Canadian Nuclear Safety Commission | Commission canadienne de sûreté nucléaire

OECD/NEA/CSNI Workshop IRIS_2012 Improving Robustness Assessment Methodologies for Structures Impacted by Missiles,

Neb Orbovic, P. Eng., Technical Specialist
Andrei Blahoianu, P. Eng., Director

OECD/NEA WG IAGE Meeting
Paris, France

April 8th- 12th , 2013

nuclearsafety.gc.ca

IRIS_2012 Workshop - INTRODUCTION

- IRIS_2012 benchmark, held in Ottawa, ON, Canada, in October 2012, is based on recommendations of IRIS_2010 workshop, held in Paris, France, in December 2010,

- Post-test numerical simulation is performed with the goal to:

  - Improve existing models used in IRIS_2010,

  - Propose simplified methods.

IRIS_2012 Workshop - INTRODUCTION

- CNSC organized the workshop with technical support from the University of Toronto

- The structure of the IRIS_2012 Organizing Committee is kept similar to IRIS_2010, with permuted roles of CNSC and IRSN and IAEA as invited member:

  - CNSC, IRIS_2012 Chair, Neb Orbovic,
  - IRSN, IRIS_2012 Co-Chair, Francois Tarallo,
  - IRSN, IRIS_2012 Co-Chair, Jean-Mathieu Rambach,
  - IAEA, Kenta Hibino, Member,
  - CNSC, Andrei Blahoianu, Member,
  - OECD/NEA, Alejandro Huerta, IRIS_2012 Secretary,

IRIS_2012 Workshop - INTRODUCTION

- The members of IRIS_2012 Scientific Committee are:
  - Prof. Dr. Jorge Riera (UFRGS, Brazil),
  - Prof. Dr. Ted Krauthammer (University of Florida, USA),
  - Prof. Dr Frank Vecchio (University of Toronto, Canada),
  - Prof. Dr. Friedhelm Stangenberg (University of Bochum, Germany),
  - Dr. Norbert Krutzik (Consultant, Germany),
  - Dr. Jorma Arros (MMI Engineering, USA),
  - Dr. Len Schwer (Consultant, USA),
  - Dr. Alain Rouquand (CEA Gramat, France).

IRIS_2012 Workshop - INTRODUCTION

Goals of IRIS_2012

- Improving existing models used in IRIS_2010
  - Use of a single set of material properties (e.g. concrete unconfined compressive strength, concrete tensile strength).
  - The results of tri-axial concrete tests and Brazilian test are provided to calibrate concrete constitutive models.
- Proposing simplified models
  - No guidance nor requirement are provided regarding simplified models; any simplified approach is welcome.

IRIS_2012 Workshop - INTRODUCTION

- IRIS_2012 benchmark consist of four different simulations:

  1. Tri-axial concrete tests provided by IRSN/UJF,
  2. Brazilian concrete test,
  3. Post-test simulation of Bending VTT-IRSN test,
  4. Post-test simulation of Punching VTT-IRSN-CNSC test.

IRIS_2012 Workshop - INTRODUCTION

- Based on recommendations of IRIS_2010 workshop the participants of IRIS_2012 should:

1. Use a single set of concrete material properties (unconfined compressive strength, tensile strength),
2. Use tri-axial and Brazilian tests to calibrate concrete constitutive models,
3. Improve existing models for Flexural and Punching tests (if there is an existing model) performing sensitivity studies,
4. Propose simplified analytical/numerical models
5. Report on lessons learned based on 1. to 3.
6. Based on 1. to 4. provide suggestions for future work.

IRIS_2012 Workshop - INTRODUCTION

IRIS_2012 Participants

- 26 teams from 10 countries and one international organisation performed the simulations,

- 26 teams are from 20 different organisations:
  - IRSN, France, is represented with 4 teams,
  - VTT, Finland, is represented with 3 teams,
  - CNSC, Canada, is represented with 2 teams.

IRIS_2012 Workshop - INTRODUCTION

No.	TEAM NAME	ORGANIZATION	COUNTRY
1	AERB	Atomic Energy Regulatory Board	India
2	ANATECH	Anatech	USA
3	BARC	Bahba Atomic Research Centre	India
6	CANDU	CANDU Energy	Canada
7	CEA	Commissariat à l'énergie atomique et aux énergies alternatives	France
8	CNSC Team I	Canadian Nuclear Safety Commission	Canada
9	CNSC Team II	Vector Analysis Group	Canada
10	EDF	Électricité de France	France
11	ENSI	Swiss Federal Nuclear Safety Inspectorate	Switzerland
12	F4E-IDOM	Fusion for Energy - IDOM	International
13	Fortum	Fortum	Germany
14	GRS	German Reactor Safety Authority	Germany
15	IRSN Team I	Institut de radioprotection et de sûreté nucléaire	France
16	IRSN Team II	Institut de radioprotection et de sûreté nucléaire	France
17	IRSN Team III	Institut de radioprotection et de sûreté nucléaire	France
18	IRSN Team IV	Institut de radioprotection et de sûreté nucléaire	France
19	JNES	Japan Nuclear Energy Safety Organization	Japan
20	KINS	Korean Institute of Nuclear Safety	South Korea
21	NRC-SNL	United States Nuclear Regulatory Commission / Sandia National Laboratories	USA
22	NRI	Nuclear Research Institute Res Energoprojekt Praha	Czech Republic
24	Swissnuclear	Swissnuclear	Switzerland
25	UJF	University Joseph Fourier - Grenoble	France
26	VTT Team I	VTT Technical Research Centre of Finland	Finland
27	VTT Team II	VTT Technical Research Centre of Finland	Finland
28	VTT Team III	VTT Technical Research Centre of Finland	Finland
29	Woelfel	Woelfel Group	Germany

IRIS_2012 Workshop - INTRODUCTION

SOFTWARE USED IN IRIS BENCHMARKS

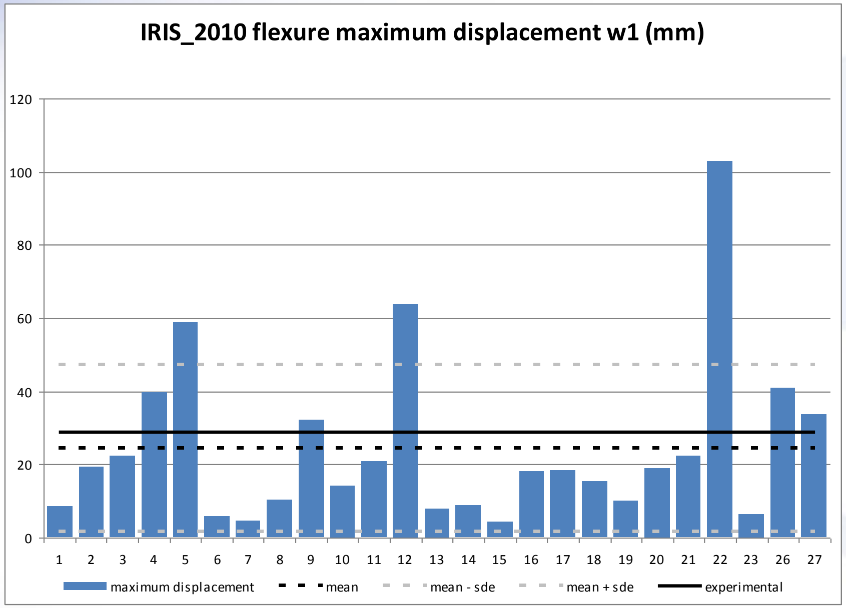
IRIS_2010	IRIS_2012
6 ABAQUS	9 ABAQUS
9 Ls-Dyna	6 Ls-Dyna
2 Autodyne	2 Autodyne
2 Europlexus	2 Europlexus
Radioss	Radioss V11
SofiStik	SofiStik
SAP	VecTor2-VecTor3
EMU	BARC in-house
ULCA	Sierra/SM2012
2D Simplified	PENTABLOC Simplified
	Simplified VTT team II
	Simplified VTT Team III

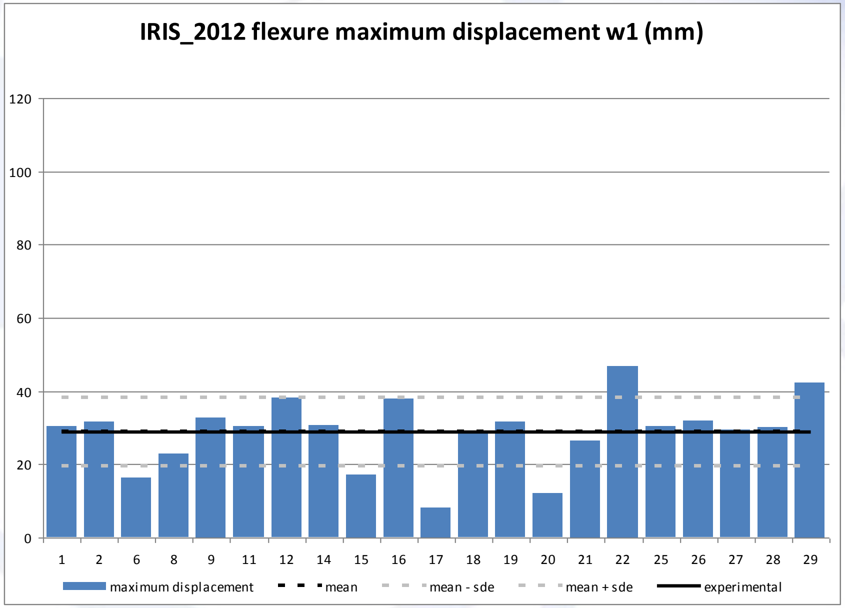
N. Orbovic, A. Blahoianu,
WGIAGE Meeting, 2013

IRIS_2012 Workshop – Simulation of Flexural Test

Comparison of IRIS_2010 and IRIS_2012 Flexural test results

Maximum displacement





IRIS_2010 C.O.V. = 97%

IRIS_2012 C.O.V. = 33%

N. Orbovic, A. Blahoianu,
WGIAGE Meeting, 2013

Flexural Test Results

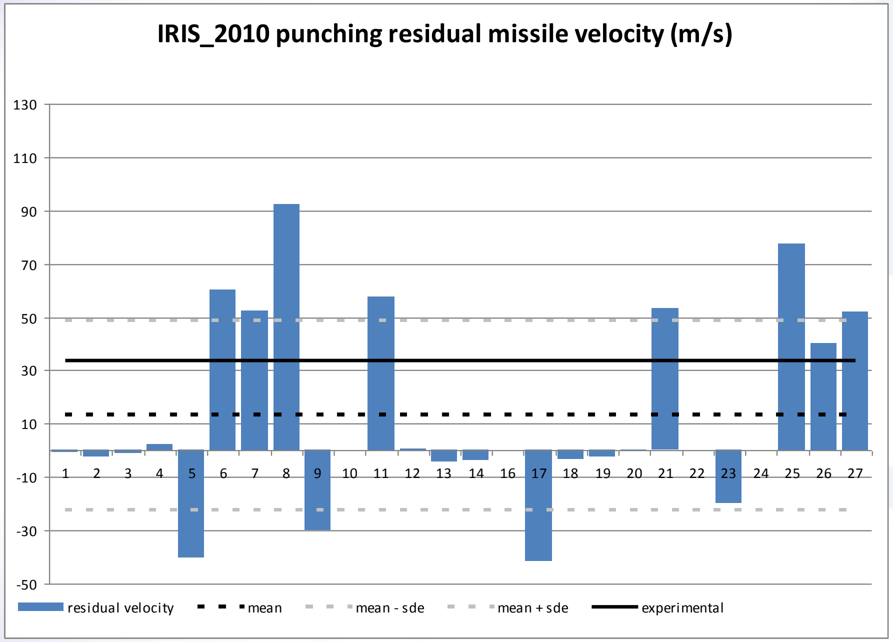
- Significant improvement in IRIS_2012 simulation of flexural test comparing to IRIS_2010. However, the scatter of results remains important (C.O.V =33%)

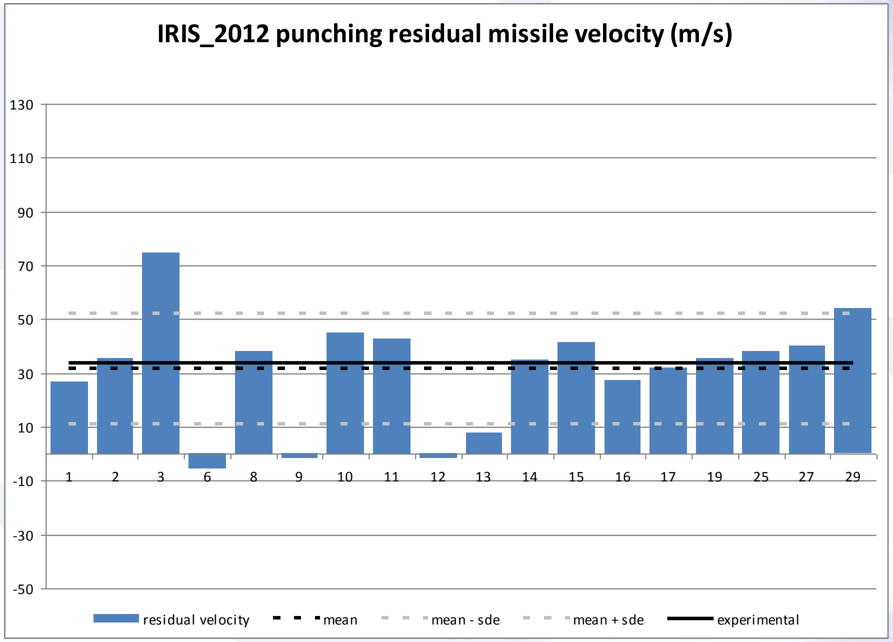
- It is not clear whether the improvement is a consequence of a better understanding of the problem and its simulation or a simple search for input parameters to mach given results.

IRIS_2012 Workshop - Summary of Punching Test Simulations

Comparison of IRIS_2010 and IRIS_2012 Punching test results

Residual missile velocity





PUNCHING TEST SIMULATION RESULTS

- Significant improvement in IRIS_2012 simulation of punching test comparing to IRIS_2010.
- 15 out of 18 teams simulated perforation with residual velocity very similar to the test result,
- Induced vibrations of stiff slab under look more difficult to simulate.

IRIS_2012 Workshop – Syntheses of Punching and Flexural Tests

- For both flexural and punching test - Significant improvement in IRIS_2012 simulation of punching test comparing to IRIS_2010.
- It is not clear whether the improvement is a consequence of a better understanding of the problem and its simulation or a simple search for input parameters to mach given results.
- The good news is that different software, or/and different concrete models within the same software, can be calibrated to get similar results which are close to results the results of two different tests with two different failure modes (flexure and punching).
- The question is if thus calibrated models would give good results in blind simulations of different tests.

IRIS_2012 Workshop - Syntheses

SIMPLIFIED MODELS

- The understanding of what a simplified model is, varies from one team to another:

1. Simplified FE models (SANDIA NL, UofT/CNSC),
2. Analytical or semi-empirical models (one, two or three degrees of freedom or empirical formulas: IRSN, VTT, EDF),

IRIS_2012 Workshop - Syntheses

SIMPLIFIED MODELS (Cont.)

Both approaches are important.

1. Simplified FE models:

   To find acceptable and accurate methods for industrial applications of this kind of simulations,

2. Analytical or semi-empirical models :

   To have a simple way (with hand calculation or an excel sheet) to approach the problem, to assess structural behaviour and to have a means to check the results of complex models.

IRIS_2012 Workshop – Syntheses of Simplified Model Simulation Results

SIMPLIFIED MODELS

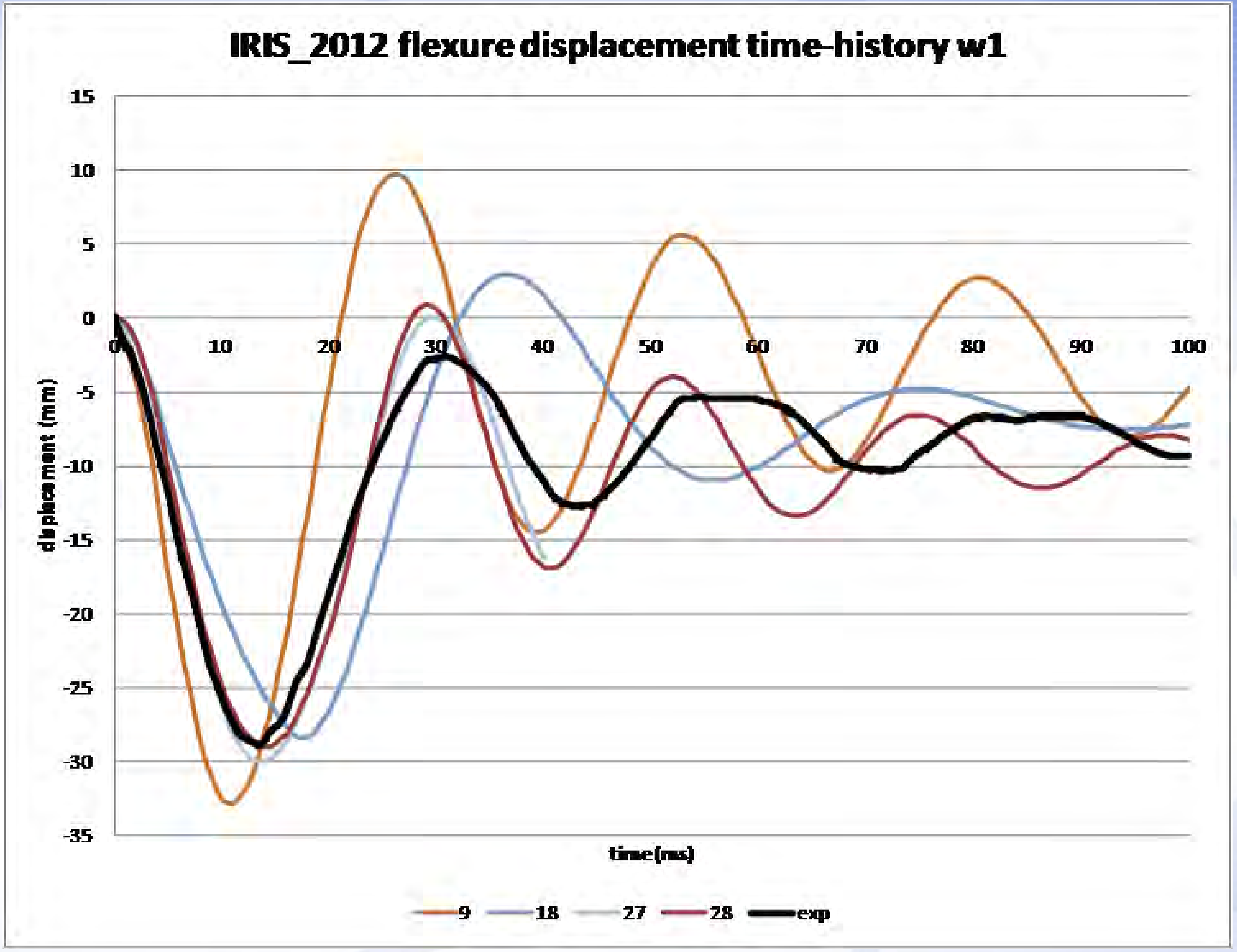
- Only five teams presented different kinds of simplified approaches,
- The level of accuracy of the results is comparable to those obtained with complex models.
- The spirit of IRIS_2010 recommendation is that each team presents two types of simulations: a FE model (with hundreds of thousands of elements) a simplified model, and to compare results.

