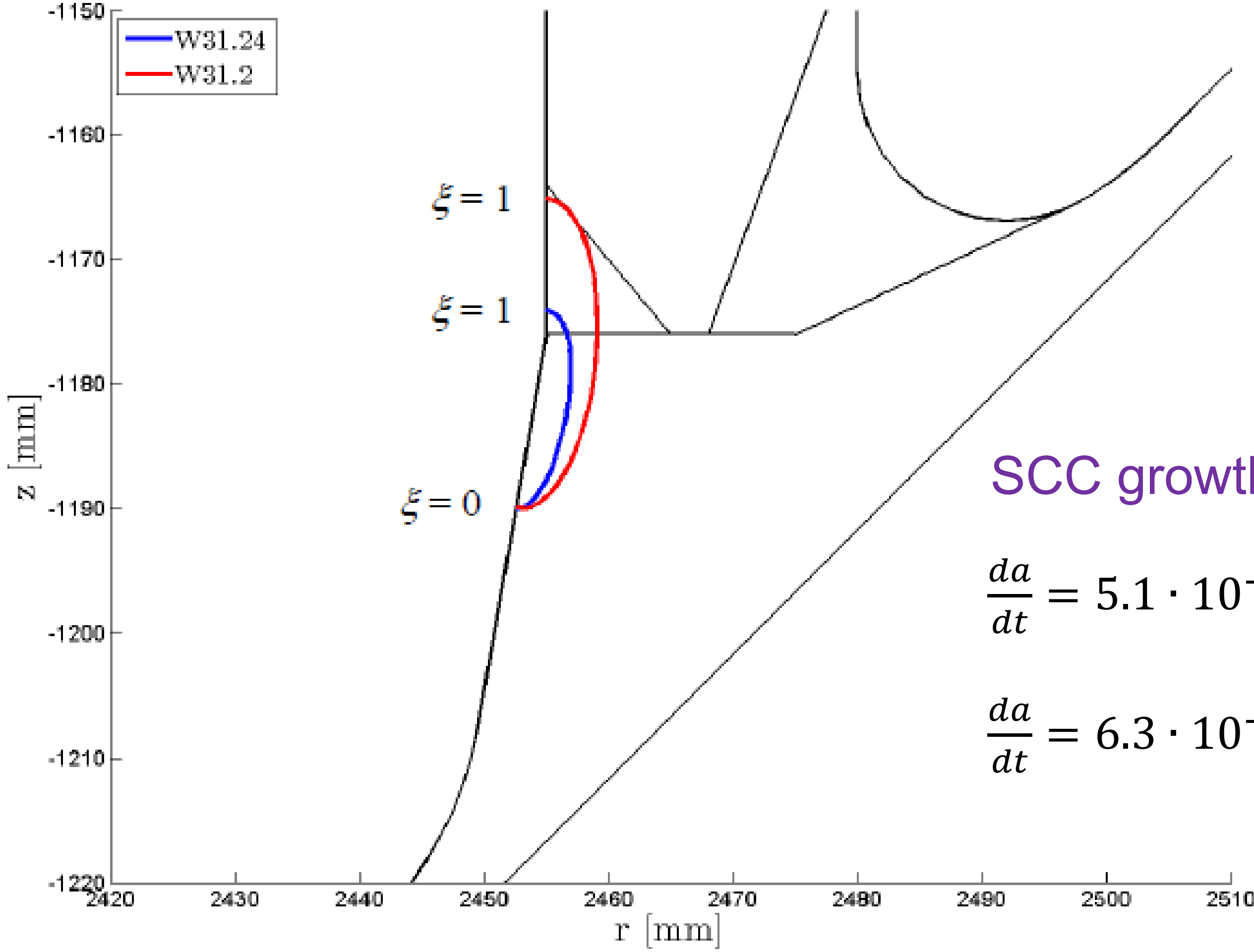
Hoop stress due to WRS and a thermal transient



Detailed 3D-FEM was performed for the crack growth analysis



Postulated initial crack sizes for the growth analyses

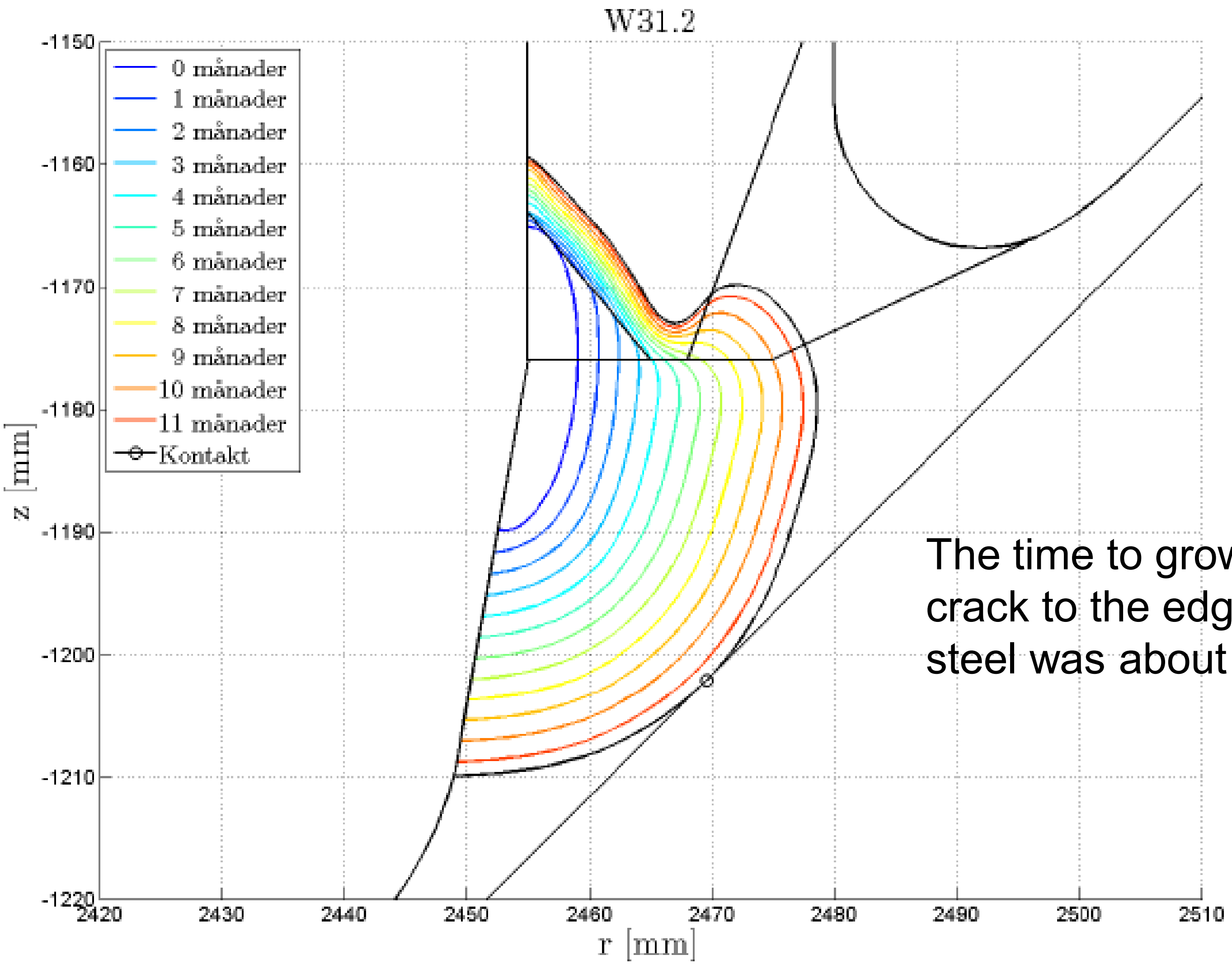


SCC growth in Alloy 182

$$\frac{da}{dt} = 5.1 \cdot 10^{-14} \cdot K^{4.8} \text{ , mm/s}$$

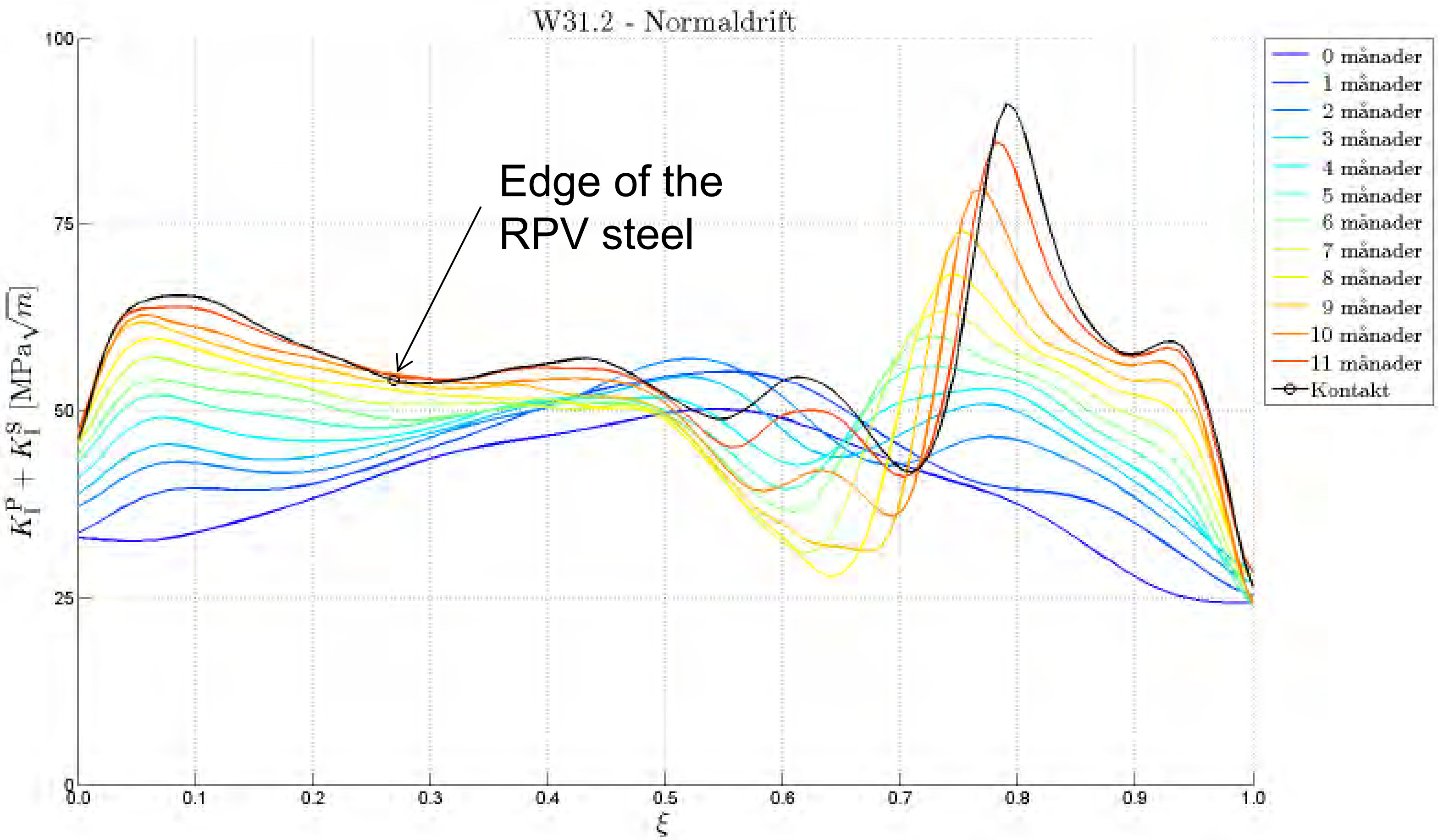
$$\frac{da}{dt} = 6.3 \cdot 10^{-7} \text{ mm/s for } K \geq 30 \text{ MPa}\sqrt{\text{m}}$$

Crack growth as function of time

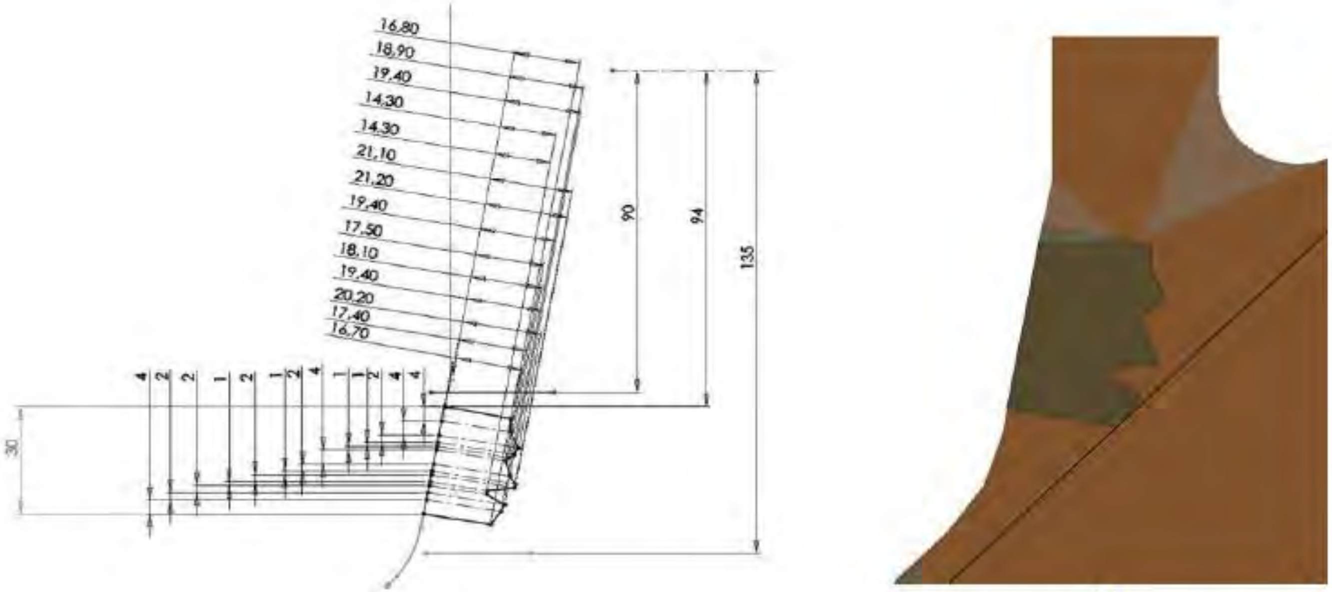


The time to grow the initial crack to the edge of the RPV steel was about 12 months

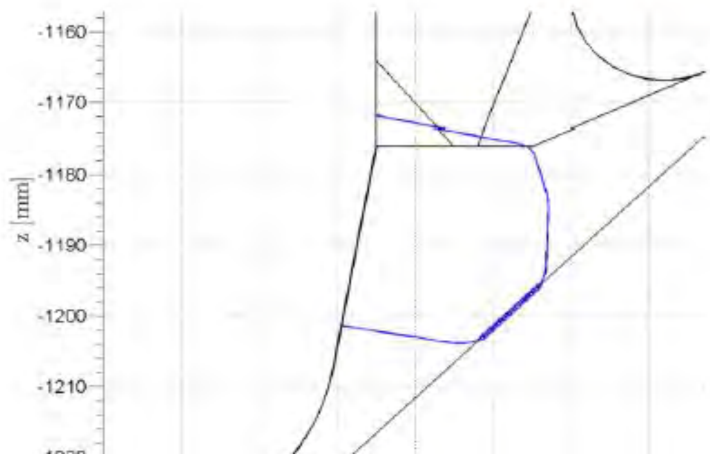
Total stress intensity factor along the crack front at normal operation as function of time



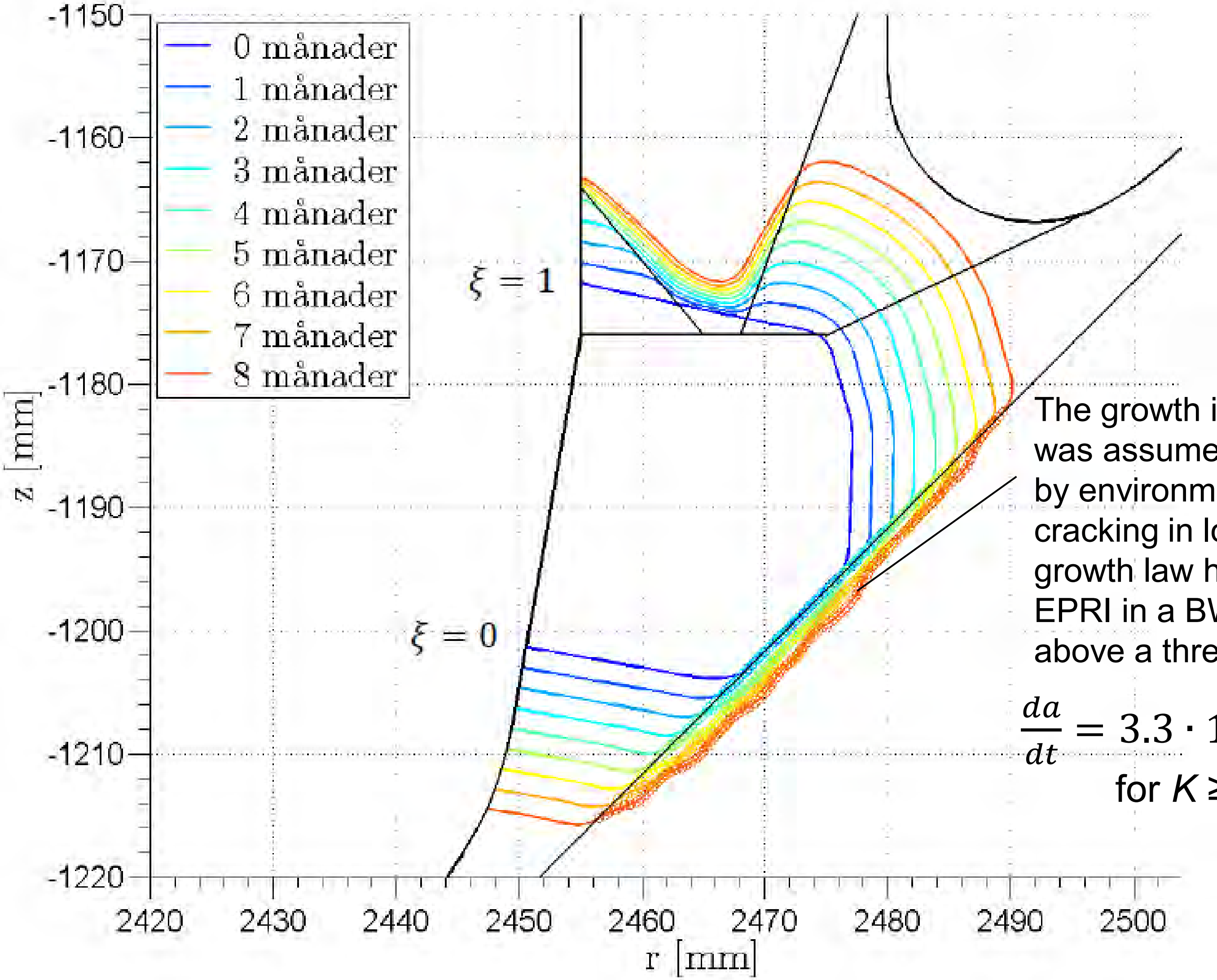
Detailed UT-measurements of the crack depth



(a) (b)



Estimated crack growth from the maximum detected crack size as function of time

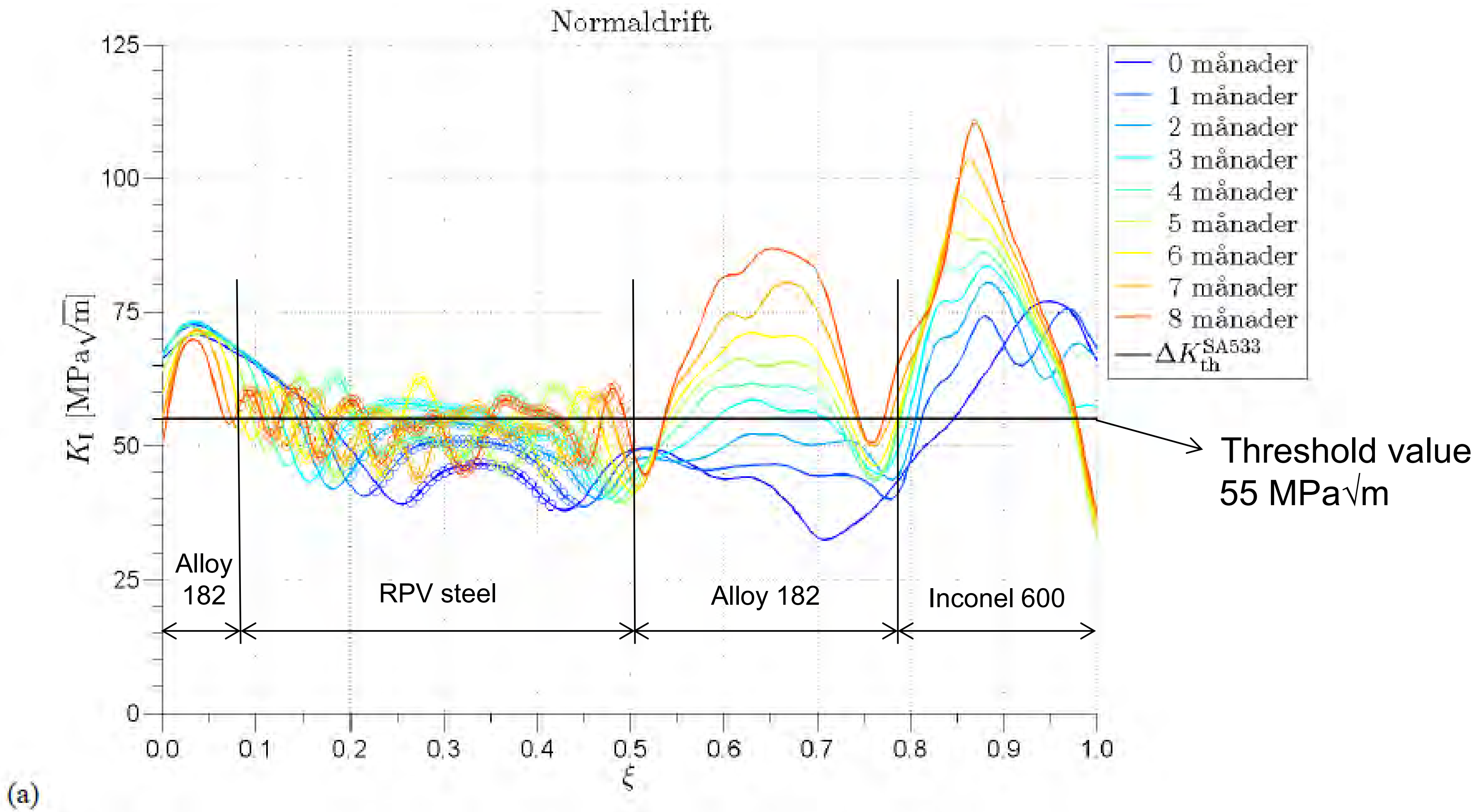


The growth in the RPV steel was assumed to be governed by environmental assisted cracking in low-alloy steel. A crack growth law has been developed by EPRI in a BWRVIP-program, valid above a threshold of 55 MPa√m

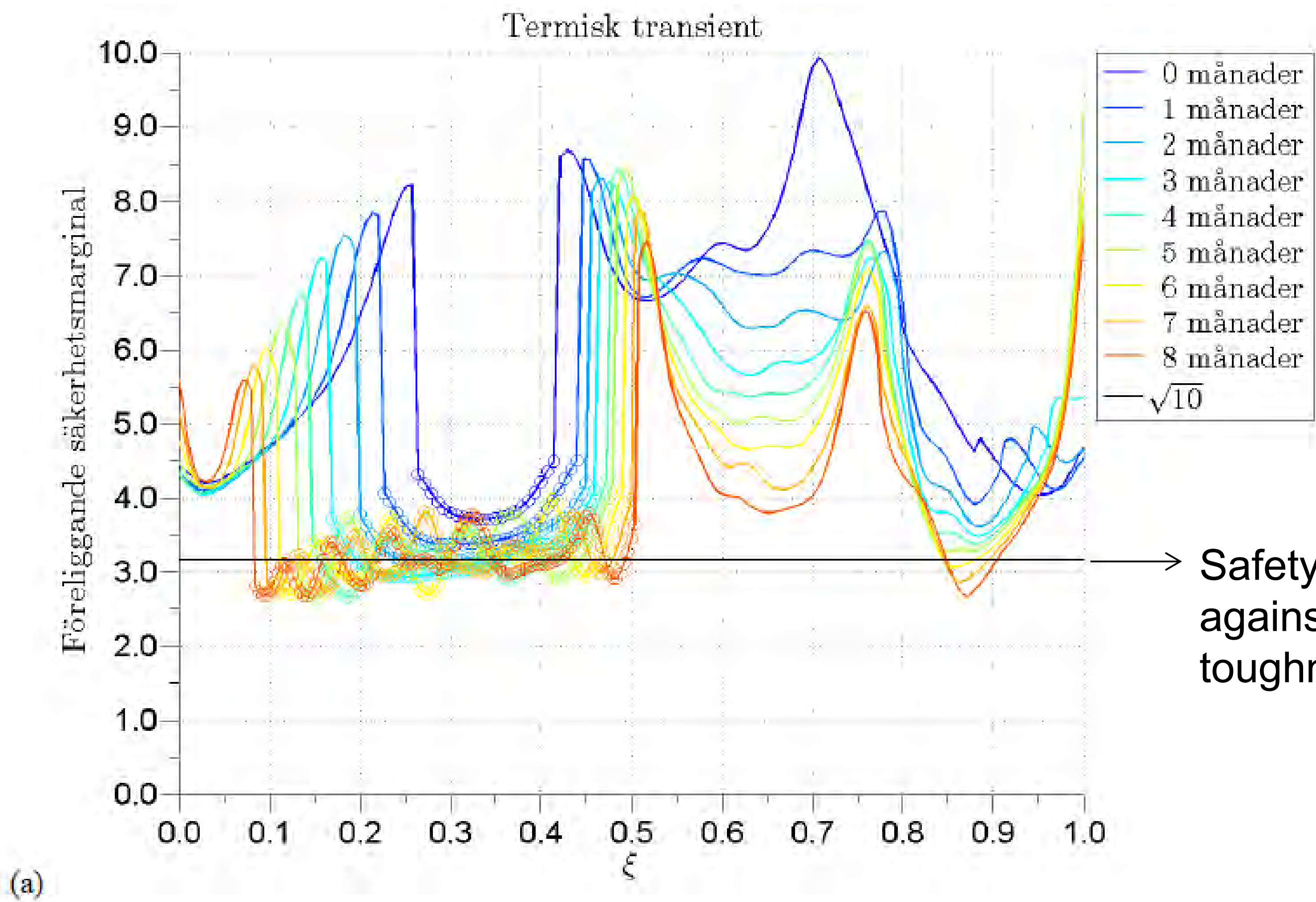
$$\frac{da}{dt} = 3.3 \cdot 10^{-14} \cdot K^{4.0} \text{ , mm/s}$$

for $K \geq 55$ MPa√m

Total stress intensity factor along the crack front at normal operation as function of time

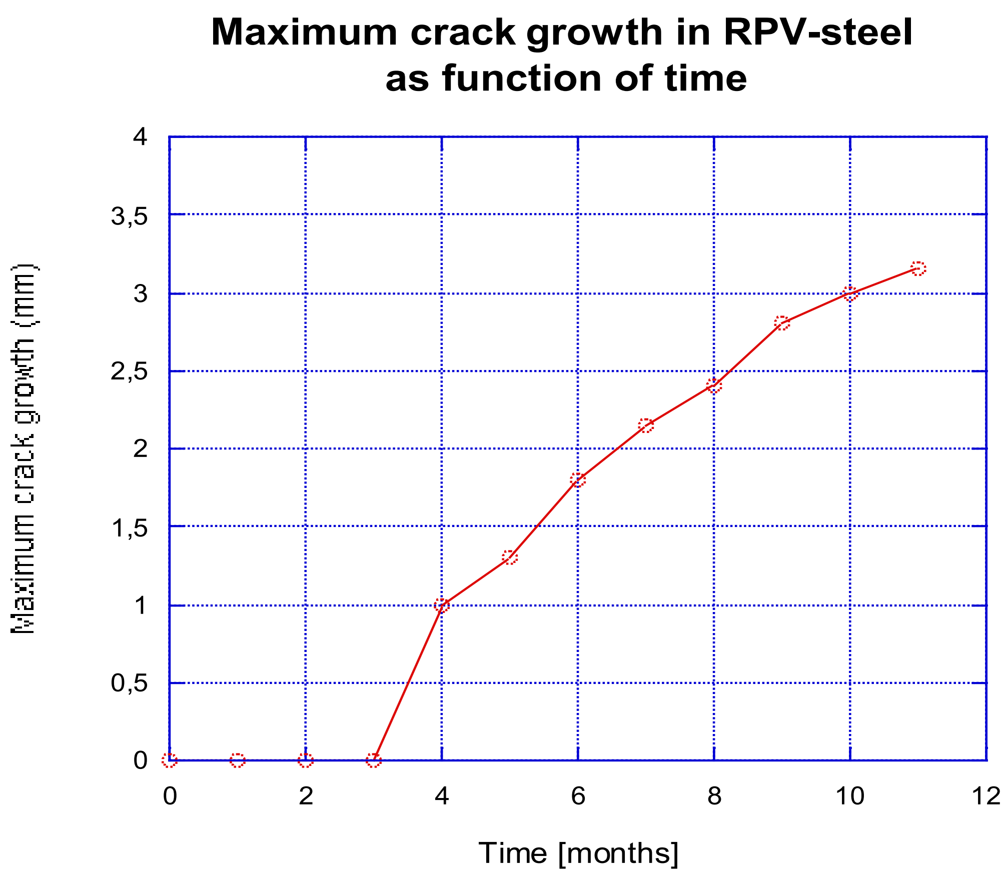


Safety factor against RPV fracture toughness along the crack front as function of time, during a thermal transient

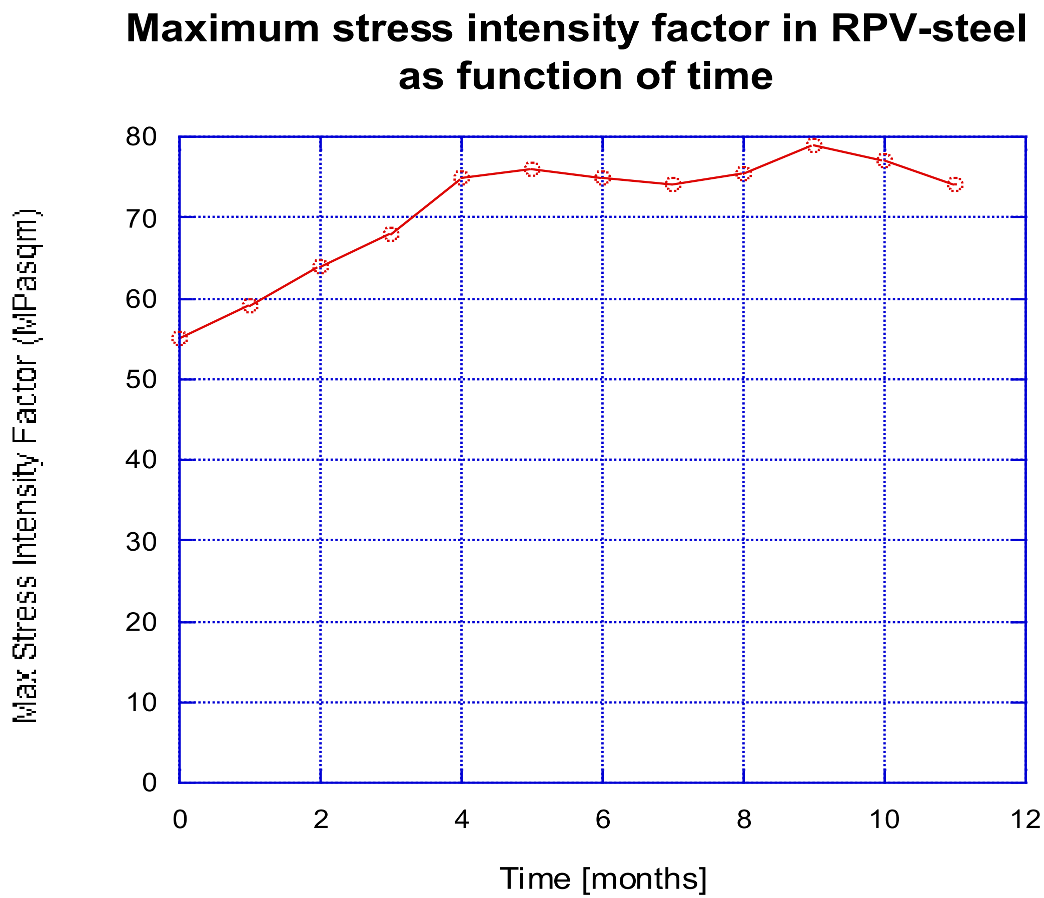


Safety factor √10 against the fracture toughness

Maximum crack growth in the RPV-steel



Maximum stress intensity factor in the RPV-steel



Regulatory views and decisions

- During the analysed time interval, the risk should be small for the detected cracks to grow significantly into the RPV steel. A condition for this judgement is to have a continuing good water chemistry.
- Even if a limited crack growth into the RPV steel cannot be excluded, such cracks are still judged to be acceptable.
- The NPP owner must remove the cracks during the next outage (July 2013).