IRIS_2012 Workshop - Summary of Simplified Model Simulation Results

Simplified models

Flexural test

- 9 CNSC Team II (UofT),
- 18 IRSN Team IV,
- VTT Team II
- VTT Team III



N. Orbovic, A. Blahoianu, WGIAGE Meeting, 2013

IRIS_2012 Workshop - CONCLUSIONS

General conclusions

- The most important is the understanding of physical phenomena based on test results,
- Continues process of understanding, quantification and modeling of the governing phenomena requires an important effort, time and *eye-burning*,
- Use of a simplified model or a set of simplified models should be mandatory in this type of simulations,
- Constant validation of simulations (simplified or complex) using test results is required to achieve credible predictions.

IRIS_2012 Workshop - CONCLUSIONS

Future objectives:

- Continue to improve the confidence in simulation results,
- Improve the knowledge of physical phenomena based on test results,
- Develop means for accurate prediction of induced vibrations

IRIS_2012 Workshop - RECOMMENDATION

FOLLOW-UP OF THE WORKSHOP IRIS_2012

- Complete numerical simulation report with the lessons learned from the comparison of IRIS_2010 and IRIS_2012 by June 2013,

RECOMMENDATIONS OF THE BENCHMARK IRIS_2012

- Continue to enhance existing FE models: the target criterion for scatter could be C.O.V < 20%,
- Present simplified analysis parallel to FE simulations with engineering attitude.
- Next phase will be focused on induced vibrations.

IRIS_2012 Workshop

Spirit of IRIS_2012 Workshop

- Three days of good presentations and very interesting discussions,
- A big improvement, comparing to IRIS_2010 workshop, regarding the attitude of participants and the involvement in discussions,
- The Organizing Committee is encouraged to continue with OECD/NEA IRIS Research Program.

Thank you

nuclearsafety.gc.ca

Post-tensioning methodologies for containment building: greased or cement grouted tendons – consequences on monitoring, periodic testing and modelling activities

NEA/IAGE meeting

Concrete Group

April 2013

MM Gallitre & Varpasuo

Organization, content & technical progress

Summary of the future document 1/2

1°) Context & objective

2°) General Philosophy Safety & advantages and drawbacks

3°) Feedbacks

non nuclear field

nuclear field

4°) Standards and codes of practices

5°) Design

Design basis (greased/grouted)

Design extension (*beyond design*)

Pressure load case (greased/grouted)

Aircraft aspect (greased/grouted)

Summary of the future document 2/2

6°) Construction

- Greased
- Grouted
- HDPE

7°) In Service Inspection

- Greased
- Grouted
- HDPE

8°) Conclusion

Technical Progress 1/2

	Writer	Checker	Draft progress
Objective & context	Gallitre	Varpassuo	100%
Safety requirement	Rambach	Gallitre	100%
Non nuclear feedback	Jackson	Barré	0
Nuclear feedback	Touret	Mitmarta	80%
Standard & codes	Homayoun	Debattista	20%
Design basis	Homayoun	Barré	90%
Design extension/pressure	Varpassuo	Ghavamian	90%

Technical Progress 2/2

	Writer	Checker	Draft progress
Design extension/aircraft	Tarallo	Varpassuo	50 %
Construction/greased	Jan	Mitmarta	25 %
Construction/grouted	Touret	Kim	40 %
Construction/HPDE	Isard	Mitmarta	40%
ISI/greased	Mitmarta	Jan	90
ISI/grouted	Courtois	Techner	80
ISI/HPDE	Isard	Mitmarta	25

Technical Findings (1/3)

<u>Design:</u>

Design basis / design extension conditions

- One question is the necessity or not to represent the sliding of the ungrouted cables
- Findings have been given to help the modelling of the containment, depending on pressure level (design basis or design extension conditions)

<u>Aircraft crash:</u>

General considerations will be given

Technical Findings (2/3)

<u>Construction</u>

critical points will be precised such as:

- the mock-up for grouting process validation
- the temporary measures to be taken (corrosion)
- material requirements
- performance criteria (advantages and limits)

Technical Findings (3/3)

In Service Inspection:

- What strategy for ISI?

ISI of grouted and non grouted are different but consensus is met to propose solutions for both technologies and for both following requirements:

  - Integrity test
  - Tightness test

- Interpretation of monitoring data
- Need of feedback for Fiber Optic

Some fisrt conclusions

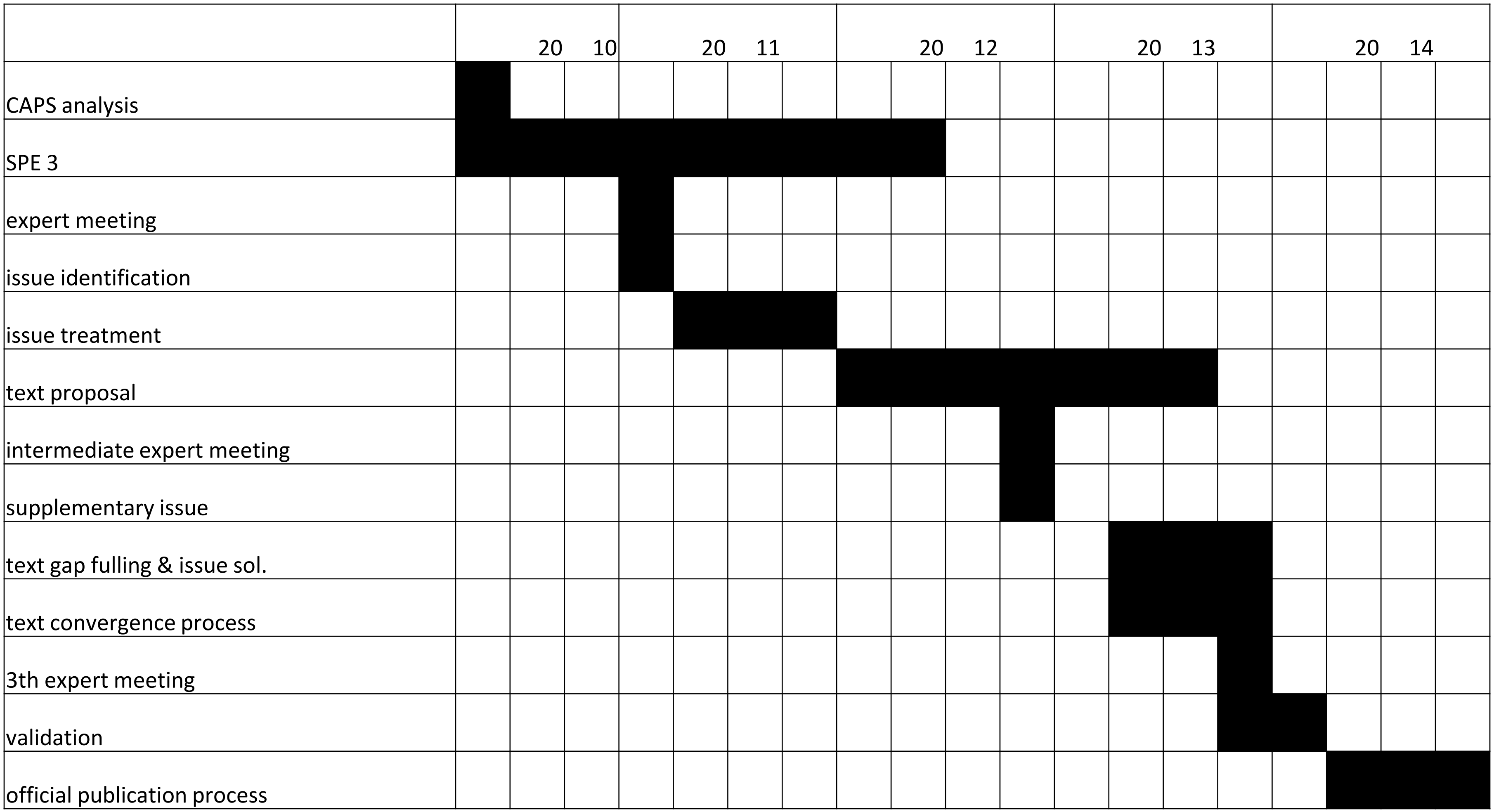
The experts meeting were very helpful:

- To share « precious » information
- to met concensus on possible strategies

But some further work has to be done to put on paper every idea and to validate it/group

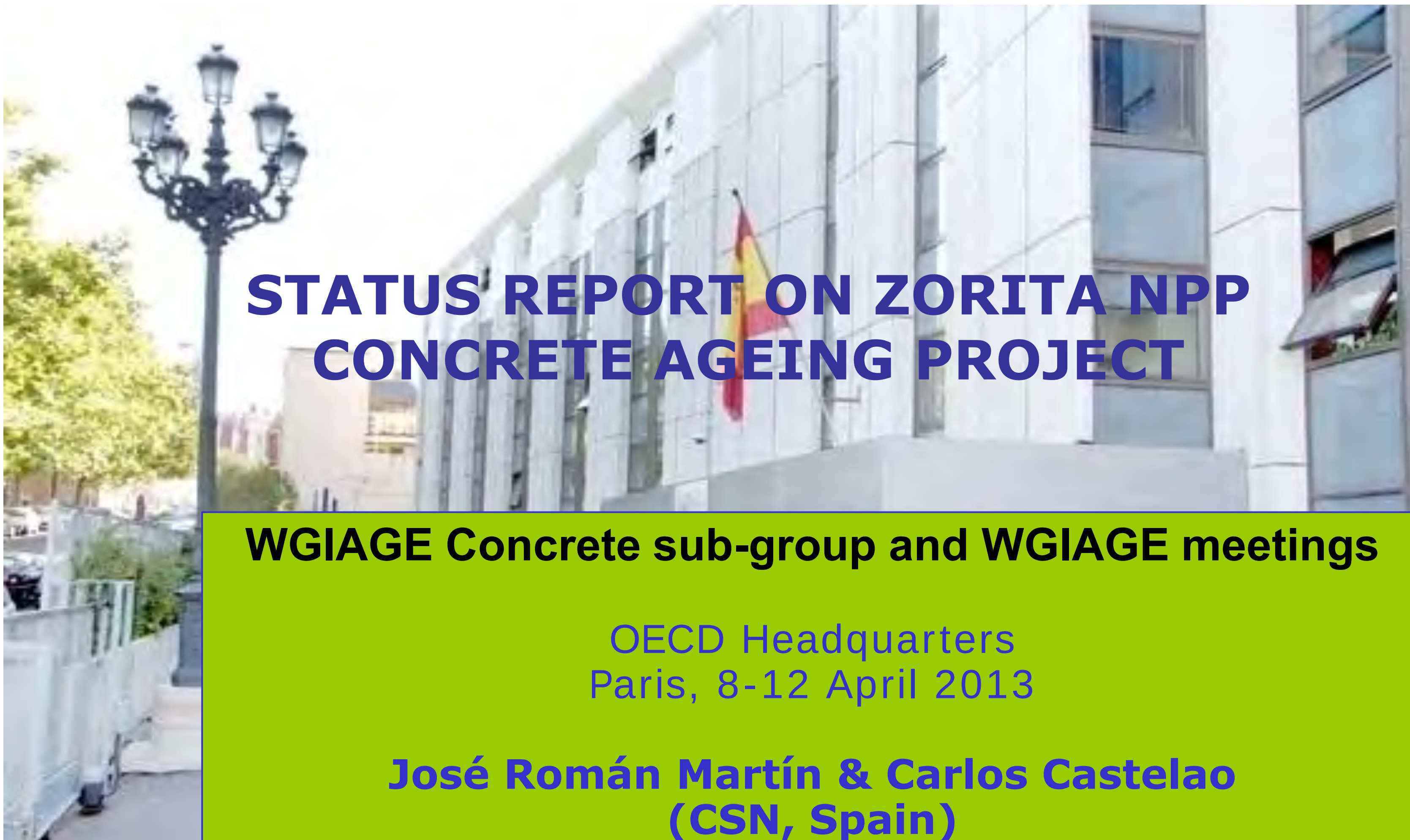
Pentti and Etienne thank everybody for the work done and for the last tasks to be done

General forecast schedule



CSN CONSEJO DE SEGURIDAD NUCLEAR

CEIDEN



STATUS REPORT ON ZORITA NPP CONCRETE AGEING PROJECT

WGIAGE Concrete sub-group and WGIAGE meetings

OECD Headquarters
Paris, 8-12 April 2013

José Román Martín & Carlos Castelao
(CSN, Spain)

CONTENTS

1.- BACKGROUND

- Zorita NPP consists of a PWR reactor, 1 loop, W design. 160 Mwe output.
- It was commissioned in 1968, and it operated until 2006.
- On february 2010, ENRESA took the responsibility for decommissioning the station.

José Cabrera NPP (Zorita)

CSN CONSEJO DE SEGURIDAD NUCLEAR

1.- BACKGROUND (cont.)

CEIDEN

- The Consejo de Seguridad Nuclear (CSN), Spanish regulator, is aware of the need of more research on the area of effects of high radiation (neutron+gamma) and temperature on concrete structures.
- Zorita's concrete structures could be of great value for this purpose, folllowing the steps of the ZIRP project about reactor vessel internals:
    - High neutron fluence in a commercial reactor
    - Synergy with the ZIRP project.
- CSN started in 2009 to meet with national organizations to check the feasibility of a project to obtain material from the Zorita concrete structures.
- In January 2010 the CEIDEN (Spanish Strategic Platform on Nuclear R&D) Working Group on Zorita Concrete was set up.

1.- BACKGROUND (cont.)

The project will be fund by the following partners:

- ENRESA (National Company of Nuclear Waste), in charge of the dismantling process of Zorita NPP
- ENDESA (utility)
- Gas Natural Fenosa (utility), licensee during operation of Zorita NPP
- IETCC (Instituto Eduardo Torroja de Ciencias de la Construcción), R&D lab.
- CSN (Consejo de Seguridad Nuclear, Spanish Regulator)

- IETCC is the lead laboratory and Gas Natural Fenosa provides technical support as licensee during operation.
- The project will be opened to other national or international organizations

1.- BACKGROUND (cont.)

Zorita NPP operating conditions:

- Inlet Coolant Temperature 282 ºC
- Core Average Temperature 293 ºC
- Core Exit Temperature 305 ºC
- Reactor Coolant System Pressure 140 kg/cm^2
- The plant was converted to up-flow 8 years before decommissioning
- 26,5 EFPY (Effective Full Power Years) of operation

1.- BACKGROUND (cont.)

Synergy with ZIRP (reactor vessel internals project)

- In ZIRP it was determine that the reactor vessel internals had an accumulated fluence of 58 dpa in some areas. No other commercial reactor has accumulated such fluence level, so far, at least under decommissioning stage.
- So, it is expected that some concrete areas surrounding the reactor vessel have also high values of neutron fluence and gamma radiation.
- This makes these structures of high value to perform tests to determine the irradiation effects.
- However, temperature is not so high, but enough to determine its effects on concrete properties.

2.- SCOPE

Steps followed:

- Bibliography review, GAP analysis, on the degradation of concrete structures on NPPs.

- As a consequence of the Gap Analysis, it was agreed that the effects of the following stressors on concrete will be studied:

  - High levels of neutron and gamma radiation

  - High temperature for long periods

  - Boric acid

  and

  - NDT on liner under concrete slab

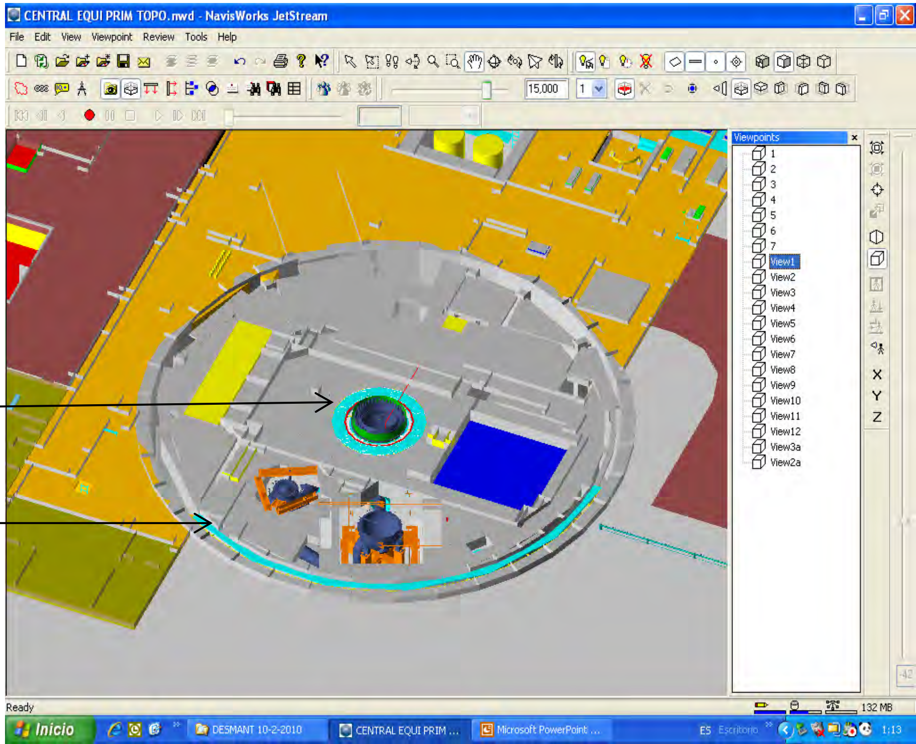
2.- SCOPE (cont.)

Structures to be considered:

- Biological shielding
- Spent fuel pool/transfer channel
- Containment building (as reference)
- Steel liner under concrete slab (NDT)

High-density concrete

Normal-density concrete



Biological shielding. Core samples

- The main purpose of the project is to test:
  - Areas of high irradiation
  - Areas of high temperature
  - Areas with a combination of high irradiation and temperature, if it is the case

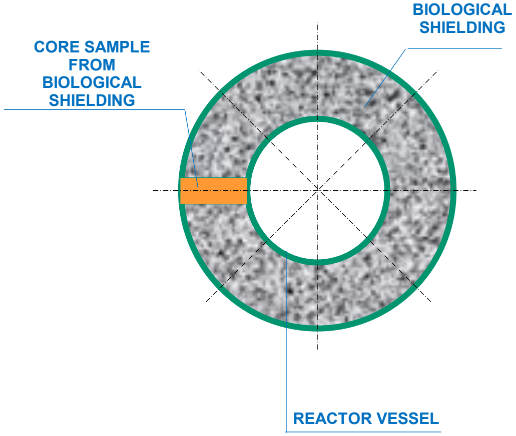
- It is anticipated that these areas are:
  - Those of the biological shielding closer to the fuel elements (for irradiation), or
  - Those closer to the outcome branch of the reactor coolant system (for temperature),

  and have to be clearly identified.