Full structural weld overlay on BWR Feedwater nozzle in Switzerland

K. Germerdonk

18th Meeting of the IAGE Sub-Group Metals, 09-10 April 2013

Round table discussion

Crack in BWR feedwater nozzle

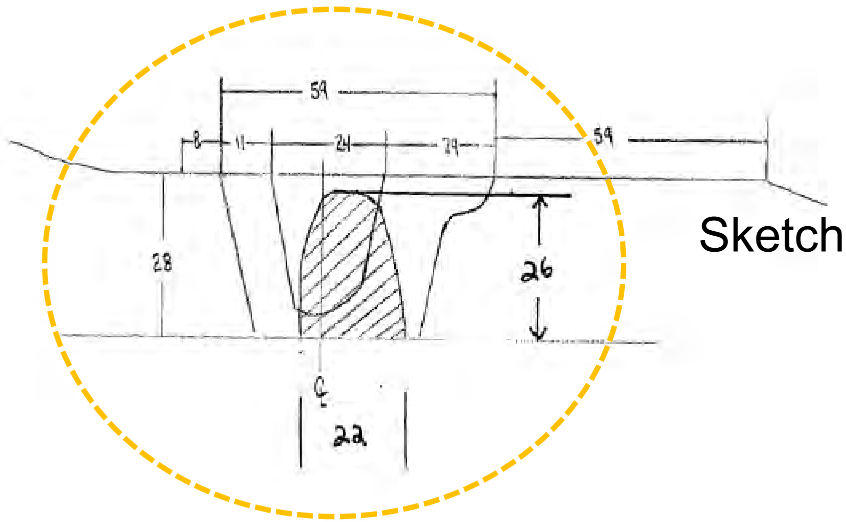
NPP Leibstadt (KKL)

- BWR-6 (GE), in operation since 1984
- HWC-OLNC-water chemistry since 2008

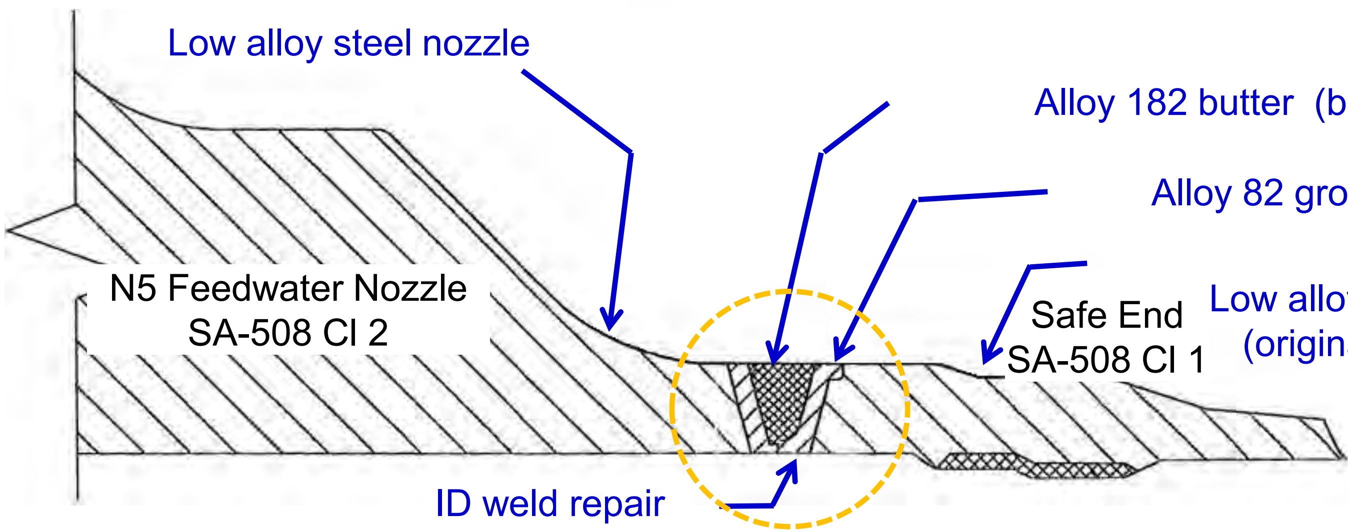
- UT on 6 FW-nozzles during outage 2012
- Qualified UT procedure based on PDI
- Last UT-inspection in 2004
- No „relevant“ indication reported in 2004

Axial flaw on N5 at 150° found in 2012,
92 % through wall

Crack in BWR feedwater nozzle N5



Sketch of axial flaw detected



3

UT-inspection: 2004 vs. 2012 results:

2004 evaluation of UT:

- An indication was detected
- Characterized as embedded without contact to the inner surface
- Embedded indications are not reported

➡ Not characterized as relevant indication

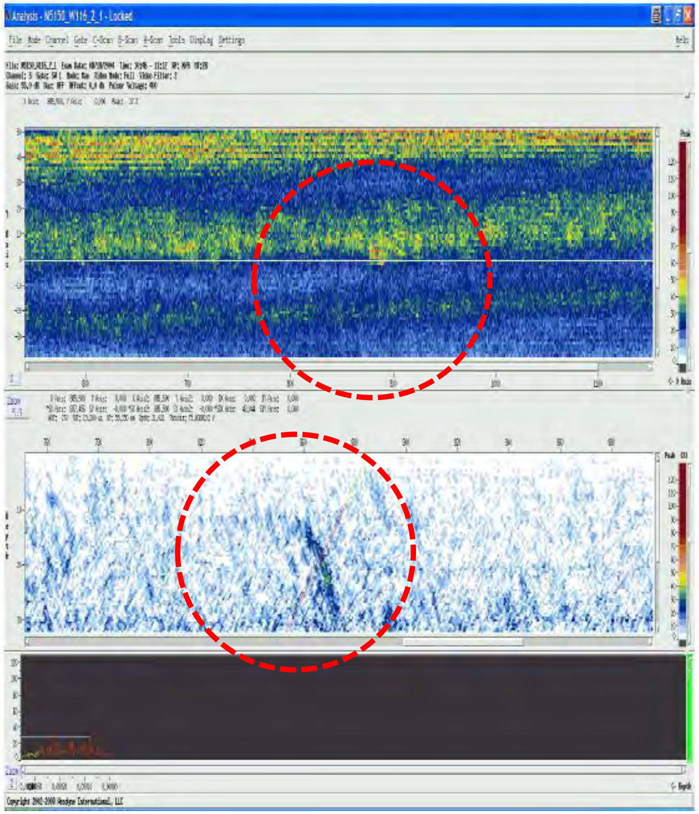
2012 re-evaluation of 2004 UT data:

- Crack depth 15 mm in 2004
- Crack growth 9 mm in years

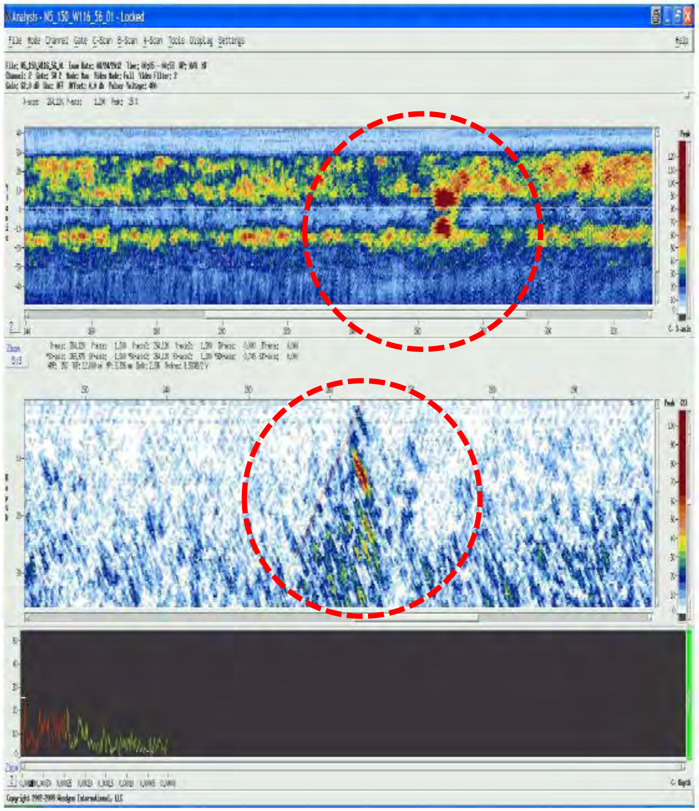
➡ **about 1.1 mm/y IGSCC in Alloy 182**

UT-inspection: 2004 vs. 2012

45 Degree 2.0 MHz CCW 2004



45 Degree 2.0 MHz CCW 2012



Decision for FSWOL-Repair of Crack

- **Full structural weld overlay (FSWOL)**
- Code Case ASME N-740-2
- Local repair 2 layers (52M)
- 7 layers (52M) FSWOL
- FSWOL approved by ENSI as temporary repair measure (until 2015)

Main Issues of concern to ENSI and KKL related to FSWOL-repair

First FSWOL according to ASME N-740-2 on RPV-nozzle in Switzerland!

- Temperbead welding
  - e.g. HAZ properties
- Peening and leak seal welding
  - Potential that leak could not be sealed
- 52M weldability problems
  - e.g. Hot cracking
  - Oxide inclusions,
  - etc.
- NDE demonstration

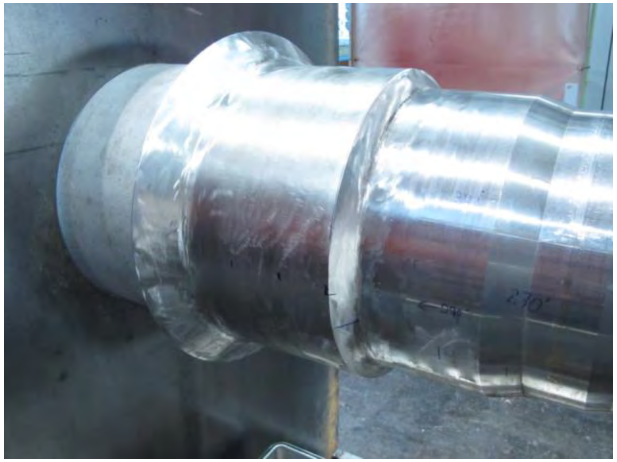
KKL initiated:

- Manufacturing of several specific mock-ups
- Detailed residual stress and crack growth analysis

Full size mock-up

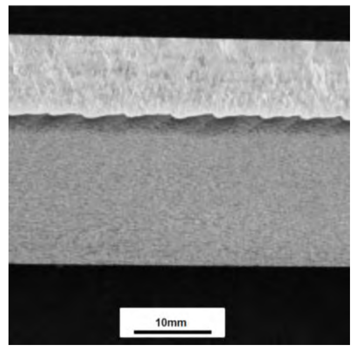
Full size mock-up for welding, acceptance NDE, and PSI/ISI demonstration

- Machined CS round bar with same nominal ID and OD dimensions as N-5 DMW including all tapers and transitions
- Mock-up was water backed

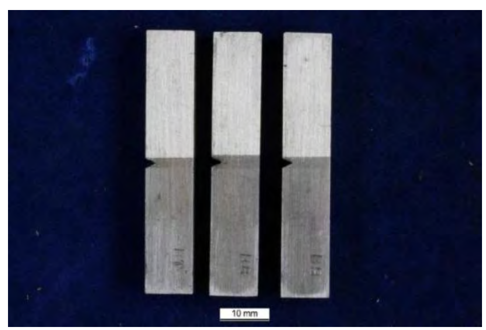


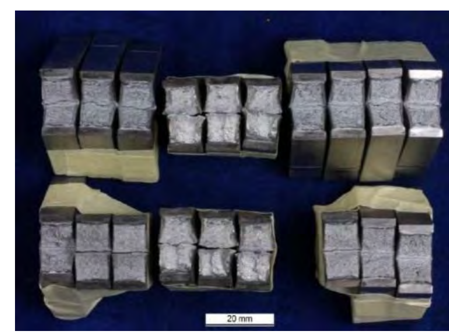
Mockup for metallography and mech. testing

– Purpose was to demonstrate temperbead welding (4 layers of 52M deposited by temperbead process
– Sections cut out and removed for metallography and mechanical testing



metallography

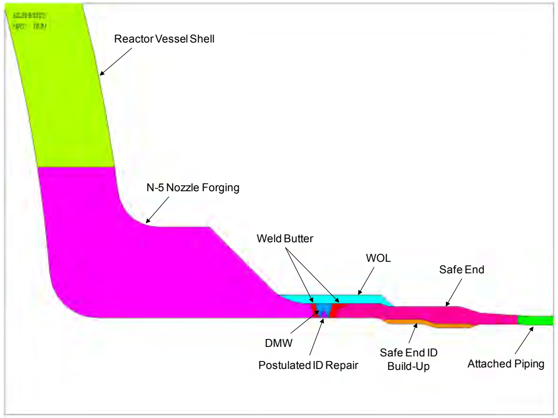




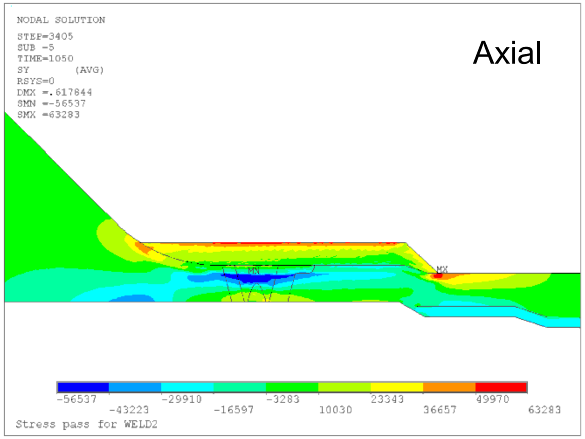
mechanical testing

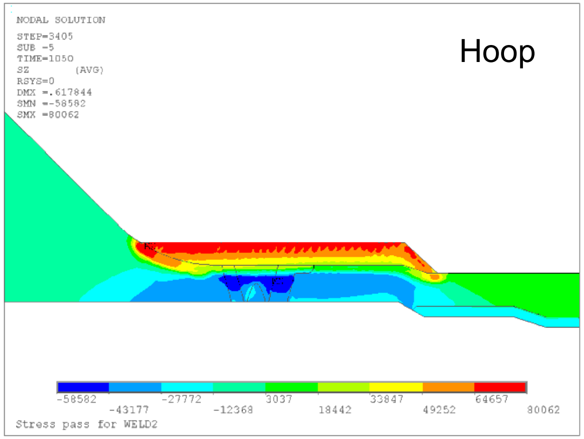
Stress analysis on FSWOL

Effect of FSWOL on stress distribution was analyzed



2-D axis-symetric FE model





Compressive stress on inner portion of SCC susceptible weld material at operating condition

Summary

- FSWOL repair according to ASME codecase ASME N-740-2 was performed successfully on a feedwater nozzle, comprehensive validation work on several mock-ups confirmed required welding quality;
- A detailed root cause analysis related to the N5-crack finding and review of existing ISI-program on nickel base alloys at KKL is ongoing;
- ENSI approved FSWOL as a temporary measure for three years (until 2015).

The 18th WGIAGE Metal sub-group meeting
OECD Convention Center, Paris, France, 9-10 April 2013

Seawater Intrusion Event at Hamaoka Unit 5

April, 2013

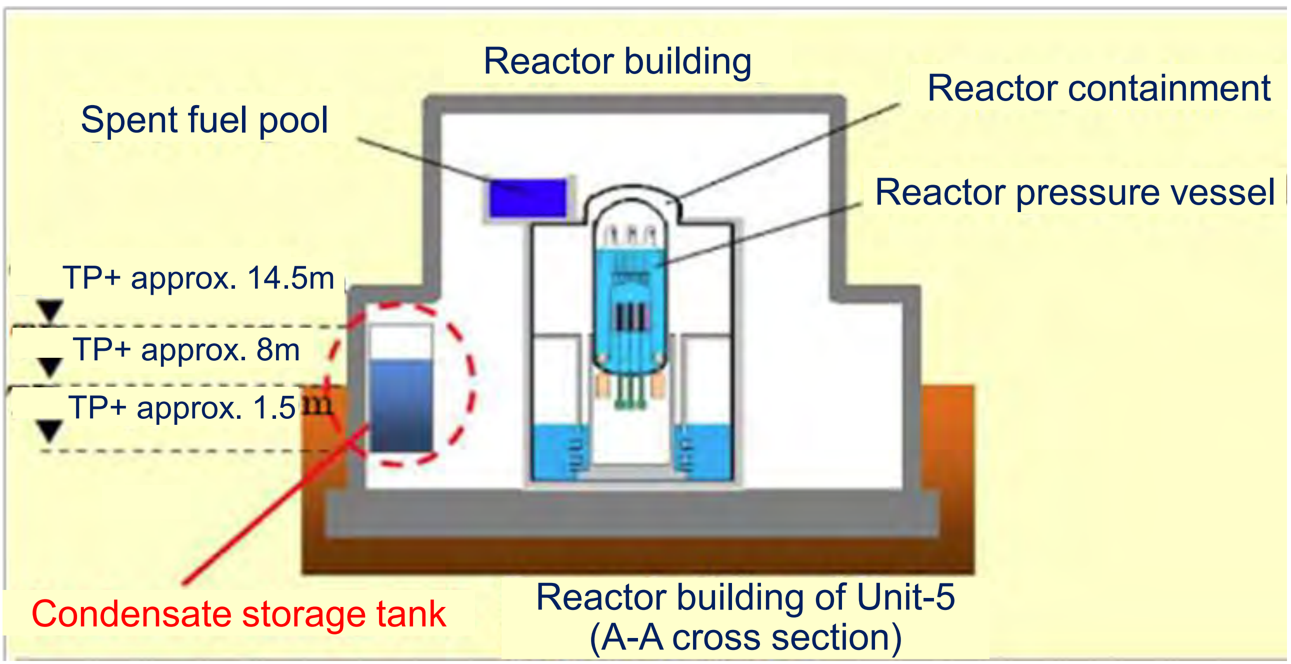
Masakuni Koyama
Japan Nuclear Energy Safety Organization (JNES)

Plant Description (Hamaoka Unit 5)

- Plant : Hamaoka Nuclear Power Station Unit 5
- Licensee : Chubu Electric Power Co., Inc.
- Reactor Type : ABWR
- Gross Electric Capacity : 1,380 MWe
- Date of Commercial Operation : January 2005

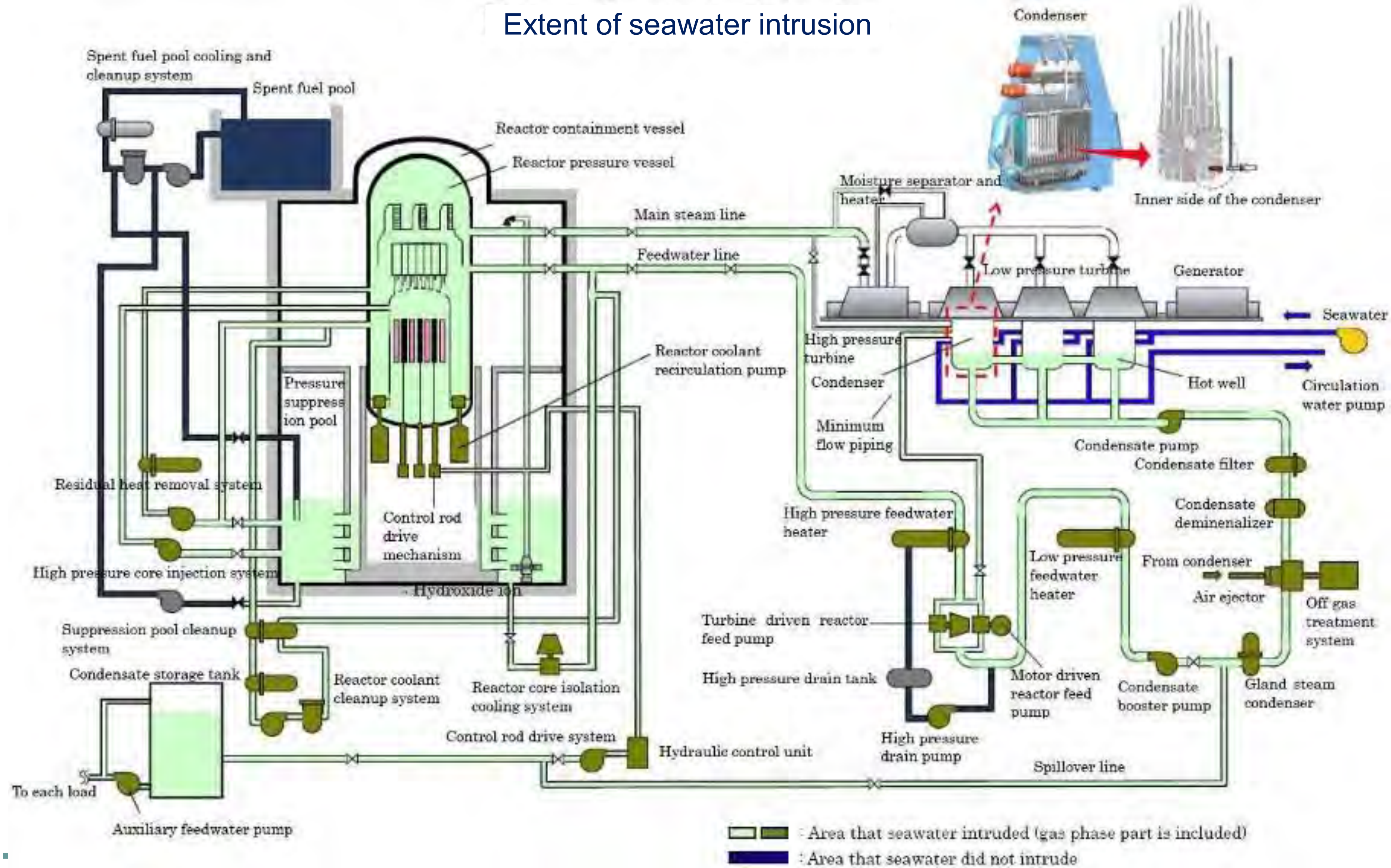
Seawater Intrusion Event at Hamaoka Unit 5

- Chubu Electric Power Co. reported identification of corrosion holes in the liner of the condensate storage tank (CST) on Hamaoka Unit 5 (ABWR) on 30 Mar. 2012
  - ✓ A total of 40 corrosion holes identified on the wall and bottom of CST
  - ✓ Causes of corrosion holes are estimated due to seawater intrusion into the CST under the condition that crud accumulated in the tank and the crevice corrosion.



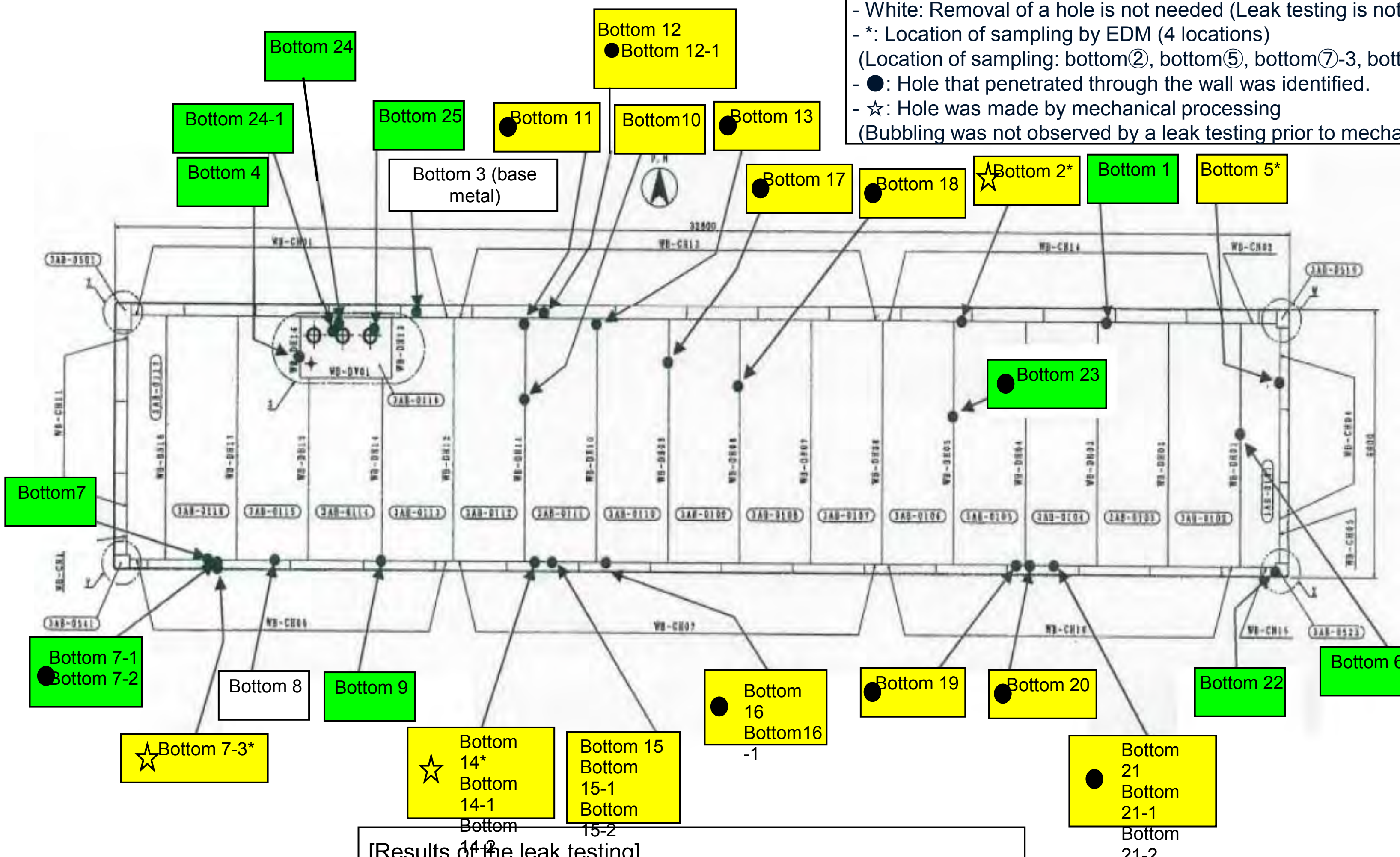
Affected systems and components at Hamaoka Unit 5

Extent of seawater intrusion



Corrosion of the bottom of the Condensate Storage Tank

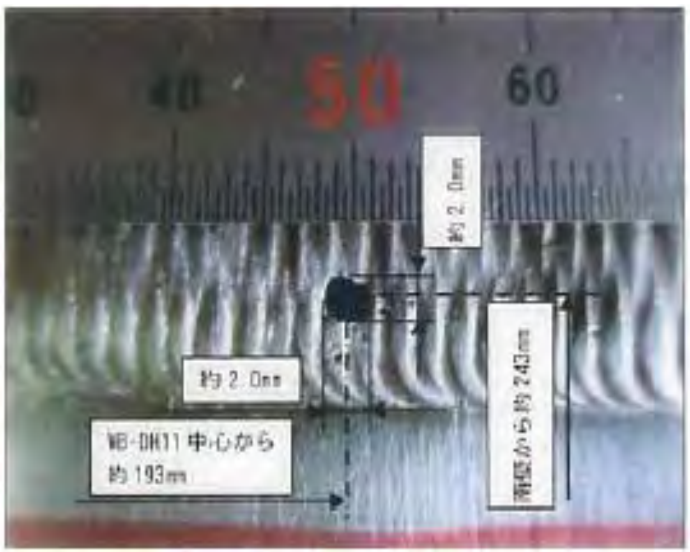
Meaning of colors and symbols:
- Green: Vacuum bubble leak testing is conducted prior to removal of a hole
- Yellow: Vacuum bubble leak testing is conducted during removal of a hole
- White: Removal of a hole is not needed (Leak testing is not applicable)
- *: Location of sampling by EDM (4 locations)
(Location of sampling: bottom②, bottom⑤, bottom⑦-3, bottom⑭)
- ●: Hole that penetrated through the wall was identified.
- ☆: Hole was made by mechanical processing
(Bubbling was not observed by a leak testing prior to mechanical processing)



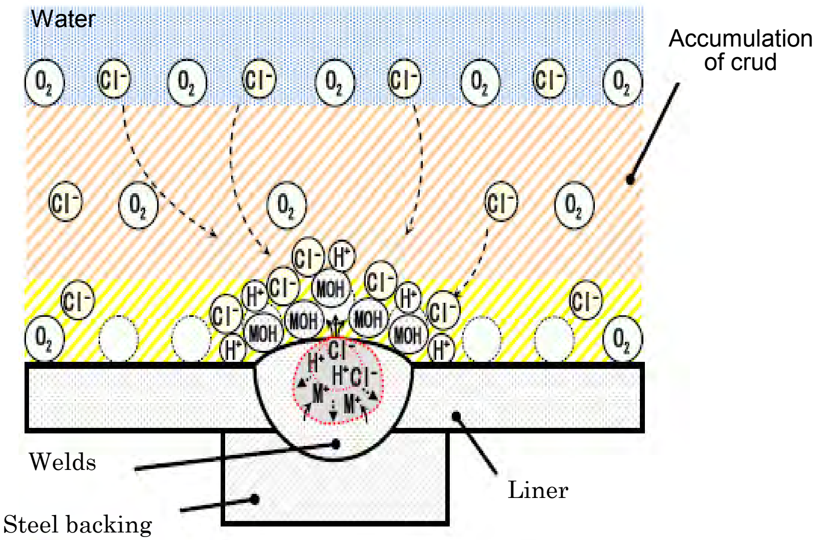
[Results of the leak testing]
Bubbling (●) was observed at 11 of 26 grooves for welding.

Corrosion of the bottom of the Condensate Storage Tank

Corroded hole butt welding part (No.1)
(hole depth: 4.89mm)
Locations: Bottom 14



(Illustration*)



: The region where effect of crevice generated due to accumulation of crud is shown as an image.

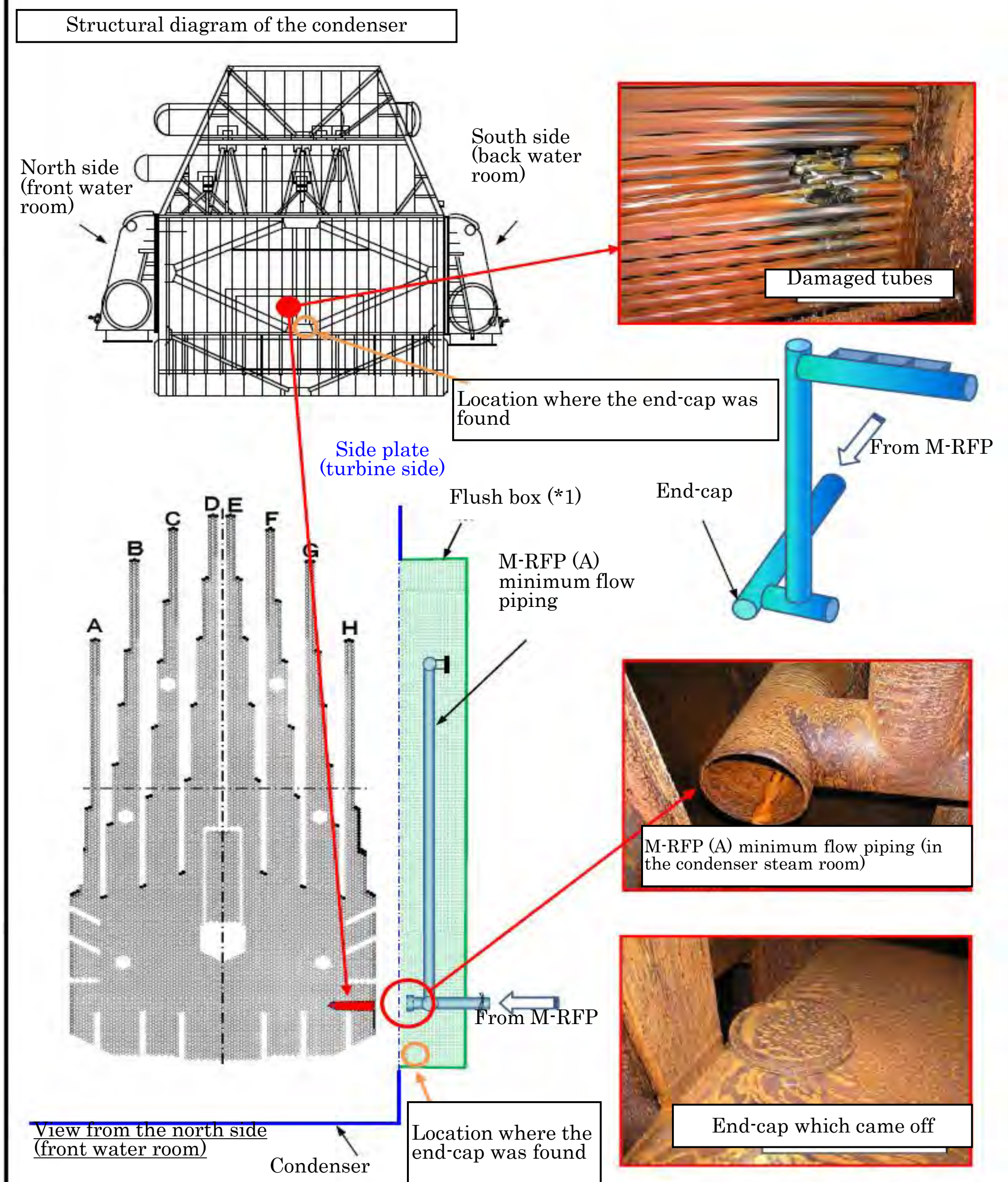
Seawater Intrusion Event at Hamaoka Unit 5

- Cause of Seawater Intrusion
  - ✓ At 16:30 on May 14,2011, during plant shutdown, the condenser tubes failed and seawater intruded into the facility.
  - ✓ Tubes failure was caused by a failure of a blank plate called the End-cap of MD-RFP minimum flow piping, which induce the jet flow to condenser tubes.
  - ✓ End-cap failure was due to the combination of initial crack during welding, high cyclic stress by MD-RFP operations.

JNES

Cause of Failure at End Cap and Condenser tubes

Status of damage of tubes in the condenser steam room (A-1)

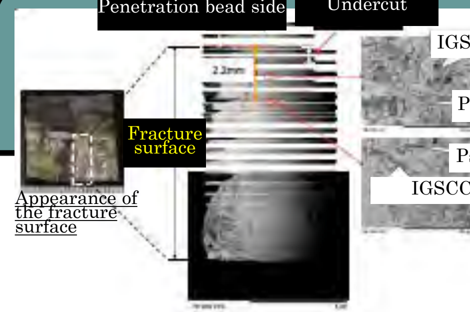


*1) This is a space for dispersing fluid energy by flushing (or reducing pressure and boiling) the high energy fluid before flowing into the condenser.

Cause of Failure at End Cap and Condenser tubes

Mechanism of damage of condenser tubes

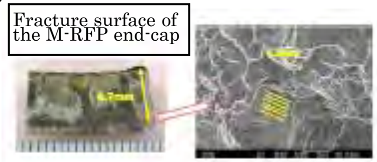
Crack generated at the welded end-cap (Occurrence of the initial welding defect)



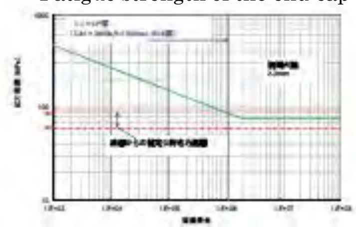
As a result of fracture surface observation, **undercut** (initial welding defect) **and low temperature crack** (pseudo cleavage fracture surface) were found.

Crack developed at the welded end-cap

Fracture surface of the M-RFP end-cap



Fatigue strength of the end-cap



As the **striation** was observed, it is estimated that stress greater than fatigue limit was applied to the end-cap and crack developed. Stress greater than fatigue limit is estimated to be 60-90MPa taking into consideration both effects of structural characteristic of the end-cap (stress of 77MPa was applied to the end-cap in the case that inner pressure is 1MPa) and acoustic resonance phenomenon (amplified 3.2-fold).

Remained sound welded end-cap is reduced and could not withstand the fluid force, causing it to come off.

As the plastic deformation was observed in the end-cap, it is estimated that remained sound part of the end-cap reduced due to the crack development and it could not withstand the fluid force, causing it to come off.

The end-cap came off and fluid ejected from the concerned part, causing tubes to be damaged

Damaged test tubes

Based on the fact that damaged tubes are just in front of the M-RFP minimum flow piping whose end-cap came off and test tubes were damaged in the "Tube damage test" which simulates the actual machine, it is estimated that the end-cap came off and a condenser tube was damaged by a jet flow from there, causing the condenser tubes to be damaged.

Seawater Intrusion Event at Hamaoka Unit 5

-Corrective measures-

- ✓ Structure of the end-cap is changed so as to be able to adopt butt welding.
- ✓ The holes on the CST will be repaired by welding etc.
- ✓ In preparation for a large amount of seawater intrusion in the condenser, response procedures* for prevention of intrusion to RCS will be specified in the plant manual.

  *: if conductivity over-scale, Spillover line valve close ~ Reactor shutdown ~ RCIC or HPCI start / Condenser Isolation ~ Feed water & Condensate water systems stop

Seawater Intrusion Event at Hamaoka Unit 5 (additional information)

April, 2013

Masakuni Koyama
Japan Nuclear Energy Safety Organization (JNES)

Historical background related seawater intrusion event of Hamaoka NPP unit-5

- 14 May 2011

During operating process after a reactor shut down at unit-5 of Hamaoka Nuclear Power Plant that was requested to stop its operation, event of large volume of seawater intrusion or about 400 m^3 into reactor facilities from condenser was occurred. Chubu Electric Power Company (Chubu EPC) preformed a investigation of cause of the event and a check and evaluation of effects of intrusion on related facilities.

- 30 March 2012

Nuclear Industry and Safety Agency (NISA) received a event report on through wall pitting of condensate storage tank of important safety related facility and requested the utility to conduct investigation of effect of seawater intrusion into reactor facilities.

- 25 April 2012

Chubu EPC submitted to NISA the 1st interim report on investigation of effects of seawater intrusion into reactor facilities in Hamaoka unit-5.

- 28 May 2012

Chubu EPC submitted to NISA a document on cause and corrective actions on confirmation through wall pitting of condensate storage tank that was reportable events.

- 25 July, 10 August, 23 August and 6 September 2012

NISA held some public meetings on investigation of effects of seawater intrusion into reactor facilities in Hamaoka unit-5

- 30 January 2013

Chubu EPC. Submitted to Committee of Nuclear Regulation Agency (NRA) the 2nd interim report on investigation of effects of condenser tube failure in Hamaoka unit-5.

- 1 February 2013

The 1st meeting of Task Group on Monitoring and Evaluation for Seawater Intrusion at Hamaoka unit 5 (TT-MESIH-5) was held by NRA's study team.

Task Group on Monitoring and Evaluation for Seawater Intrusion at Hamaoka unit 5 (TG-MESIH-5) as of February

● **Objectives of the Task Group**

○ Events of large volume of seawater intrusion into reactor facilities occurred at Hamaoka unit-5 on 14 May 2011:

- Unique event in the world
- Concerns of possible effects of intrusion on reactor facilities

○ To conduct appropriate safety regulation to the events,

- to monitor and evaluate the effect investigation of seawater intrusion and the maintenance and management performed by the utility,
- to assesse the effects on safety of reactor facilities and
- to accumulate the knowledge effective to further safety

○ Therefore, to monitor and evaluate the effect investigation of seawater intrusion and the maintenance and management in appropriate manners,

- Task Group on Monitoring and Evaluation for Seawater Intrusion at Hamaoka unit 5 (TG-MESIH-5) was established, composing of experts on corrosion and water chemistry

● **Issues to be considered at the moment**

○ Regarding the events, the interim report was summarized based on experts' discussions about check and investigation status carried out by the utility in September 2012.

○ Referring the report, the TG-MESIH-5 is

- to assess the check and investigation status again conducted by the utility,
- to study the event analysis of effects of seawater intrusion and integrity evaluation of reactor facilities,
- to monitor and evaluate further countermeasures carried out by the utilities.

Items of study in TG-MESIH-5 as of 1 February 2013

1. Integrity confirmation of reactor facilities

○ To identify methods of integrity confirmation of facilities considering seawater effects, to review relationship of the methods and technical standards and inspection, and to study further actions necessary to future maintenance and management.

○ Furthermore, to study countermeasures to possible phenomena seemed to be appeared in future.

(1) Confirmation methods of each facility

(2) Countermeasure methods for potential events seemed to be observed in future

2. Check & survey necessary to integrity confirmation

○ Conducting appropriate integrity confirmation, to evaluate what kinds of phenomena and effects have occurred and what degree of extent has been affected, based on checking actions.

○ Obtaining knowledge necessary to maintenance and management of facilities such as continuity and development of effects due to seawater, to conduct analysis and to clarify occurrence mechanism for identified events and their effects.

(1) Status of check for each facility

(2) Countermeasure methods for potential events seemed to be observed in future

(3) Understanding effect mechanism necessary to conduct further check and investigation of the facilities and to maintain and manage them.

● Referring status of check and investigation and discussions at TG-MSIH-5, items of study will be revised if necessary.

Items of study in TG-MESIH-5 as of 1 February 2013 (1/3)

1. Integrity confirmation of reactor facilities	(1) Confirmation methods of each facility	a. Identification for requirements of current technology standards and methods of the confirmation b. Appropriate confirmation methods for technology standard compliance considering these events c. Procedures and methods of integrity confirmation for components difficult to check (including ones with narrow gap area where it is difficult to conduct to access and check) d. Consideration of conservativeness (in quantitative and objective manners) in conducting tests under environmental conditions simulated ones of the plant. e. Confirmation of technology standard compliance for each facility f. Status of consideration and application for replacement and reuse of parts if necessary. g. Additional actions and checks necessary to further maintenance and management
	(2) Countermeasure methods for potential events seemed to be observed in future	a. Identify potential events seemed to be observed in future b. Methods to check and to confirm integrity of each facility considering these events

Items of study in TG-MESIH-5 as of 1 February 2013 (2/3)

2. Check & survey necessary to integrity confirmation	(1) Status of check for each facility	a. Events related to corrosion, occurrence of rust, incrustation of foreign substances for each facility and structure identified by checks and effects of the events on functions of each facility b. Validity of sample size on sampling check and scope of check for components important to safety c. Situation of record keeping for checked record (including check sheets, photos and movies as a reference) and obtained sample (including sampling water, parts machined out) d. Status of conducting mock-up tests and tests under environmental conditions simulated ones of the plant
	(2) Extents of effects on reactor facilities and status of their countermeasures	a. Extent and degree of effects on reactor facilities, based data of check and tests for each facility and structure (including results of evaluation by mock-up test and simulation) b. Confirmation of time trend of water chemistry after the event occurrence c. Occurrence condition of crevice corrosion d. Method of purifying and cleaning for each facility and situation of resume and reposition for then e. Plans for further check and investigation

Items of study in TG-MESIH-5 as of 1 February 2013 (3/3)

2. Check & survey necessary to integrity confirmation	(3) Understanding effect mechanism necessary to conduct further check and investigation of the facilities and to maintain and manage them	a. Basic idea of even calcification due to materials such as metal, non-organic and organic material and their degradations such as corrosion of metal in electrolyte of water, bimetallic corrosion, crevice corrosion, pitting corrosion, stress corrosion cracking, microbiological corrosion and so on. b. Change in water chemistry such as dissolved oxygen and hydrogen, crud type, ion type, heavy metal, non-organic material, organic material, marine creatures and microscopic organism, and change in their concentrations and degrees c. Temperature history such as high temperature stage during reactor shut down procedure and low temperature condition at plant shut down after seawater intrusion. d. Development of phenomena such as corrosion and degradation due to irradiation, that is corrosion development due to radiation degradation of seawater e. Consideration of effects of corrosion rate, microbiological corrosion, crud and so on. f. Study of potential extent subject to effects for each occurrence mechanism.

Further schedule

- February 2013: The 1st meeting of Task Group on Monitoring and Evaluation for Seawater Intrusion at Hamaoka unit 5 (TT-MESIH-5)
  - Review of check situation and basic idea of further study
- April – May 2013: The 2nd meeting of TG-MESIH-5
  - Confirmation of check situation after the 1st meeting.
- After that, the meeting of TG-MESIH-5 will be held as necessary based on situation of check and investigation conducted by the Chubu EPC

Meeting schedule of TG-MESIH-5 and check schedule by Chubu EPC

		2012 FY	2013 FY	2014 FY
Monitoring and evaluation by NRA Committee		Set up TG-MESIH-5 ▽	(TG-MESIH-5 to be held as necessary)	
Check by Chubu EPC	Core internals		(interruption)	
	Fuel			
	FW, CW system			

(Note) Chubu EPC has plans to interrupt check of reactor vessel and core internals of Hamaoka unit-5 from Dec. 2012 to Mar, 2014, due to transferring spent fuels of unit-1 and unit-2 to spent fuel pool in unit-5. Check of feedwater and condensate system and fuel is scheduled to be continued.

The 36 Meeting of the Integrity and Ageing of Components and Structures Working Group (WGIAGE)
The 18th WGIAGE Metal sub-group meeting
OECD Convention Center, Paris, France, 9-12 April 2013

Operational time limit extension regulation system in Japan

April 2013

Masakuni Koyama

Japan Nuclear Energy Safety Organization (JNES)

Contents

- Background
- Outline of regulation system
- Basic policy
- Remaining actions
- References
- Reference materials
  - ○ Flow of ageing management technical evaluation and ageing management implementation
  - ○ Evaluation of ageing mechanism (example)
  - ○ Status of validity confirmation of ageing management technical evaluation of nuclear power plants

Background

● 11 March 2011

The grate east Japan earthquake hit Japan.

● 27 June 2012

Amended Nuclear Regulation Act was promulgated.

- New regulation against severe accidents
- Regulation system based on the latest scientific / technical knowledge
- 40-years operation limit for NPPs

● 27 February 2013

○ Secretariat of Nuclear Regulation Authority (NRA) issued position paper, entitled " Consideration of operation period extension permission relegation system".

○ It said that regarding operational time limit extension regulation system introduced under the Amended Nuclear Regulation Act in July 2013, based on a direction at Committee of NRA on 20 February 2013 and considering relationship with existing ageing management regulation system, it promotes to prepare necessary Government Order and rules on a policy as follows;

[http://www.nsr.go.jp/committee/kisei/20130227.html]

Outline of regulation system

(1) Operational time limit extension regulation system

- Legally defines the time of NPP's operation to 40 years for the past date of its per-service inspection.
- The Nuclear Regulatory Authority (NRA), however, can give permission to extend operational time limit of NPPs by certain time period (not longer than 20 years, which will be defined by the Government Order) only once.
- This permission is to be given only if NPPs comply with technical standards developed by the NRA, which examines its safety considering ageing of nuclear facilities in a long time operation.

(2) Ageing management regulation system

- Every-ten-year ageing evaluation of components and structures and development of long term maintenance policy for reactor facilities that has been 30 years since operating commissioning are mandated and the policies were subjected to operational safety program permission.
- The framework of the system is stipulated by the Nuclear Regulatory Authority Rules and detailed evaluation items are provided by NRA's bylaw.
- Under the system so far, we have experiences of regulatory reviews and confirmation of the validity for the ageing technical evaluations of 17 units for 30 years since operation commission and 3 units for 40 years.

Basic policy

(1) Operational time limit extension regulation system

● **Basic policy of extended periods**

Terms of years defined by Government Ordinance will not exceed upper limit of 20 years that is stipulated by the Act.

Specific extended terms will be determined based on reviewing each units.

● **Basic policy for regulatory system design**

(1) In making judgments if operational time limit extension is valid or not,

Current status of the plant should be assessed in detail and implementations of check and confirmations of degradation status reflecting latest knowledge are required for portions where were not confirmed every 13 month regulatory periodic inspection and the every 10 years ageing management regulatory system after 30 year operation for un-replaceable critical components and structures such as reactor pressure vessel and concrete structures,

(2) Regarding degradation evaluation,

Evaluations of items for ageing mechanism in conducting ageing management regulatory system are required the utilities to carry out in well conservative manners reflecting latest knowledge and to assess to ensure that technical standards will be satisfied during the extended periods.

(3) As for extension permission,

Making judgment for consistency of “back-fitting regulatory system”, compliance with latest technical standards at permission stage and compliance with such standards considering degradation during extended period are confirmed.

Basic policy

(2) Relationship of ageing management regulation system

● Ageing management regulatory system allows only one time of permission based on the evaluation at 40 years.

For example, even if the case that the premises of permission were not complied without conducting appropriate maintenance management would be occurred, there would be no effective countermeasures under such system.

● In order to secure an appropriate safety during extended periods, it is required to conduct maintenance managements including monitoring of degradation status, checking, refurbishments and replacements.

● For this reason, it is effective way that it forced utilities which enter into extended periods to establish **long term maintenance management policy** and to pledge it based on **operating safety program**. Therefore, the ageing management regulatory system will be applied.

● In particular, the regulatory authority will require utilities to submit an application of operating safety plan including **long term maintenance management policy** along with submission of **application of operating period extension** and it will prepare the rules necessary to ageing management regulatory system.

Remaining actions

- **Regarding draft of government orders and rules,**

  Based on the discussion of Commission of NRA, Secretariat of NRA prepares the draft to seek public comments in April and to put the draft into effect along with enforcement of amended Nuclear Regulation Act in July.

- **As for bylaw on detailed system implementation,**

  It will be prescribed in order of precedence according its contents.

  Partial modification of the bylaw on ageing management regulatory system will be conducted along with the enforcement of amended Nuclear Regulation Act in July, including to require evaluation of effects on facilities of long term shutdown and to clarify and evaluate affects on plants experienced earthquake and/or tsunami.

References

[1] Operational time limit extension regulation system, Secretariat of Nuclear Regulation Authority (NRA), The 21rd meeting of the commission of NRA, 27 February 2013 (in Japanese), http://www.nsr.go.jp/committee/kisei/h24fy/20130227.html

Flow of ageing management technical evaluation and aging management implementation

Reference material-1

Confirmation by the government

Review of validation of AMTA and Long-term Maintenance Management Policy

Evaluation of technical validation AMTE (JNES)

Confirmation of the validation and approve (OSP)

Confirmation in strictly manner during PI, PSMR, SI etc.

Ageing management technical evaluation (AMTE) implemented by electric utilities

Flow of AMTE (for 30 year)

Evaluate all facilities with safety functions

Identify ageing mechanisms
(SCC, fatigue, neutron embrittlement, pipe wall thinning, etc.)

Identify characteristics of ageing mechanism and evaluate integrity assuming 60 years services

In addition to normal maintenance plan, to develop "Long-Term Maintenance Management Policy" including necessary ageing managements

Reflect

Additional requirements in AMTE before 40 years of operation

1. Verification of AMTE conducted before 30 years of operation
2. Evaluation of operational experiences
3. Evaluation of effectiveness of long-term maintenance program established before 30 years of operation

Implementation of aging management

Surely Implement additional maintenance measures based on LTMMP every operational cycle

OSP : Operational safety program
PI : Periodic inspection
PSMR : Periodic safety management review
SI : Safety inspection

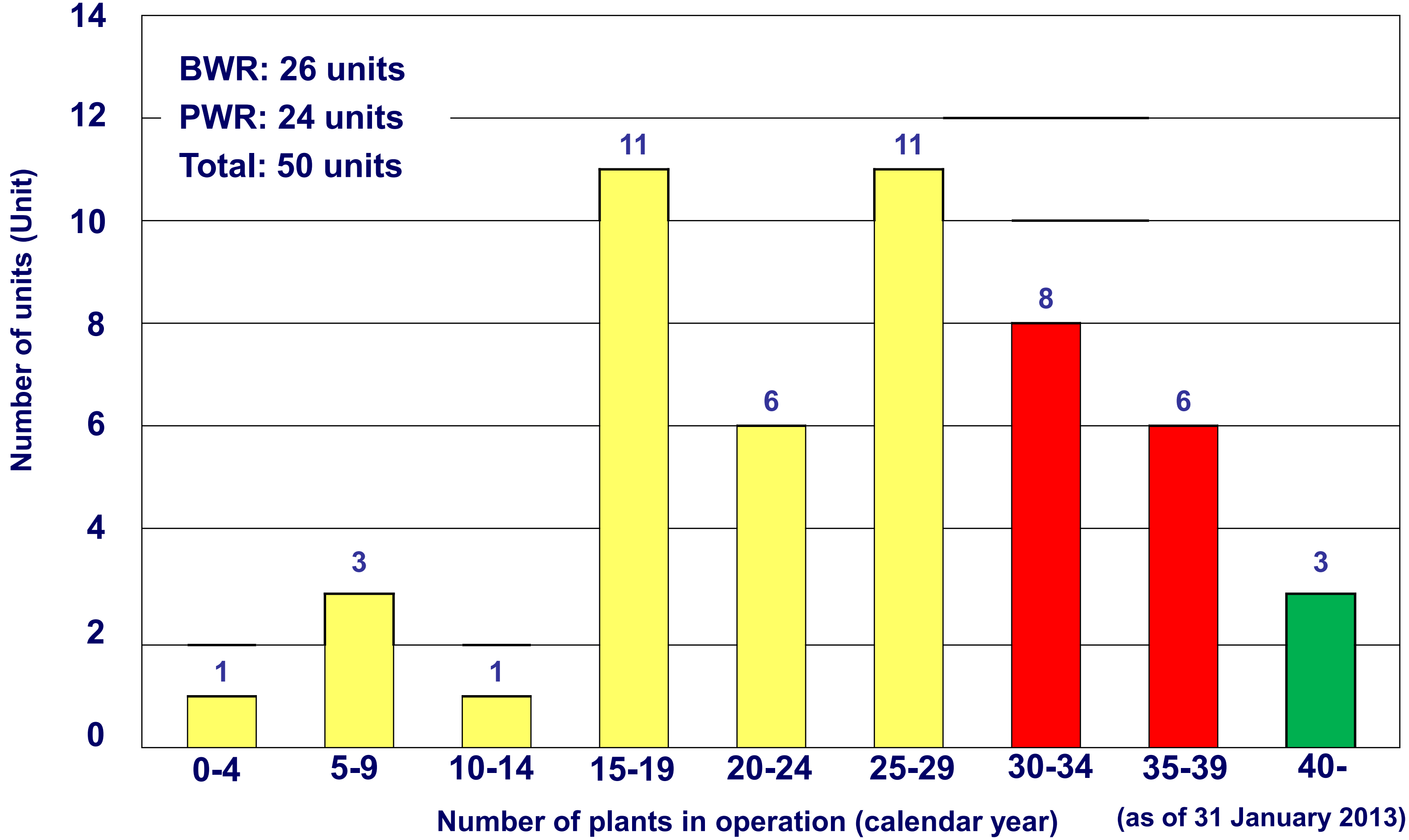
Evaluation of ageing mechanism (example)

Reference material-2

- Ageing management technical evaluation
  - Ageing mechanism
    - ① Low cycle fatigue — Reactor vessel, pressurizer, pump, piping etc.
    - ② Neutron irradiation embrittlement — Reactor vessel
    - ③ Irradiation assisted corrosion cracking — Core internals
    - ④ Thermal ageing for duplex stainless steels — Stainless steel casing for piping, pump, valve etc.
    - ⑤ Insulation degradation for electric and I & C apparatus — Electrical cable etc.
    - ⑥ Strength reduction and shielding performance reduction for concrete — Concrete structure
    - ⑦ Stress corrosion cracking (Except IASCC) — Reactor vessel, pressurerizer, steam generator etc.
    - ⑧ Pipe wall thinning — Carbon steel piping, low alloy piping etc.
    - ⑨ other ageing mechanisms — Vessel corrosion, piping corrosion etc.
  - Seismic behavior — Considering accelerated ageing degradation such as anchor bolt corrosion, piping corrosion etc.

Status of validity confirmation of ageing management technical evaluation of nuclear power plants

Reference material-3



- Operation years and number of units of NPPs -

- : Completed AMTE after 40 year operation (3 units, excluding 1F1)
- : Completed AMTE after 30 year operation (17 units including plants for 40 year operation AMTE, excluding 1F1, 2, 3, 4 and H1, 2.
- : To be prepared for AMTE

AMTE: Ageing management technical evaluation, 1F: Fukushima-Daiichi NPP, H: Hamaoka NPP

IAM BRNO

Institute of Applied Mechanics Brno, Ltd.

Steam Generator DMW repair

Lubomír Junek, Ph.D.

Jiri Ždárek, CSc.

10.04.2013

Czech NPPs of WWER type

IAM BRNO

NPP Dukovany

4 Units / 471 MWe per Unit

27 years in operation from 1985

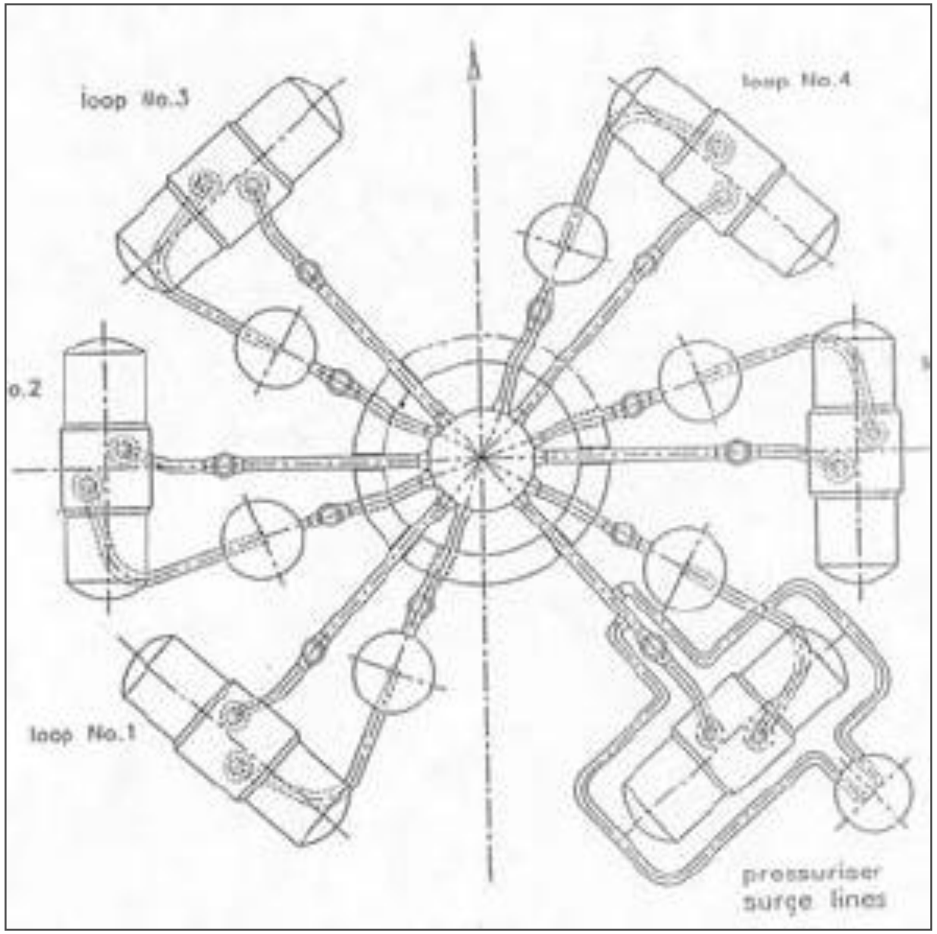
NPP Temelin

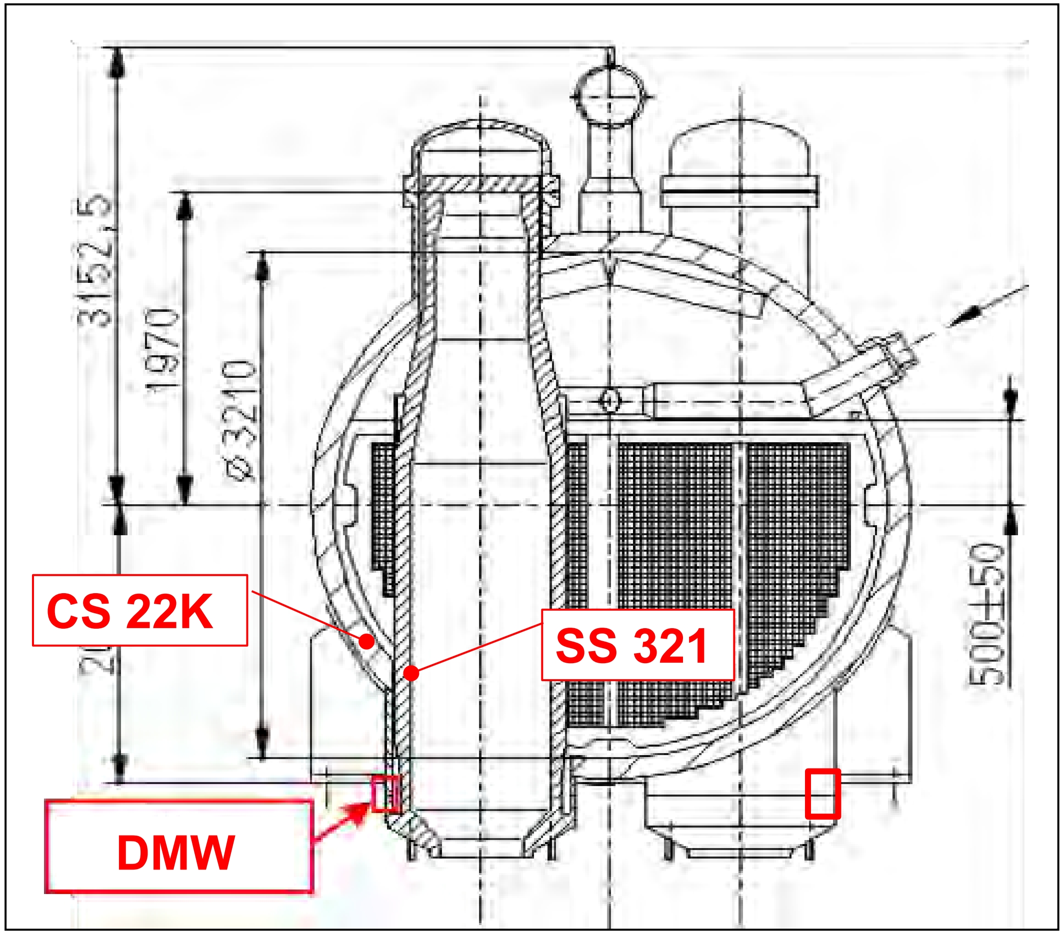
2 Units / 1000 MWe per Unit

10 years in operation from 2002

1

NPP Dukovany





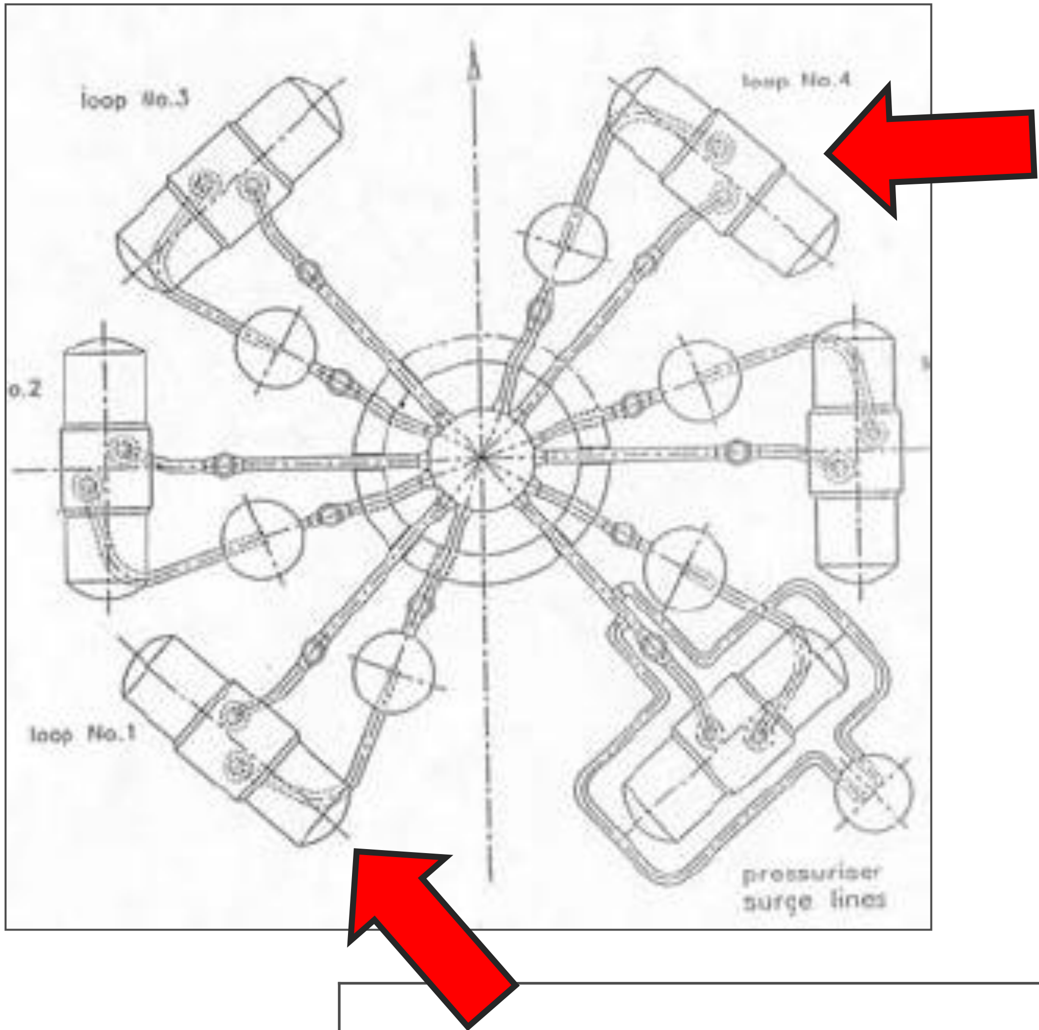
6 Loops per Unit (6 SGs)

2 DMW per SG

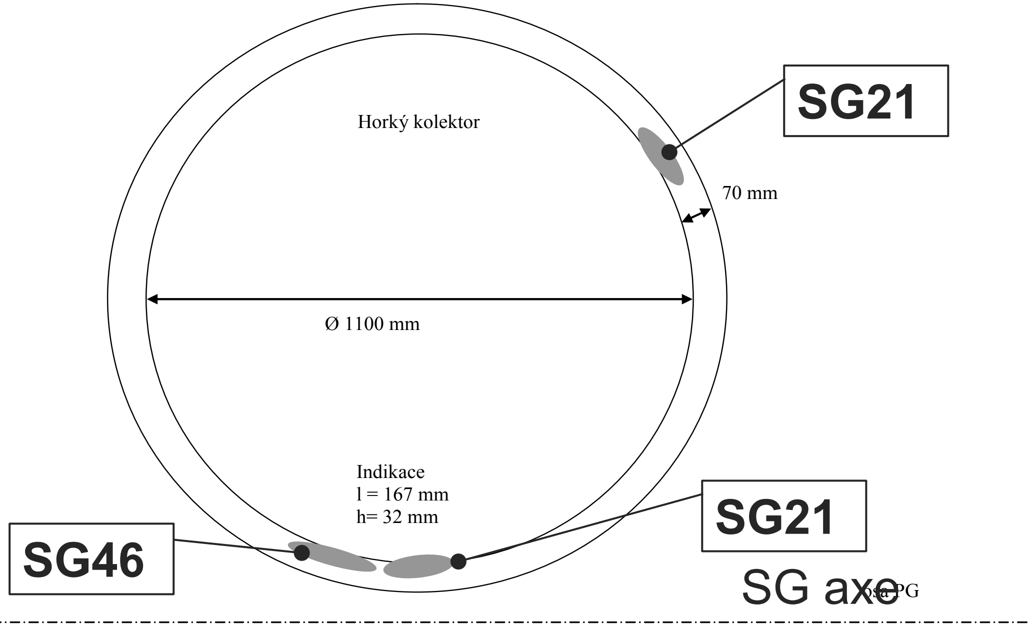
12 DMW per Unit

48 DMW on NPP Dukovany

Indications on SG DMW



SG 46		
2011	L = 167 mm	H = 32 mm
2012	L = 210 mm	H = 45 mm



SG 21				
2010	L = 79 mm	H = 23 mm	L = 91 mm	H = 26 mm
2012	L = 84 mm	H = 29 mm	L = 108 mm	H = 31 mm