- Some spare cores will also be drilled, to perform accelerated irradiation (EPRI's suggestion)

2.- SCOPE (cont.)

Biological shielding. Core samples (irradiation) identification:



- Symetry of the irradiation level around the fuel elements is 1/8
- This means that only 8 core samples with the maximum level of irradiation can be obtained
- For their identification, the same methodology of the ZIRP project will be used:

Neutron and Gamma sources calculation

- ➢ Compilation of 29 cicles of operation (38 Years – 26.36 EFPY):

  Load patterns, cicle duration, power and burnup montly history, moderated high and low leaks configuration, upflow-downflow configuration of core bypass

Biological shielding. Core samples (irradiation) identification:

Neutron and Gamma sources calculation (cont.)

- Compilation of 594 fuel elements:

  Enrichment, operating cicles history, initial and final burnup for each operating cicle, HIPAR y LOLOPAR.fuel elements characteristics

- Reproduction of power and burnup history of 29 operating cicles. Representative state of each operating cicle
- Neutron sources (intensity and spectrum) of the 29 operating cicles
- Gamma sources (intensity and spectrum) of the 29 operating cicles
- Isotopic composition of fuel elements

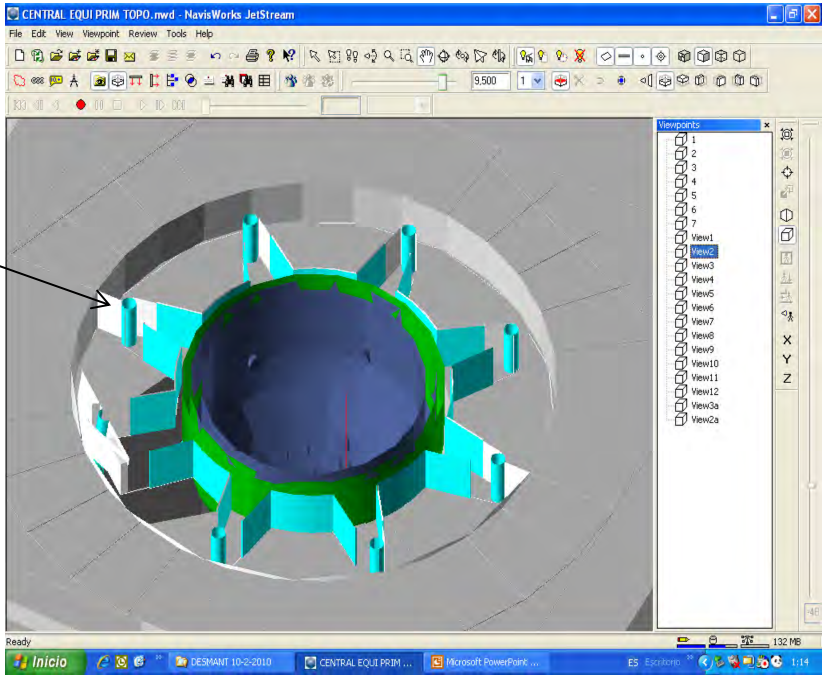
- The output of these calculations will serve to identify the most irradiated areas of the biological shielding

2.- SCOPE (cont.)

Biological shielding. Core samples (irradiation) identification

- Also a rough estimation of fluence and gamma radiation will be obtained from the instrumentation wells and from operating experience.

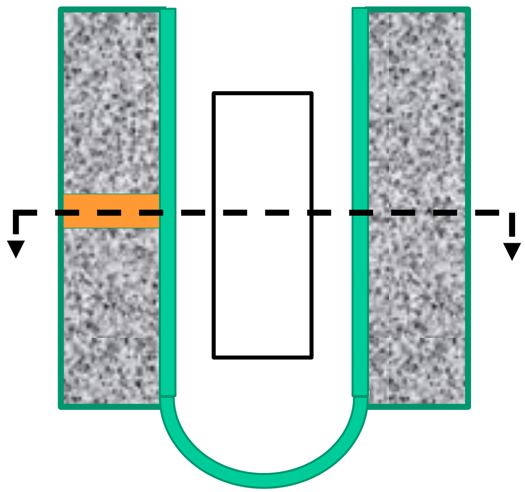
Instrumentation wells

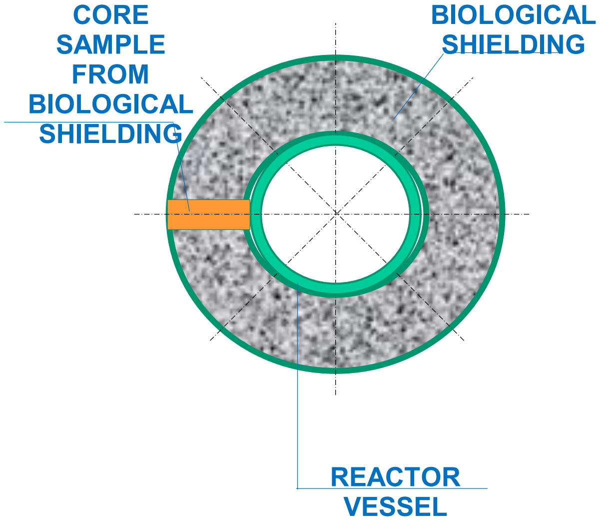


2.- SCOPE (cont.)

<u>Biological shielding. Core samples (irradiation)</u>:

- The biological shielding will be drilled in-situ, in the identified locations, from reactor vessel side or from the opposite side, still to be decided





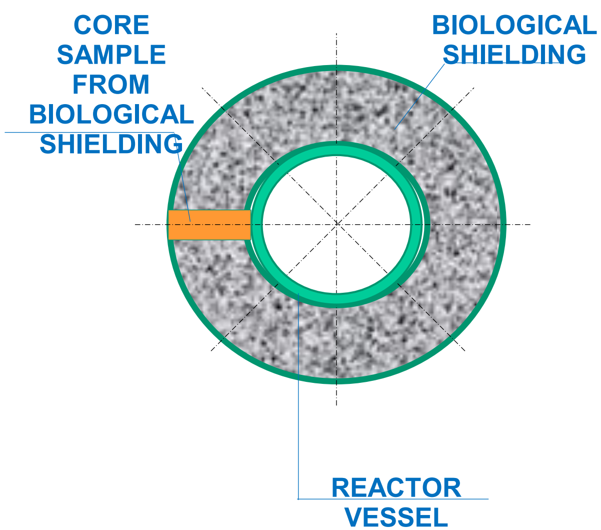
CSN CONSEJO DE SEGURIDAD NUCLEAR

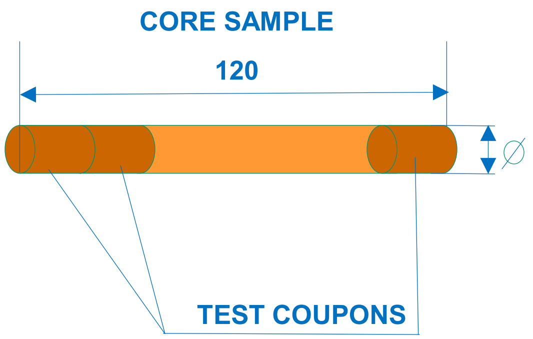
2.- SCOPE (cont.)

CEIDEN

Test coupons (irradiation):

- From each core sample several test coupons will be obtained:
  - The closest to the reactor vessel
  - One from the opposite side
  - Some intermediate ones
- This approach will also allowed to determine the attenuation effects
- Dimensions of test coupons still to be decided





Biological shielding. Core samples (temperature):

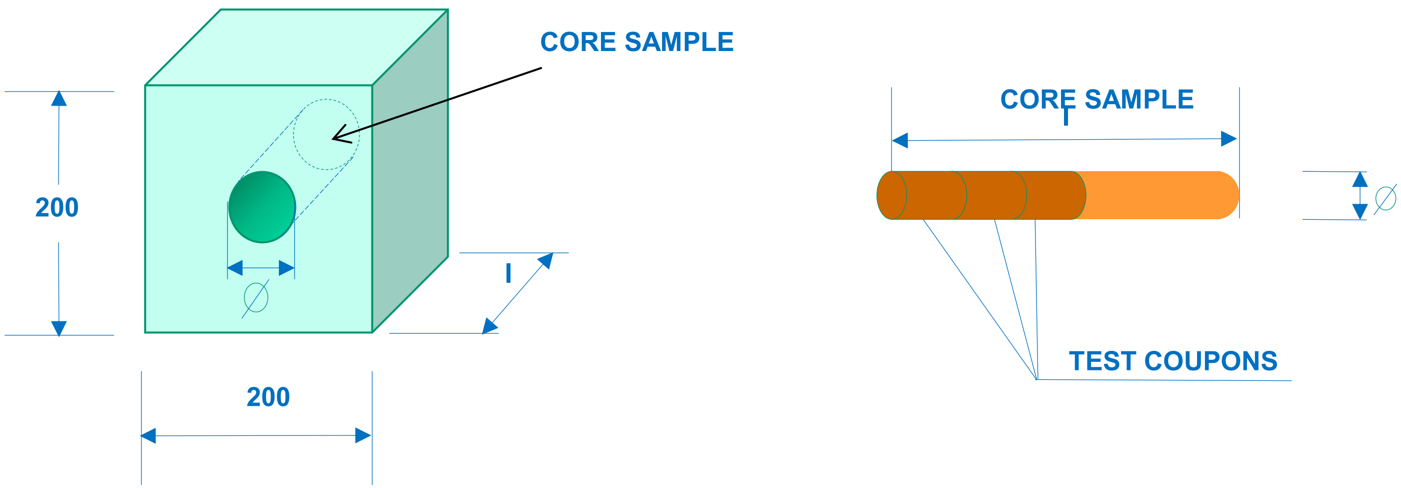
- It is anticipated that these areas are those closer to the outcome branch of the reactor coolant system, in the biological shielding.
- This has to be confirmed through calculations, in the same way as for the irradiation, and through analysis of operating experience.
- The extraction process will be the same as for the previous core samples.
- Test coupons will be prepared from the core samples, as already explained.

Biological shielding. Core samples (irradiation+temperature)

- With the previous data on irradiation and temperature, decision will be taken on whether to drilled or not some other core samples with the "most representative" combination of both parameters.

Spent fuel pool/transfer channel. Core samples (boric acid):

- The core samples of the spent fuel pool/transfer channel will be identified through analysis of operating experience:
    - Boric acid leaks during operation or refuelling outages
- Blocks of 2x2 meters will be cut
- As in the previous stage, core samples will be drilled from the blocks and from these cores tests coupons will be cut.



2.- SCOPE (cont.)

Containment. Core samples (reference):

- Some coupons will also be drilled from the containment, at different locations and heights, from the internal and external sides
- The results of these tests will be used as reference

NDT on liner under concrete slab

- Within the scope of the project it is also included the performance of NDT in some areas of the liner.
- IETCC will apply a specific technique to analyse the state of the liner in the selected areas.
- Destructive analysis of these areas will be then performed in order to verify the results obtained.

2.- SCOPE (cont.)

What is not included?

- The performance of any NDTs on thick concrete structures with the aim of their validation

CEIDEN is however open to any suggestion on this issue, as these structures could be very useful to validate NDT techniques.

3.- ON-GOING ACTIVITIES

Taking into account that it is expected that cores will be removed in year 2015, the following information is being collected:

- Drawings
- Concrete specifications (high and normal density),
- Operating experience, including Integrated Leak Rate Tests (ILRTs) performed
- Early tests on concrete, and
- Any other information that can be located.

- Also, chemical analysis on samples taking from some areas will be performed in order to characterize the concrete

4.- TESTING

Testing

- Test coupons coming from the biological shielding, the spent fuel pool/transfer channel and containment will be tested in IETCC research lab or in ENRESAS's facilities, in case they are contaminated.
- The envisaged tests to be carried out are those that the international community has already identified as neccesary to determine the temperature and radiation damage in concrete structures.
- This tests will include:
  - Mechanical
  - Microestructural
  - Others, still to be defined
- Two sets of tests have also been defined for characterization of concrete and liner corrosion

5.- STEPS TO FOLLOW

CEIDEN considers that after collecting the information of interest, at least the following protocols/procedures should be prepared and approved during year 2013:

- Fluence calculations (same methodology as in ZIRP)
- Temperature calculations (same as in ZIRP+operating experience)
- Identification of the exact locations of cores to be removed (synergy with ZIRP)
- Removal of cores
- Labelling of cores
- Storage
- Shipping to testing facility
- Testing performance
- Quality Assurance system of each organization will be followed

Deliverables. Reporting

- IETCC R&D lab will be in charge of compile, analyse and perform all neccesary work in order to draw conclusions on the effects of irradiation and temperature on concrete structures.
- Open literature will be taking into account.

Collaborative project

- The Spanish partners in this project wish to open it to other potential participants, at national and international level, once the agreement of the actual partners is in place.
- EPRI has already be in touch with the working group and will be contacted once the Spanish agreement be signed.
- NEA countries have also been informed through the Spanish delegate on the Subgroup on Concrete of the IAGE/CSNI working group.

8.- COMMITTEEs

Technical Committee (TC) members:

- Carmen Andrade (Chairperson), IETCC
- José Román Martín (Secretary), CSN
- Manuel Ordoñez, ENRESA
- Eduardo Serra, ENDESA
- Pedro Ortega, Gas Natural Fenosa
- Eduardo Más, CSN
- Carlos Castelao, CSN

Gas Natural Fenosa Engineering is supporting the TC work in recovering data and documentation from Zorita NPP.

- A Steering Committee with a representative from each participating organization is also in place.

Thanks for your attention

Concrete Degradation of the Containment of Tihange 2 NPP in Belgium

Status as of December 2012

OECD Nuclear Energy Agency

Committee on the Safety of Nuclear Installations

IAGE – Concrete subgroup

08 April 2013

Tchien Minh TANG
Safety Analysis - Structural Mechanics
Bel V, Technical Safety Organization
Subsidiary of the Federal Agency for Nuclear Control
Rue Walcourt 148
BE-1070 Brussels
Belgium
Office: +32 2 5280 249
tchienminh.tang@belv.be

Short history

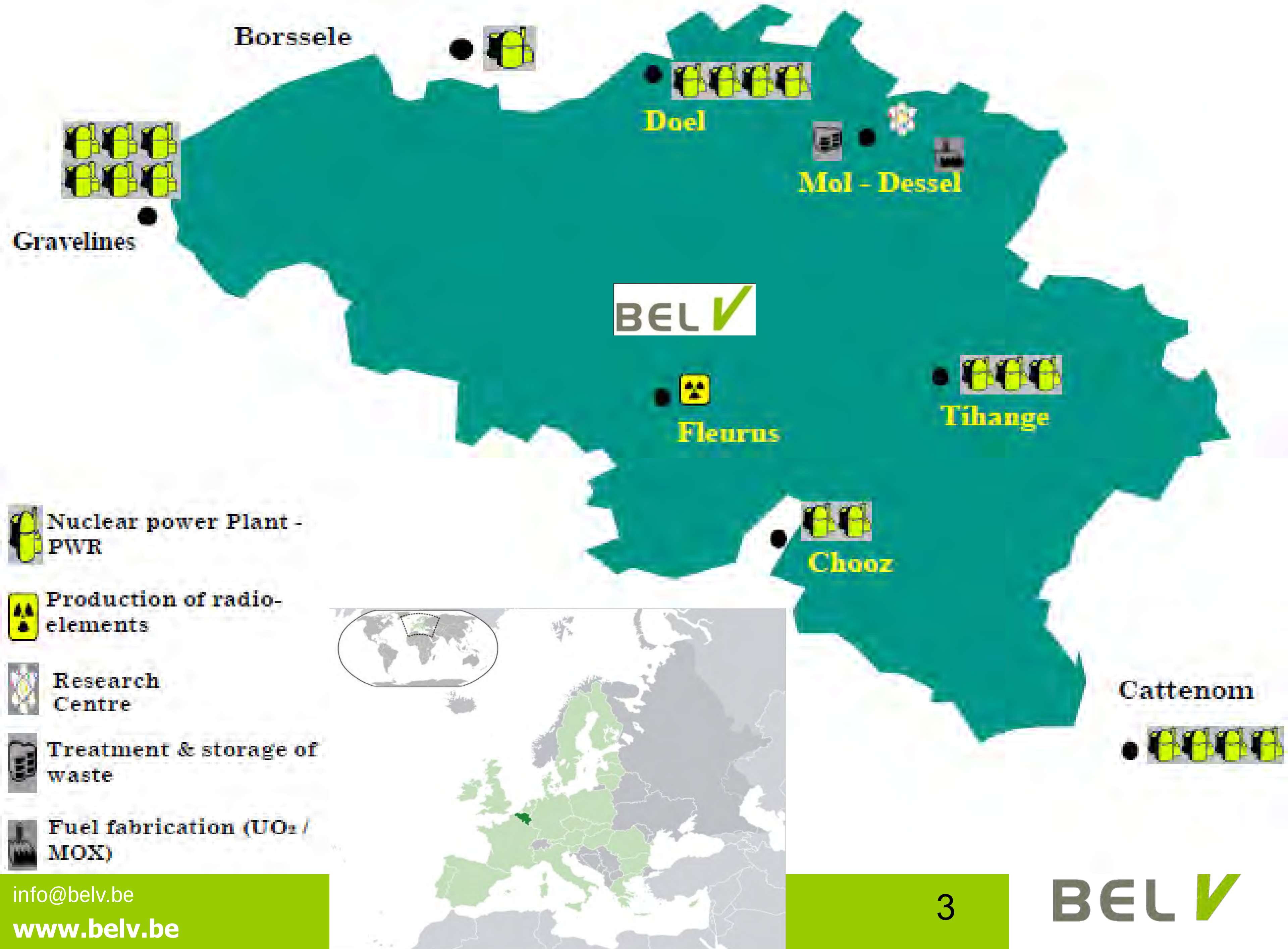
2008: Concrete degradation on the external face of the containment building of Tihange 2: spalling of the concrete and corroded rebars.

2009: Identified causes: general concrete carbonation and alkali-silica reaction (ASR).

2010: Concrete repair started. Works partially delayed till 2012 due to poor weather conditions.