Content of Presentation

Welding technology of SG DMW repair was prepared

Repair technology was qualified

SG 46 was repaired (19.11.2012 – 10.12.2012) – 20 days

DM can be described

NDT (UT PE) inspection was qualified, PAUT is accepted

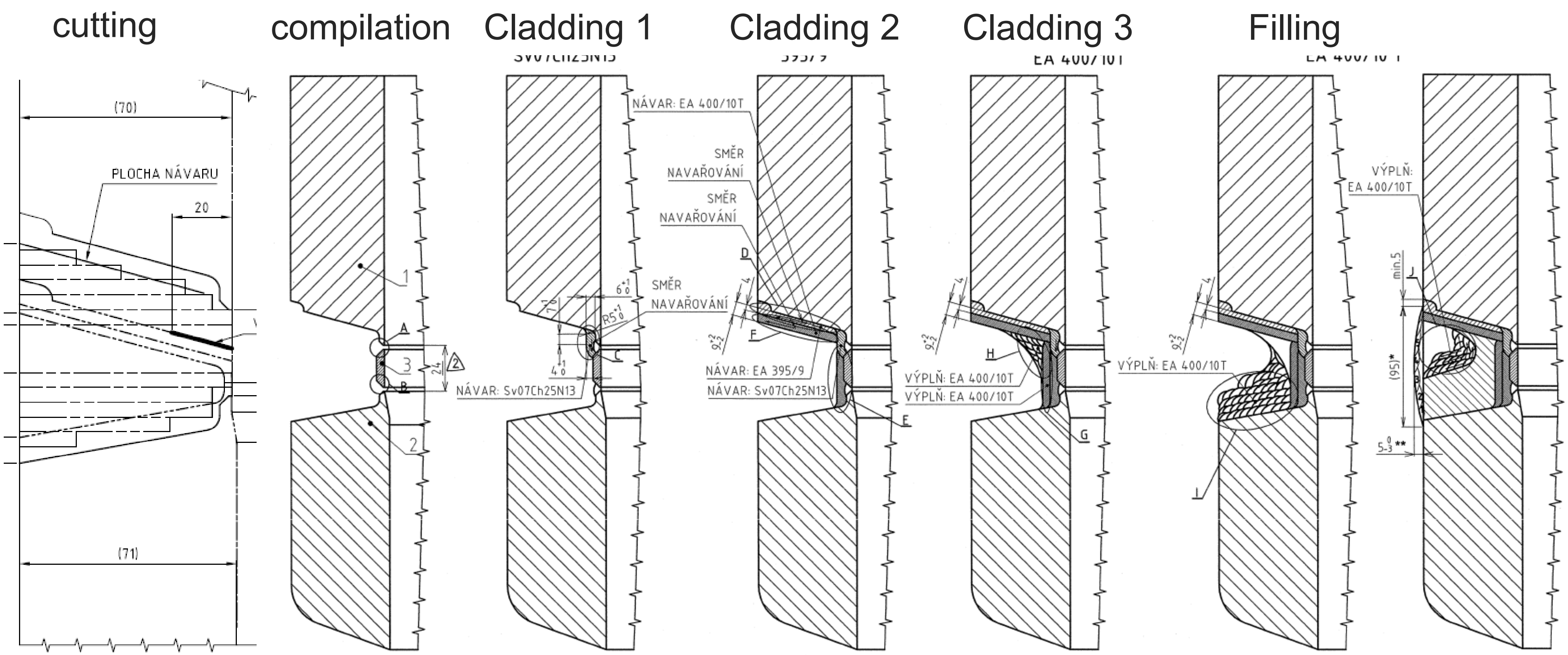
Action plan of SG repair depend on detected indications

SG21 was started repaired (4.4.2013)

Actions from 2012

Welding SG DMW Repair Technology

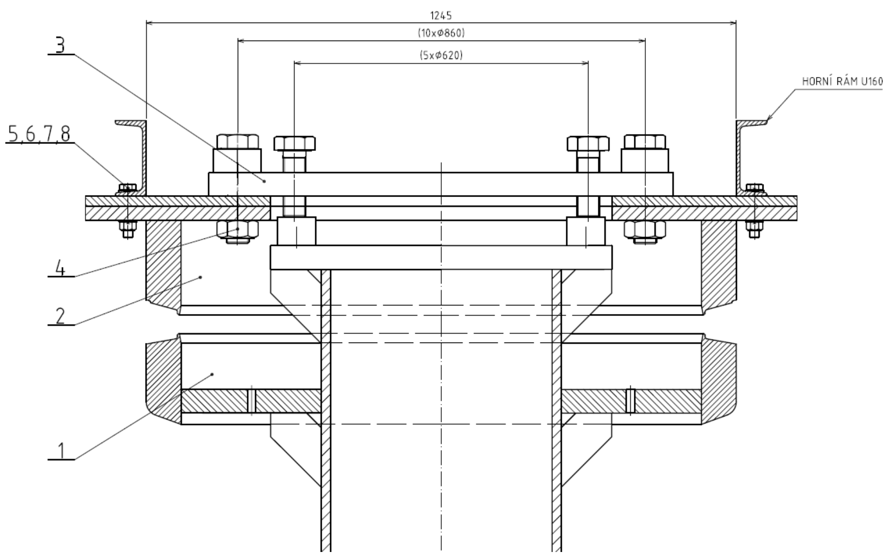
IAM BRNO



The same technology as in Russia

Repair DMW Qualification (EU Standard)

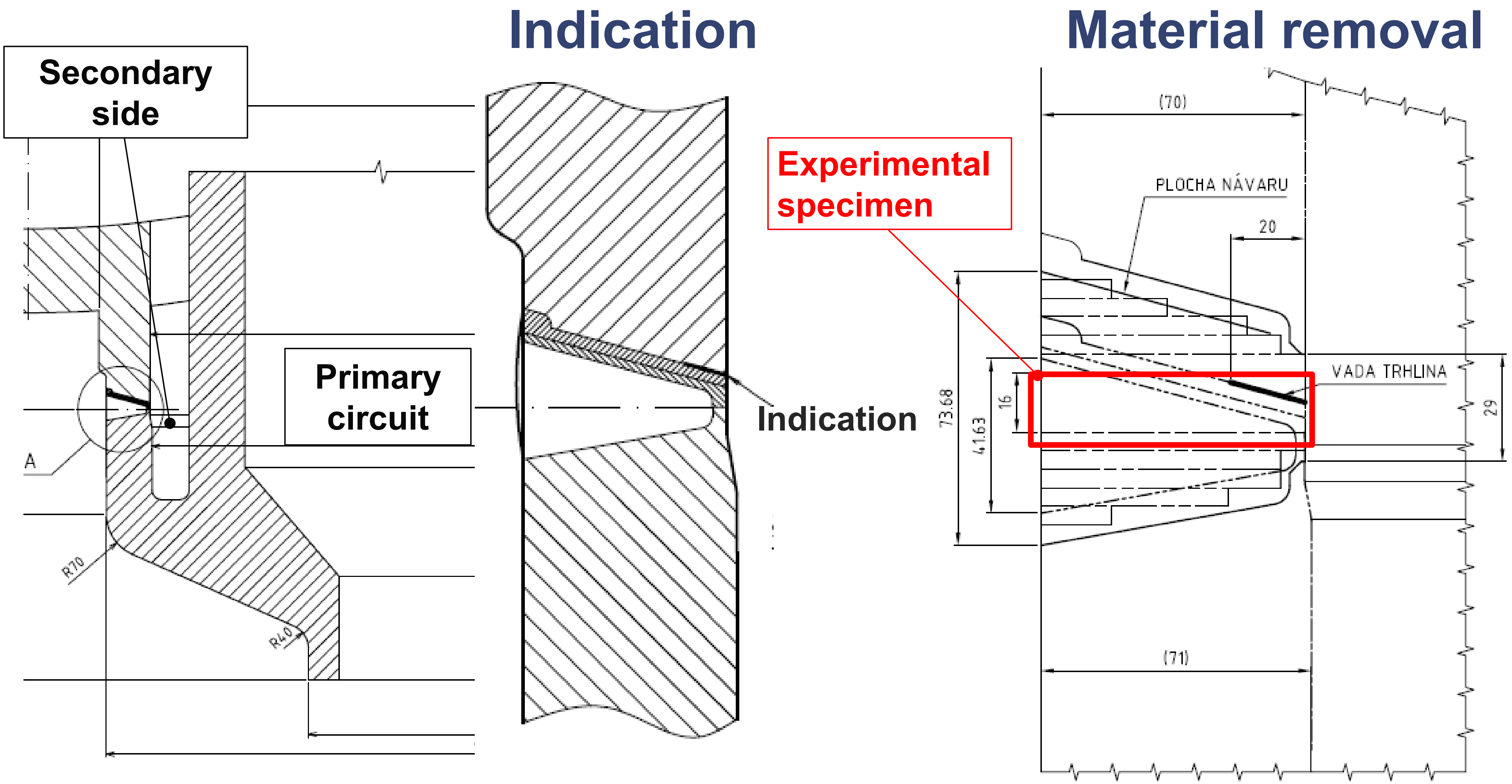
IAM BRNO



Scale 1: 1

Removal of Original Material

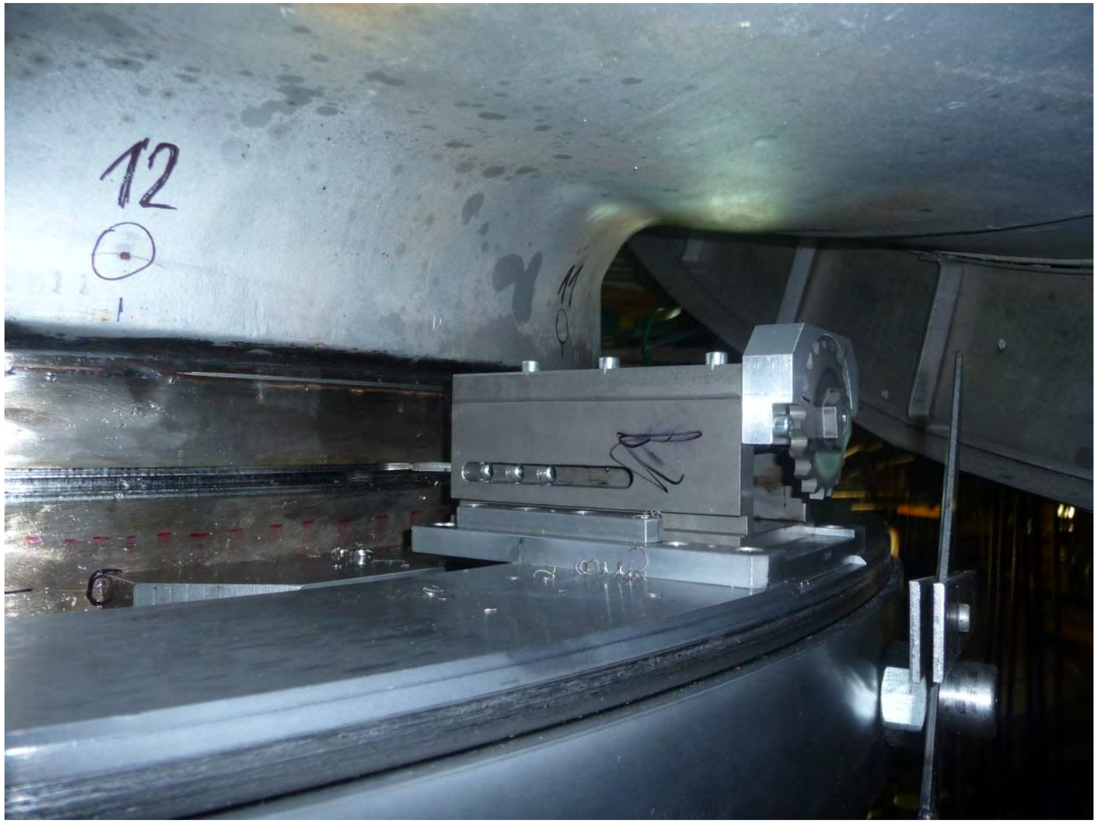
IAM BRNO

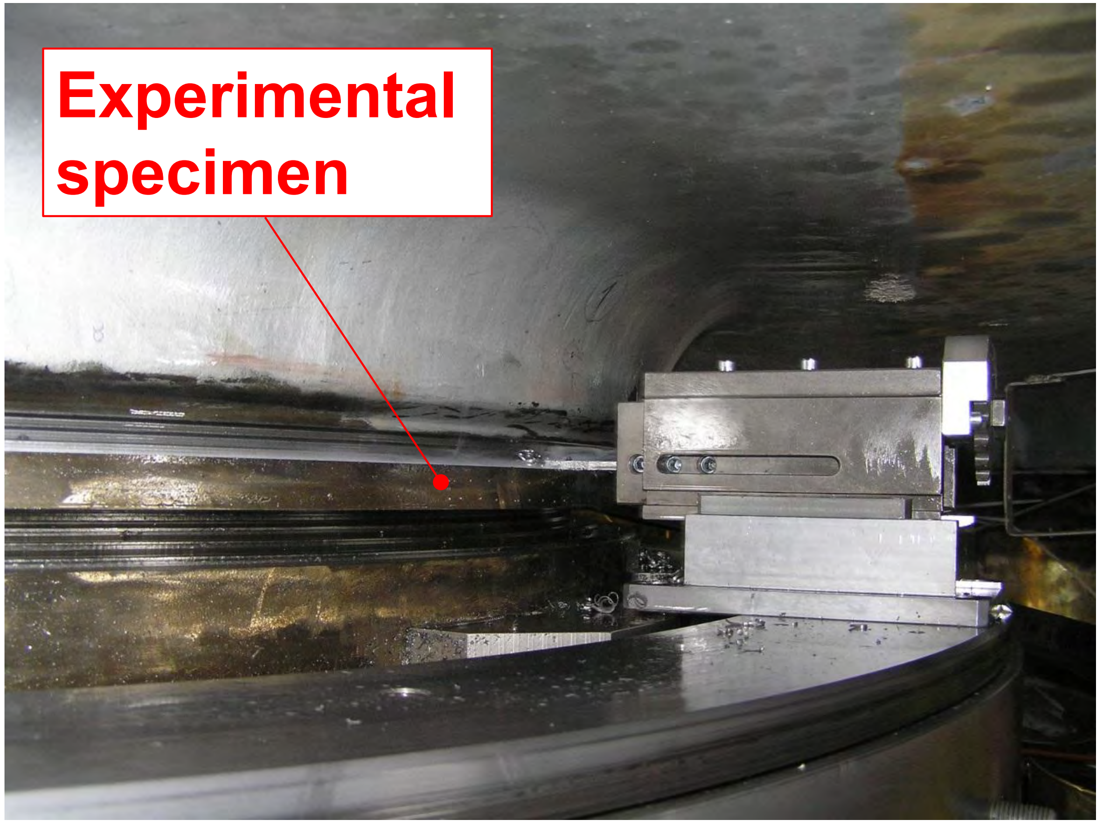


Removal of Original Material

IAM BRNO

Cutting Machine



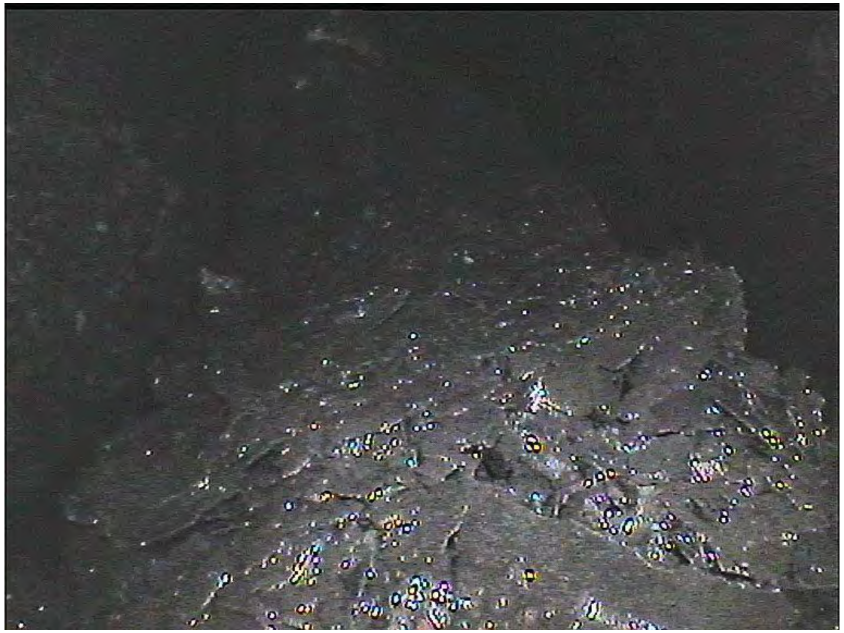


Confirmation of Indication

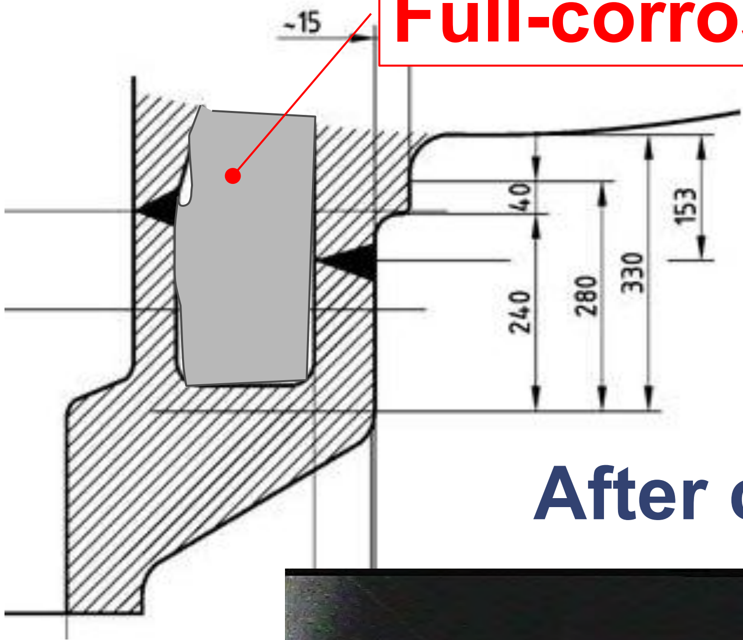
IAM BRNO

Depth will be confirmed on SG21

Corrosion Products in “KARNAN” and Cleaning



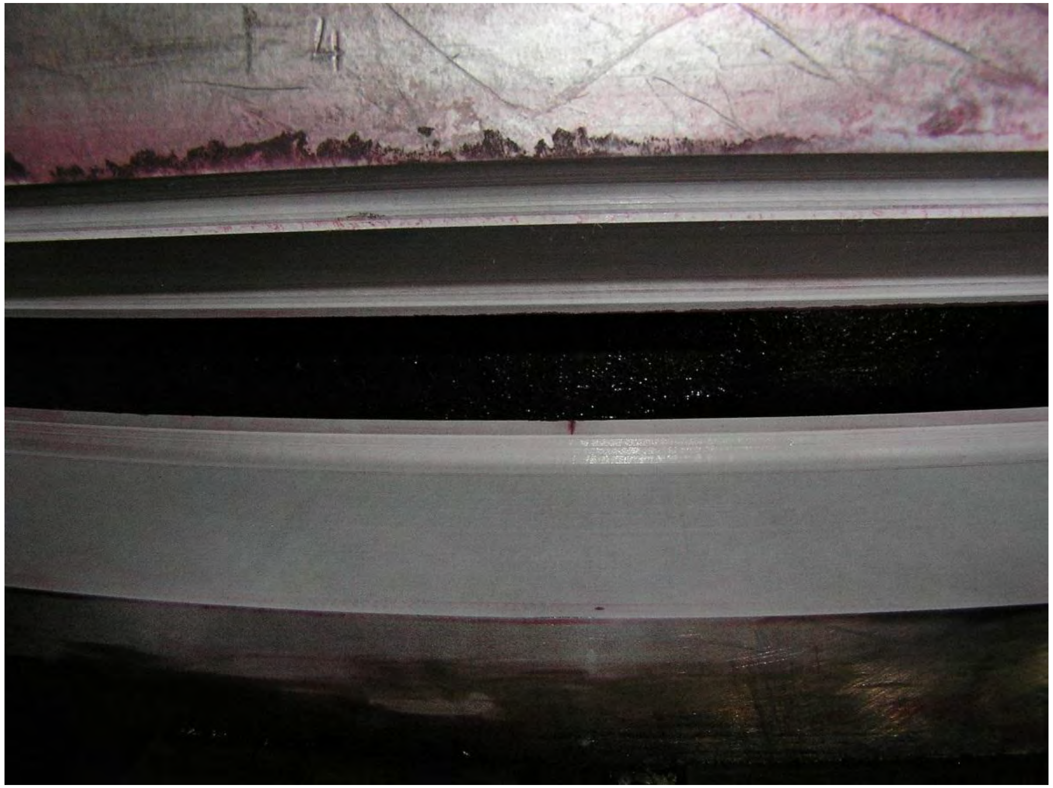
Full-corrosive deposits

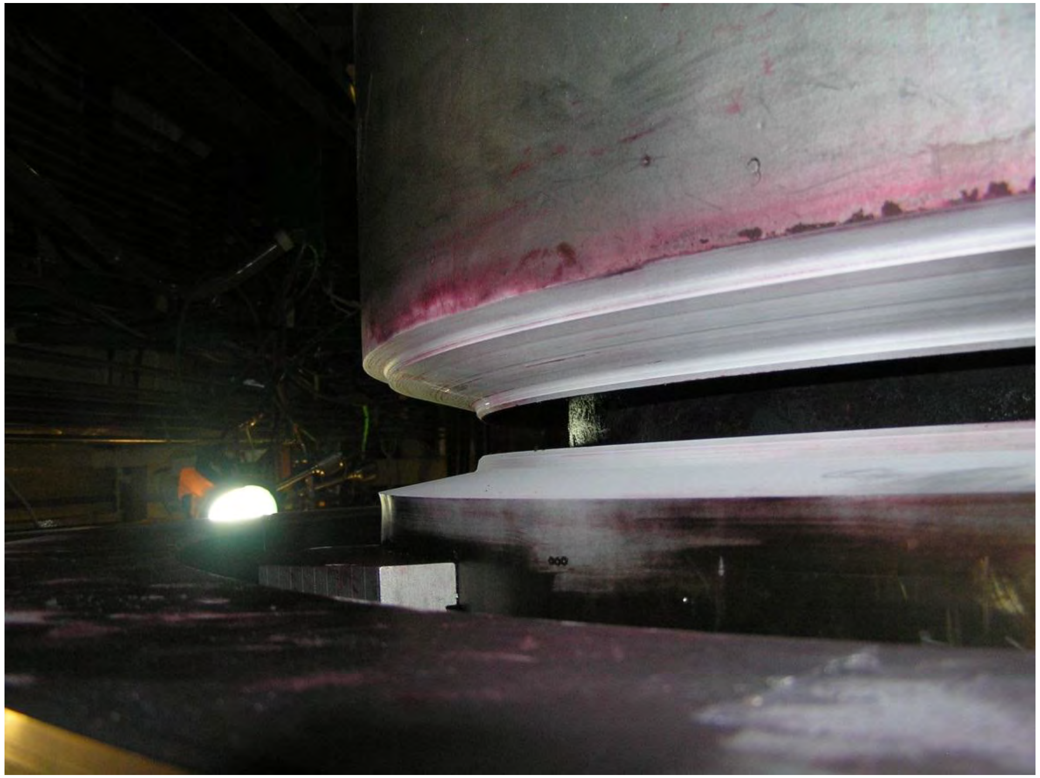


After cleaning

Before welding

New weld area





Welding Technology Repair of SG DMW

New root of DMW weld

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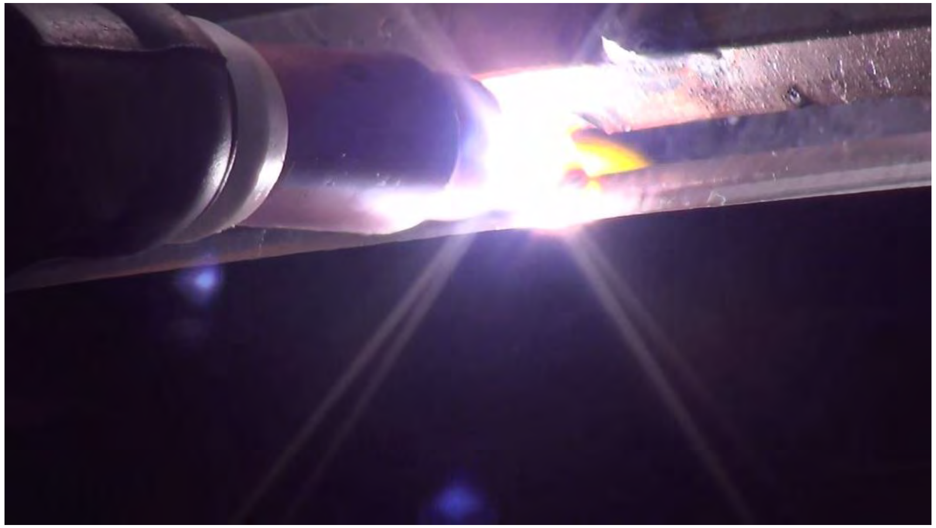
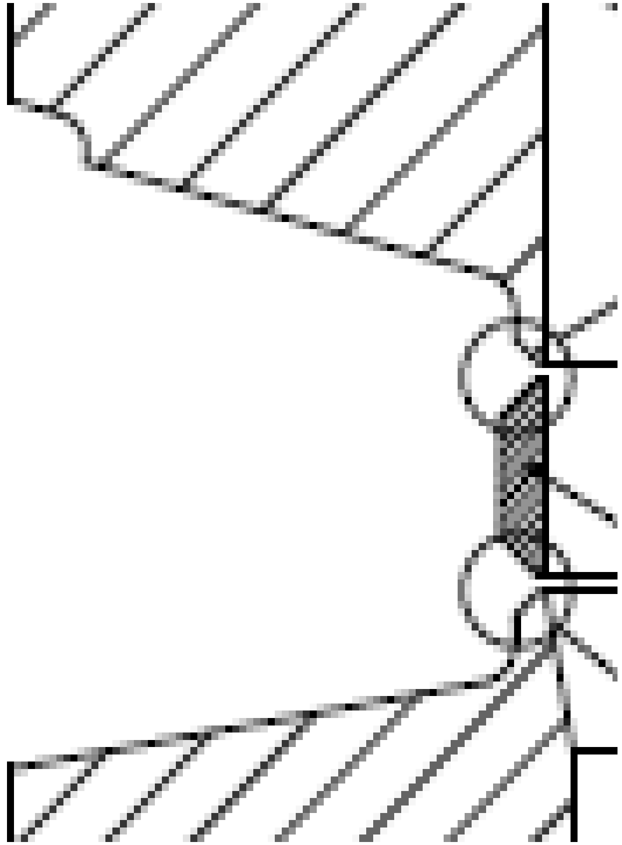
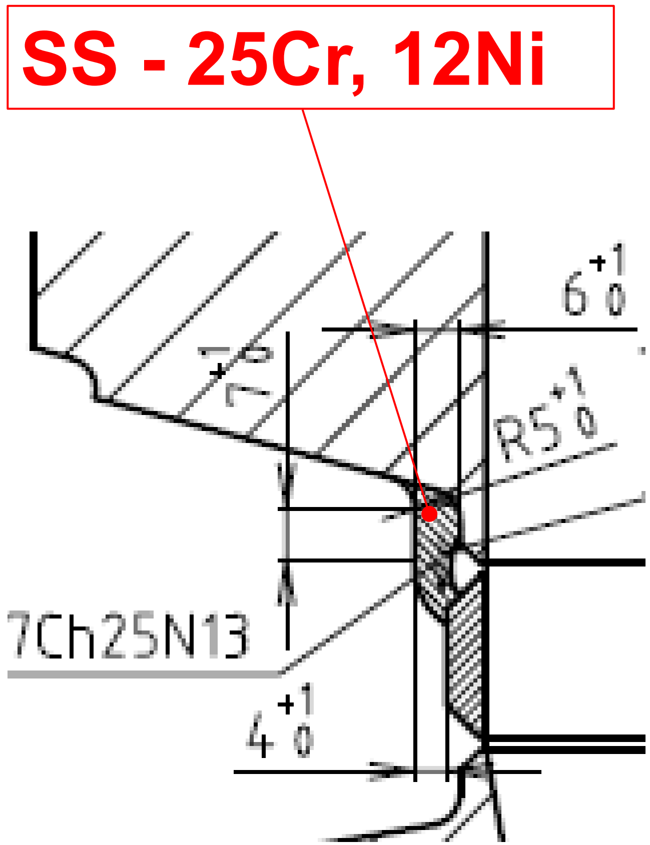
Material with 25Cr, 12Ni (SS 321) instead 25Ni, 12Cr





Welding Technology Repair of SG DMW

New root of DMW weld



Filler Welding

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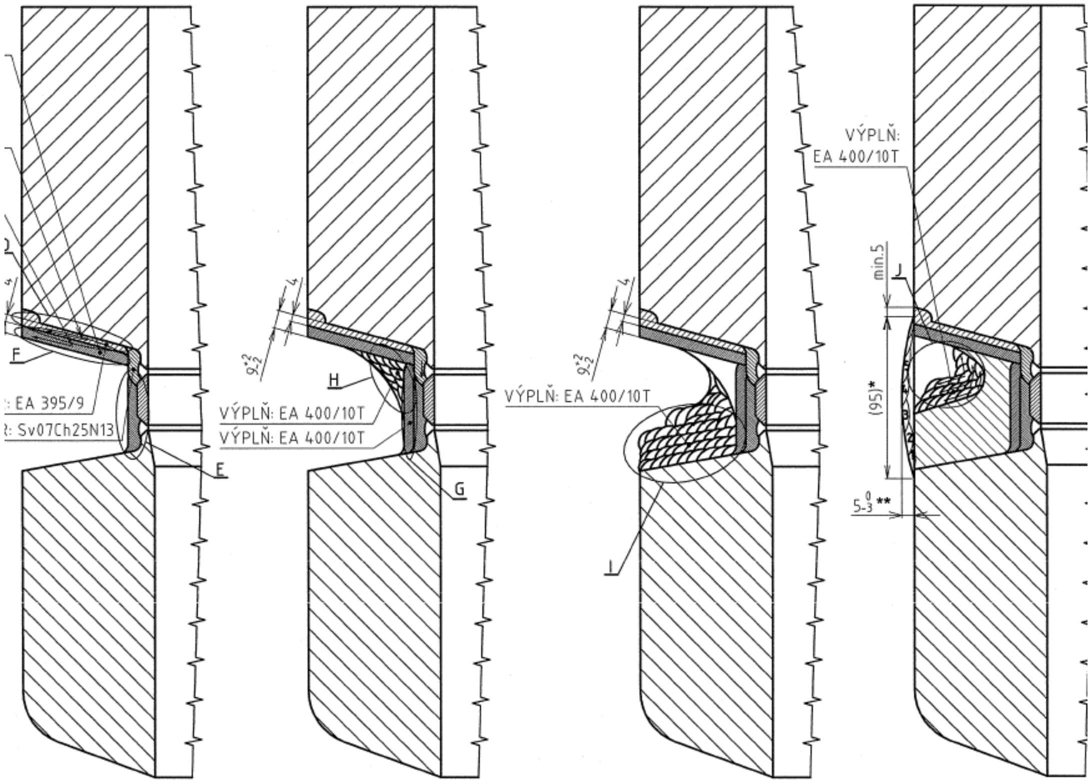
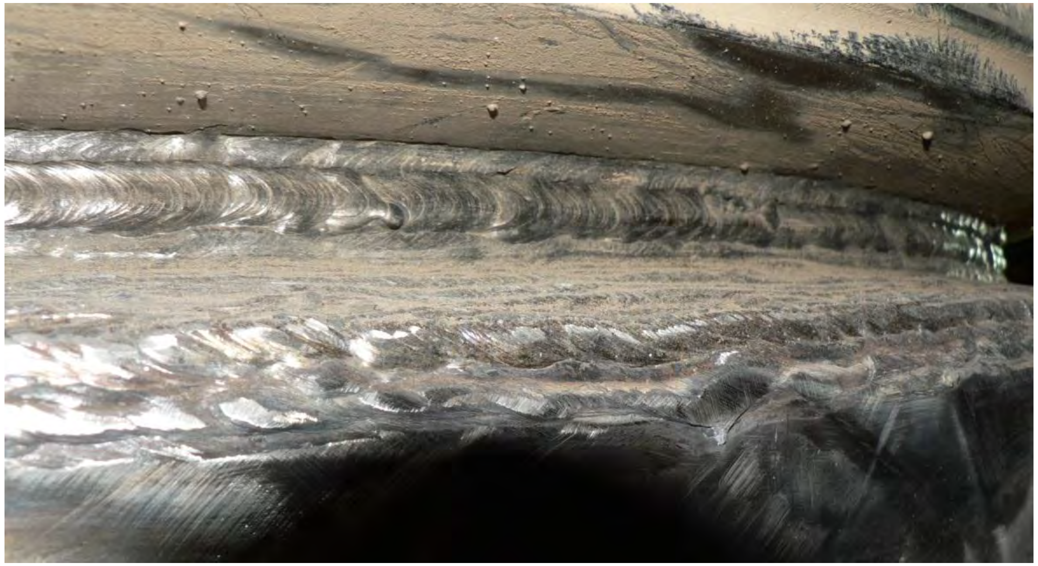


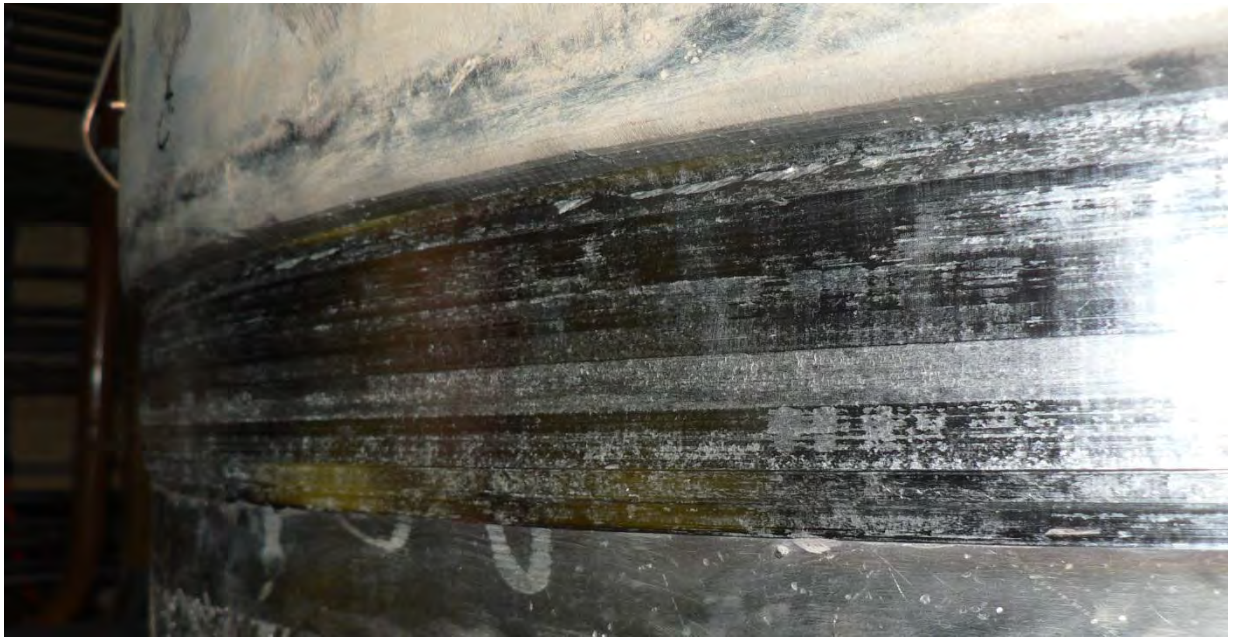


Welding Technology Repair of SG DMW

New SG DMW after Repair

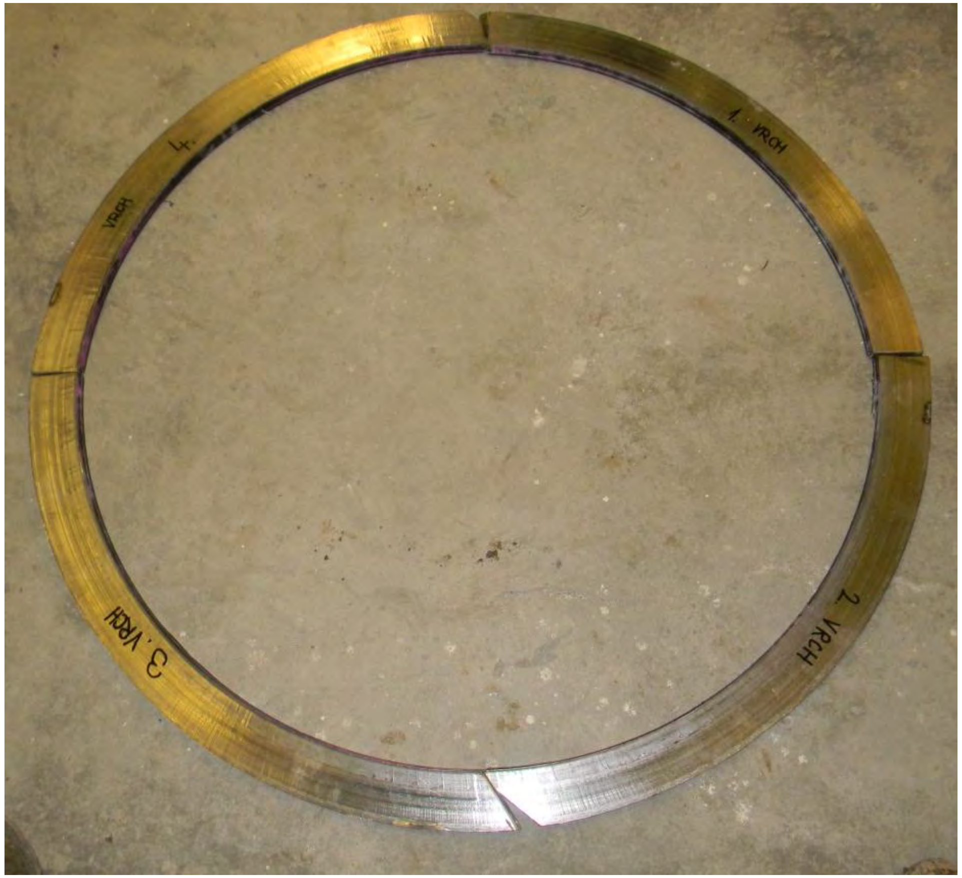
IAM BRNO

New DMW

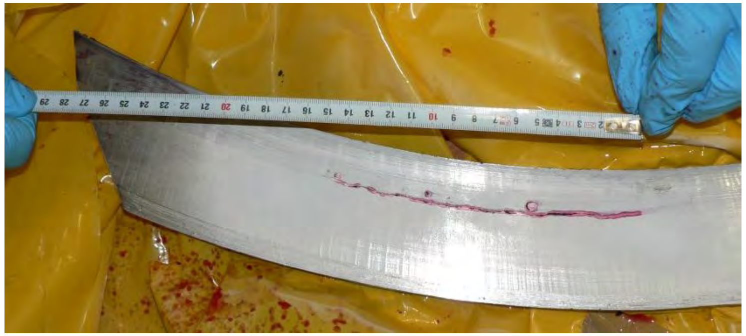


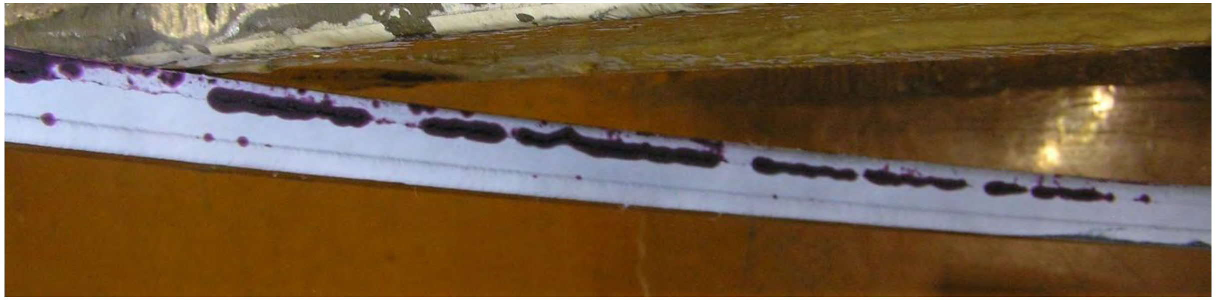
20 days

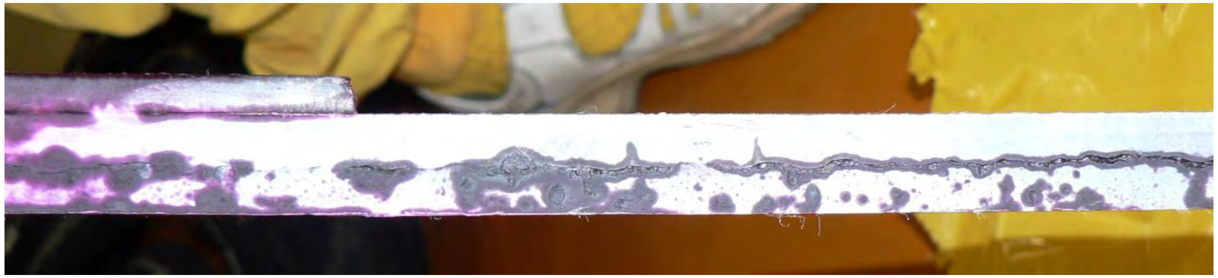
Degradation Mechanism Determination



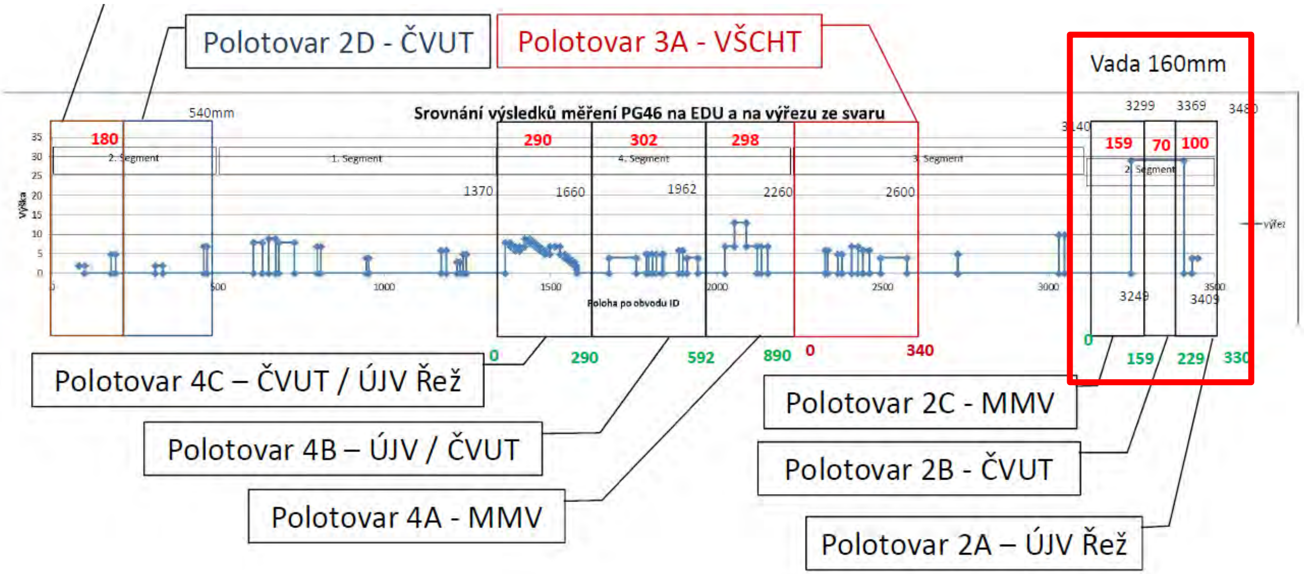
Material experimental program was prepared

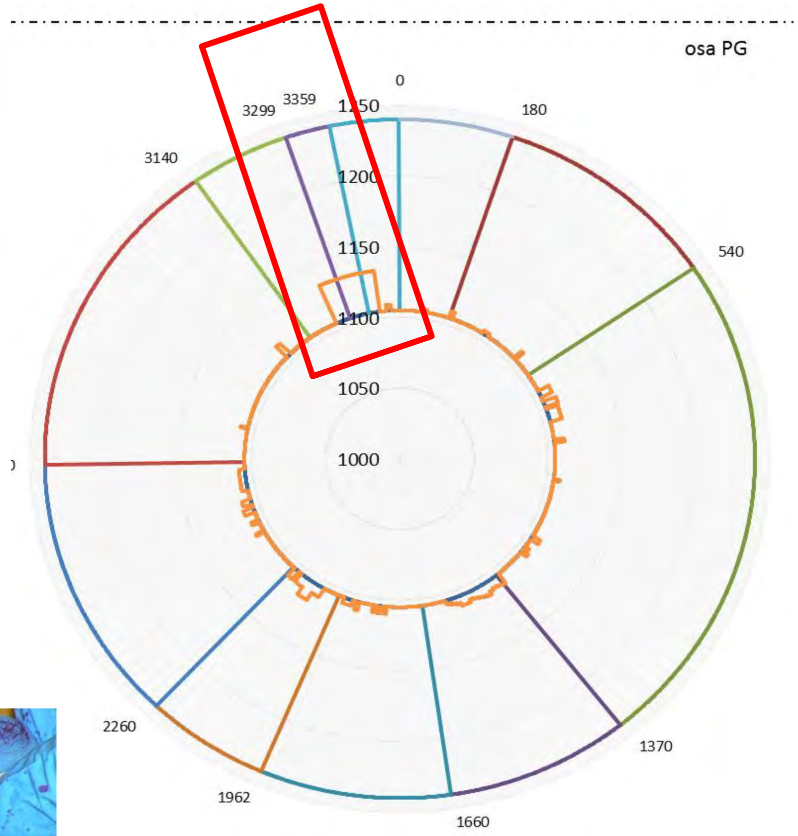


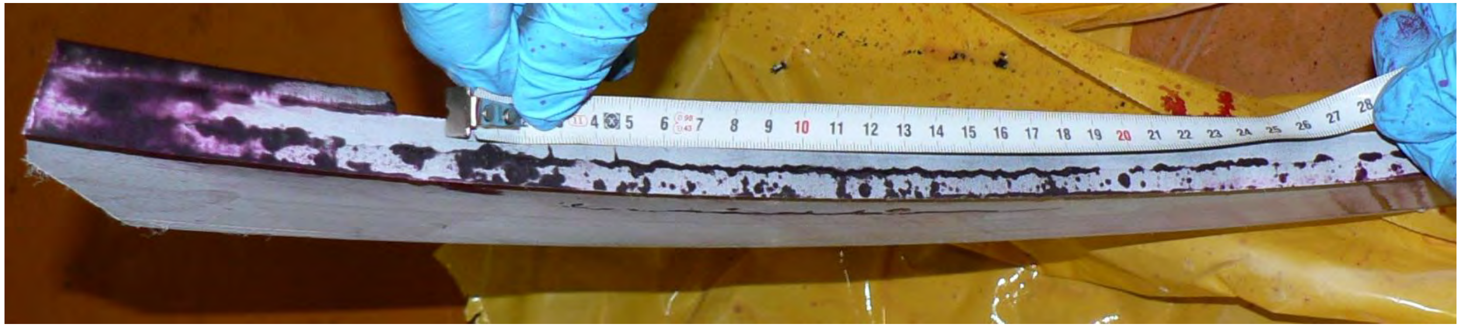




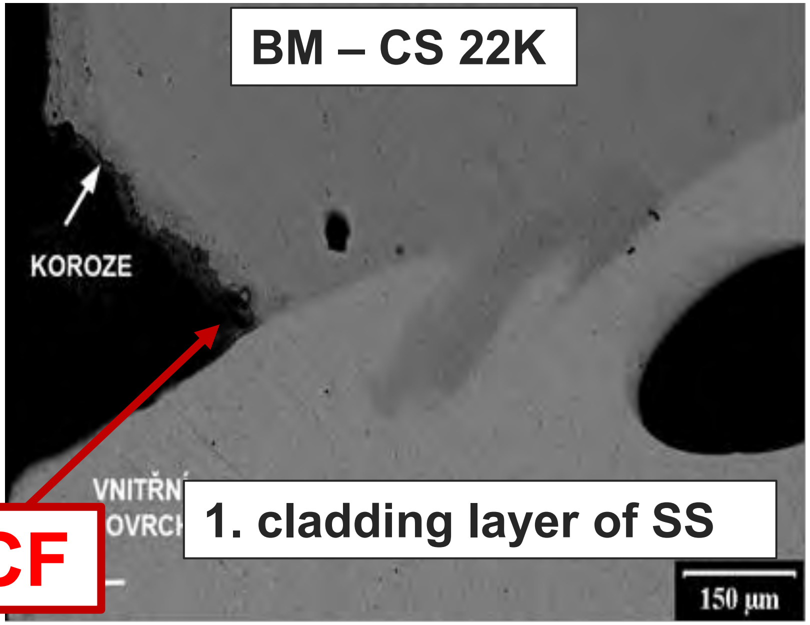
Defects Validation on Specimen



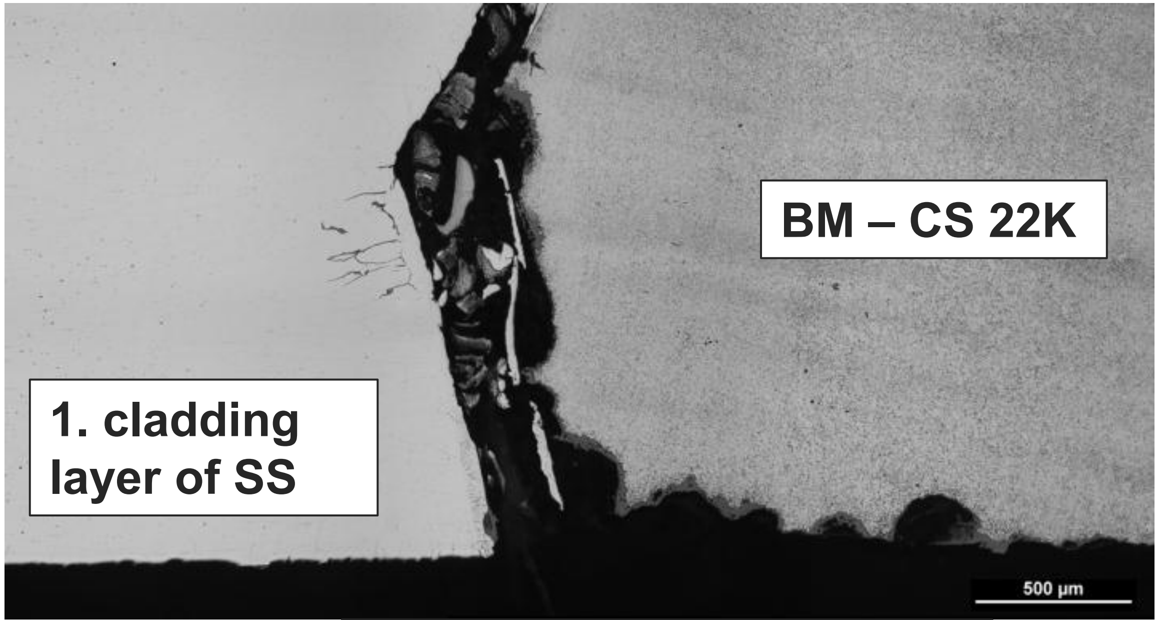




Initiation Process - Electrochemical Corrosion

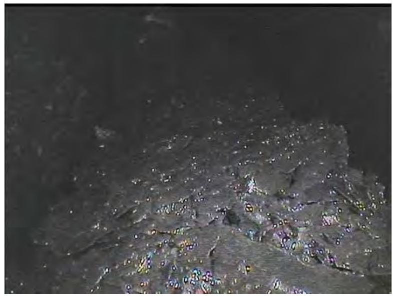


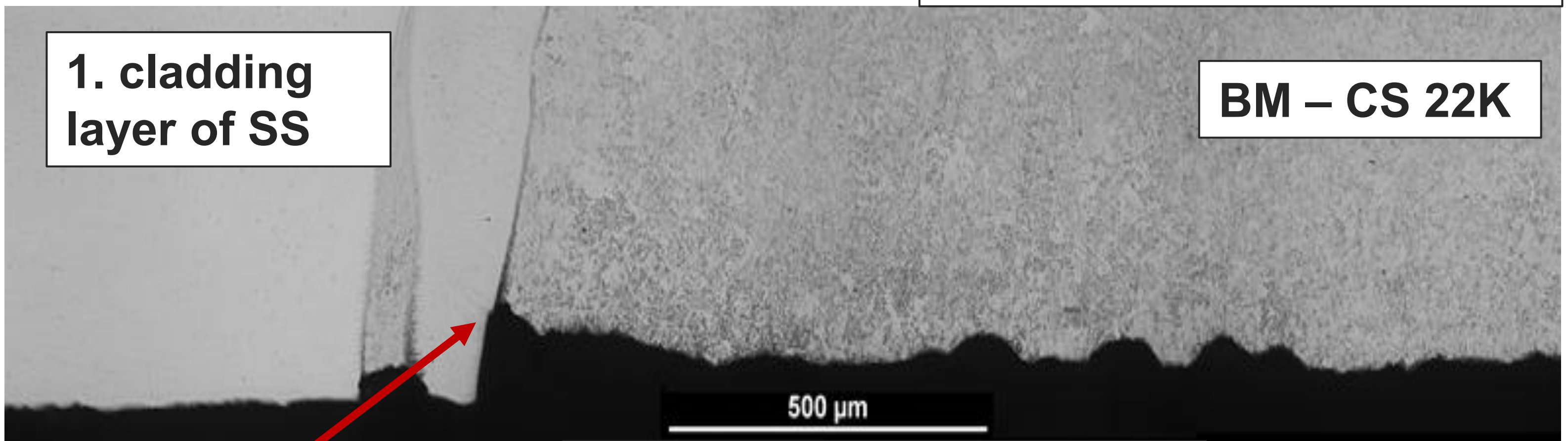
SCF



Anodic Dissolution of CS 22K

Inner Surface

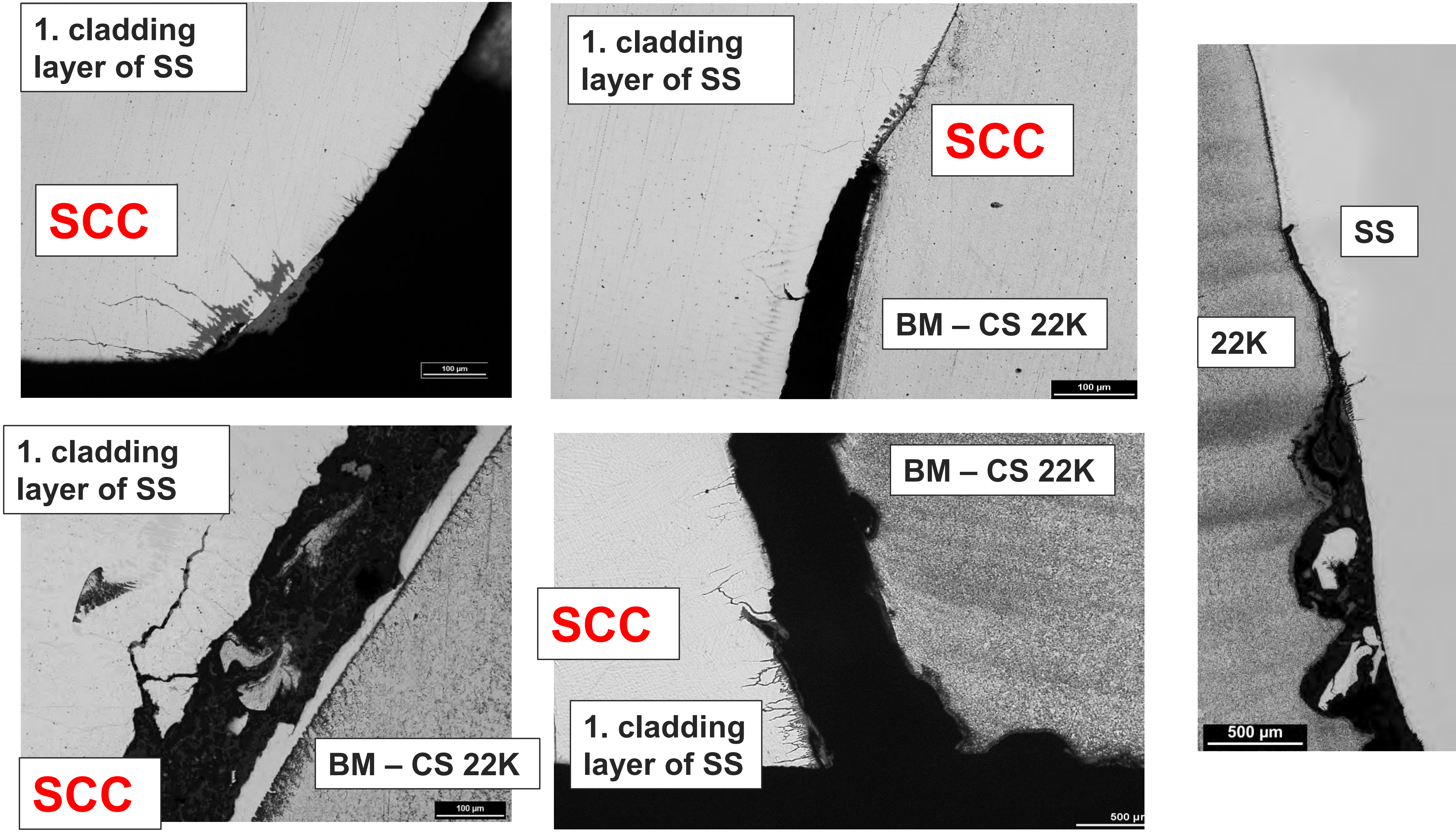




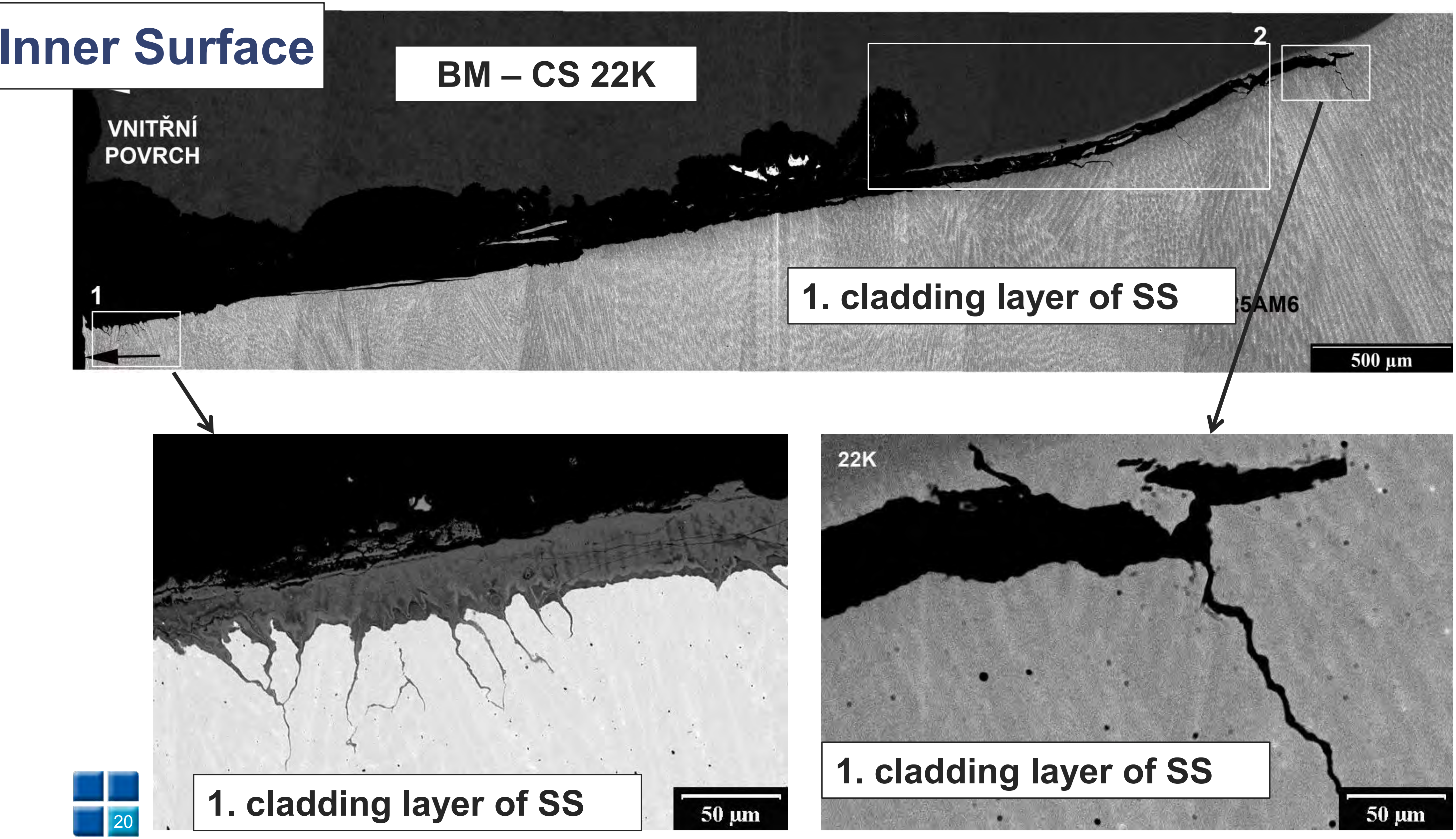
SCF

Inner Surface

Dominant Degradation Mechanism - SCC

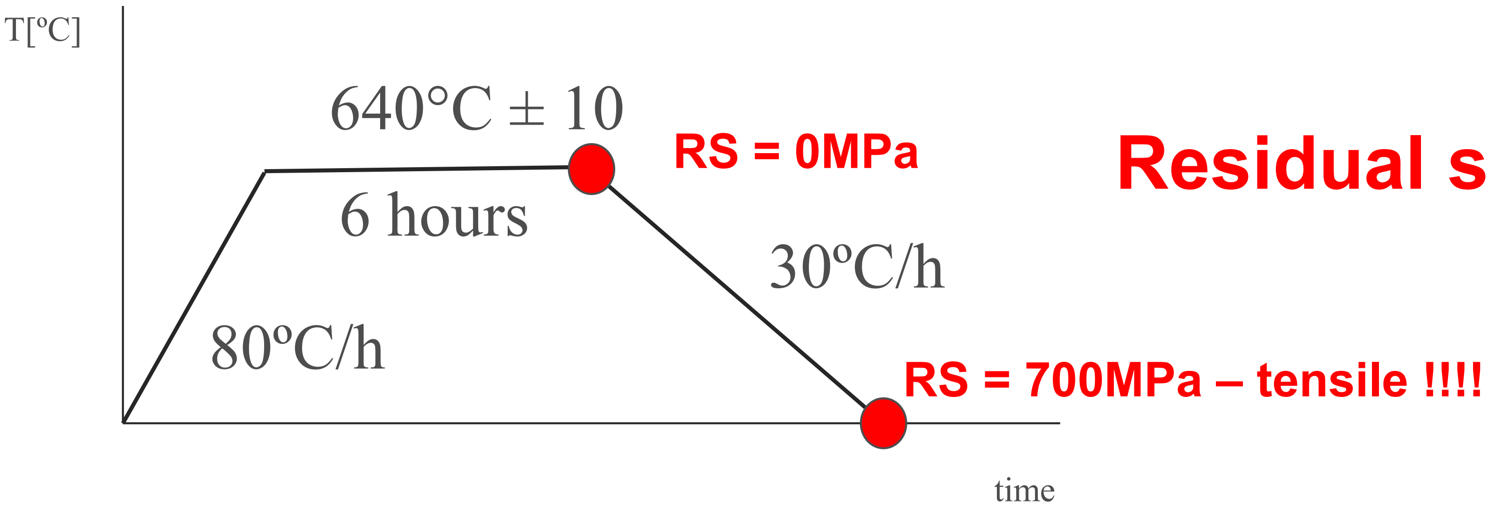


DM - SCC



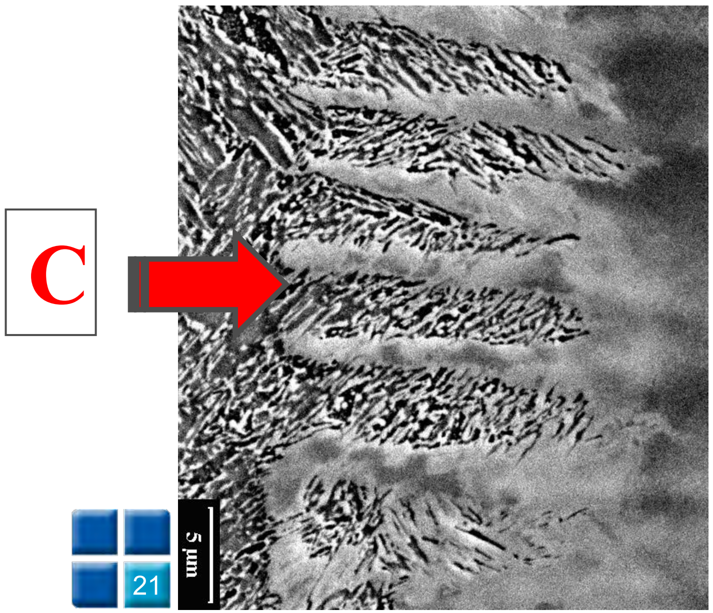
Inner Surface
BM – CS 22K
VNITŘNÍ
POVRCH
2
1
1. cladding layer of SS
500 µm
22K
1. cladding layer of SS
50 µm
1. cladding layer of SS
50 µm

SCC Main Factors (1) - design mistakes



Residual stresses

PWHT application on DMW during manufacturing



C diffuse into first layer – non-stabilize steel (depth approx. 50µm)

$M_{23}C_6$ on grain border

Steel with higher content of P

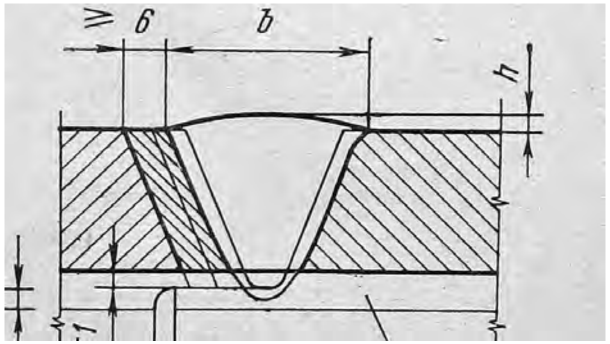
Increase sensitivity on SCC

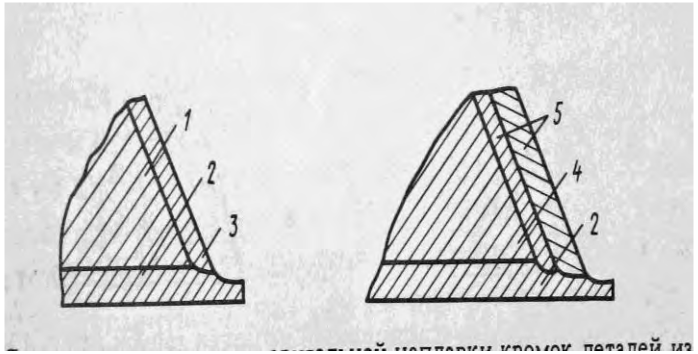
SCC Main Factors (2) – operation aspects

Strain loading due to thermal expansion differences (app. 33%)

Non-standard corrosion medium (under corrosion deposits) and medium contact with material sensitive on SCC (material of 1. cladding).