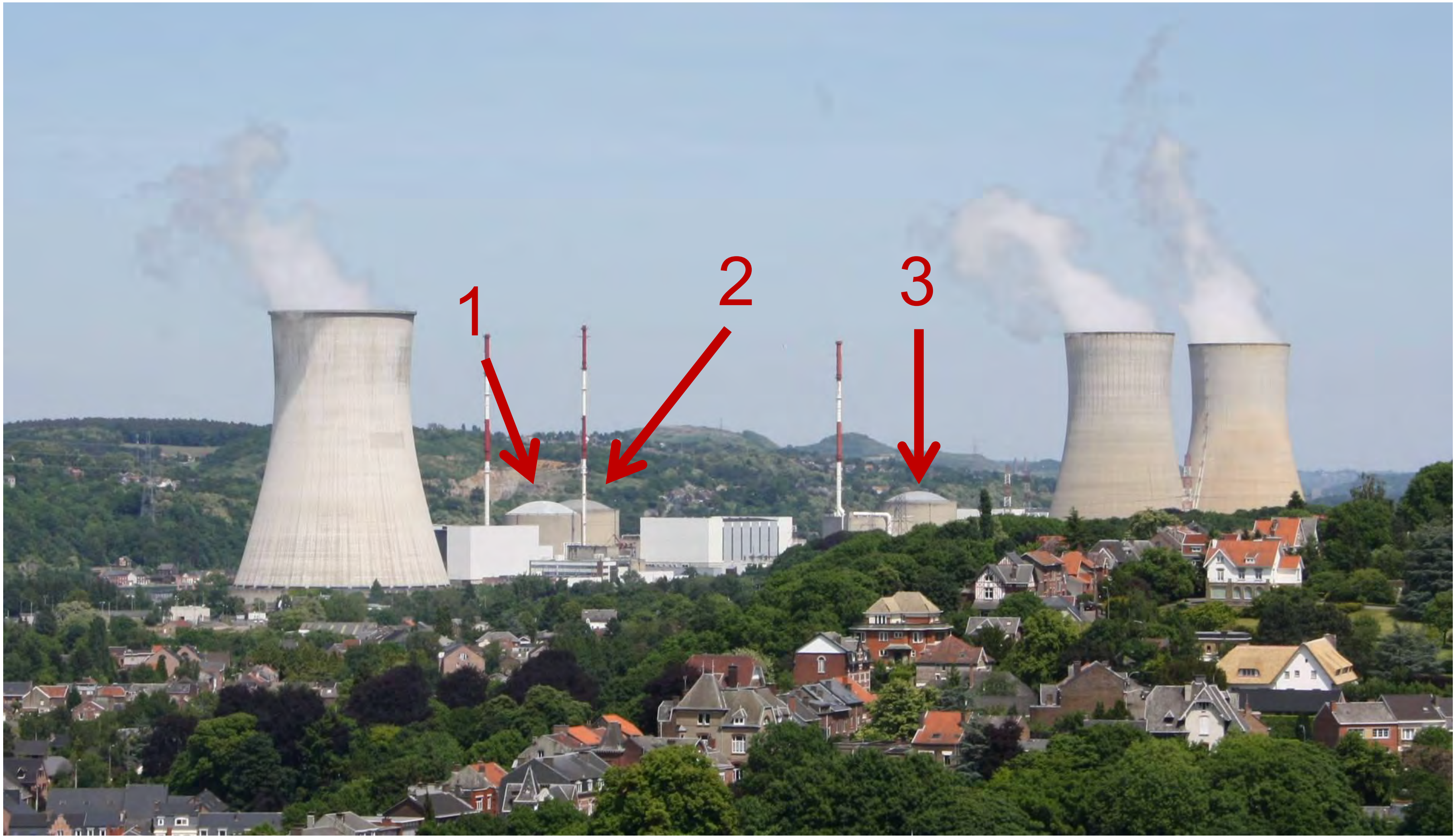
2012: New analyses revealed the predominance of ASR. Repair techniques changed after the assessment of the initial repairs.

2013: Further analyses are foreseen.


Borssele
Doel
Mol - Dessel
Gravelines
BEL V
Fleurus
Tihange
Chooz
Cattenom
Nuclear power Plant - PWR
Production of radio-elements
Research Centre
Treatment & storage of waste
Fuel fabrication (UO_2 / MOX)

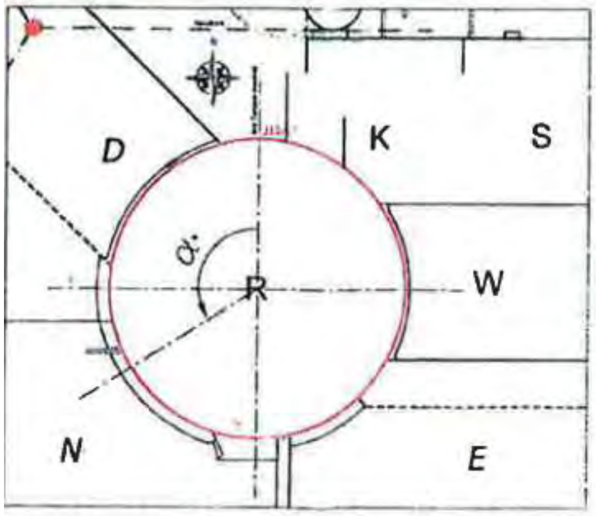
Doel NPP

Tihange NPP



Reactor building of Tihange 2

- Double containment with annular space
- External containment :
  - Reinforced concrete
  - Cylindrical with a dome
  - Thickness: 120 cm, diameter: 51 m height: 50 m
  - Built in 1977 and surrounded by auxiliary buildings
  - Designed to withstand safe shutdown earthquake (SSE) of 0.17 g and aircraft (91 tons) crash



Characteristics of the concrete: CEM-III + limestone aggregates

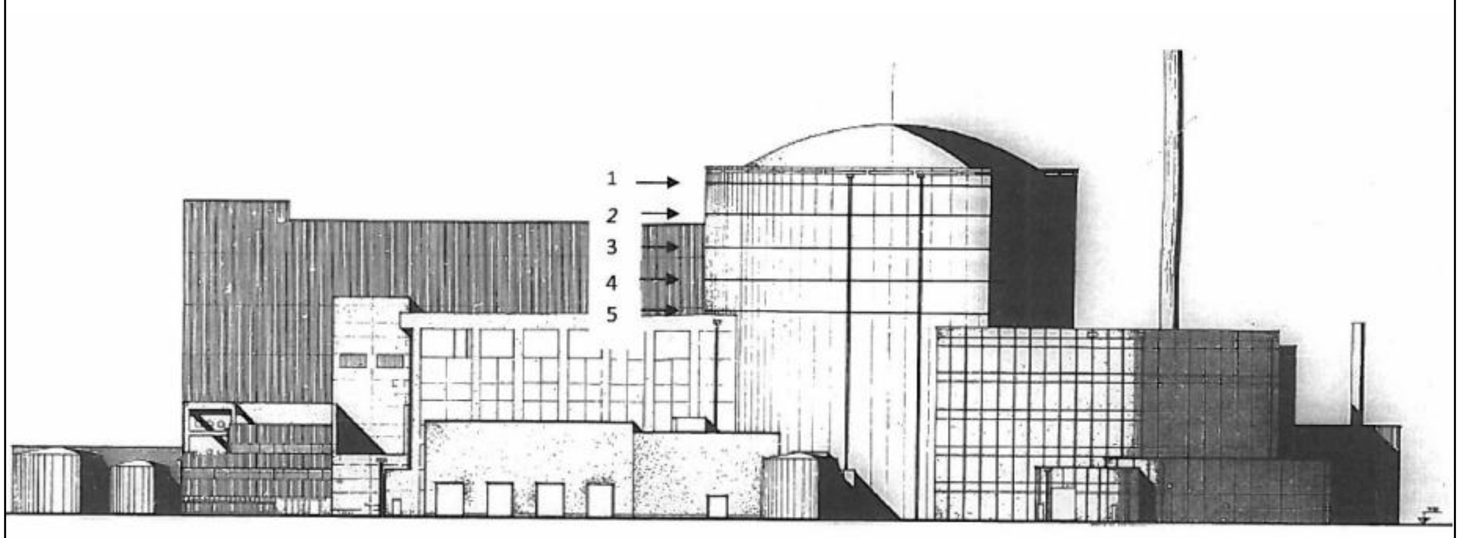
Crushed limestone	For 1 m^3 of concrete
Gravel 7/20	940 kg
Gravel 2/7	220 kg
River sand	615 kg
CEM-III (HK 400: OPC-BFS)	375 kg
Water	185 kg
Water-to-cement ratio	0.49
Compression strength on cube	350 kg/cm^2

Concrete degradation mechanisms

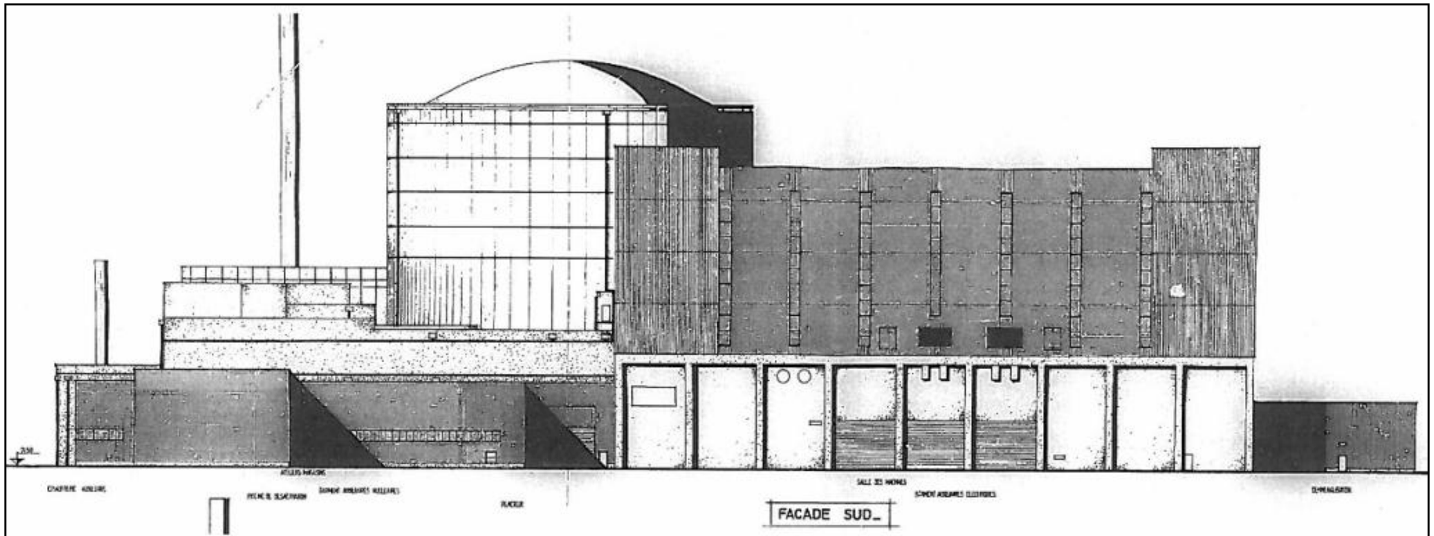
Spalling of concrete above the corroded rebars on the external face of the containment building of Tihange 2 (observed during in-service inspection, 2008)

Possible multiple causes:

- **Insufficient concrete cover** above the rebars
- **Advanced carbonation front**:
  - rebars corrosion followed by spalling of concrete surface
  - Particularly along cracks
- **Alkali-silica reaction (ASR)**
  - Reduced mechanical properties of the concrete, + white traces and presence of alkaline silica gel

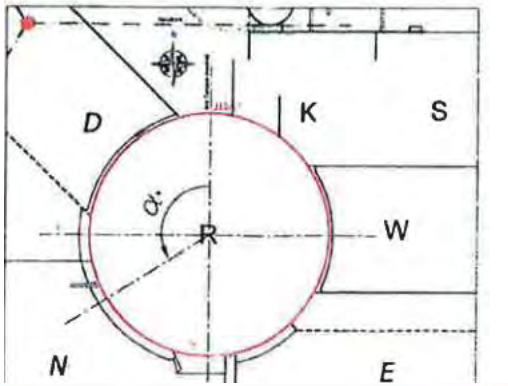


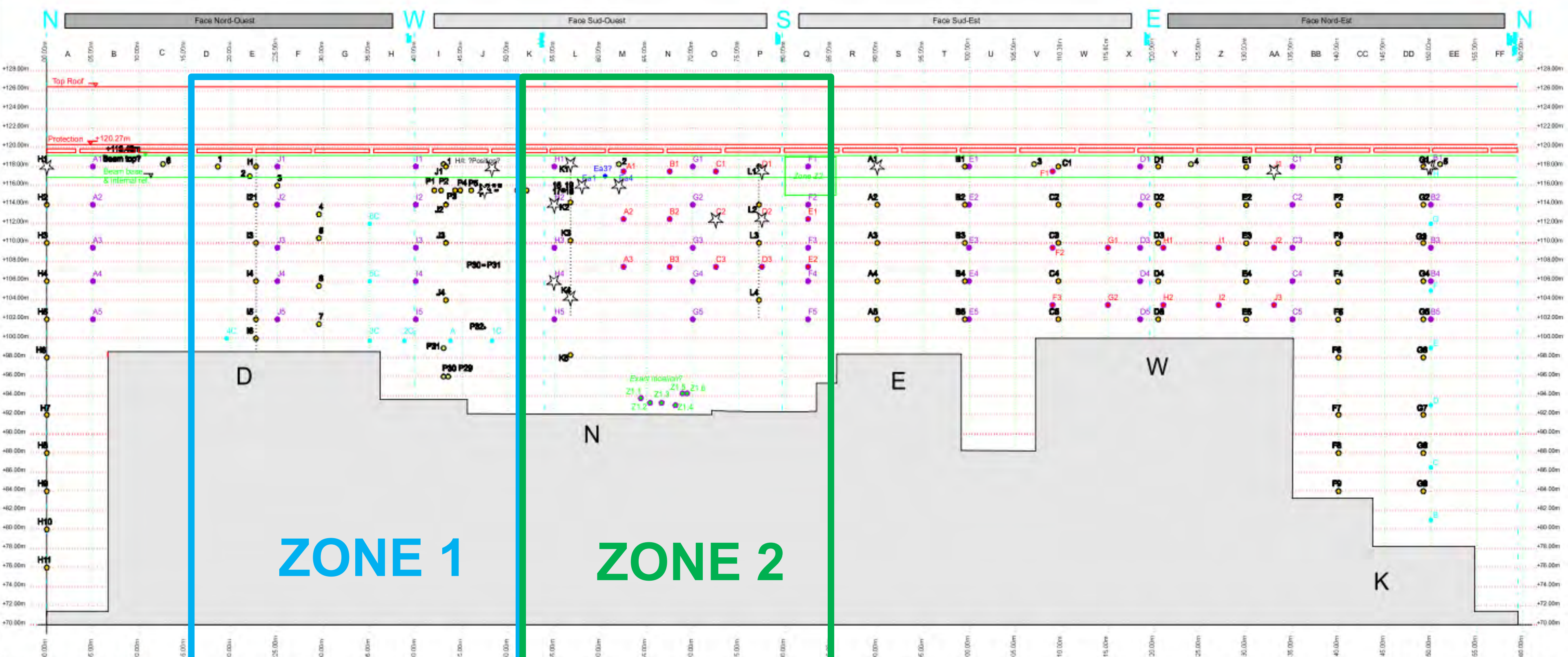
FACADE NORD DE TIHANGE 2



FACADE SUD DE TIHANGE 2

Unfolded view of the surface of the containment building





Campagnes d'échantillonnage & rapports de tests 2009-2012

A	(14 samples) LMC/09/082/	2009-1
A1	(55 samples) LMC/09/082/2	2009-2
Ea1	(4 samples, INTRADOS) LMC/09/082/3	2009-3
A1	(26 samples) GMC/10/002/8 - LMC/11/182/1 & LMC/11/182/2	2011-1
Z1	(14 samples - Test repairs) GMC/10/002/12 - LMC/11/182/3	2012-1
Z1	(7 samples) - After repairs GMC 10/002/13 - LMC/11/182/4	2012-2
A1-L1	GEOS 2012 Extrados - Pxx = Intrados	2012-3
1-6	OXAND 2012 Pxx = Intrados(?)	2012-3

Positions incertaines à lever: 1C (GEOS), PXX intrados OXAND & GEOS, P20-29 Dôme GEOS

Concrete degradation Tihange 2 NPP

Three phases in the repair campaign

First phase in 2010-2011:

- **Zone 1** (¼ of the perimeter) was repaired
  - Non-reinforced shotcrete was applied.
  - Assessment in 2012 revealed that the adhesion of this shotcrete layer is lower than expected.

Zone 1, already repaired

Three phases in the repair campaign

Second phase in 2012:

- **Zone 2** (¼ of the perimeter) where concrete was being scrapped
  - Will be reinforced with shotcrete
  - Installation of a steel grid (not shown in the picture) that will be anchored in the base concrete is envisaged.

Zone 2, scrapped on 1/4

Three phases in the repair campaign

Third phase in 2013:

- **Zone 3** (the remaining zone) is still to be assessed:
  - Less severe degradations
  - Repair method still to be defined

Zone 3, not yet repaired

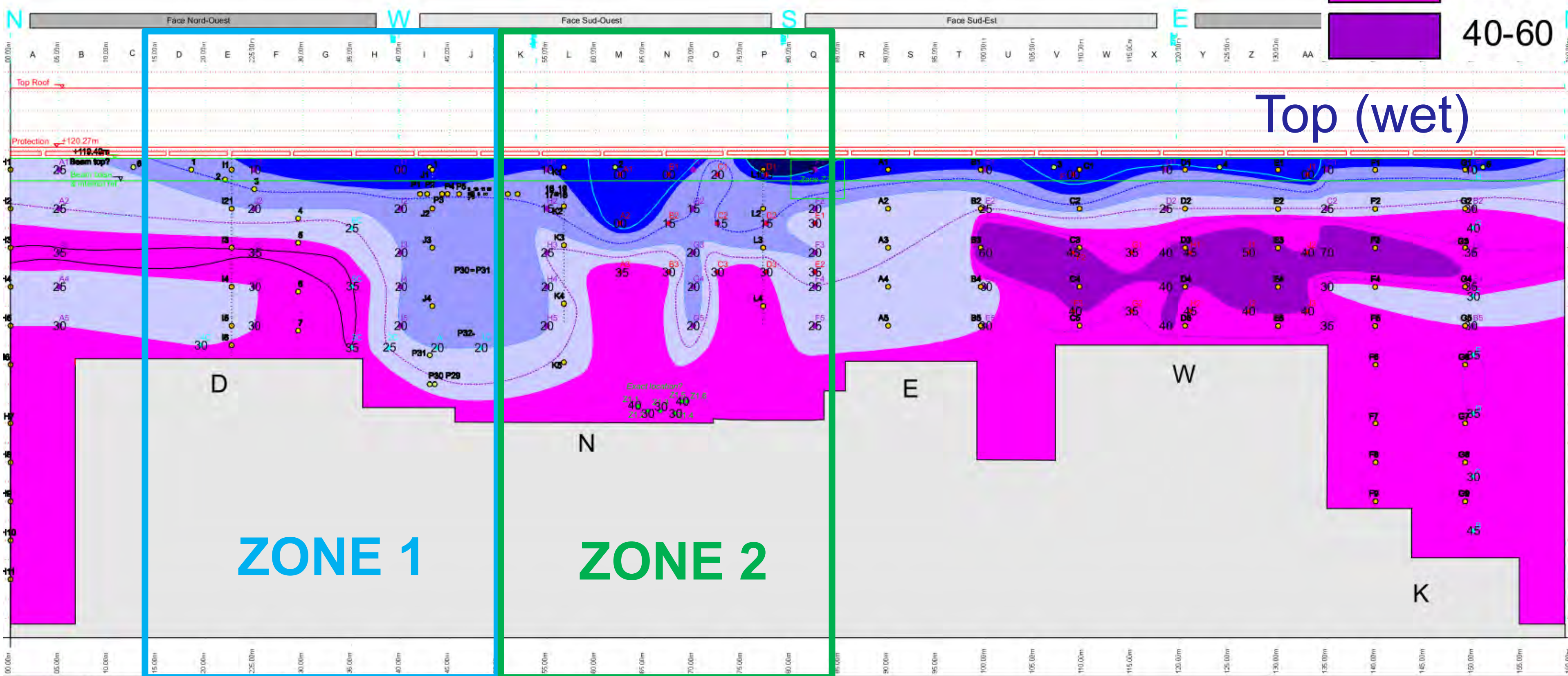
Results of the assessment in 2012

- Unexpected size of the degraded surface and the affected depth
  - 50% of the surface
  - With a maximum depth of 30 cm on 5% of the surface
- New analyses confirmed the predominance of ASR. The ASR is mainly observed at the top of the containment (upper 6 m). ASR and carbonation are two degradation mechanisms mutually exclusive.
- Measured porosity of the concrete: between 6 and 9%. Target value is 5%.
- New assessment of the structural integrity performed by the operator.