Stress concentration factors on the crack tip





Russian standard 1972	Russian standard 1989
VVER 440MW	VVER 1000MW

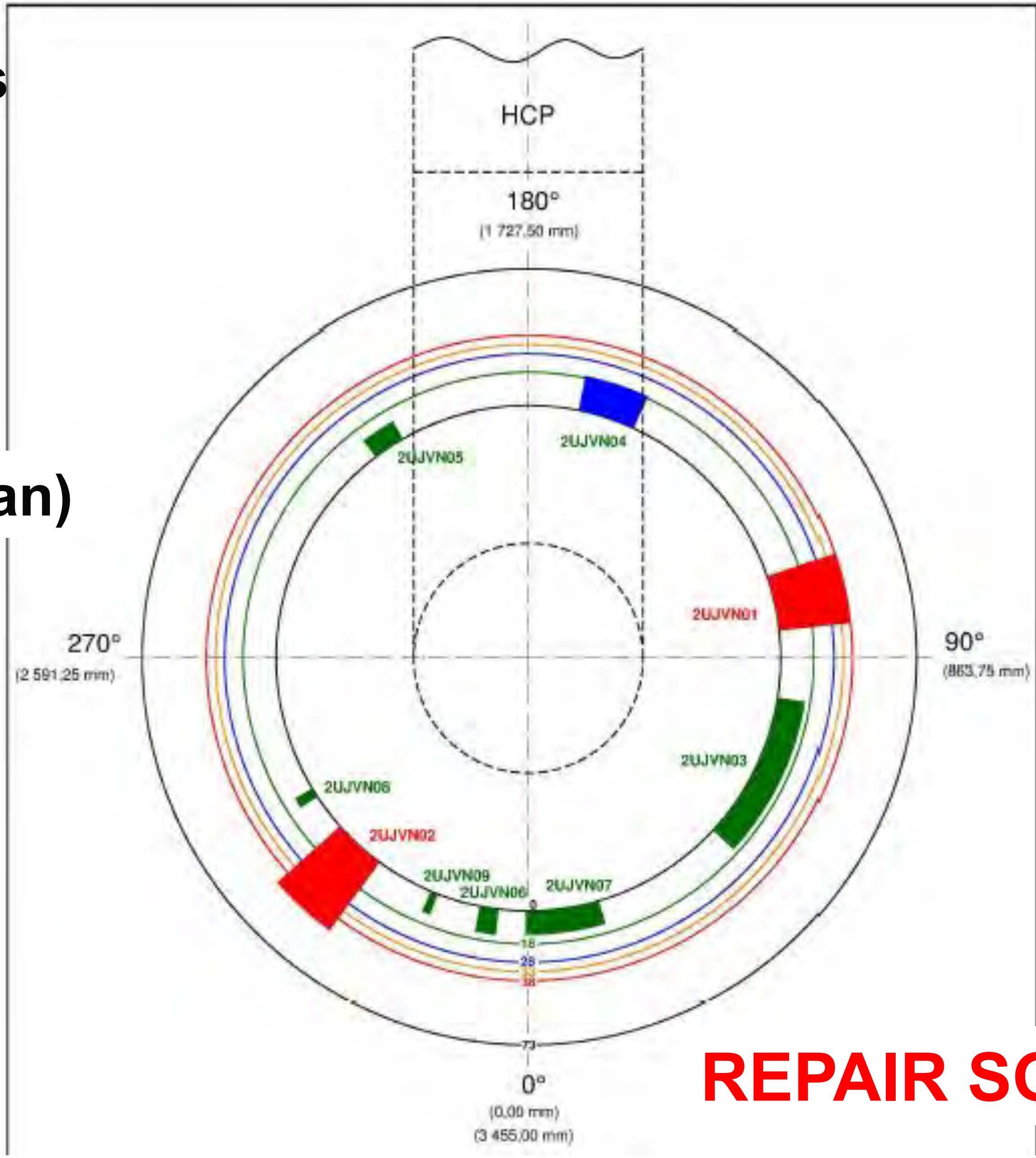
UT Qualification PE, PAUT (SG21)

Till 2012 – two indications

(red color)

Key for detection (action plan)

Next activities in this area



REPAIR SG21

Action plan for SG in NPP Dukovany

IAM BRNO

Corrosion deposits inside – new aspect for us. To clean all SGs by special technology (high pressure water)

NDT inspection all SG DMW during 2012 and 2013 (last SG in 08/2013)

To correct NDT inspection interval in accord with table (next slide)

To prepare plan of SG repair on the base UT results (one next SG now)

Improve UT inspection – we are not satisfy (training specimen, specimen for blind test)

Action Plan of SG Repair

IAM BRNO

depth dz [mm]	Action			
dz < 18	Inspection in accord with NDT plan			
dz ≥ 18	SG is included to plan of SG repair			
	NDT inspection is changed			
Interval NDT correction				
18 < dz ≤ 28	In accord with crack velocity interval is correct:			
	≤ 1,5 mm/y.	≤ 5 mm/y.	≤ 10 mm/y.	≤ 15 mm/y.
	3 year	2 year	1 year	Repair
28 < dz ≤ 33	In accord with crack velocity interval is correct on:			
	≤ 1,5 mm/y.	≤ 5 mm/y.	≤ 10 mm/y.	≤ 15 mm/y.
	2 year	1 year	repair	repair
33 < dz ≤ 38	Repair			
	Repair in next outage in any cases only (expert team)			
dz > 38	Repair			
dz = 42	Critical depth			

Conclusion

IAM BRNO

We are able to describe main DM – combination electrochemical corrosion with SCC

Corrosion deposits inside – key aspect for degradation (electrolyte for electrochemical corrosion)

The problem is general for all SG VVER 440MW (Slovak, Hungry, Finland, Ukraine, Russia)

UT inspection is key for crack detection - we would like to share experience and information in this field. **Discussion is welcome.**

Main Events 2012 in Belgian NPPs

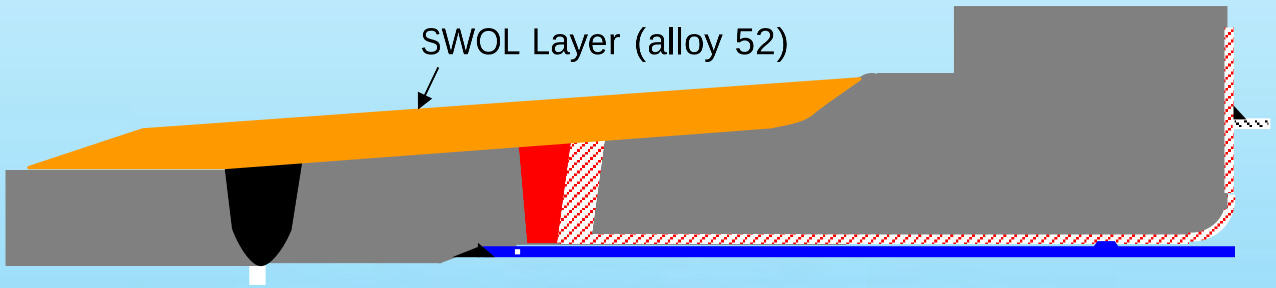
18th WGIAGE Metal Sub-group Meeting, Paris, 9-10/04/2013

OVERVIEW

- RPV Hydrogen Flakes (seperate presentation)
- Inconel welds
  - ✓ Pressurizer nozzles to safe-end welds
  - ✓ RPV head penetrations
  - ✓ BMI penetrations
- Baffle bolts
- Flow-Accelerated Corrosion
- Fire water loop replacement

INCONEL WELDS (1)

- Pressurizer surge nozzle to safe-end weld (14")
  - ✓ Full structural weld overlay (FSWOL) performed in Doel 3 in June 2012 and in Doel 4 in October 2012 by AREVA
  - ✓ Pre-emptive measure to reduce susceptibility to PWSCC of the weld in Alloy 82/182
  - ✓ Same operation performed in Tihange 3 in 2010 and in Tihange 2 in 2011



INCONEL WELDS (2)

- ✓ Dual wire process

Weld overlay in progress

Final aspect

INCONEL WELDS (3)

- Pressurizer nozzle-to-spray line weld (4”), nozzle-to-safety valve lines weld (6”) and nozzle-to-discharge line weld (6”)

  ✓ FSWOL selected for mitigation in Tihange 2 and Doel 3&4

  ✓ Qualification in 2012-2013

  ✓ Application

    - in 2014 in Tihange 2/Doel 3
    - In 2015 in Doel 4

INCONEL WELDS (4)

- RPV head penetrations

  ✓ Tihange 3 and Doel 4:
    - New heads ordered in 2010 and currently under fabrication
    - Replacement scheduled in 2015

  ✓ Doel 1:
    - BMV & UT inspection in November 2012: Growth of indications that had already been found in 2005 (PWSCC)
    - Safe operation until definitive shutdown (2015) justified by calculations

  ✓ Doel 2&3, Tihange 2:
    - BMV & UT inspections respectively in 2011, 2010 and 2009: Nothing to report

INCONEL WELDS (5)

- BMI penetrations
  - ✓ BMV inspections:
    - 2012: Doel 1, 2 & 4: No service-induced degradation reported
    - Next inspections:
      - ✓ Doel 3 and Tihange 3 in 2013
      - ✓ Tihange 1 & 2 in 2014
  - ✓ Impact of the Gravelines 1 event on the future inspection program has to be discussed with the Safety Authorities

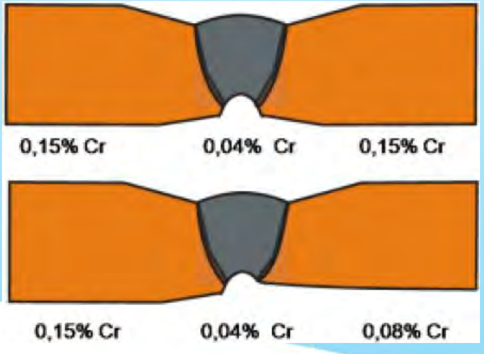
BAFFLE BOLTS

- Inspection in Doel 3 in June 2012
  - ✓ All 960 5/8" baffle-to-former bolts examined by UT
  - ✓ All bolts are sound
- Next inspections
  - ✓ Tihange 2 in 2014
  - ✓ Inspections foreseen in Doel 4 and Tihange 3 around 2015

FLOW-ACCELERATED CORROSION

- Checworks
  - ✓ Predictive calculations performed with Checworks for the 7 units on the main lines of the secondary loop (i.e. main steam, feedwater, extraction steam)
  - ✓ No significant FAC detected on the most susceptible components highlighted by Checworks up to now

- Time-Of-Flight Diffraction (TOFD) technique used since 2011
  - ✓ Purpose: Investigate a dozen of welds per outage to detect weld root corrosion
  - ✓ First use in Tihange 2: Identification of 4 welds where the criteria were exceeded
    - -> Preventive cut and replacement
    - Mechanical tests did not show evidence of FAC
  - ✓ No exceeded criteria in the other units since then



REPLACEMENT OF FIRE WATER LOOP

- Buried piping of the fire water loops in Tihange
  - ✓ Carbon steel and cast iron piping currently being replaced by piping in glass-fiber reinforced epoxy (GRE) in Tihange 2 and 3
  - ✓ Approximately 5000 m to be replaced (until 08/2013)
  - ✓ 12", non ASME
  - ✓ Linear parts attached by Key-Lock male and female mechanical joints
  - ✓ Elbows attached by male and female adhesive-bonded joints

Electrabel
GDF SUEZ

REPLACEMENT OF FIRE WATER LOOP

- Inside piping of the fire water loops in Tihange
  - ✓ Reactor building
    - ASME part, piping replaced by steel
  - ✓ Other buildings
    - Non ASME part, replacement by piping in GRE
    - 12km to be replaced in Tihange 1, 2 & 3
    - Linear parts and elbows attached by Key-Lock mechanical joints
    - Work planned until 2016

Electrabel GDF SUEZ

TRACTEBEL Engineering GDF SUEZ

Thank you for your attention

Some SCC Events in the US

Robert Tregoning
David Alley
U.S. Nuclear Regulatory Commission

CSNI WGIAGE Meeting
8 – 12 April 2013

Agenda

- SCC in Refueling Water Storage Tanks
- SCC in Control Rod Drive Mechanism (CRDM) Housings
- SCC in Seal Cap Enclosures

Refueling Water Storage Tanks

- 3 recent instances of cracking
- At least 1 SCC

Refueling Water Storage Tanks

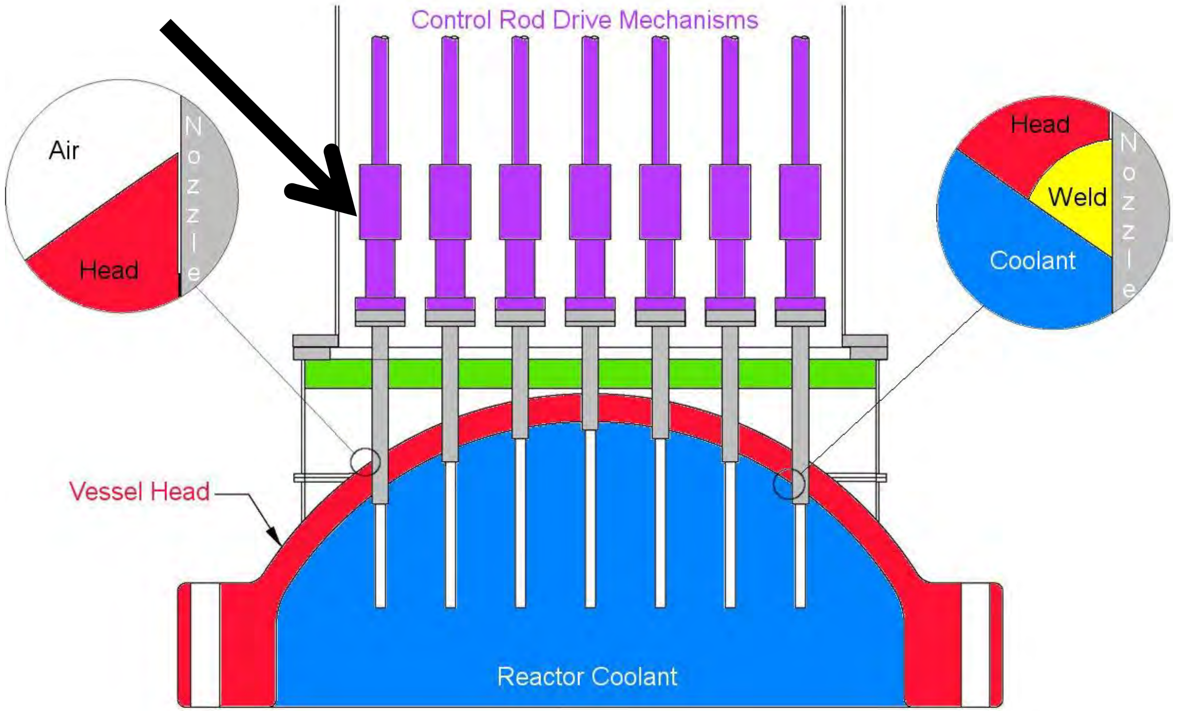
- Chloride SCC
  - Near tank bottom
  - OD originated
    - Consider
      - Ground Water
      - Deicing salt
  - Temp < traditional Chloride SCC limit

Refueling Water Storage Tanks

- Resolution
  - Tanks repaired
  - Information Notice being prepared

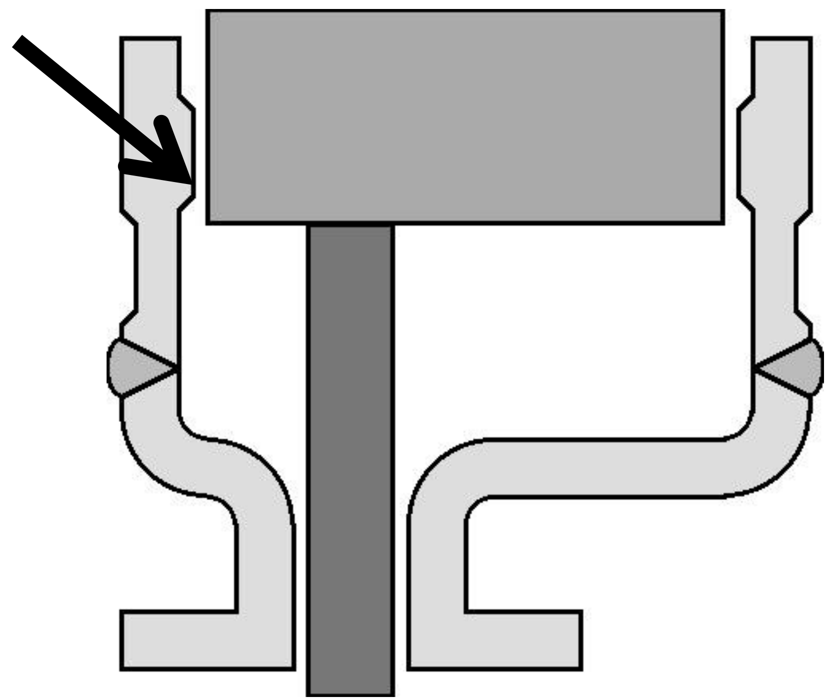
CRDM Housing

Leak well above reactor head



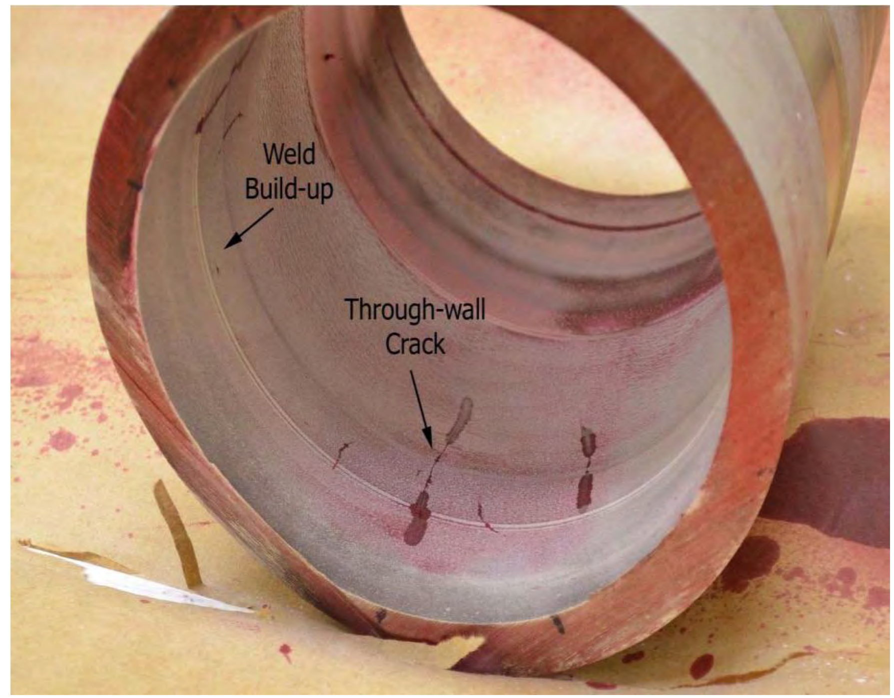
CRDM Housing

- Rack and Pinion CRDM Mechanism
- Only two plants built this way
- 316 stainless
- Crack at internal (non pressure boundary) weld



CRDM Housing

- 9 Cracks
  - 1 through wall
  - All axial
  - In two groups
  - 1 rub mark



CRDM Housing

- Resolution
  - Housing Replaced
  - 8 others inspected
    - No defects
  - NRC reviewing Plant root cause evaluation
  - Additional inspections planned in 2013 and in the future

Seal Cap Enclosure Cracking

Rob Tregoning
David Alley
John Tsao
U. S. Nuclear Regulatory Commission

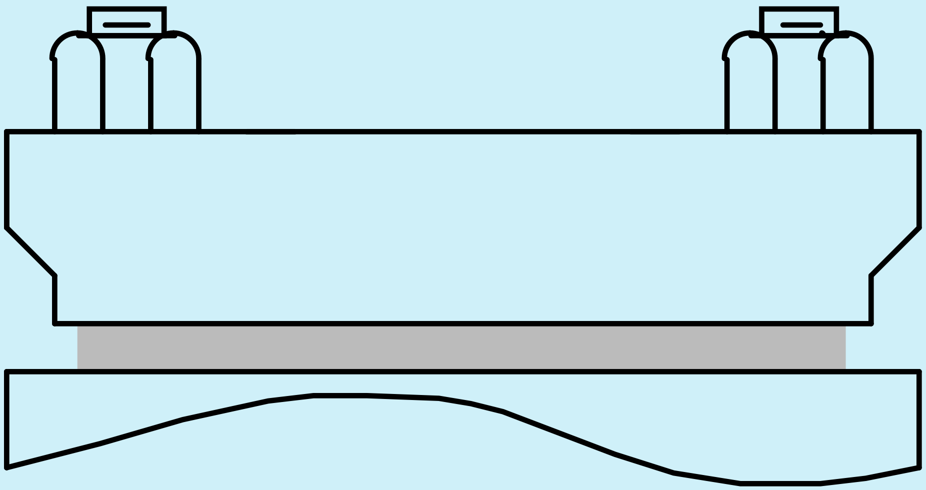
CSNI WGIAGE Meeting
8 – 12 April 2013

Agenda

- Introduction
- Problem
- Operating Experience
- Laboratory studies
- Resolution

Introduction

- Component under consideration
  - Reactor coolant system check valves
    - 304, 316 body and bonnet
    - A286 bolts
    - Metallic/graphite gasket



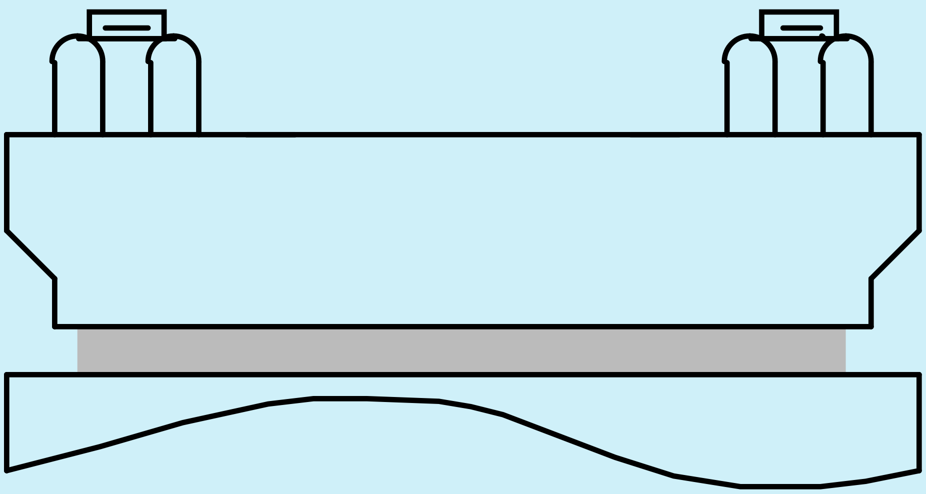
Introduction

- A286 (UNS S66286)
  - Precipitation hardened austenitic stainless
  - SA453, Grade 660, or SA638, Grade 660
  - Corrosion resistance similar to 300 series stainless

	Cr	Ni	Mo	Co	Va	Ti	C	Fe	Mn	Si
MIN	13.5	24.0	1.0	-	0.1	1.9	-	-	-	-
MAX	16.0	27.0	1.5	1.0	0.5	2.35	0.08	Bal	2.0	1.0

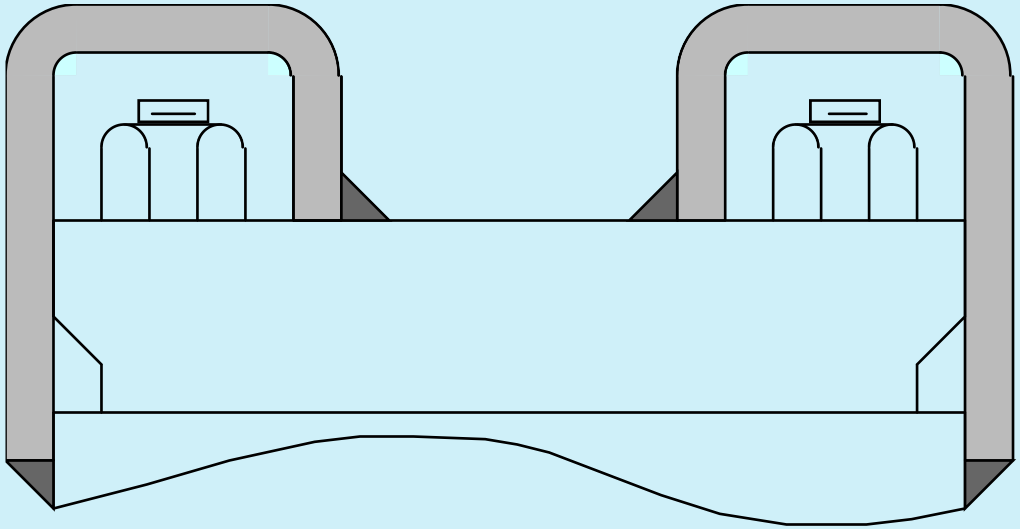
Introduction

- Problem
  - Leakage through body/bonnet gasket
  - Valves located so that repair requires plant shutdown and substantial draining of system
    - Not a trivial exercise



Introduction

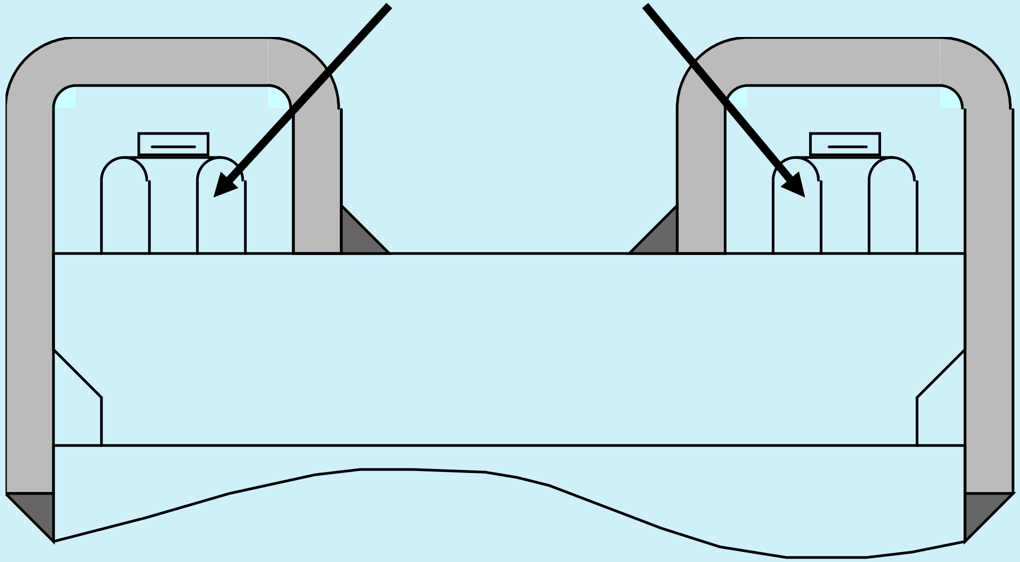
- Initial solution
  - Install seal cap enclosure
    - Leak barrier only
    - Not designed to be pressure boundary



Introduction

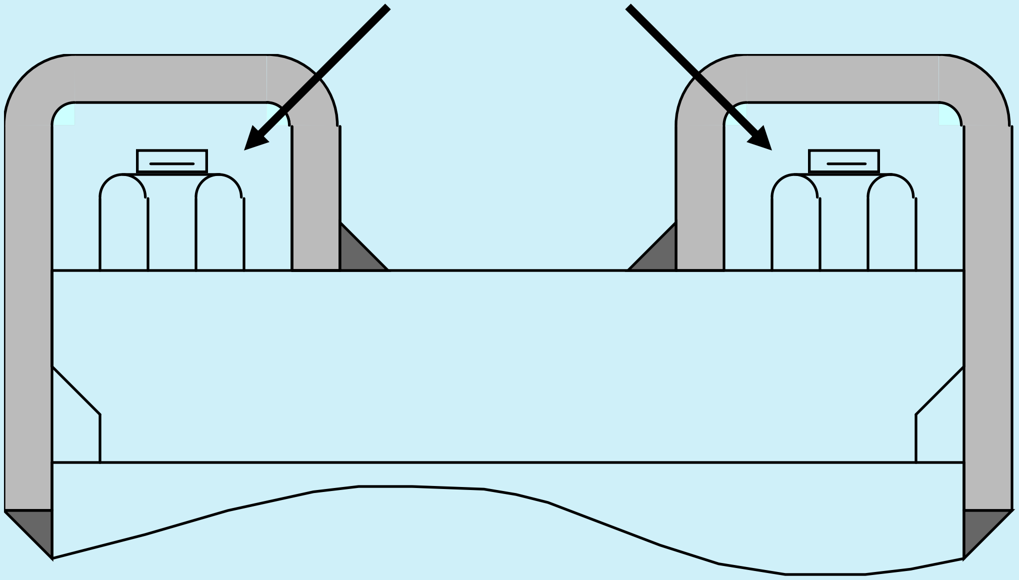
Problem

- ASME code requires inspection of class 1 pressure boundary
  - Bolts secure pressure boundary
  - Bolts are hidden



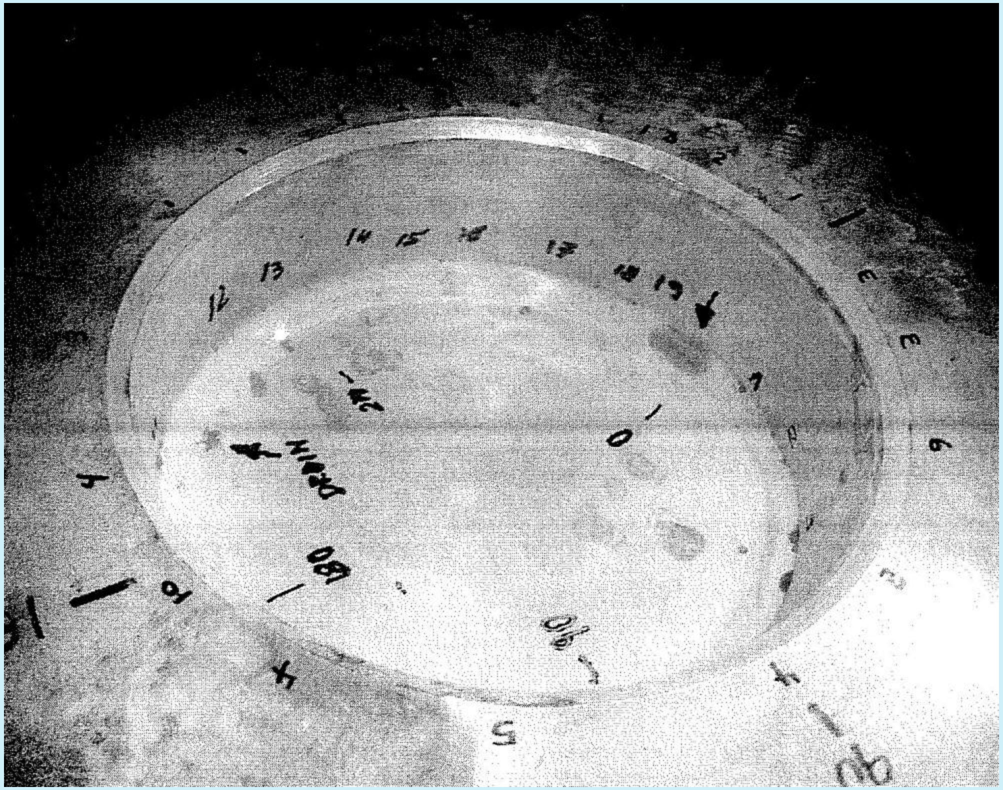
Problem

- A-286 has cracked
  - High stress
  - Hot oxygenated water
    - Seal cap design ensures oxygenation of water
    - Oxygen in enclosure higher than any level tested



Problem

- Cracking of 300 series stainless steels and welds



Operating Experience

- Seal Cap Enclosures and/or cracking of A-286 bolting
  - Five Events
  - NRC Information Notices IN 90-68, IN 90-68 Supplement 1, and IN 2012-15

Operating Experience

- Reactor internals bolting
  - Core barrel and lower thermal shield bolts
    - 4 Plants
    - 1981-1984

Operating Experience

- Pump turning vane bolts
  - 5 of 23 bolts cracked

Operating Experience

- More turning vane bolts
  - 3 of 4 cap screws cracked
  - 1 parted completely
    - Identified as a loose part

Operating Experience

- Seal Cap Enclosure #1 (Plant A)
  - 1985 – 1987 enclosures installed
  - 1992 redesigned enclosures installed
    - No evidence of degradation of bolting or old enclosures
  - 2002 leakage from one enclosure weld
    - Repaired

Operating Experience

- Seal Cap Enclosure #1 (Plant A) (cont.)
  - 2004 leakage from another enclosure
    - Enclosure opened, bolts examined
      - 3 of 12 failed
        - » IGSCC
    - All enclosures removed/valves repaired

Operating Experience

- Seal Cap Enclosure #2 (Plant B)
  – Seal Cap Enclosures installed on several valves
  – 1999 and 2010 steam and/or boric acid observed near valves – no leaks detected

Operating Experience

- Seal Cap Enclosure #2 (Plant B) (cont.)
  - Boric acid observed
  - Seal weld repaired

Operating Experience

- Seal Cap Enclosure #2 (Plant B) (cont.)
  - 2012 more boric acid
    - Enclosure top removed
    - Bolts inspected
      - UT/Torque
      - No degradation
    - Enclosure reinstalled

Operating Experience

- Seal Cap Enclosure #2 (Plant B) (cont.)
  - 2012 - Began repairing / replacing valves
    - Tested bolting prior to removal
      - No cracking identified

Laboratory Studies

- Babcock and Wilcox, Brookhaven National Lab
  - Stresses ↑ - Cracking ↑
    - May be a threshold stress
  - Oxygen ↑ - Cracking ↑
    - Tested up to 1200 ppm oxygen
    - Oxygen in Seal Cap Enclosures exceeds levels used in testing substantially

Laboratory Studies

- Conclusions regarding A-286 material
  - B&W
    - Reduce stress
    - Use less susceptible material
  - Brookhaven
    - Do not use due to cracking susceptibility

Resolution

- Issue requires resolution
  - Cracking observed in lab and field
  - Cracking could result in pressure boundary failure
  - Environment will be high oxygen
    - Air trapped in enclosure
    - Higher oxygen than any tests
  - Stress threshold may not exist

Resolution

- Nuclear power plants are currently in the process of addressing the issue of seal cap enclosures through inspections and/or removal.