Carbonation depth seems to depends on exposure to moisture (unfolded view of the surface)

Depth /surface (mm)

- 00-10
- 10-20
- 20-30
- 30-40
- 40-60



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1	(7 samples) - After repairs GMC 10/002/13 - LMC/11/182/4	2012-2
1-L1	GEOS 2012 Extrados - Pxx = intrados	2012-3
6	OXAND 2012	2012-3

CARBONATATION: CARTE D'ISOVALEUR

(Avant réparation, hors de l'influence des fractures augmentant ces valeurs)

->Données fiables et interprétatbles (zonations évidantes)
-> Tendances verticales :faibles valeurs localisées sur la couronnes supérieure
-> Limites verticales (associées aux armatures) importantes dans la zone SW
-> Tendances en focntion de l'azimut :
- Fortes valeurs clairement localisées à L'Est
- Faibles valeurs localisées à l'Ouest

Depth /surface (mm)

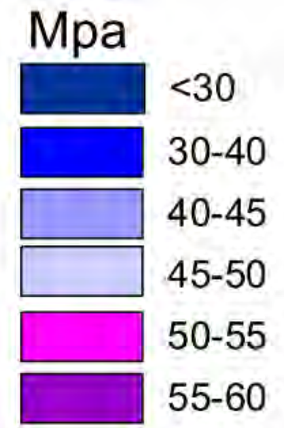
- 00-10
- 10-20
- 20-30
- 30-40
- 40-60

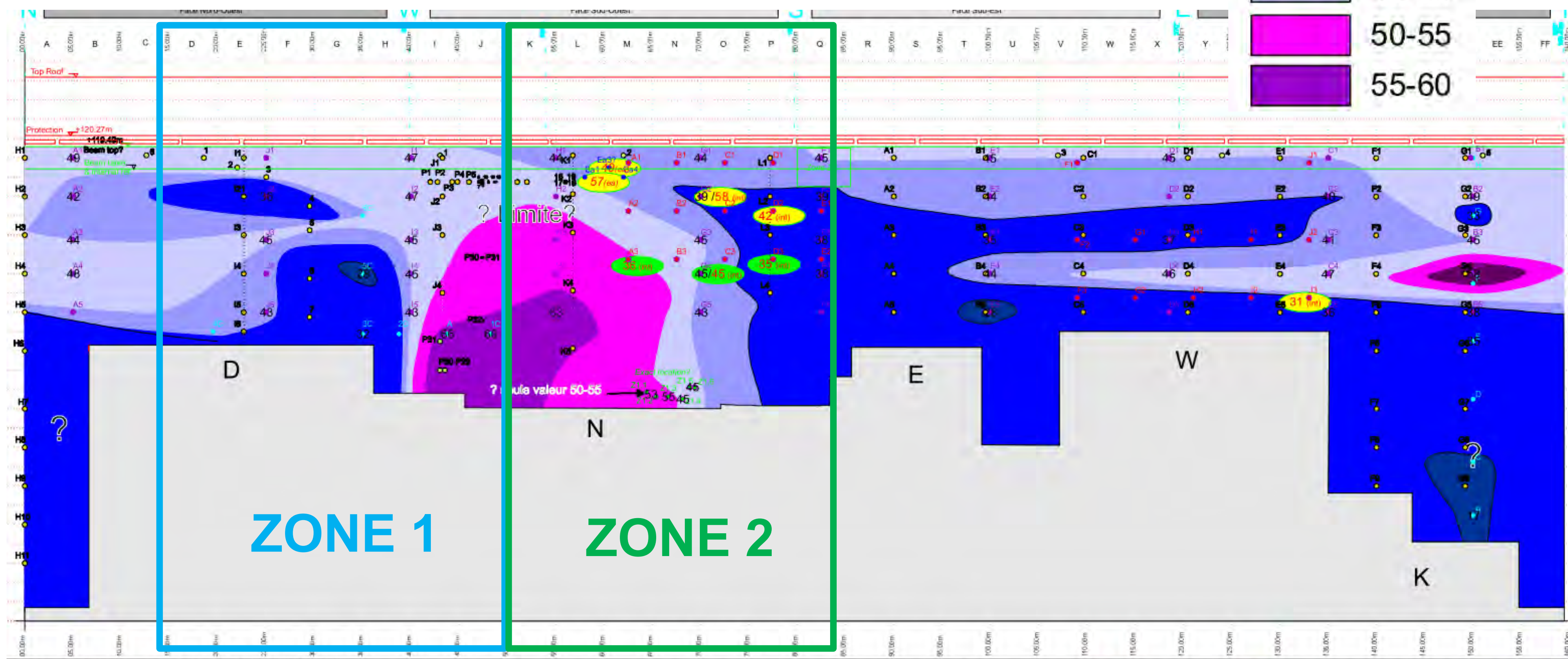
Bottom (dry)

Concrete carbonation

- Carbonation depth seems to be **correlated** with the **moisture exposure** (relative humidity)
- More carbonation in the drier zones:
  - At the bottom of the containment building
  - In the East direction (less exposed to wind/rain)
- Carbonation is **anticorrelated with ASR**
  - Less carbonation at the top (6 m) where ASR is dominant (wetter conditions)

Concrete compression strength (correlated with carbonation)





Campagnes d'échantillonnage & rapports de tests 2009-2012

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RESISTANCE EN COMPRESSION : CARTE D'ISOVALEUR

Test suivant EN12390-3 (2003) -
Diamètre carotte 75mm, longueur 80mm, vitesse 1N/mm2.s
(inclus les faiblesses initiales en surface => conservatif)

-> Zonation peu contrainte (densité de données suffisantes)
-> Tendances horizontales clairement dominantes, excepté dans la zone SW
-> Hautes valeurs à limites verticales dans la zone SW
-> Correlation avec les anomalies de carbonatation
- Limites verticales zone SW : Forte compression <-> Faible carbonatation
- Limite horizontale sone NE : Faible compresion <-> Forte carbonatation

G3 38 / 58 Valeurs en Mpa : Externe / Interne

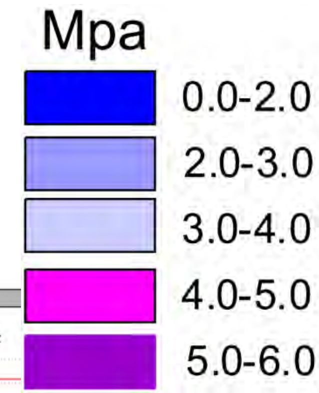
Mpa
<30
30-40
40-45
45-50
50-55
55-60

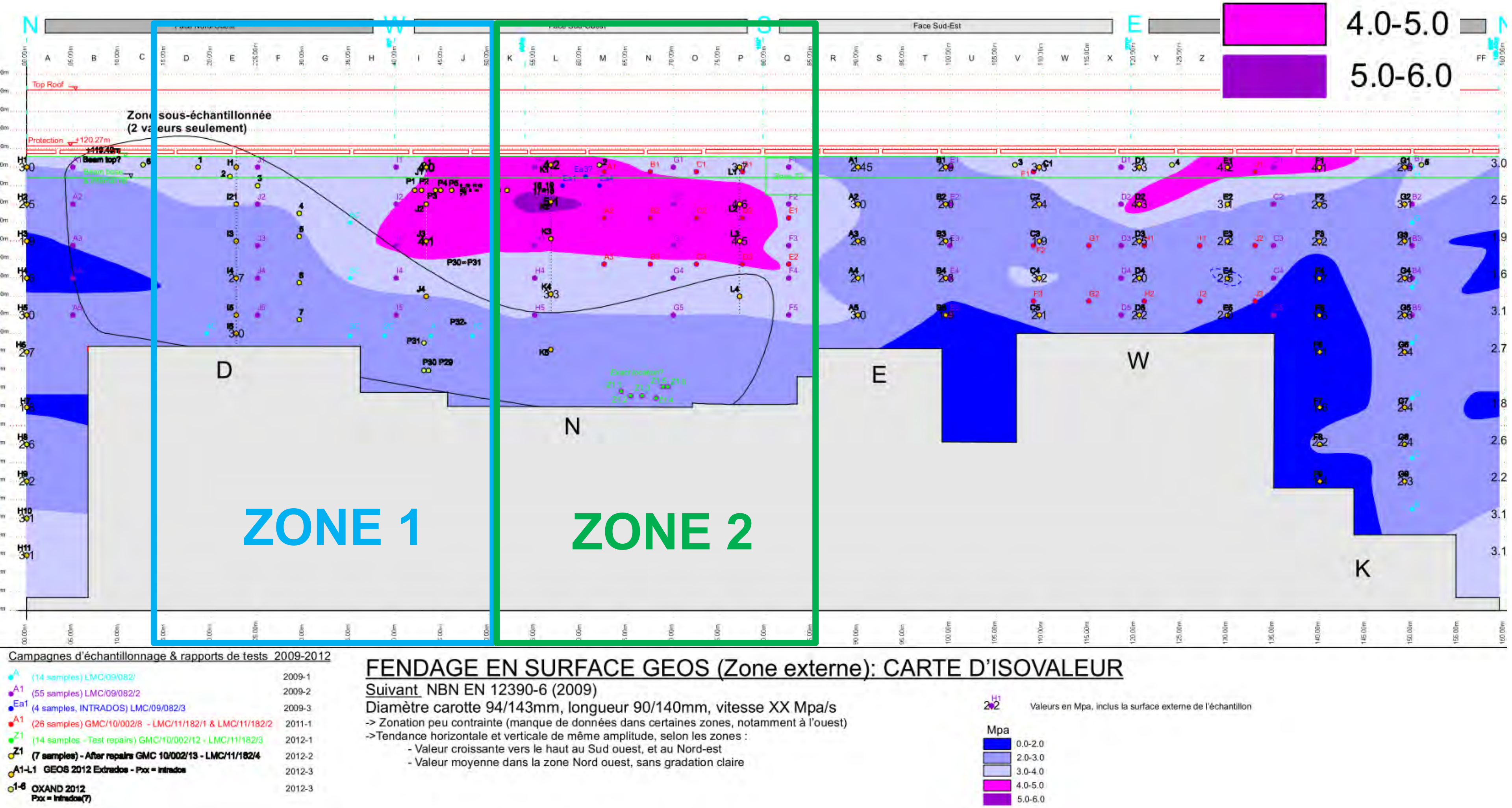
Remarque : Tous les échantillons n'ont pas les mêmes faiblesses de sub-suface. L'interpretation doit prendre en compte cette hétérogénéité
Le détail des fractures initiales ne peut être argumenté (données perdues)

39 /58 (int) Valeurs augmentant vers l'intérieur ou dans l'espace annulaire

7.0/7.0 Valeurs égale où diminuant vers l'intérieur

Indirect tensile strength test – external face of the concrete cores



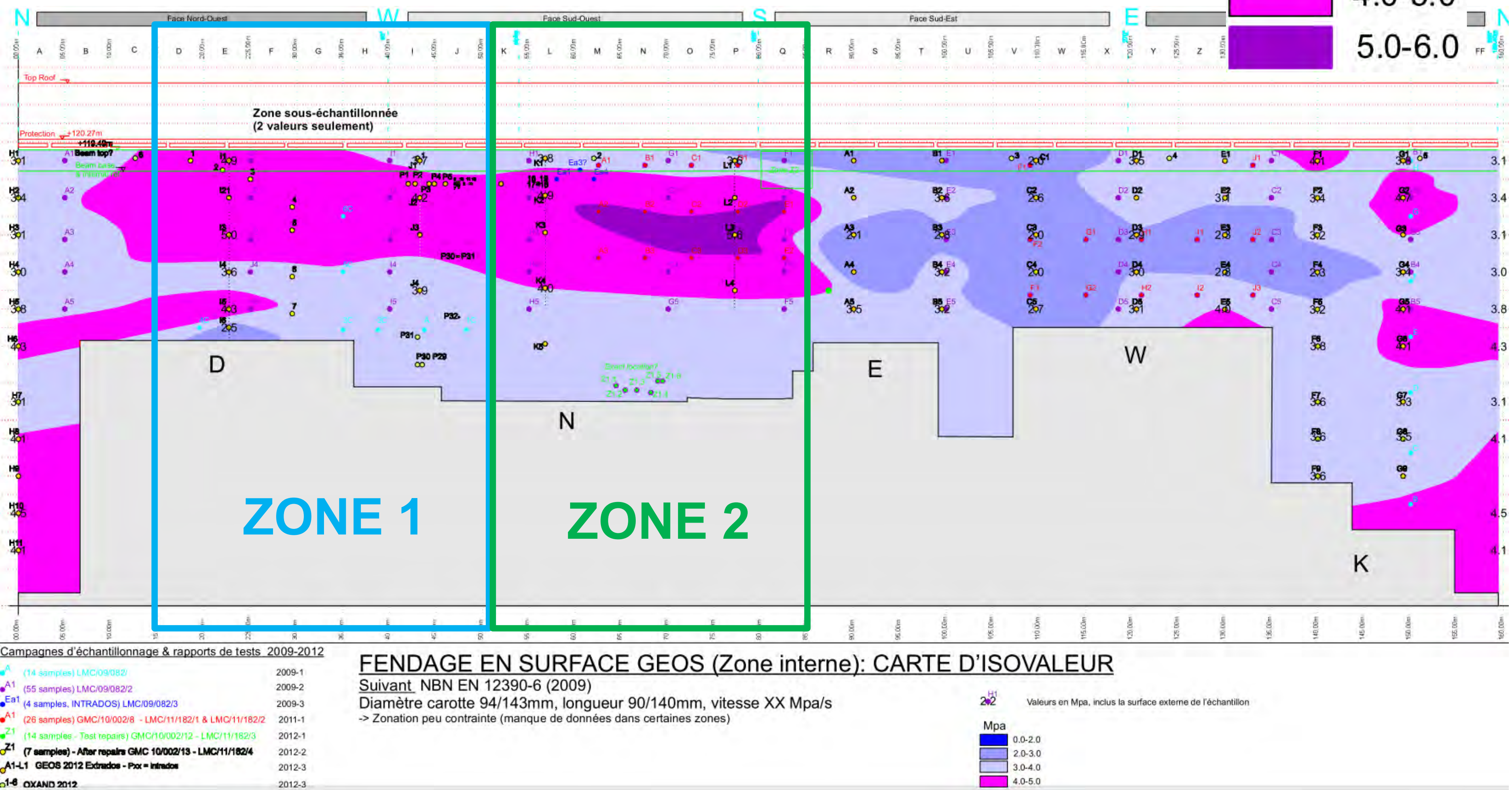


Source: Electrabel and Oxand

Indirect tensile strength test – internal face of the concrete cores

Mpa

- 0.0-2.0
- 2.0-3.0
- 3.0-4.0
- 4.0-5.0
- 5.0-6.0



info@belv.be
www.belv.be

ASR is one of the main issue

Root causes of ASR identified as of December 2012 are:

1. Presence of **reactive chalcedony (SiO_2 am)** in the river sand used as fine aggregate
2. **Water infiltrations** from the top of the containment building due to standing water and privileged exposure to poor meteorological conditions that could have favoured ASR
3. **Secondary ettringite formation** also observed in air bubbles in cement:
   - Possible concomitant occurrence of delayed ettringite formation (**DEF**) ? (to be confirmed by new analyses)

Complementary tests

Tests planned on concrete cores on all zones in 2013:

- Residual swelling capacity (ASR)
- Uranyl acetate test (ASR)
- Sensitivity check on the aggregates
- More detailed compression and traction tests
- Scanning electron microscopy (SEM)
- Ultrasonic analysis of the concrete
- Petrographic analysis of the limestone aggregates to determine their origin (Tournaisian or Visean limestone ?)
- Search for delayed ettringite formation (DEF)

Acknowledgment

Data and technical information have been provided by the operator of Tihange NPP: **Electrabel** and his different contractors

- Tractebel
- ULg
- Geos
- Oxand
- (...)

Thank you for your attention.

Any questions?

I am looking forward to hearing any Operating Experience Feedback about these issues.

Related issues:

- IN 2011-20: Concrete degradation by alkali-silica reaction, Seabrook NPP
- IN 2013-04: Davis-Besse shield building, construction during blizzard of 1978

Alkali–silica reaction (ASR)

The alkali–silica reaction (ASR) is a reaction which occurs over time in concrete between the highly alkaline cement paste and reactive non-crystalline (amorphous) silica, which is found in many common aggregates.

The ASR reaction is the same as the pozzolanic reaction, which is a simple acid-base reaction between calcium hydroxide, also known as Portlandite, or ($Ca(OH)_2$), and silicic acid (H_4SiO_4, or $Si(OH)_4$).

ASR can cause serious expansion and cracking in concrete, resulting in critical structural problems that can even force the demolition of a particular structure.

The mechanism of ASR causing the deterioration of concrete can be described in four steps as follows:

- The alkaline solution attacks the siliceous aggregate, converting it to viscous alkali silicate gel.
- Consumption of alkali by the reaction induces the dissolution of Ca^{2+} ions into the cement pore water. Calcium ions then react with the gel to convert it to hard C-S-H.
- The penetrated alkaline solution converts the remaining siliceous minerals into bulky alkali silicate gel. The resultant expansive pressure is stored in the aggregate.
- The accumulated pressure cracks the aggregate and the surrounding cement paste when the pressure exceeds the tolerance of the aggregate.

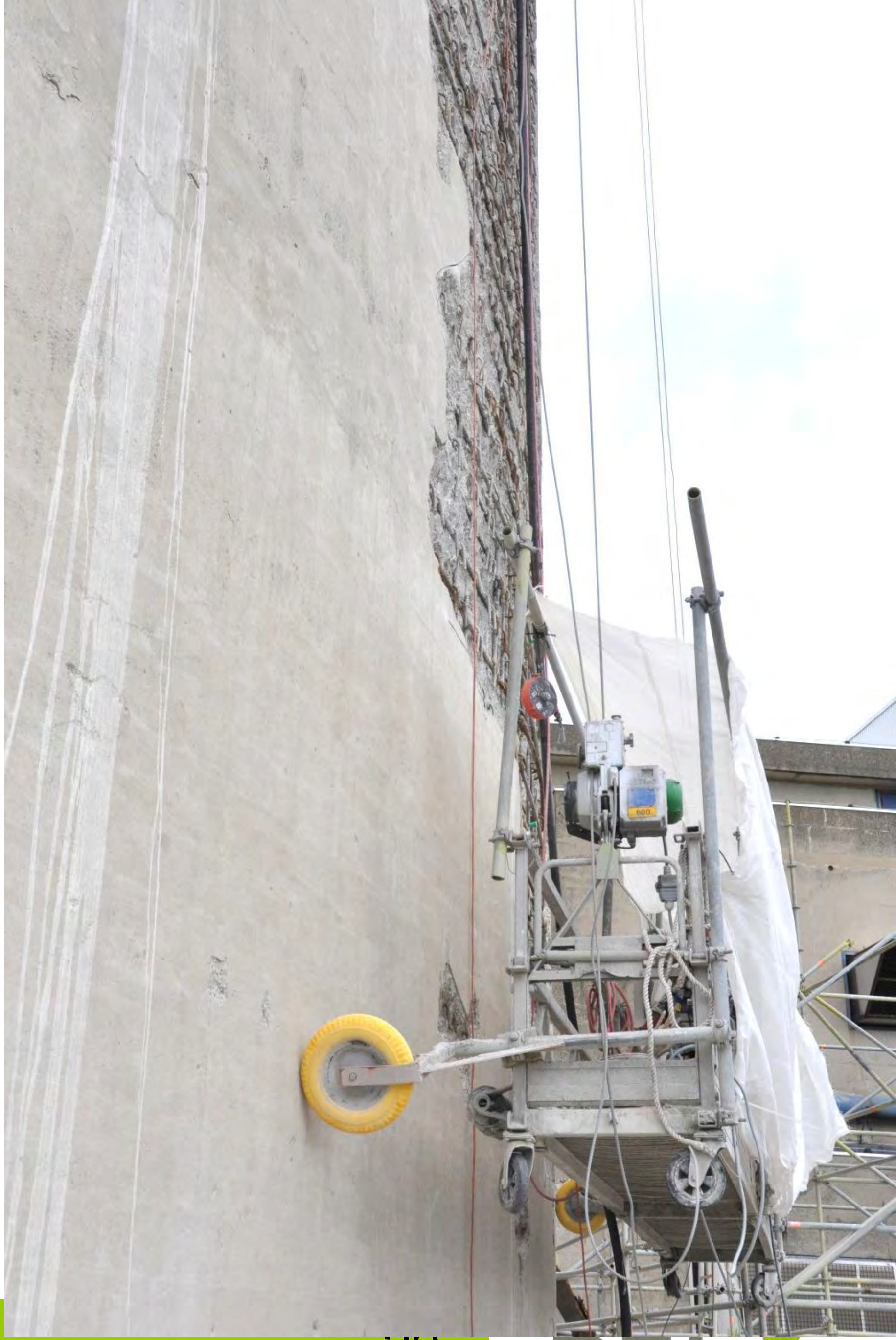
Carbonation

Carbonation or Carbonatation is a slow process that occurs in concrete where portlandite (calcium hydroxide) in the cement reacts with carbon dioxide from the air and forms calcium carbonate.

The water in the pores of Portland cement concrete is normally alkaline with a pH in the range of 12.5 to 13.5. This highly alkaline environment is one in which the steel rebars are passivated and are protected from corrosion. According to the Pourbaix diagram for iron, the metal is passive when the pH is above 9.5.

The carbon dioxide in the air reacts with the OH^- anions in the cement and makes the pore water less alkaline, thus lowering the pH. Carbon dioxide starts to carbonate the cement in the concrete from the moment the structure is made. This carbonation process starts at the surface, then slowly moves deeper and deeper into the concrete. The rate of carbonation is dependent on the relative humidity (RH) of the concrete - a 50-60 % relative humidity being optimal. If the structure is cracked, carbon dioxide from the air penetrates more easily and deeper into the concrete.

Eventually carbonation leads to corrosion of the rebars and damages to the construction.



Concrete degradation Tihange 2 NPP

Status of the nuclear energy in Belgium

In principle, no limit in time in the licence of Belgian NPP.

Periodic Safety Reviews every 10 years.

However in 2003, a phase-out law limits the life of NPPs to 40 years except in case of force majeure.

By government's decision in 2012, phase-out of nuclear energy is confirmed with scheduled shutdown of:

- Doel 1&2 in 2015
- Doel 3 in 2022
- Tihange 2 in 2023
- Tihange 1, Tihange 3 and Doel 4 in 2025
  - Extension of operation of Tihange 1 by 10 years

Doel			Tihange		
#	MWe	Connected	#	MWe	Connected
1	433	02/1975	1	962	10/1975
2	433	12/1975	2	1008	02/1983
3	1006	10/1982	3	1046	09/1985
4	1039	07/1985	**TOTAL : 5930 MWe**		

Currently 55% of the electricity is produced through nuclear power

Status of the reactor pressure vessels

As of April 2013, two Belgian reactors are shut down because of the detection of defects (hydrogen flakes) in their pressure vessels:

Doel-3: ~ 8000 defects

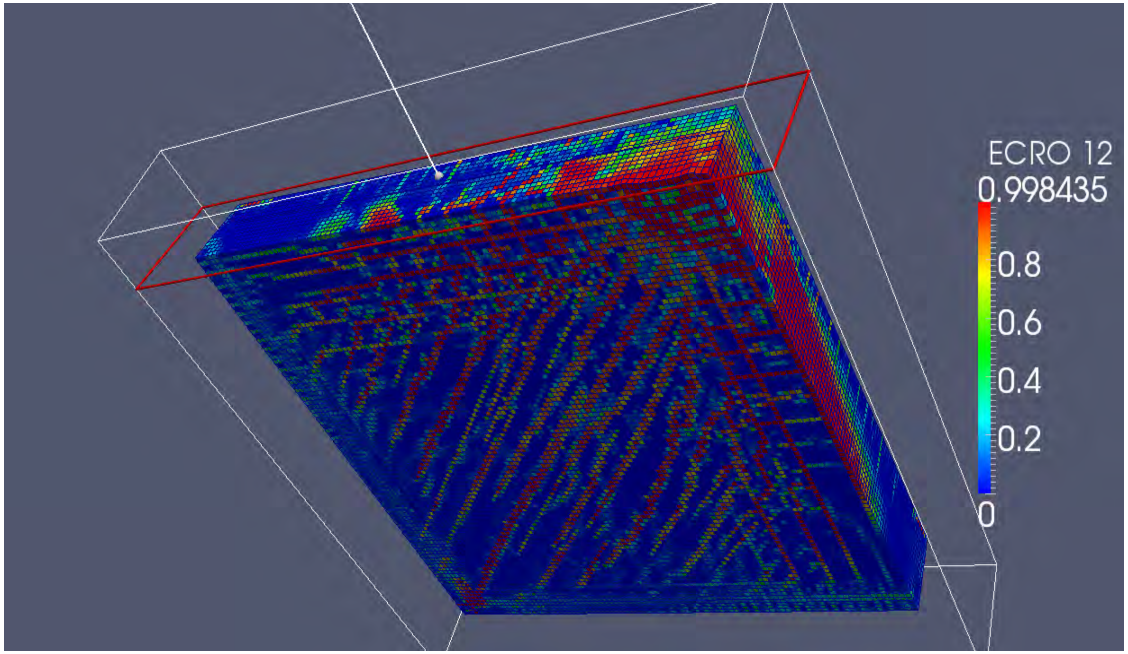
Tihange-2: ~ 2000 defects

Additional control tests are still on-going to characterise these defects and to assess the safety of these reactors.

More information are available on the website of the FANC: http://www.fanc.fgov.be/

DE LA RECHERCHE À L'INDUSTRIE

18th WGIAGE CONCRETE SUB-GROUP MEETING



Daniel GUILBAUD

APRIL 8-9TH, 2013
PARIS, FRANCE

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CONTENTS

IRIS 2012 Contribution to flexural test

Plastic damage model

Current R&D activity on mechanical behavior of concrete

Anisotropic damage model

NUCPERF 2012

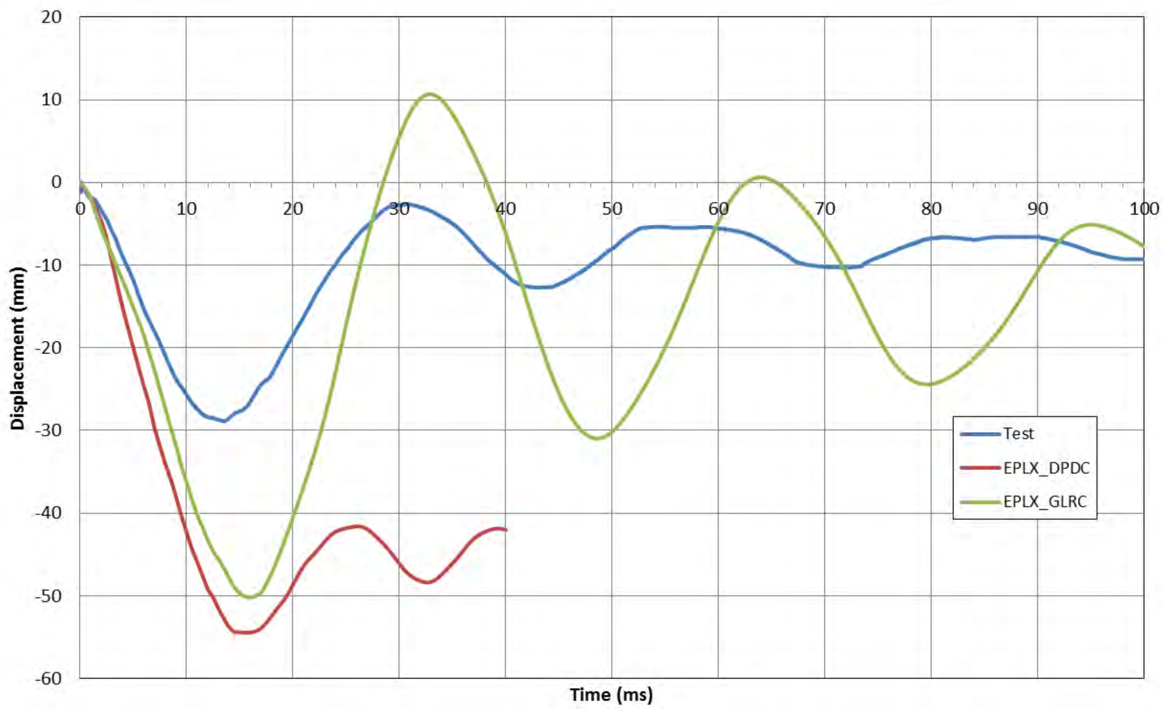
IRIS 2012 CONTRIBUTION TO FLEXURAL BENCHMARK

- EUROPLEXUS: A general finite element code for fast transient analysis of structures (with FSI),
  - Fully explicit
  - Kinematic relations can be treated implicitly (Lagrange multipliers)

- 2010 model : GLRC model: A Global Law for Reinforced Concrete , developed by EDF, was used to deal with flexural tests, because it is well suited to industrial approach.

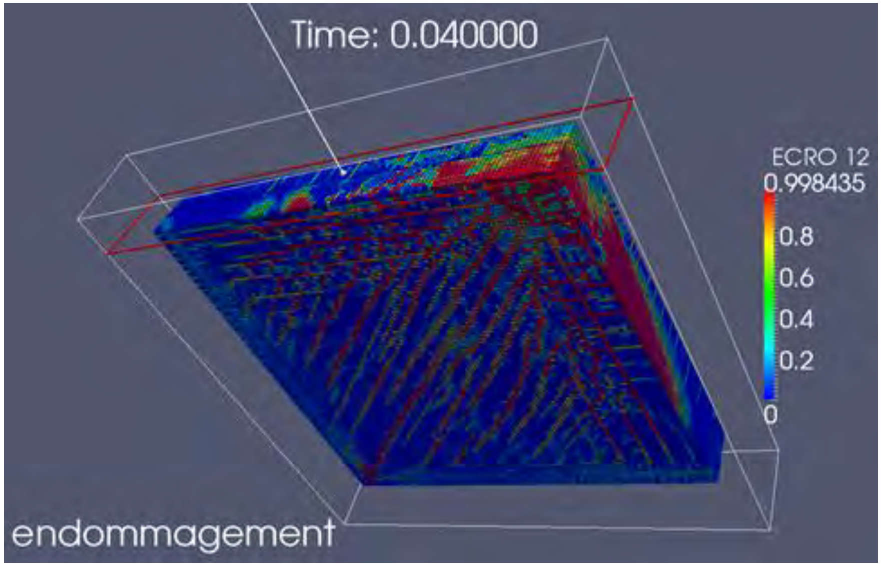
- 2012 model : DPDC : Dynamic Plastic Damage Concrete model
  - Cap model
  - Damage in tension and compression
  - Rate effect not included yet

COMPARISON WITH THE FLEXURAL TEST (1)

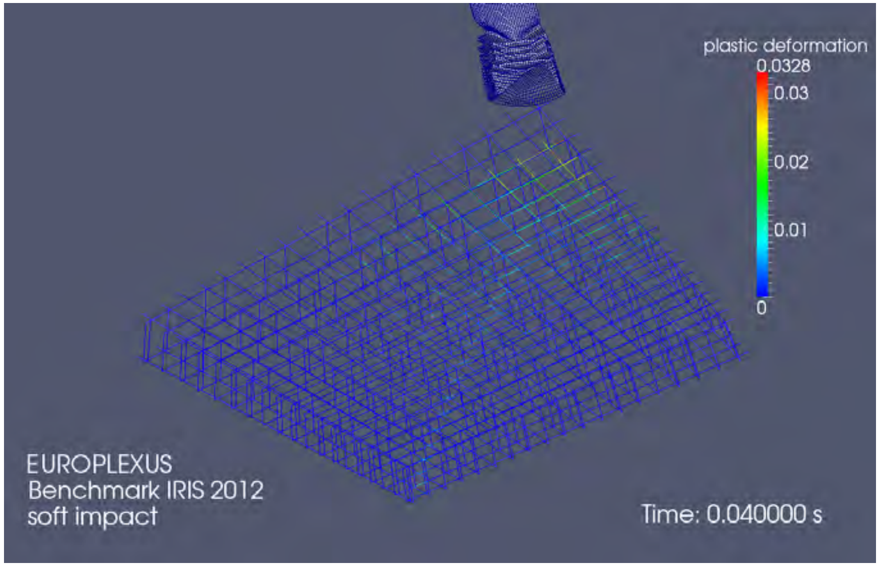


⇒Good agreement between the two models for maximum displacement.
⇒After the first peak, signatures are different probably because cracks never close in DPDC model.
⇒The two models overestimate the displacement of the slab because rate effects are missing.

COMPARISON WITH THE FLEXURAL TEST (2)



General view of damage in the slab (with a cross section showed by the red plane)



Plastic deformation of rebars is localized on back side diagonal

IRIS 2012 LESSONS LEARNED - PERSPECTIVES

- Lessons learned :
  - The DPDC model seems robust, time calculation is reasonable. The plasticity model is less obvious to implement that it seems at first glance.
  - In compression, there is a strong correlation between the $\mathcal{R}$ function and the shear band (strain localization). This point must be studied deeply.

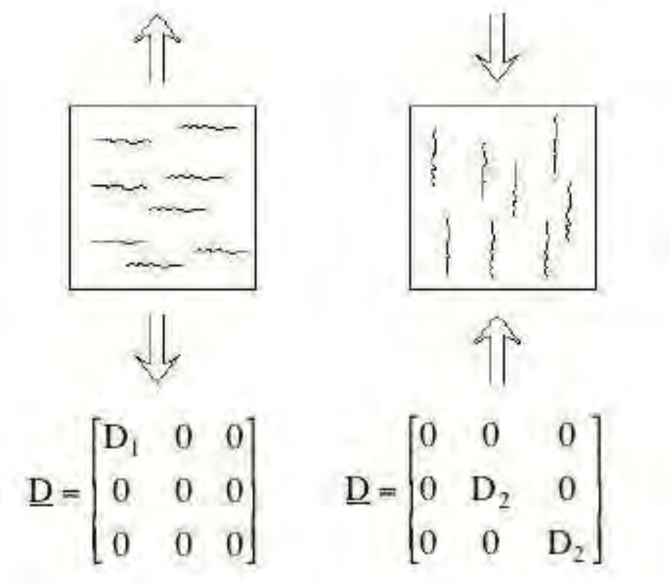
- Perspectives :
  - It seems valuable to complete the model with strain rate effect, crack closure and erosion.