Nuclear Materials Research at JRC-IET

www.jrc.ec.europa.eu

Oliver Martin

(oliver.martin@jrc.nl)

Joint Research Centre

Content

1. Joint Research Centre (JRC) – Institute for Energy and Transport (IET)
2. Activities of "Nuclear Reactor Safety Assessment (NRSA) Unit"
3. Knowledge Management (Capture)
4. Nuclear Materials Research (MATTINO)

Content

1. **Joint Research Centre (JRC) – Institute for Energy and Transport (IET)**
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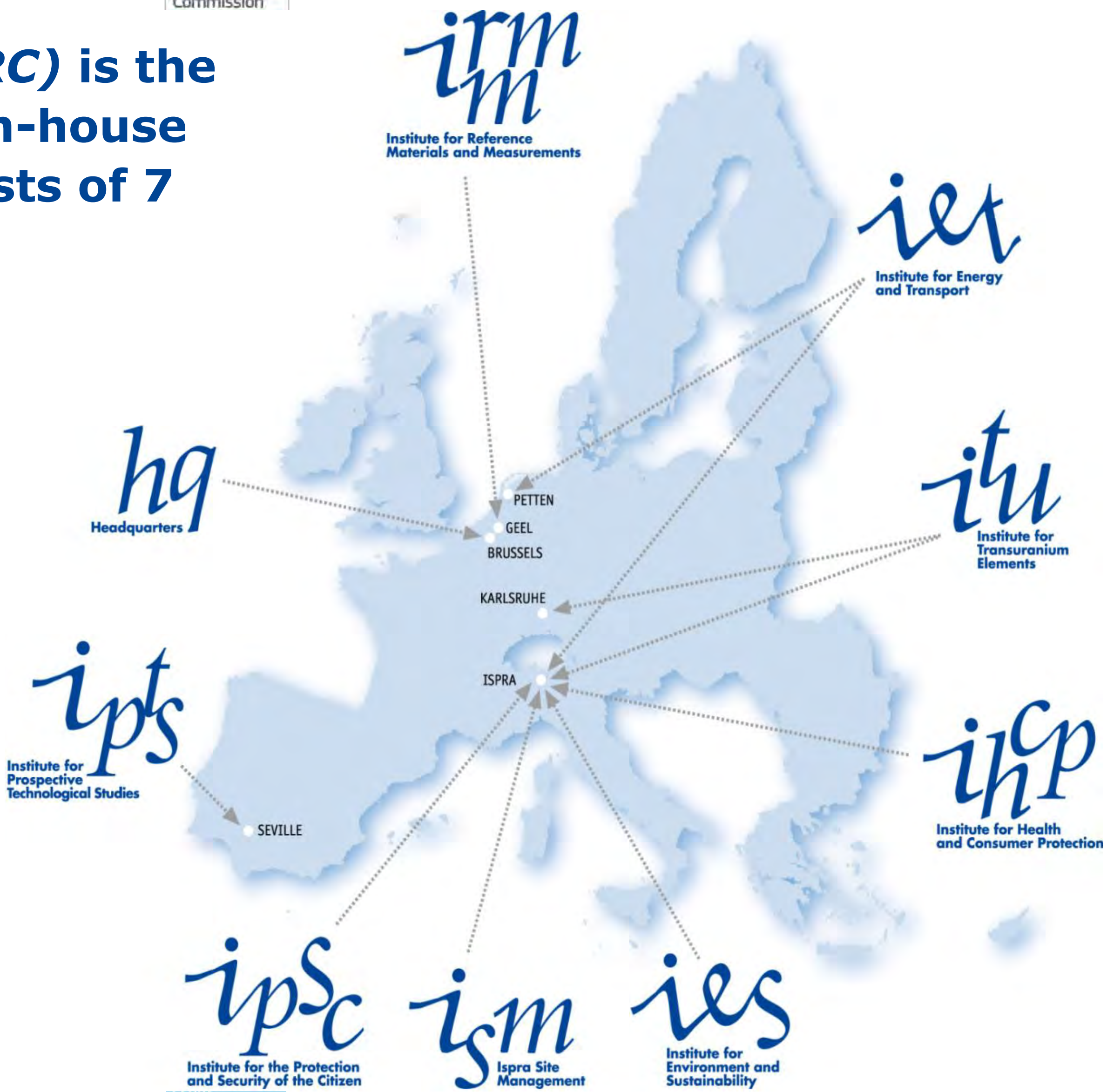
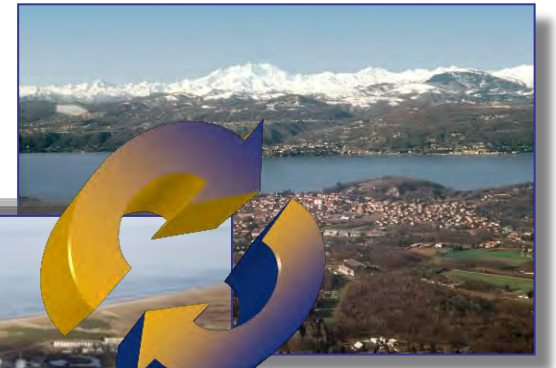
Joint Research Centre (JRC) **is the European Commission's in-house science service and consists of 7 research institutes.**

One of them is the *Institute for Energy and Transport (IET)* located in Petten and Ispra.



Institute for Energy and Transport

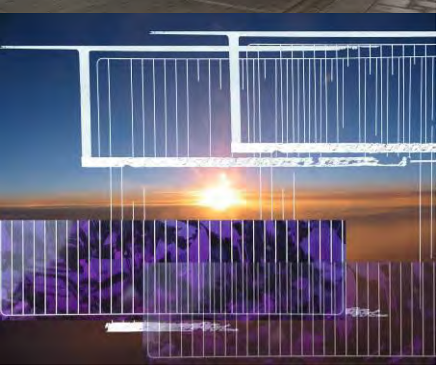


Ispra, Italy

Petten, The Netherlands

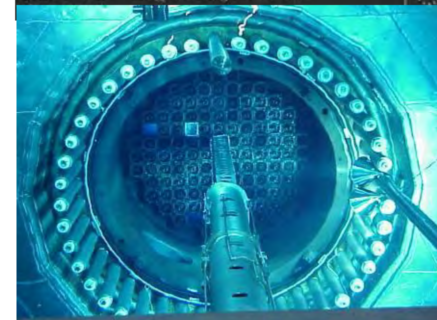
The mission of the Joint Research Centre – Institute for Energy and Transport (IET) is to provide support to Community policies and technology innovation related both:

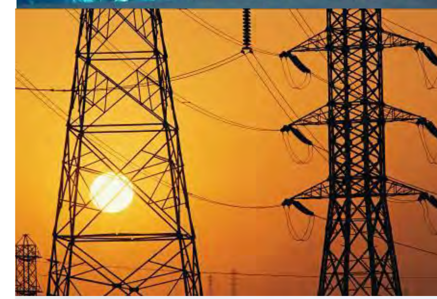
- energy - to ensure sustainable, safe, secure and efficient energy production, distribution and use and
- transport - to foster sustainable and efficient mobility in Europe.



JRC-IET Key Scientific Activities

- Renewable energy
- Sustainable & safe nuclear energy
- Security of energy supply
- Energy techno-economic assessment
- Bioenergy including biofuels
- Hydrogen and fuel cells
- Clean fossil fuel
- Sustainable transport
- Energy efficiency





Nuclear Units of JRC-IET

- Nuclear Reactor Safety Assessment Unit (F5)
  - The EU Clearinghouse on operating Experience Feedback (NUSAC)
  - Nuclear Reactor Accident Analyses and Modeling Group (NURAM)
  - Technical support to the Instrument for Nuclear Safety Cooperation (SINSAC)
- Nuclear Reactor Integrity Assessment & Knowledge Management (F4)
  - Knowledge Management (CAPTURE)
  - Nuclear Materials Research (MATTINO)

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What is the EU Clearinghouse on OE for NPPs?

A <u>centralised</u> EU initiative at the service of EU MS nuclear Safety Authorities, to improve the use of Operational Experience (OE) from Nuclear Power Plants (NPP)

- Created in 2008 with 7 participating EU MS

- Today all EU MS having NPPs are participating, together with international organisations (OECD, IAEA) and part of the nuclear industry.

EU Clearinghouse (NUSAC)

EU Safety Authorities

Members
Finland
Hungary
Lithuania
The Netherlands
Romania
Slovenia
Switzerland

Observers
Belgium
Bulgaria
Czech Republic
France
Germany
Slovakia
Spain
Sweden
United Kingdom

OEF Clearinghouse Centralized Office (JRC-IET)

Coordination & Communication
Technical & Scientific work
OE knowledge repository

Clearinghouse deliverables

International Community

IAEA

OECD-NEA

EU Technical Support Organizations

IRSN, GRS
Bel-V

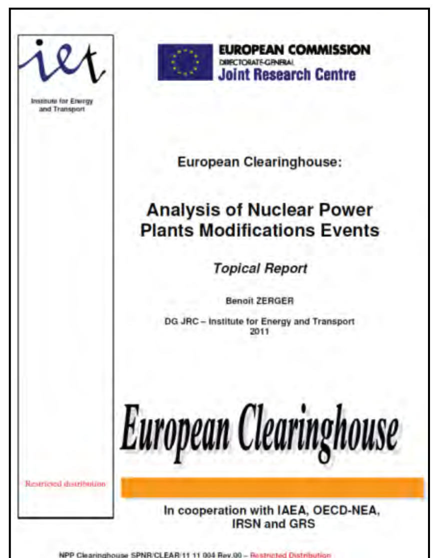
+ Areva

EU Clearinghouse (NUSAC)

European Commission

Main deliverables from the EU Clearinghouse :

- "Topical studies" about specific types of events: maintenance, nuclear fuel, construction…
- Quarterly reports about events occurring worldwide
- Organization of workshops and training about OEF
- International cooperation with IAEA and OECD-NEA
- …



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Institute for Energy and Transport

EUROPEAN COMMISSION
DIRECTORATE-GENERAL
Joint Research Centre

European Clearinghouse:

Analysis of Nuclear Power Plants Modifications Events

Topical Report

Benoit ZERGER

DG JRC – Institute for Energy and Transport
2011

European Clearinghouse

Restricted distribution

In cooperation with IAEA, OECD-NEA, IRSN and GRS

NPP Clearinghouse SPNR/CLEAR/11 11 004 Rev.00 – Restricted Distribution

+ specific actions :

- Fukushima updates
- DOEL 3

Joint Research Centre

European Commission

New accident analyses group at JRC-IET: Nuclear Reactor Accident Analyses and Modeling (**NURAM**)

- To develop within JRC strong capabilities in Severe Accident Analyses for Gen II&III and IV reactors

- To propose in collaboration with partners (EU TSO, international organisations, networks...)* possible ways to improve Severe Accident Management on EU NPPs

IRSN
INSTITUT DE RADIOPROTECTION ET DE SÛRETÉ NUCLÉAIRE

GRS
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH

IAEA

AEN NEA
OECD

Severe Accident Research NETwork of excellence

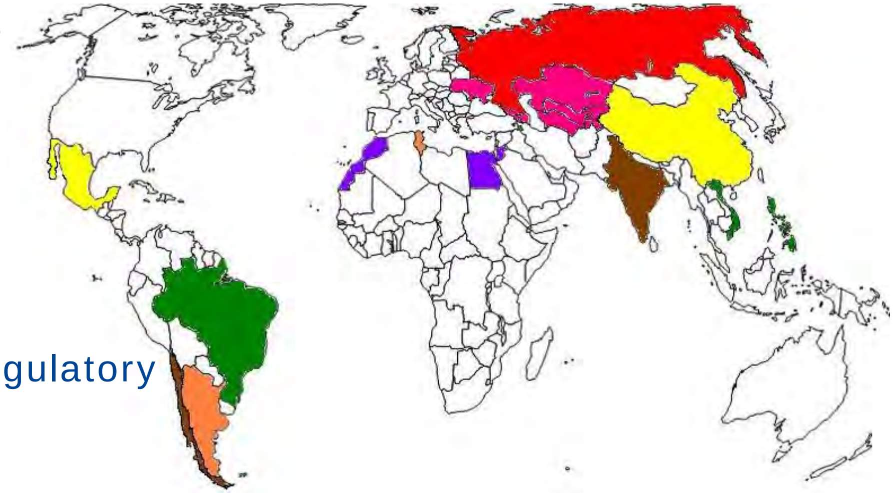
SARnet

NUGENIA
NUclear GENeration II & III Association

European Commission

Improving Nuclear Safety outside the EU

Continued technical support to the implementation of the Instrument for Nuclear Safety Cooperation (INSC) (few example below)



Support to Regulatory bodies:

- **Brazil:** Nuclear Safety Cooperation with the Regulatory Authorities of Brazil (CNEN)

Support to the Nuclear Operators

- **Ukraine:** U1.05/08 T1T2 SAMGs for VVER 440/213 and VVER 1000 in Ukraine
- **Brazil :** BR1.01/09 T1 : SAMGs for Angra 2 NPP

Starting in 2012:

- Assistance to the Armenian operator and regulator for the implementation of the Nuclear Stress Tests in accordance with the European methodology

Content

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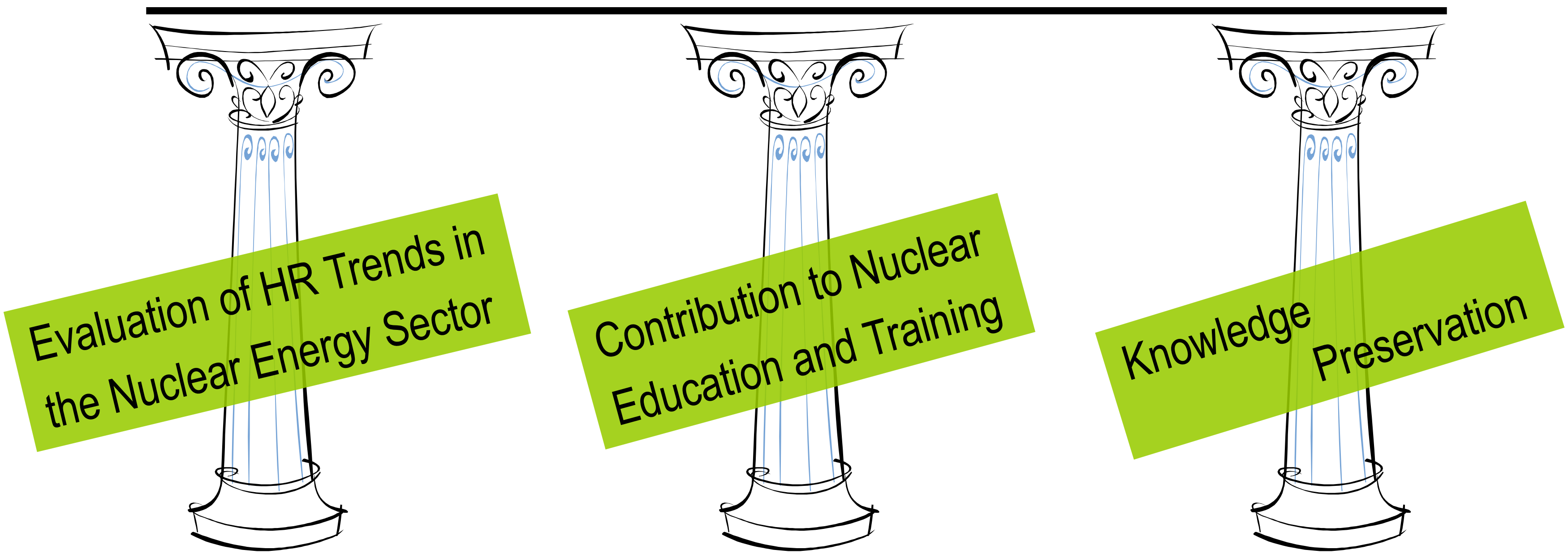
3. **Knowledge Management (Capture)**
4. Nuclear Materials Research (MATTINO)

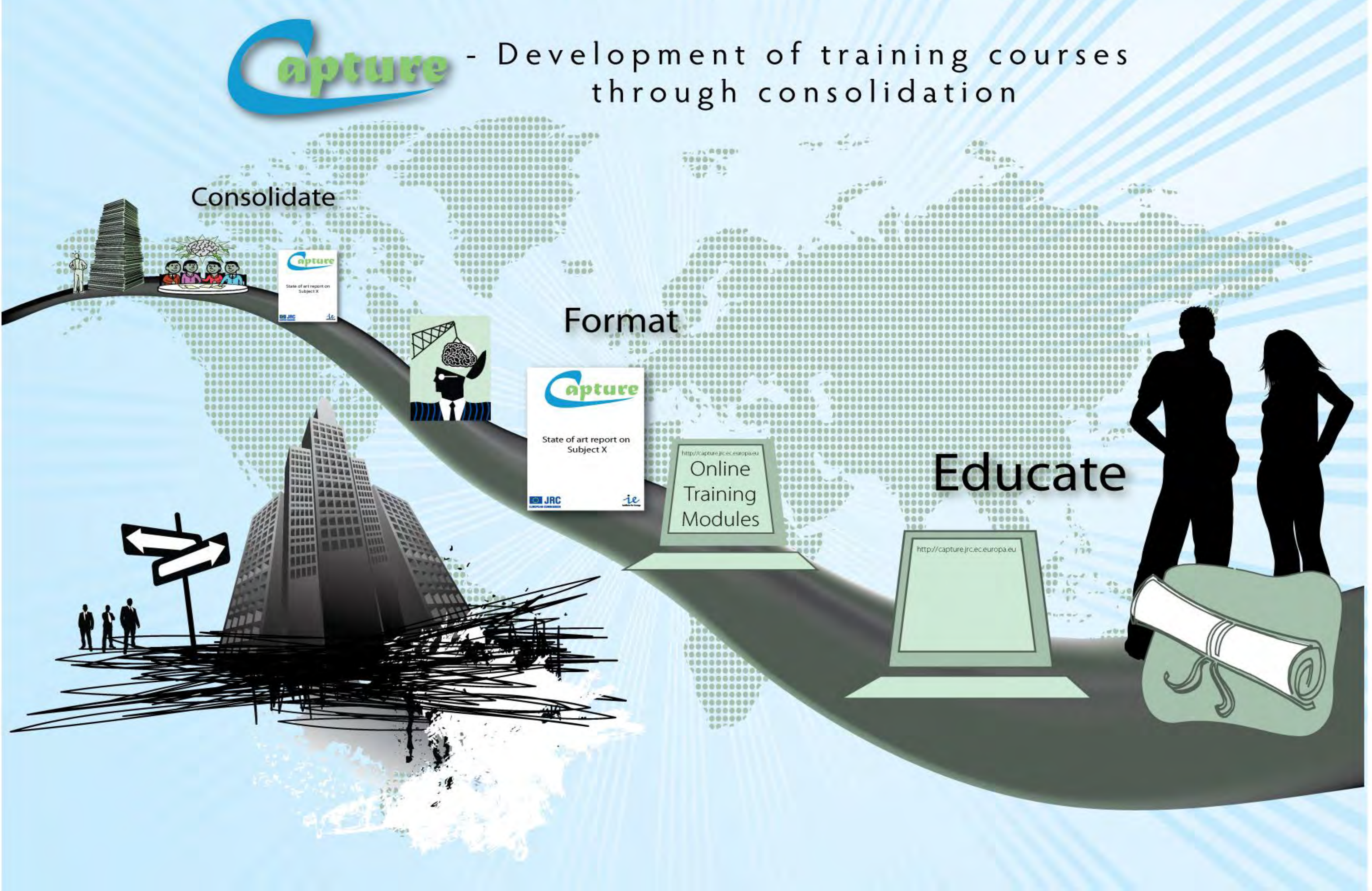
European Commission

Action on **Nuclear Knowledge Management, Education & Training**



Joint Research Centre

CAPTURE ETKM



Multimedia


WWER Reactor Pressure Vessel Embrittlement
Multimedia Trainig Course
IAEA
International Atomic Energy Agency
JRC
EUROPEAN COMMISSION
Course Modules
Start-of-Life Toughness
RPV Design Features
Irradiation Shift Prediction
Property-Property Correlation
Annealing and Re-irradiation
Material Factors
Environmental Factors
Mechanisms and Microstructural Evolution
Surveillance
Cladding
There is a huge amount of information and knowledge in WWER Reactor Pressure Vessel (RPV) embrittlement available, either published or easily available, but also publications difficult to trace. Especially those were at risk of being dispersed or lost due to a series of factors, including:
Retirement
Generational gap
Non electronic publishing in the past
Limited dissimenation possibilities
Language (many non-English publications from Eastern Europe countries)
Copyright © 2011 JRC & IAEA

Content

1. Joint Research Centre (JRC) – Institute for Energy and Transport (IET)
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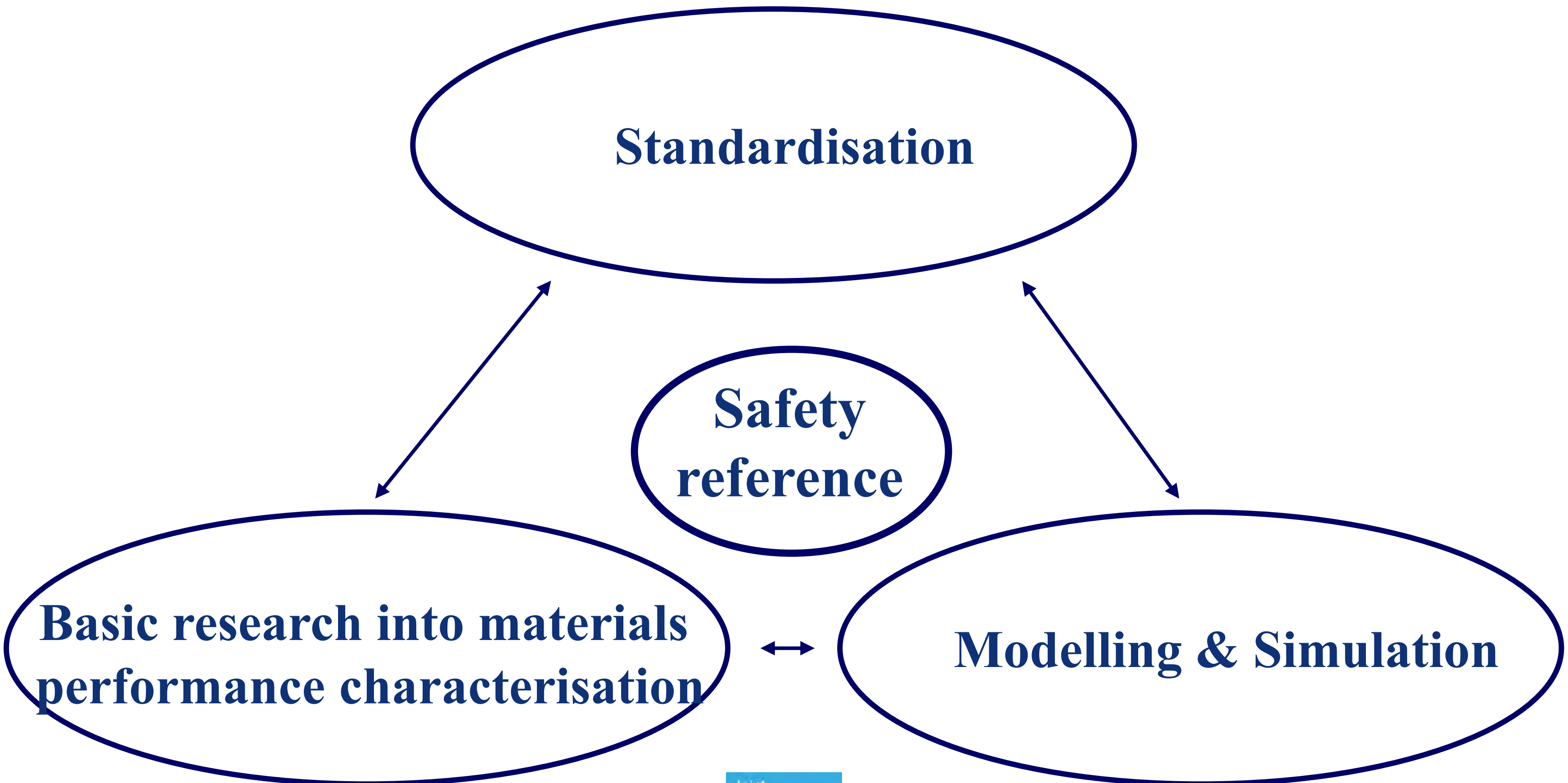
3. Knowledge Management (Capture)
4. **Nuclear Materials Research (MATTINO)**

Our Main mission is Materials and components related safety issues for present and future reactors



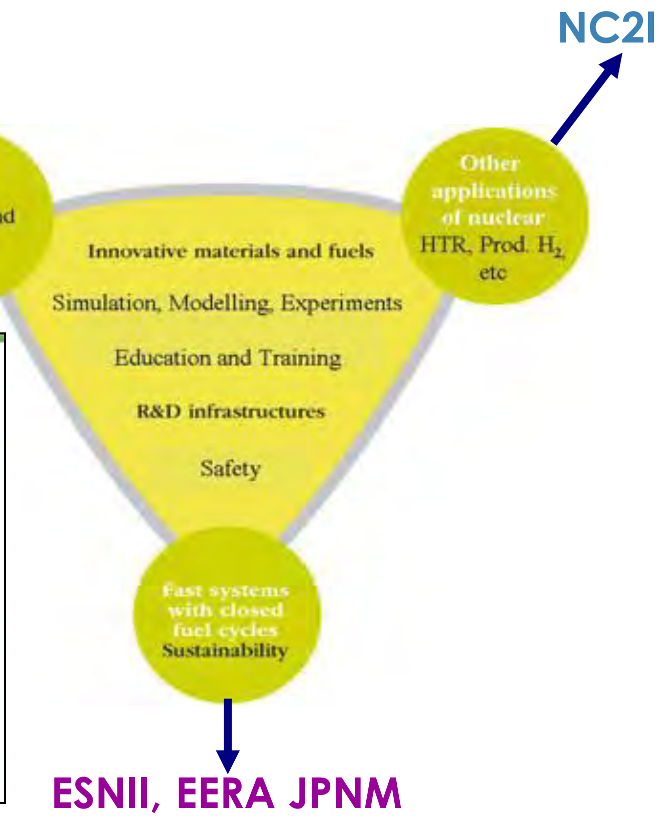
Int'l collaboration

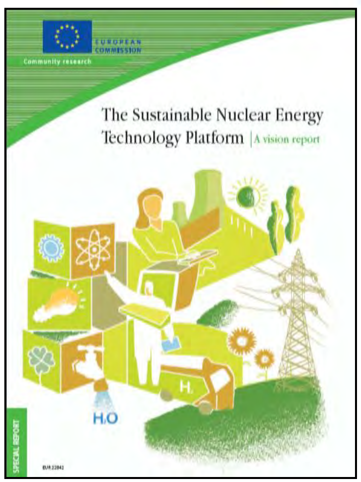
- Intergovernmental org's: IAEA, OECD/NEA
- GenIV International Forum (GIF)
- US DoE: I-NERI
- Standardisation bodies: ISO, CEN, ...
- Industries & utilities: EdF, Snecma...
- Universities: Prague, Delft, Pisa...
- TSOs: IRSN, UJV, VTT...

European Commission

SNETP Sustainable Nuclear Energy Technology Platform

Close alignment with the 3 SNETP pillars

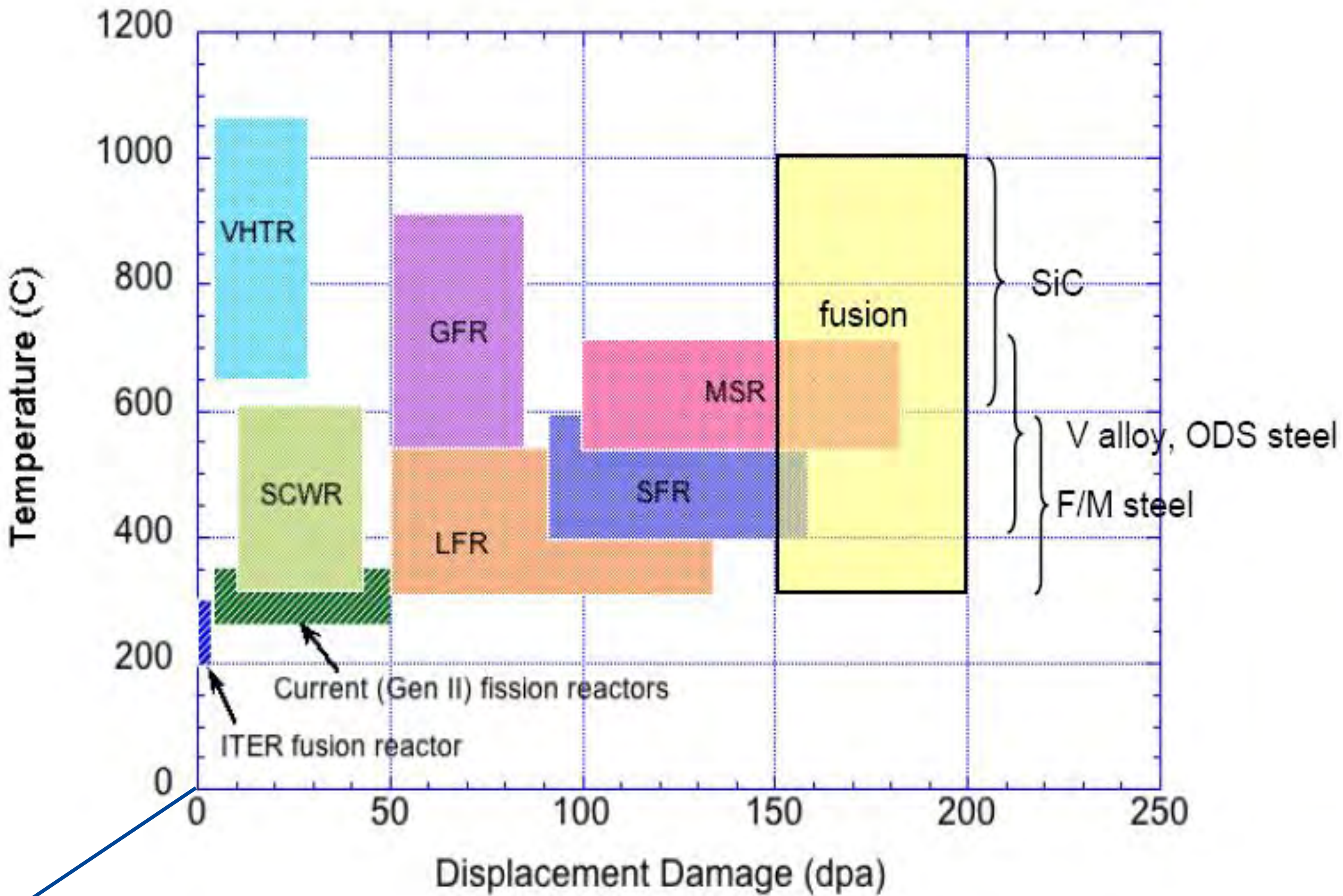




- **NUGENIA:** Maintain safety and competitiveness of today's technologies
- **ESNII/EERA JPNM:** Develop Gen IV Fast Reactors with closed fuel cycle to enhance sustainability and to minimize waste
- **NC2I:** Enlarge nuclear fission portfolio beyond electricity production (HTR, process heat)

- **SNETP defines strategies and R&D priorities (JRC contributed to SRA)**
- **The pillars will be the basis for future EURATOM funding in H2020**

Material Challenges



Corrosion

Examples of Scientific Activities

Experimental Activities

- Stress corrosion cracking
- Small Punch test
- High-cycle Fatigue
- HFR: Fuel qualification

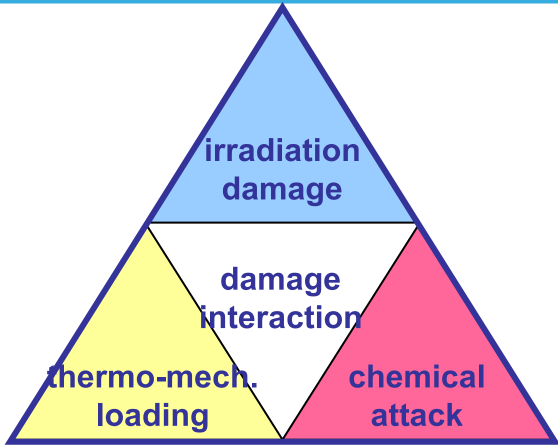
Experimental and modelling

- Thermal Fatigue of pipes
- Residual stresses in welds

Model development

- Multi-scale and physics based models
- Simulation of fuel cladding in accident scenario
- K/J value estimation and crack propagation in DMWs

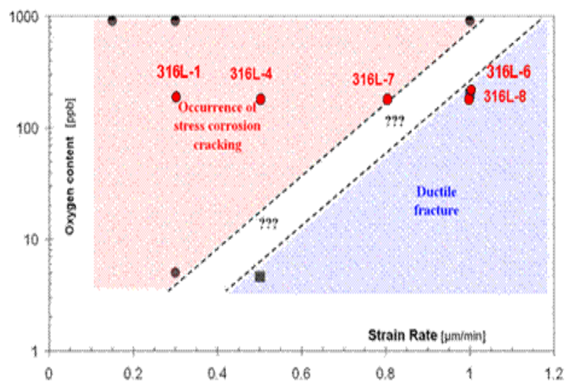
Stress Corrosion Cracking



Why?

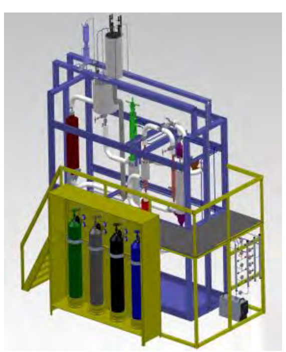
Stress corrosion cracking is a major failure mechanism in power plants

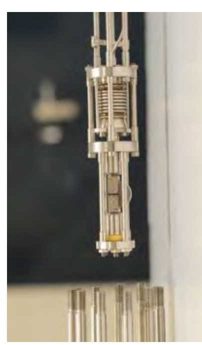
$\Rightarrow$ need for environmental testing incl. in-pile experiments, collaboration with ITU, NRG, CV Rez



3 recirculation loops with full water chemistry control, equipped for environmental mechanical tests at p_{max} = 360 bar, T_{max} = 650° C:

=> BWR, PWR, SCWR conditions





Under construction:

Liquid lead recirculation loop for

- Corrosion
- Erosion
- Stress corrosion cracking

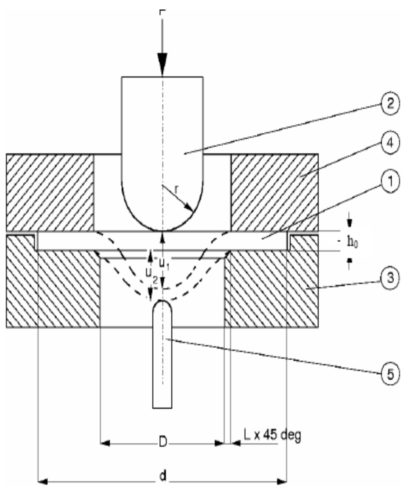
At T_{max} = 700° C, v_{max} = 5 m/s

Small Punch Test

Why?