- Open questions:
  - Is it necessary to improve strain localization if we are interested only in the global response of a structure and not in detailed cracking? And, if it is the case, how to do that?
  - Is it necessary to implement a steel concrete bond model?
  - Is an isotropic model sufficient to deal with shear ?
  - …

CURRENT R&D ACTIVITY : DADC model

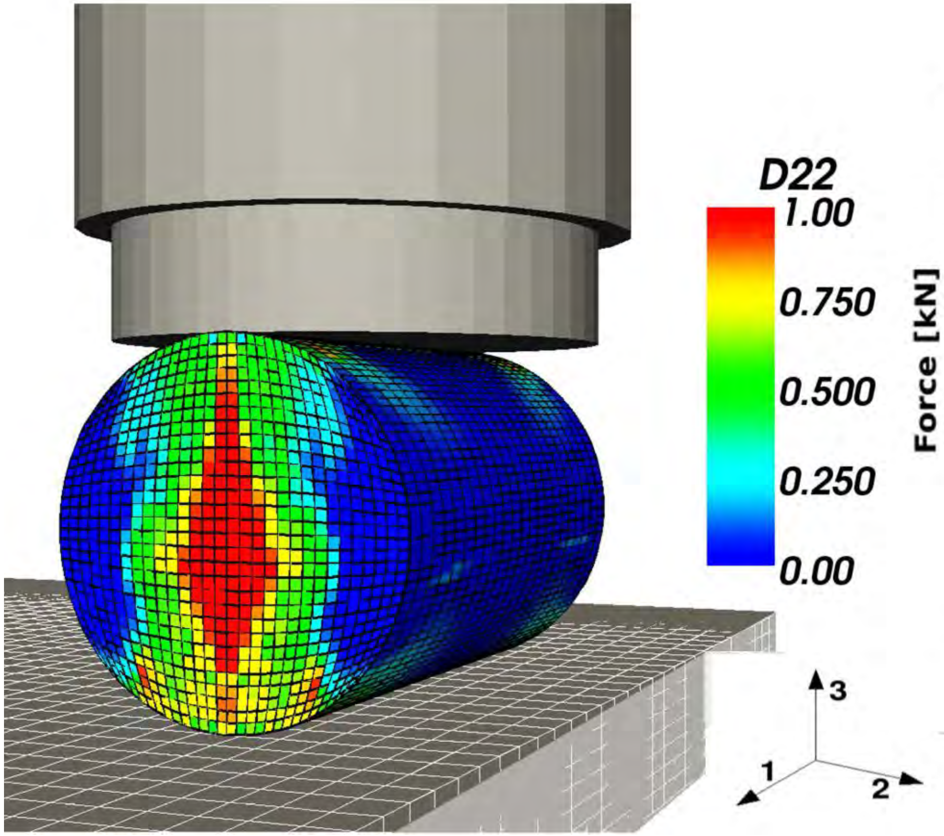
Dynamic Anisotropic Damage Concrete model for concrete under impact (thesis with ENS Cachan)

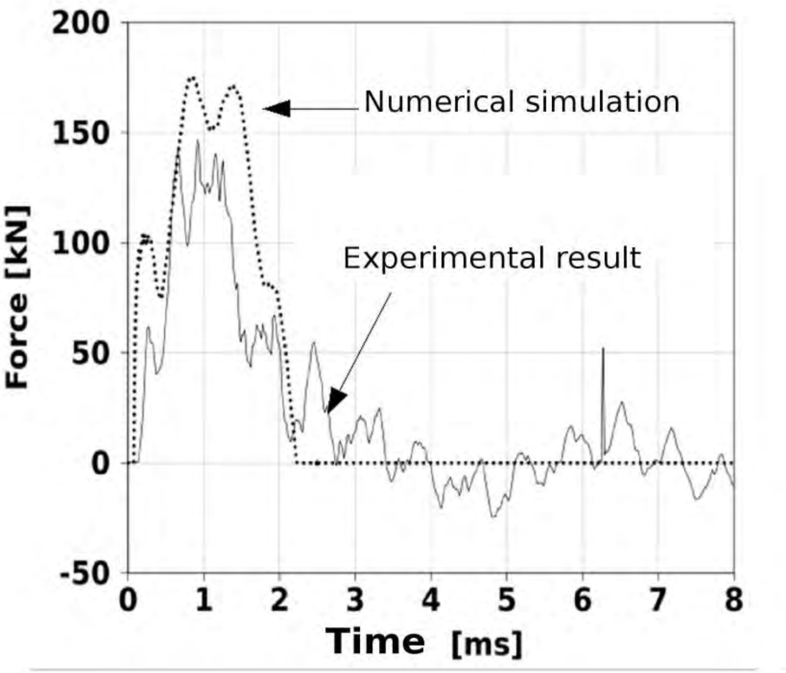
- Damage state represented by a second-order tensor D



- Damage rate proportional to the positive part of the strain tensor
- Local with visco-delay damage law or nonlocal
- Special procedure for the numerical control of rupture

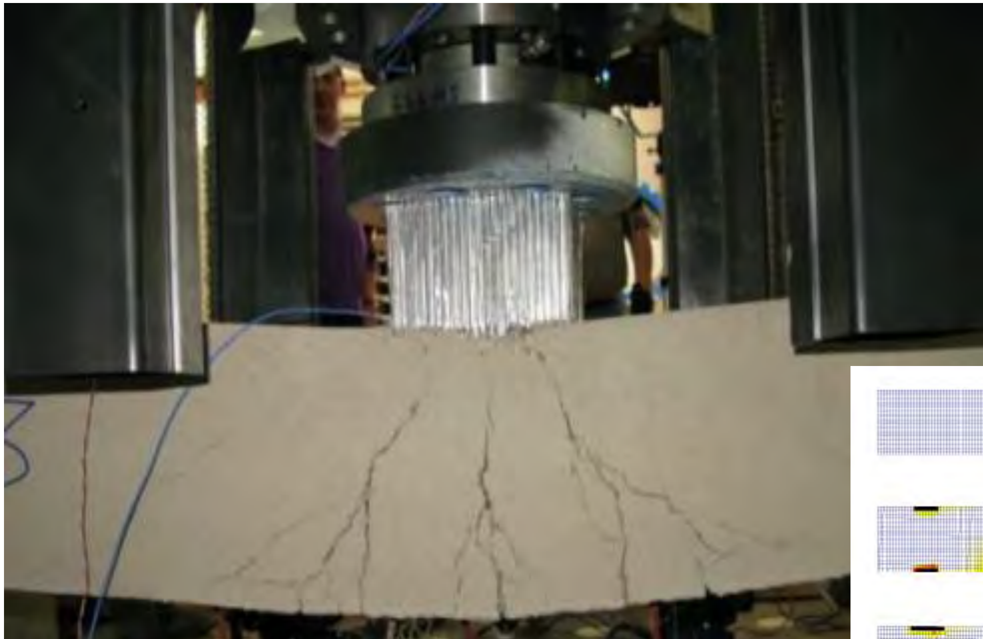
DADC model



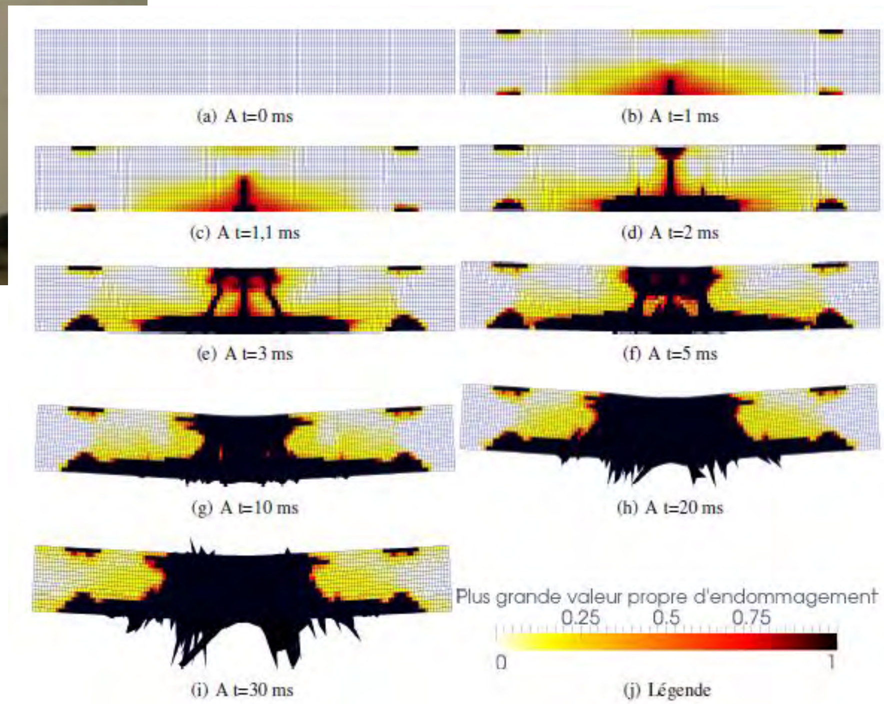


Damage map and impact force for dynamic Brazilian test

DADC : comparison with soft impact on a beam



Cracks are well reproduced at the beginning of impact …



But damage propagates too much in the neighbourhood of steels

=> Problem with tangential efforts

Organisation

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NUCPERF 2012

EFC Event 351

Long-Term Performance of Cementitious Barriers and Reinforced Concrete in Nuclear Power Plant and Radioactive Waste Storage and Disposal

12 – 15 November 2012

Cadarache - France

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<tr><td>8h50-9h15</td><td rowspan="3">Session 5 (3) - Corrosion</td></tr>
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<tr><td>10h05-10h30</td><td>break</td><td colspan="2">break</td><td>break</td></tr>
<tr><td>10h30-10h55</td><td rowspan="4">Poster Session (10)</td><td rowspan="4">Session 4 (4) -Test approach</td><td rowspan="4">Session 2 (4) - Performance and safety assessment</td><td rowspan="2">Session 5 (2) - Corrosion</td></tr>
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<tr><td>17h35-18h</td><td colspan="2" rowspan="2">Conference dinner</td></tr>
<tr><td>18h-18h25</td></tr>
</table>

71 participants

33 oral présentations

10 posters

USB Stick containing full papers distributed to the participants

At this moment, peer review of papers to be published in European Physics Journal

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T. +33 (0)1 69 08 23 60 | F. +33 (0)1 69 08 76 19

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ÚJV Řež, a. s.

COMPARISON OF CONTAINMENT CALCULATED EIGENFREQUENCIES WITH EXPERIMENTALLY OBTAINED VALUES

L.Pečínka – ÚJV Řež, a.s. Czech Republic

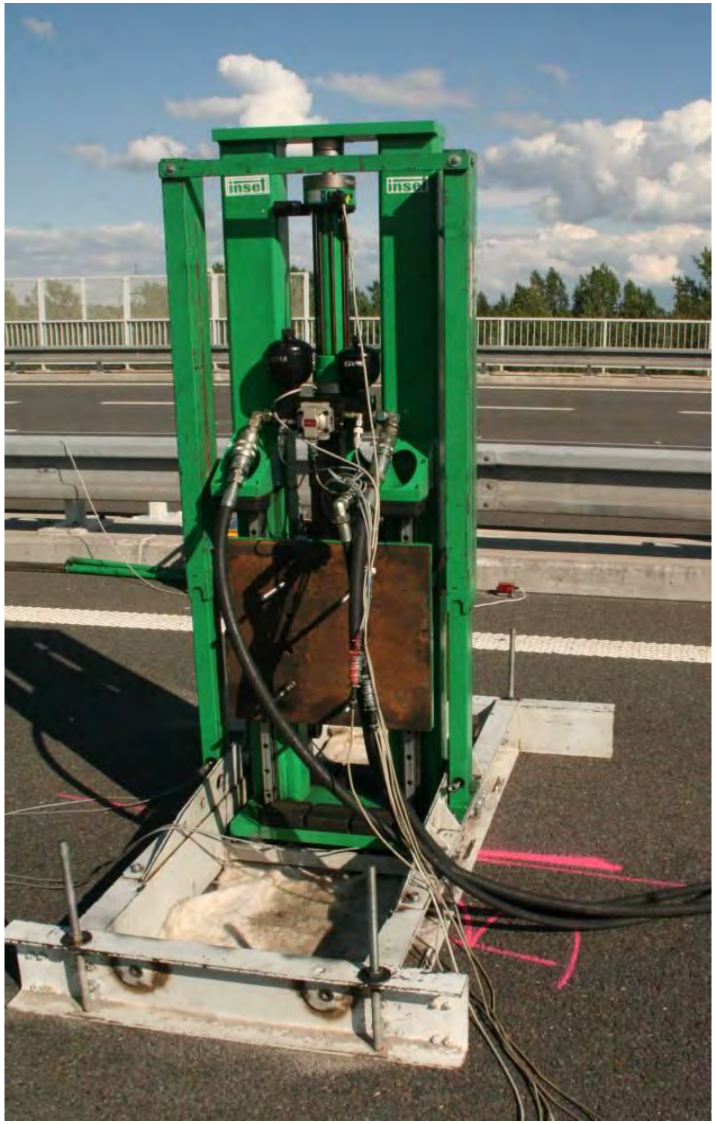
J.Stainbruch – INSET, Czech Republic

- OBJECTIVE
- THE PURPOSE OF THIS TASK IS VERIFICATION OF ANALYTIC SOLUTIONS OF CONTAINMENT FREQUENCY SPECTRA WITH EXPERIMENTALLY OBTAINED VALUES
- APPLICABLE TO PSAR AND FSAR, CHAPTER 3.7
- TESTED OBJECT: DECOMMISIONED NPP

Hydraulic vibration exciter

- vibration exciter can be used for measuring structural natural-frequencies and natural-vectors in situ
- experimentally measured values can be compared with the results of numerical modeling (measuring mass and stiffness of a whole structure)
- repeated measurements (with time period 5 - 10 years) can be used for the monitoring of temporal changes and for the estimation of concrete degradation (overall loosing of stiffness)

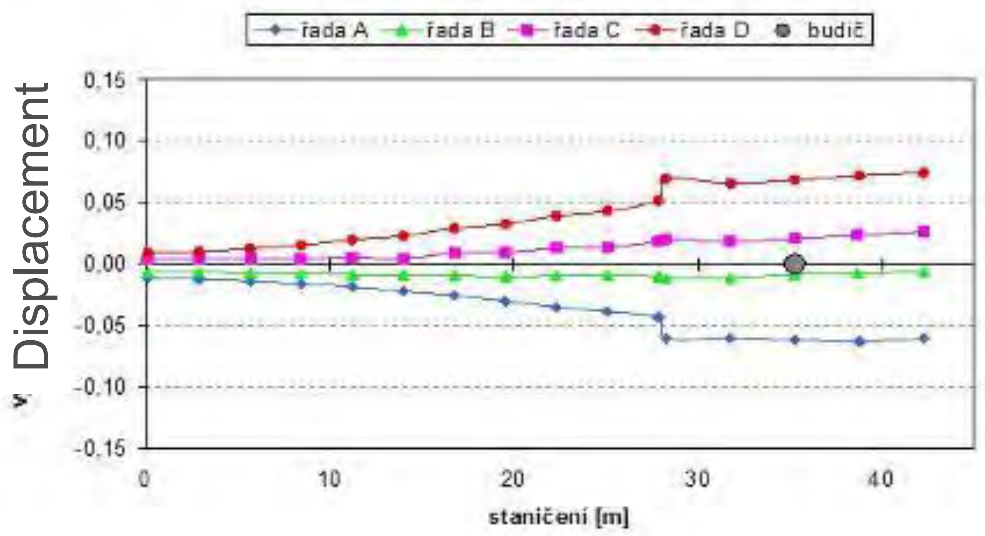
Hydraulic vibration exciter



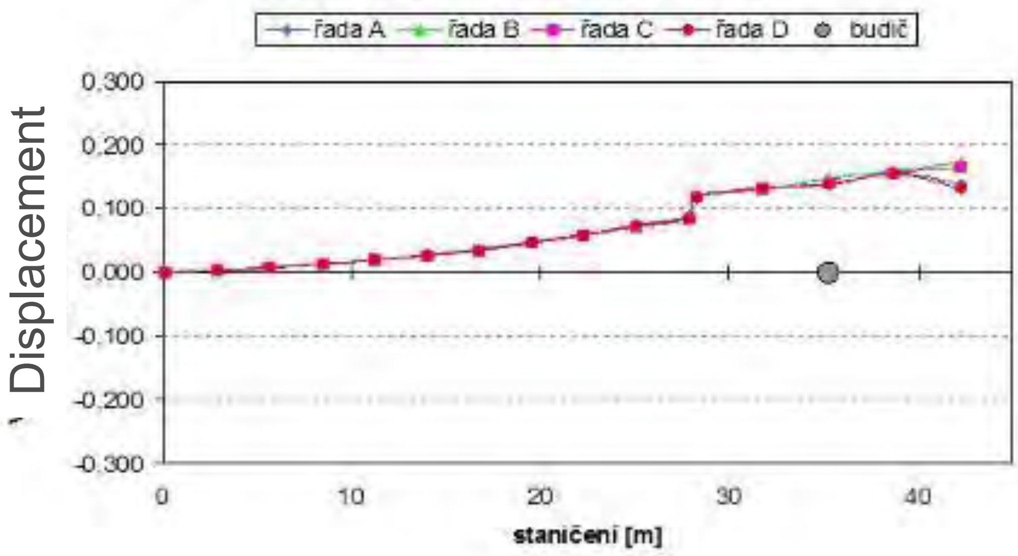
- enable horizontal or vertical movement
- frame 0.8 x 2.0 m
- frequency 1.0 – 120 Hz
- maximal force ± 10.0 kN
- weight of frame 750 kg, weight of moving mass up to 900 kg
- weight of attached engine 1200 kg

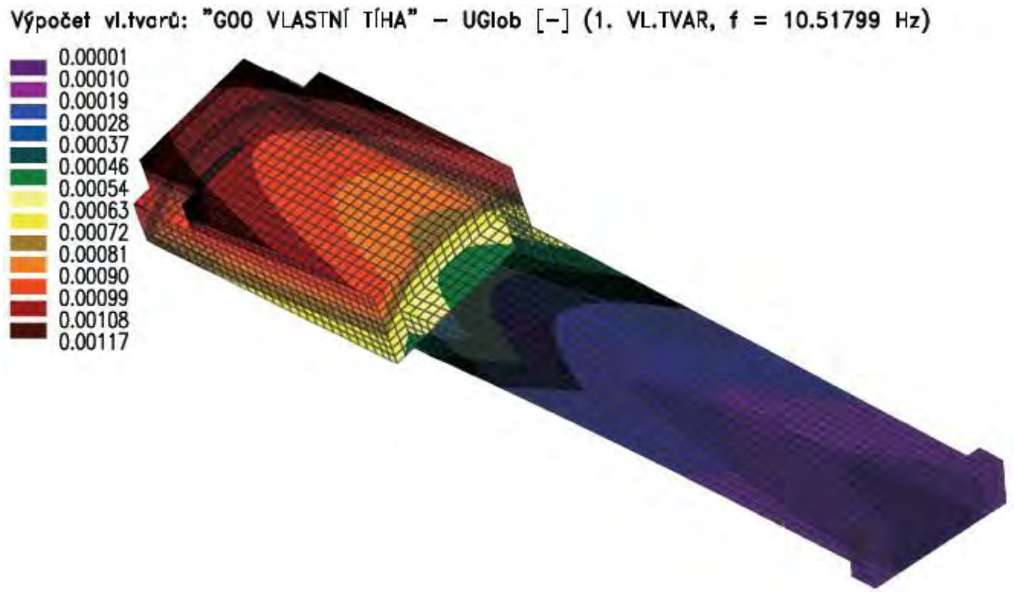
Numerical model vs. experiment

Vertical direction



Horizontal direction

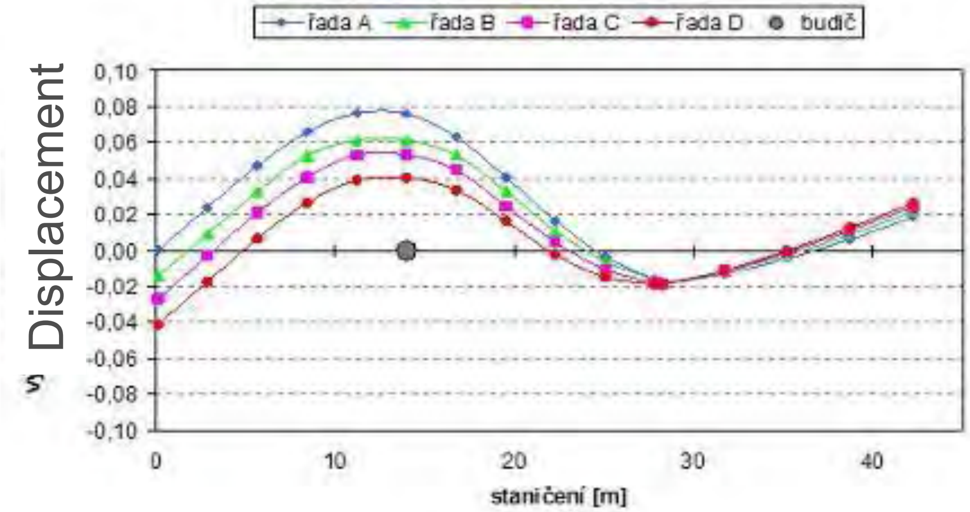




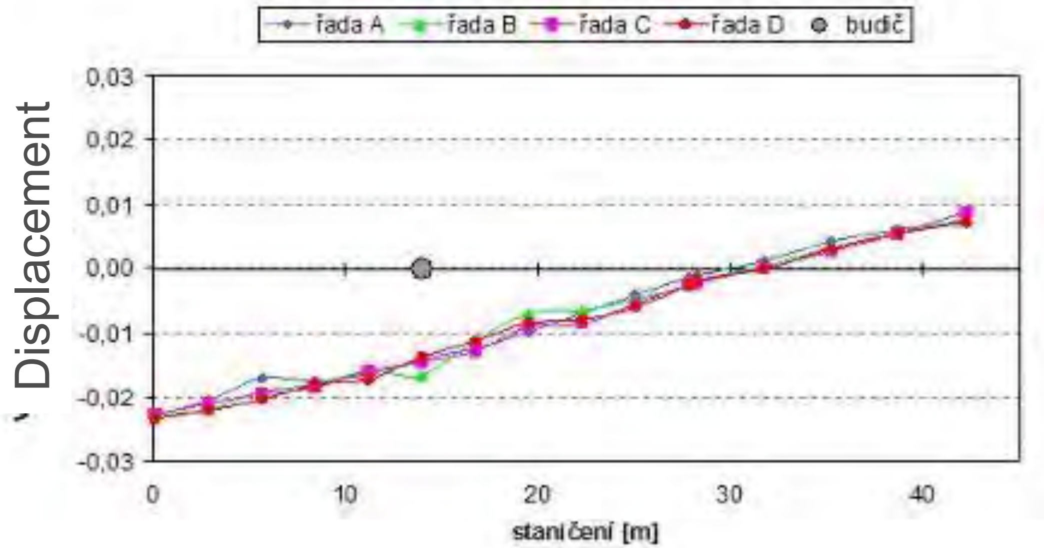
- measured frequency 10.9 Hz
- calculated frequency 10.5 Hz
- variance 3.8%