Need for fast "semi non-destructive " test method for small specimen

- Possibility to obtain creep resistance data from small amounts of material
- Characterization of material response to multi-axial loading
- Characterization of anisotropy in mechanical properties



Key
1. SP disc test-piece
2. Hemispherical ended punch
3. Lower die
4. Upper die
5. Dilatometer push rod

Main principle of small punch test

SP test specimen, 8mm diameter, 0.5 mm thickness

SP creep tests were carried out in accord with the CWA 15627 Code of Practice, at 650° C in an Ar atmosphere.

Environmental High Cycle Fatigue

Why?

Accelerated tests to simulate service life

Characteristics

Ultrasonic excitation at 20 kHz

Working gas: Ar, H_2

Pressure < 350 bar

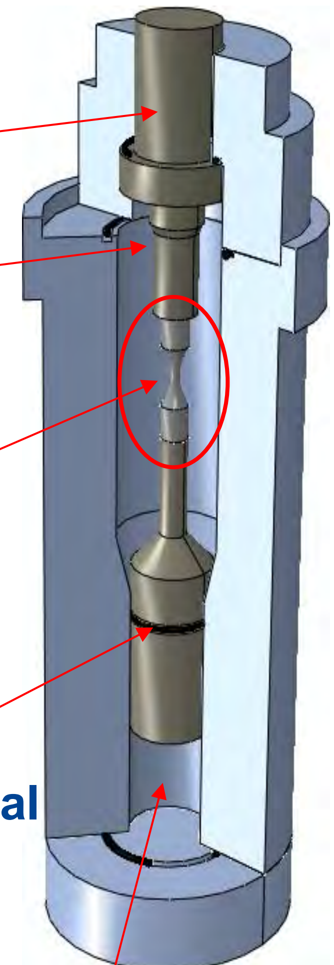
Static stress < 1000 MPa

Dynamic stress < 1000 MPa

Strain ratio R < 0.6

horn for coupling of ultrasonic waves

upper chamber



2nd horn: pressure seal

bottom chamber

Control Parameters

Constant stress σ_m (pressure diff.)

Stress amplitude σ_a (piezoelectric converter)

Joint Patent Application with Industry (Snecma)

Irradiation Testing

Why?

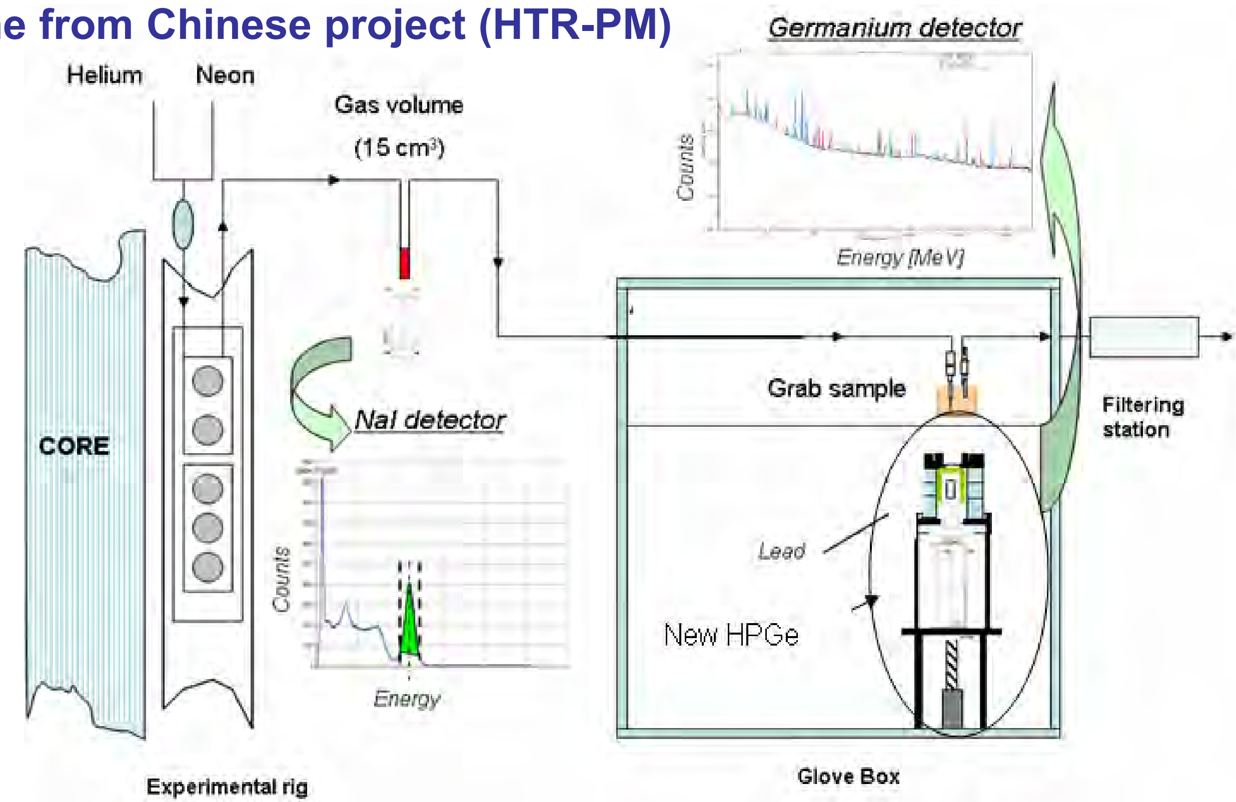
Demonstrate negligible radioactivity release for licensing

Irradiation in the HFR Petten of spherical HTR fuel pebbles
- **1050° C central temperature**
- **up to 100 GWd/tU Burn-Up**

- **Qualification for HTR fuel to be used in the demonstrator reactors**
- **300 k€ competitive income from Chinese project (HTR-PM)**

SLF principle

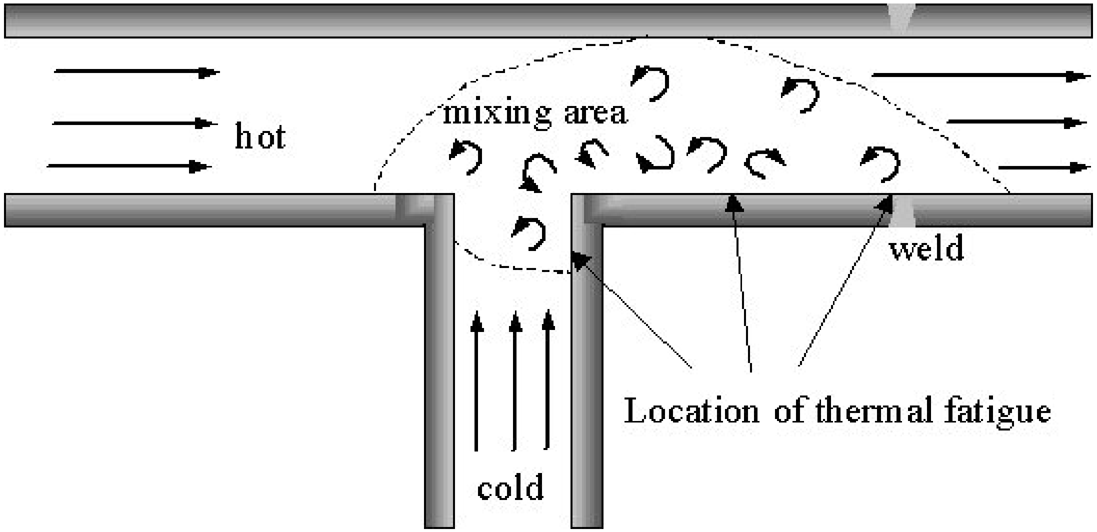
- **Monitoring of safety relevant operating parameters**
- **Maintenance of constant irradiation conditions**
- **Surveillance of fission gas release**



Thermal Fatigue in Nuclear Components

Why?

Thermal fatigue is one of the major degradation mechanisms in LWRs. Complex loadings is a main issue.

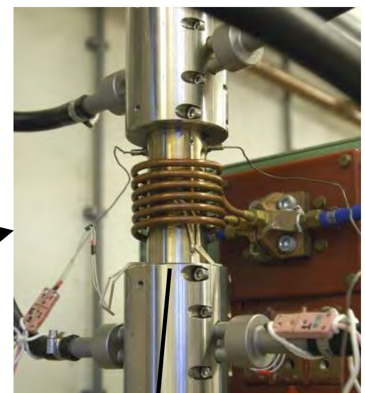


- ***Procedures for thermal fatigue initiation and propagation by replacing the load spectrum with the single frequency load that gives the shortest life***
- ***Experimental Programme to simulate thermal fatigue damage through cyclic down shocks***

Thermal Fatigue in Nuclear Components

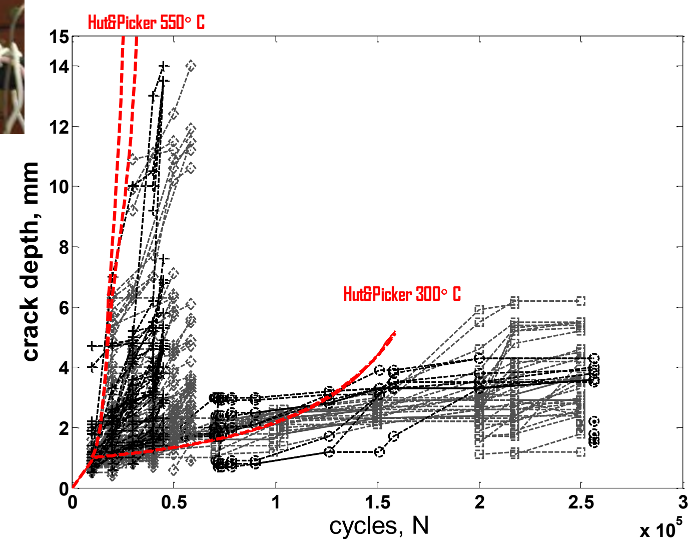
Water quenching lines

Axial loading train



Thermal loads: Induction Heating and cooling by water

Mechanical Load: Axial load (0, 50, 100 kN)

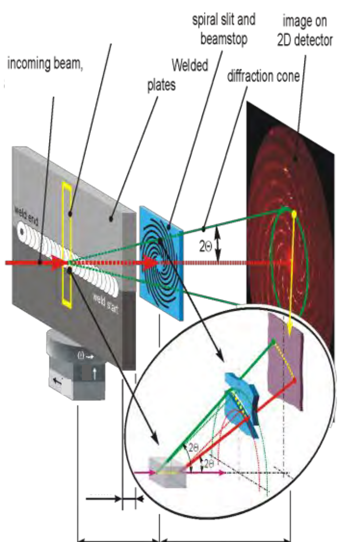


Residual Stresses in Welds

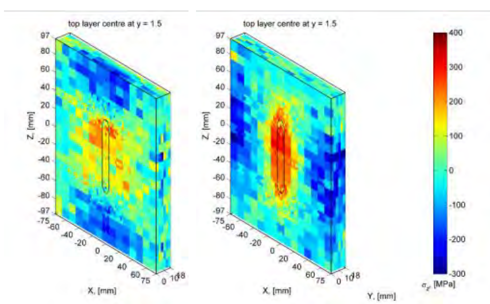
Why?
Welds are weak junctions in components.
For an assessment one need to know:
- **Residual stresses**
- **Material variability and defects**

In MATTINO we perform:

- Residual stress measurements with neutron diffraction and synchrotron diffraction
- Analyses with different levels of refinement

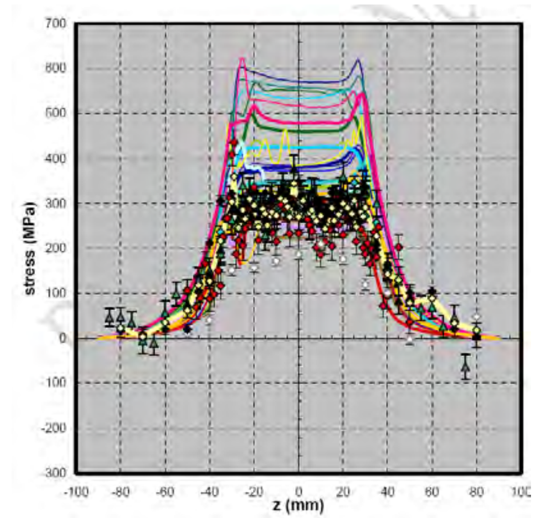


R.V. Martins et al.

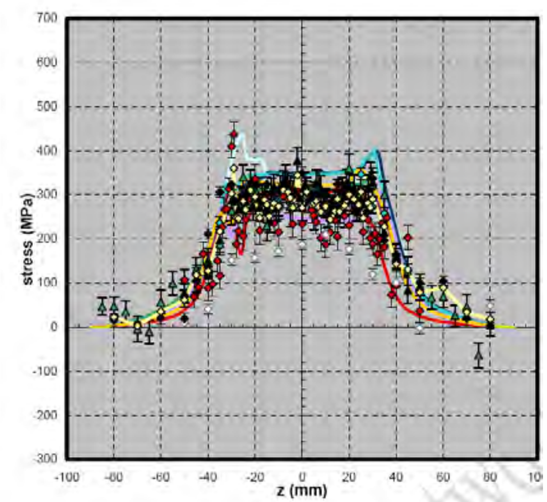


Spiral slit technique in synchrotron diffraction stress measurement:

Measured vs. computed



Initial



Refined

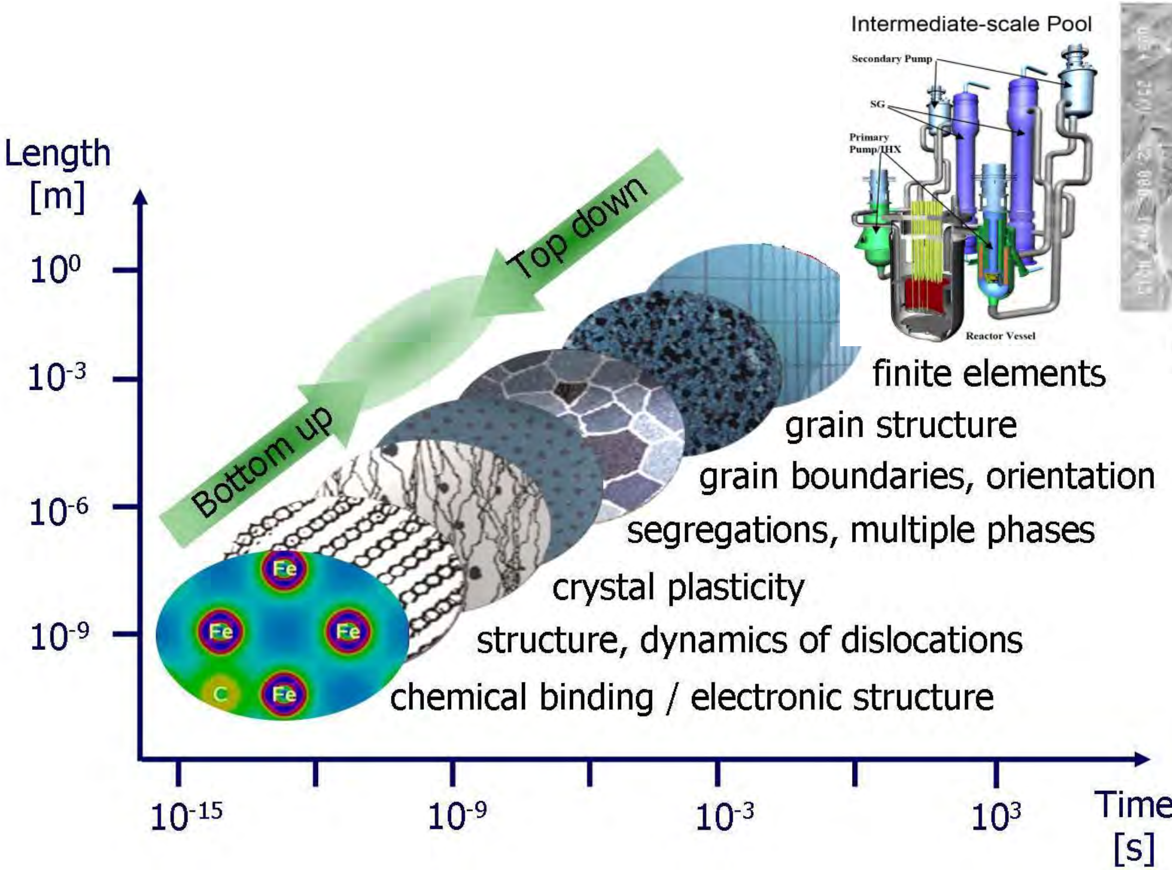
European Commission

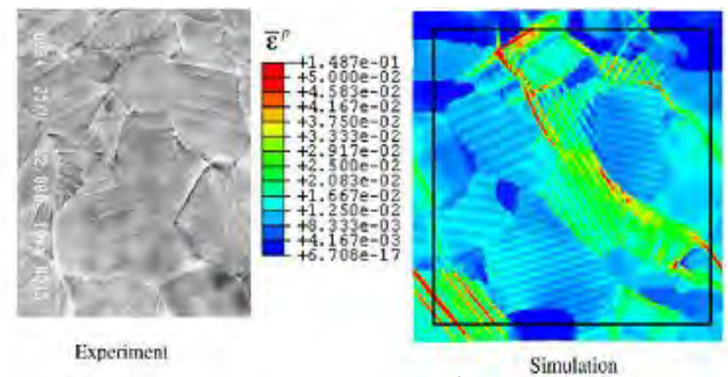
Joint Research Centre

Research Front: Multi-scale Modelling (Crystal plasticity models)

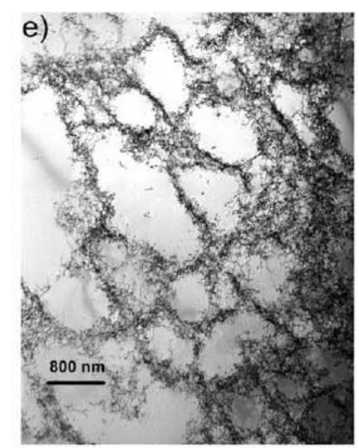
Why?

- **Material degradation occurring at different "length and time scales"**
- **Necessity to extrapolate from accelerated tests to operational conditions**
- **Basis for development of new materials (e.g. nano materials)**





Fatigue initiation and short crack growth



Dislocation patterning

IGSCC-Multiscale modelling

Surface reconstruction

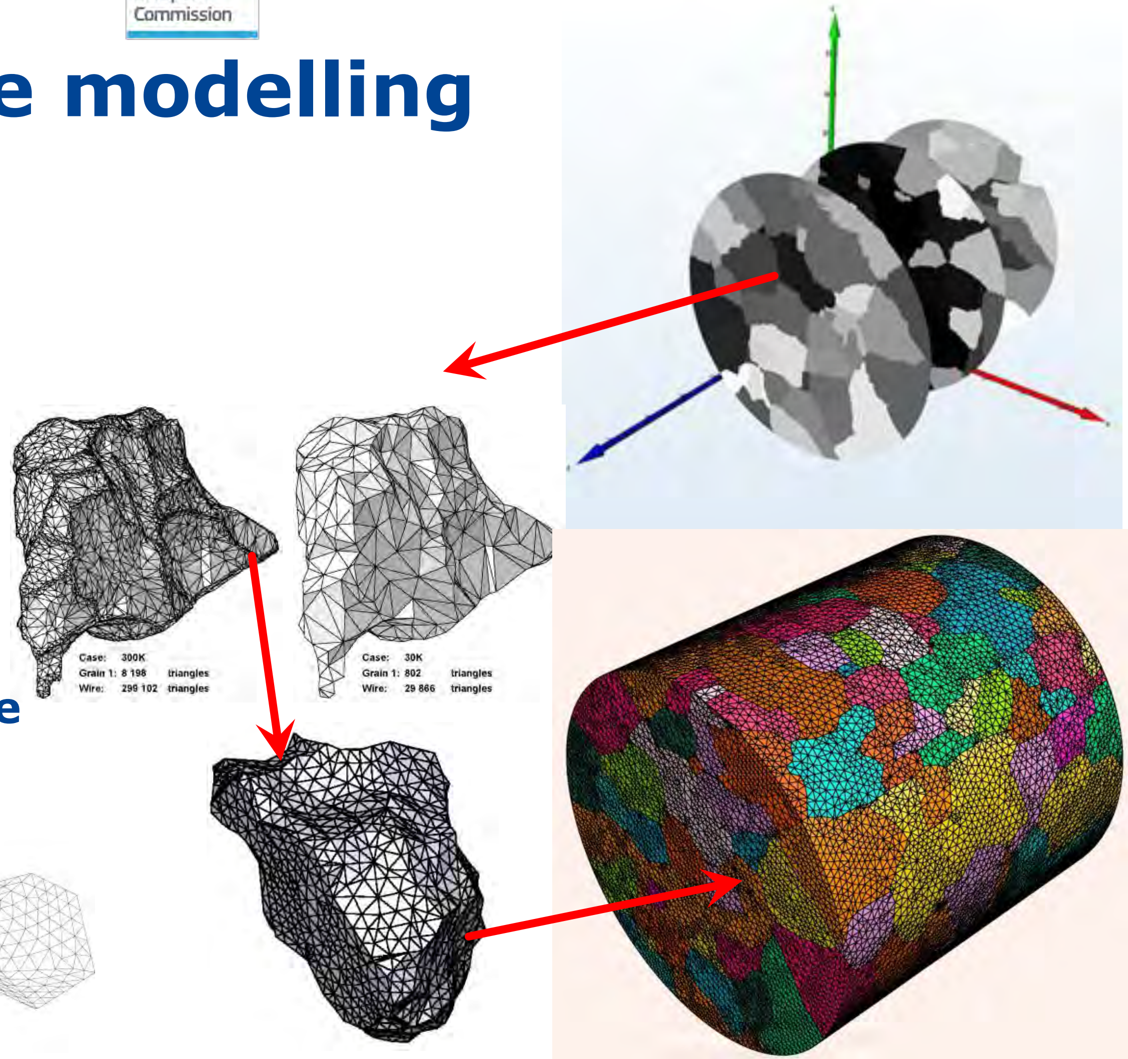
- **Real grain topology**
- **Simplification**

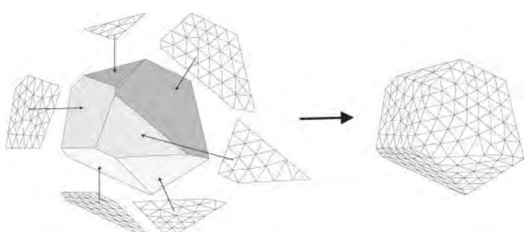
Conformal meshing:

- **Surfaces**
- **Volumes**

Constitutive models:

- **Grains: AE+CP**
- **Grain boundaries: cohesive zone**



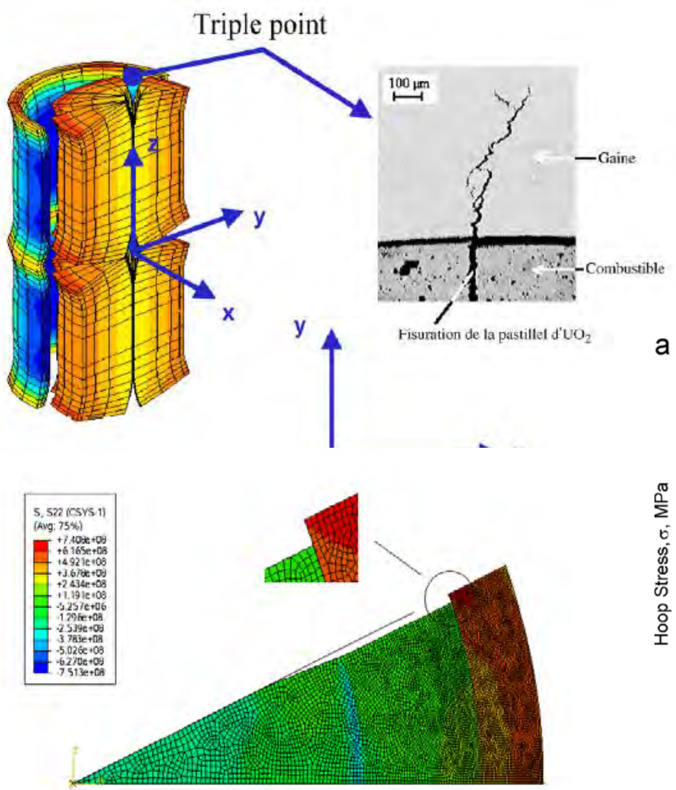


Experimental data University of Manchester: http://dx.doi.org/10.1016/j.commatsci.2010.12.014

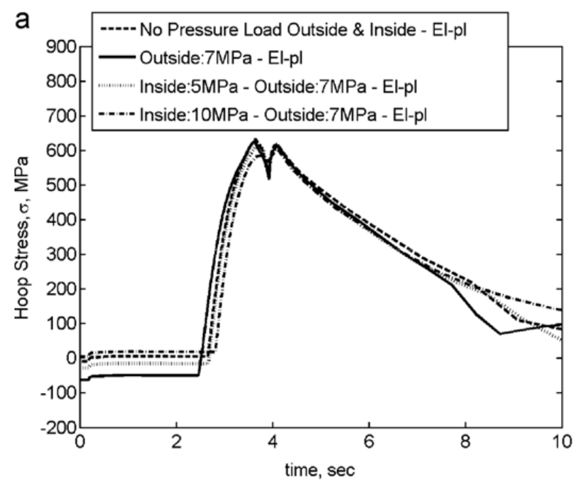
Fuel-Clad Interaction

Why? Fuel cladding is the first safety barrier. Safety assessment requires modelling of the fuel and cladding for relevant loads

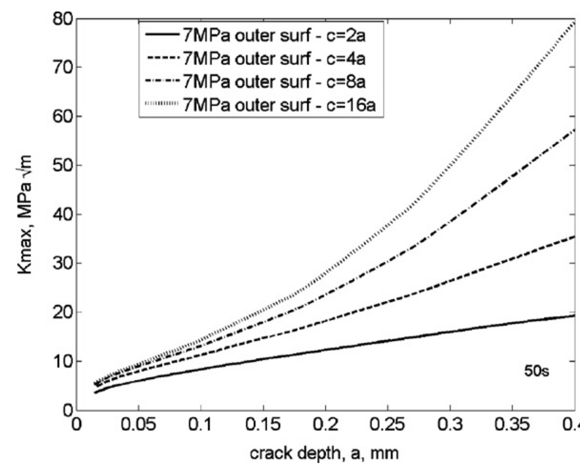
- Assessment of the behaviour of fuel pin sub-assembly blockage (GFR)
- Two-step analysis:
  - CFD → temperature transients
  - FEM fuel-pin (cracked fuel and cladding)



Von Mises stress distribution in fuel and cladding



Computed hoop strain cladding



Computed K vs. crack depth (different crack aspect ratios)

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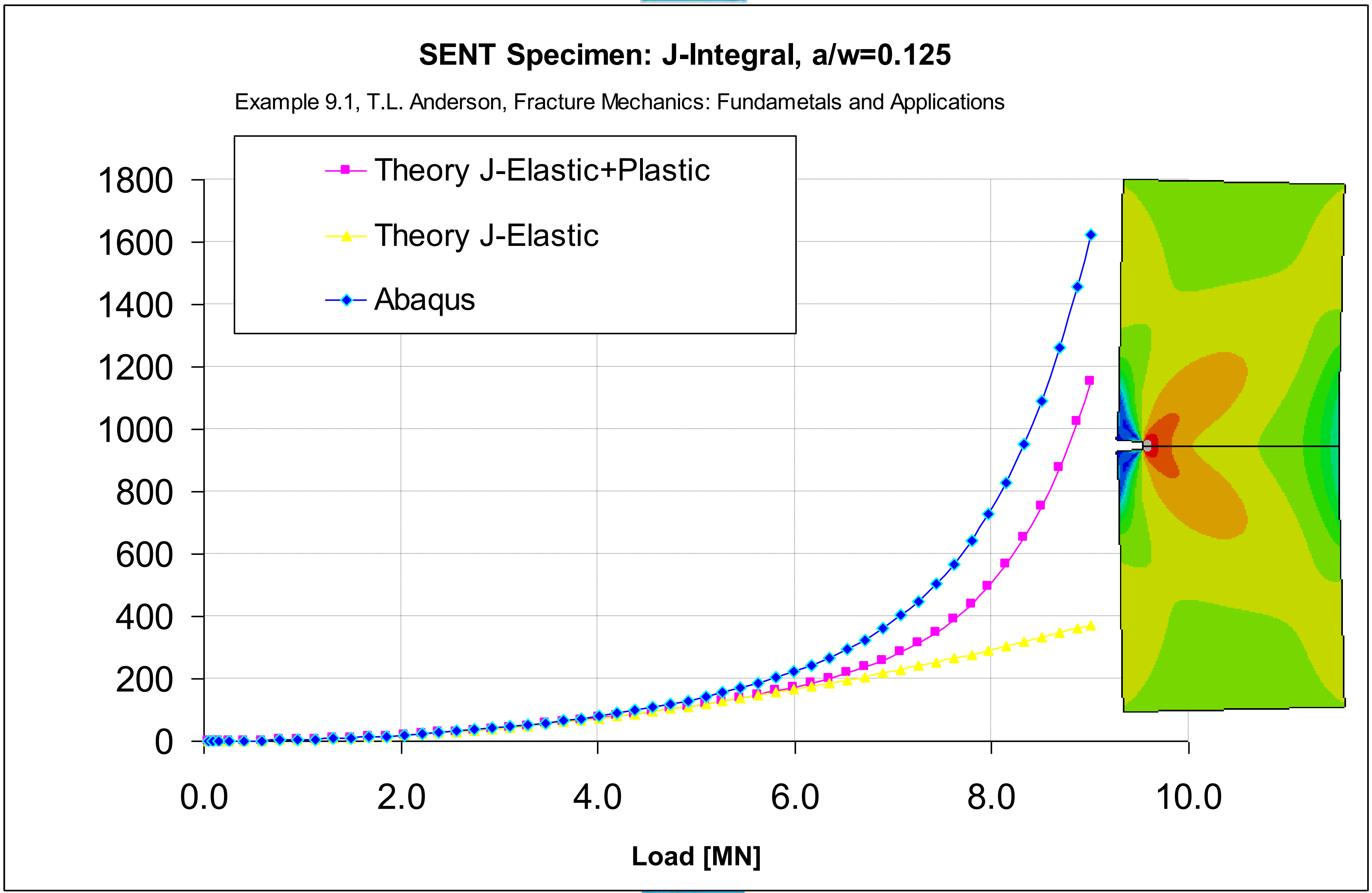
K/J value estimation in DMWs

European Commission



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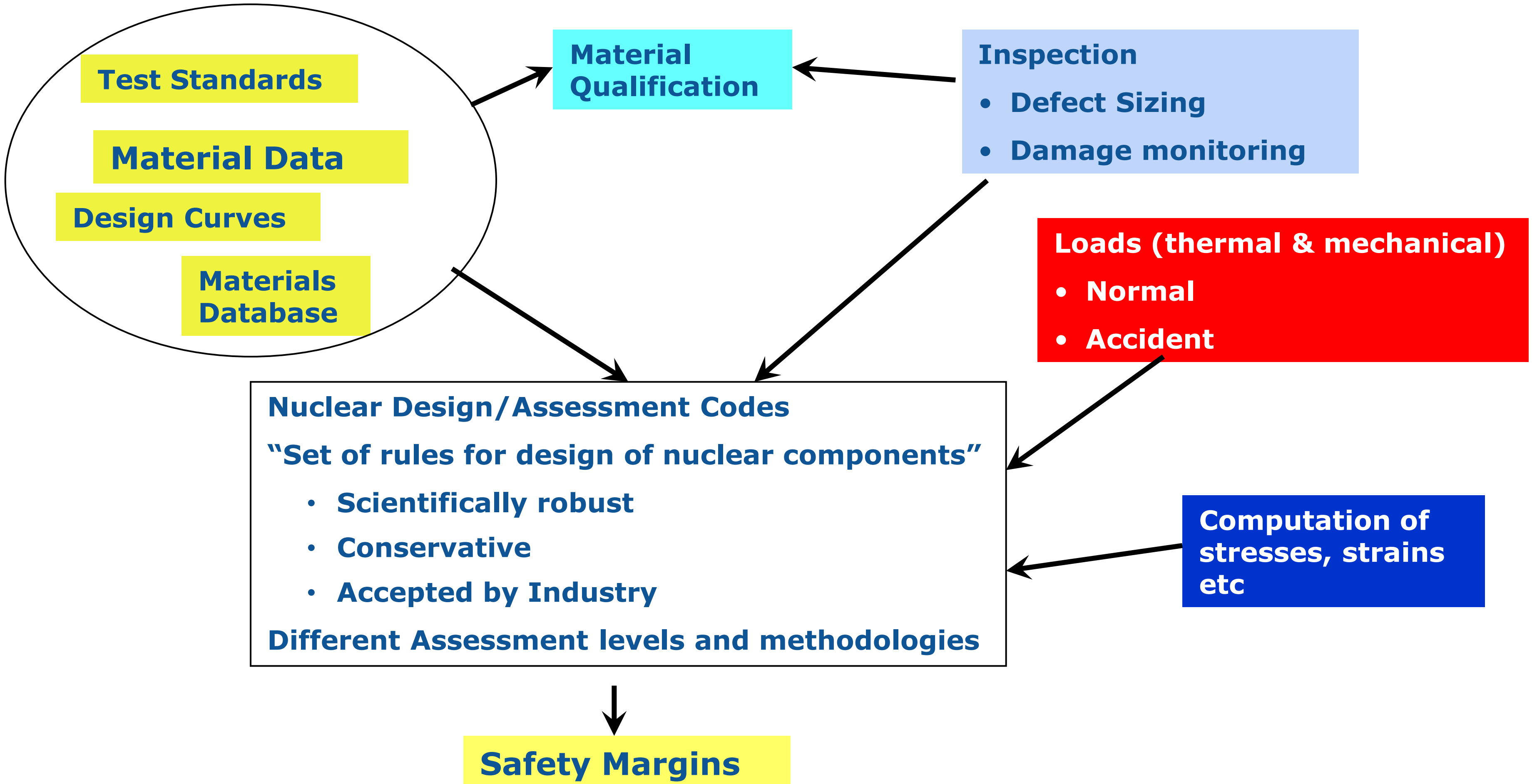
Materials Databases: Harmonization, Codes & Standards



European Commission

Thank you for Your Attention

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