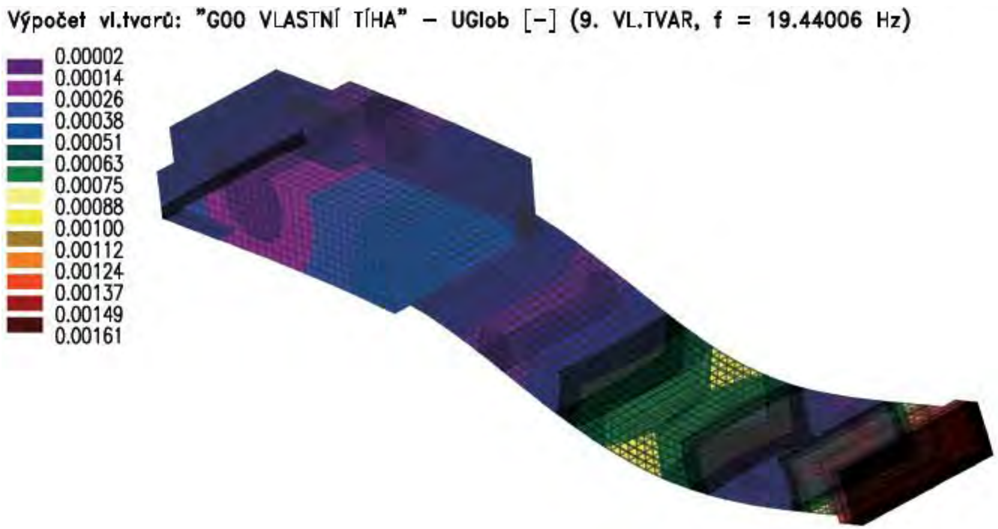
Numerical model vs. experiment

Vertical direction



Horizontal direction





- measured frequency 18.9 Hz
- calculated frequency 19.4 Hz
- variance 2.6%

Case studies

- Dynamic load test of bridges (50 bridges, e.g. Nuselsky Bridge in Prague)
- Orlik dam – ship elevator
- Tallest exhauster in Europe in Trbovlje (Slovenia)
- Power plant turbine and generator basement (PP Pocerady)

ÚJV Řež, a. s.

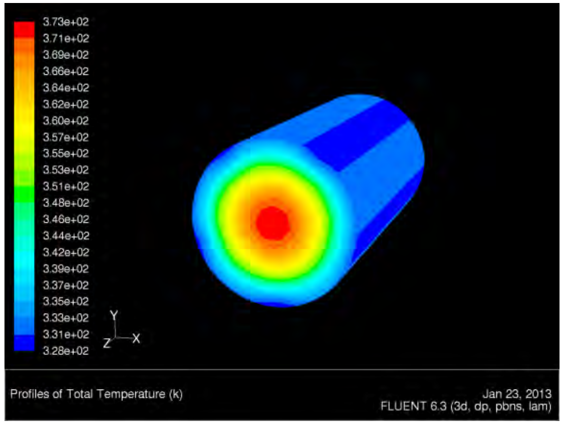
NPP concrete - UJV Rez activities

Jiří Žďárek

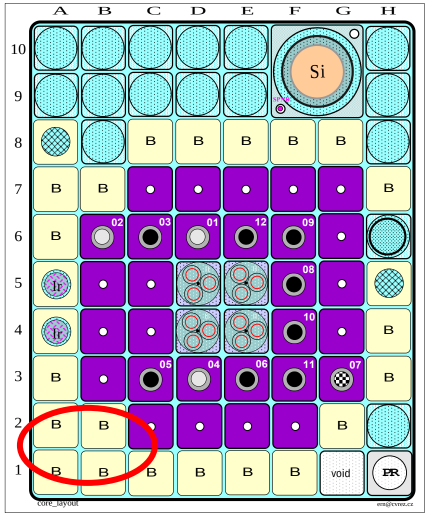
16. 04. 2013

Irradiation of concrete samples (1/2)

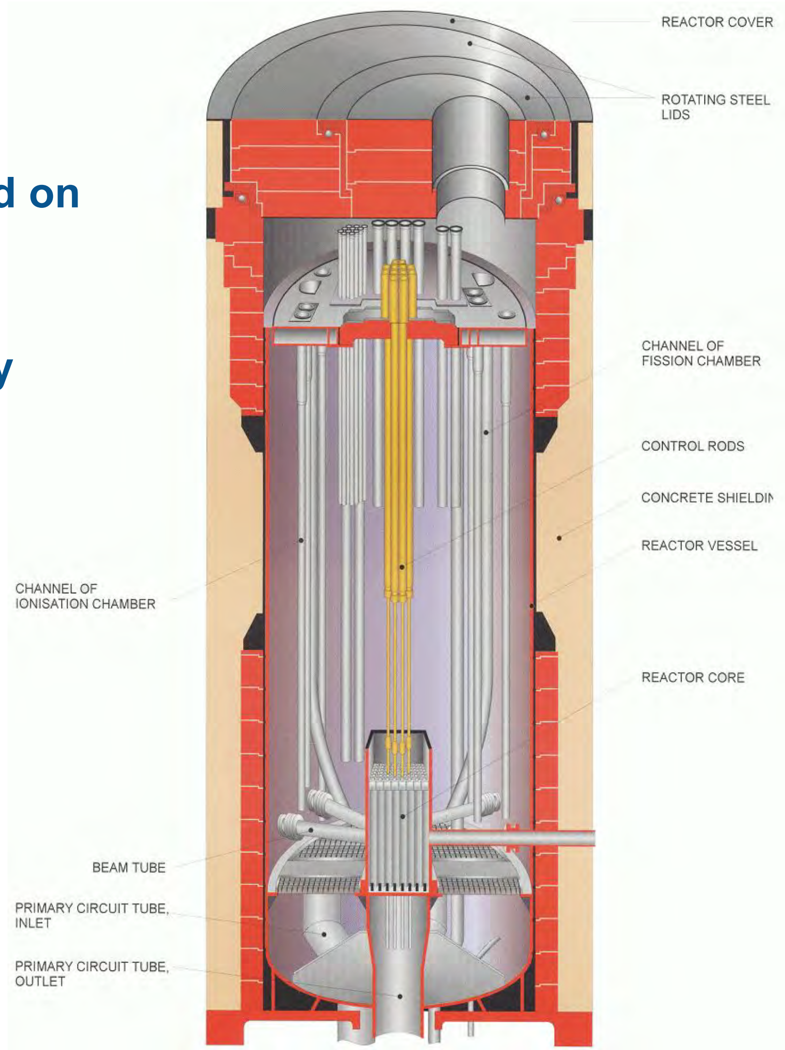
- **LVR-15 research reactor of 10 MW max. power**
- **Set of vertical irradiation channels**
- **Samples enclosed in special capsules**
- **Neutron flux a fluence determination from detectors placed on sample surface**
- **Temperature monitoring by embedded thermocouples**
- **Temperature control by passive systems - cooling primary circuit water, led inlay**
- **Gas composition and volume measurement after in-pile operation (GC-HID measurement)**



temperature field in the rig's cross section. The reactor power of 9MW



horizontal cross section of the reactor describing the position of proper irradiation channel



vertical cross section of the LVR-15 reactor

Irradiation of concrete samples (2/2)

- Cylindrical samples of 10 x 5 cm size
- Activation and thermohydraulic calculations made for accumulated fluence 1.E+19n/cm^2 and E>1 MeV
- Calculations made for 3 types of concrete: limestone, serpentite, silicate

Limestone	A (Bq)	
	7 days	30 days
Total	1,83E+11	1,19E+11

Serpentintite	A (Bq)	
	7 days	30 days
Total	9,05E+10	5,91E+10

Silicate	A (Bq)	
	7 days	30 days
Total	7,86E+10	5,09E+10

samples activity 7 and 30 days after irradiation

		Состав бетона, % массы		
		Известняковый	Серпентинитовый	Силикатный
SiO_2		28.3	34.3	69.1
CaO		26.0	9.8	10.8
MgO		9.6	30.7	0.7
Al_2O_3		3.5	1.8	8.8
Fe_2O_3		1.6	6.4	1.7
K_2O		0.6	0.1	1.6
TiO_2		0.14	0.0	0.15
Na_2O		1.1	0.06	2.7
SrO		0.03	0.013	0.036
BaO		0.03	0.01	0.1
Cr_2O_3		0.009	0.19	0.006
MnO		0.005	0.13	0.05
CuO		0.005	0.007	0.004
ZnO		0.007	0.009	0.007
ZrO_2		0.02	0.0	0.009
NiO		0.005	0.2	0.004
H_2O	Свободная	4.1	3.1	0.8
	Связная	2.0	11.3	1.1
CO_2		21.4	0.9	1.2
SO_3		0.58	0.52	0.58

chemical composition of limestone, serpentite and silicate concrete used for calculations

Mechanical properties testing

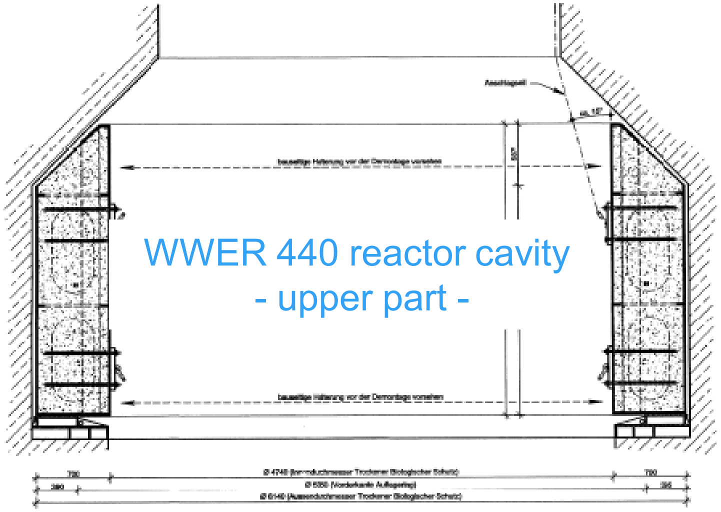
- **Laboratories of UJV Rez and Faculty of Civil Engineering of the Czech Technical University in Prague**
- **Equipment for investigation of mechanical behavior of concrete in compression, tension, flexure, dynamic loading, thermal loading, abrasion,…**

UJV Rez testing facility for compressive testing

Parameters	Maximal strength	300	kN
	Strength measurement range	3,00 - 300,00	kN
	Height of the working space	210	mm
	Piston lift	50	mm
	Size of press tables (sample size)	100 x 100	mm

Biological shielding from decommissioned NPP

- NPP Greifswald Unit 5 (Germany) – trial run stopped in 1990
- WWER 440 type reactor – PWR of russian construction
- UJV Rez in 2012 purchased a block of biological shielding as a surveillance material for complex testing
- Segment of steel lined serpentine concrete
- Documentation available – plant operation history, recipe of concrete mix, drawings,…



- **Segment 1 of RPV biological shielding**
  - Represents full construction height (2,78 m) and thickness (0,7 m)
  - In width (1,62 m) separated by parallel cuts from original ring of biological shielding
  - Segment comprise two original neutron flux measuring channels
  - Original construction and composition reference material !

Segment No.	mass [kg]	max. dose rate 0.1 m distance [mSv/h]	max. dose rate 1 m distance [mSv/h]	contamination α [Bq/cm²]	contamination β/γ [Bq/cm²]	mass specific activity [Bq/g]
1	7250	6.500E-01	1.200E-01	<4.000E-02	7.000E-01	1.720E+03

Thank you for your attention

www.ujv.cz

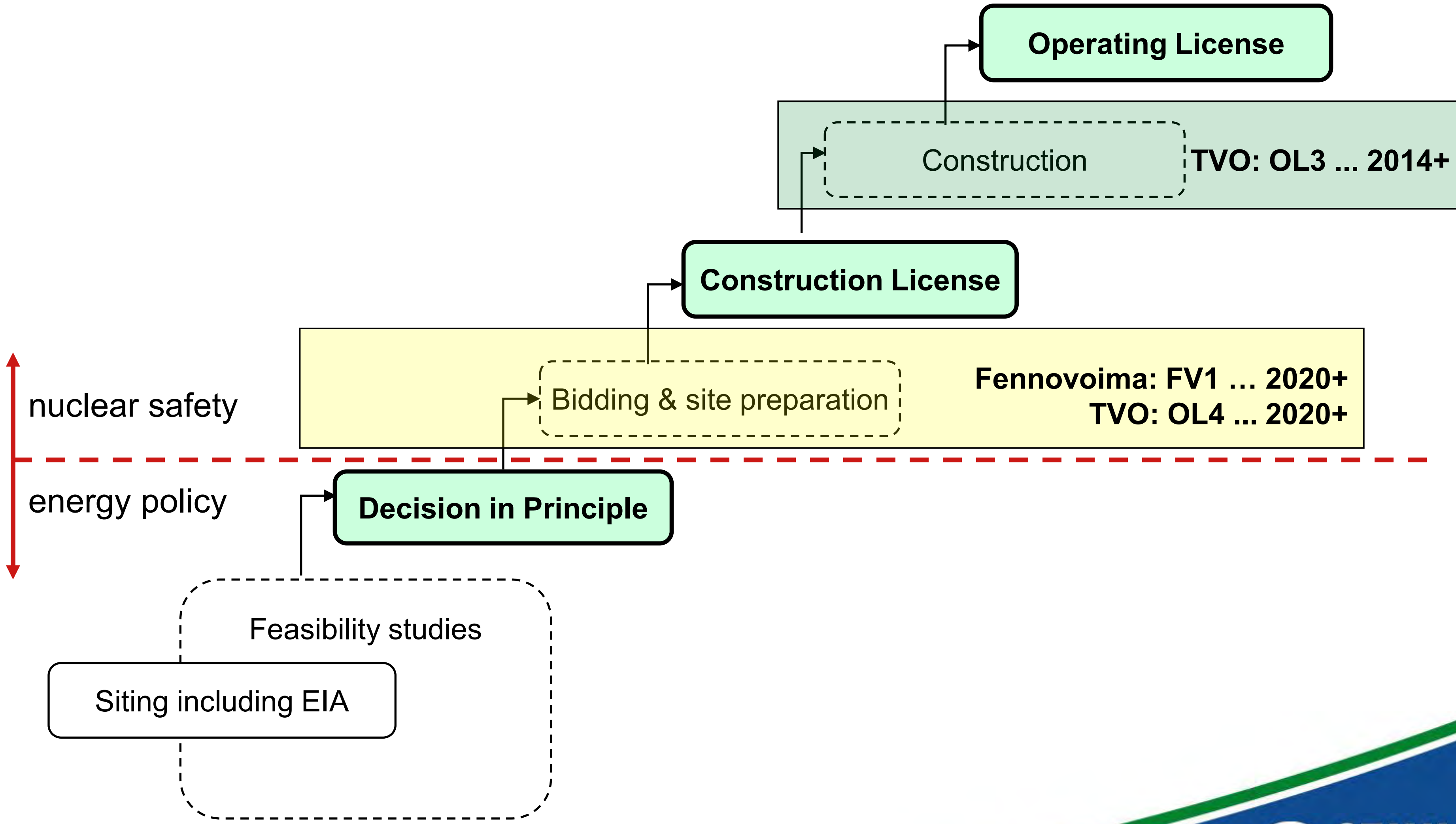
36th Meeting of the WGIAGE, Concrete sub-group meeting

NPPs under construction in Finland

April 8-9, 2013 – OECD/Paris

Pekka Välikangas

New NPPs under Licensing Steps in Finland



Status of Olkiluoto 3 (1/2)

- Construction works
  - Concreting almost completed, only secondary concreting left
  - Steel frameworks in containments under construction
  - Reactor and fuel pools passed tests
  - Cooling water channel structures commissioned
  - Office building under use of commissioning organization
- Finalizing works: almost all of 2700 rooms delivered for equipment installation
- Containment liner, design criteria under verification
  - Deformation capacity exceed ASME III criteria, updated design criteria and documentation under discussion
  - Pressure test plans for final verification
  - List of open questions to be answered before pressure tests

Status of Olkiluoto 3 (2/2)

- Beyond Olkiluoto 3 project: research for new methods for ensuring the grouting of post-tensioning cables of inner containment planned for 2013+
  - UT based tomography studies beyond design criteria
  - Research supporting corresponding maintenance planning
  - OECD/IAGE/NEA co-operation

FIN6 and FIN7

Two Construction License applicants
Five Vendors
Six type of NPPs

First NPP for Fennovoima

- Pyhäjoki is the selected site
- Simo planned to be available after the construction of first unit
- ABWR selected option for 4300 MWt size NPP
- Started negotiations with Rosatom: 1200 MWe AES-2006 PWR

PLANT	SUPPLIER	TYPE	THERMAL POWER [MWt]	ELECTRICAL OUTPUT [MWe]
ABWR	Toshiba-Westinghouse	Boiling water reactor	4300	ca. 1600
KERENA (SWR1000)	AREVA	Boiling water reactor	3370	ca. 1250
EPR	AREVA	Pressurised water react.	4590	ca. 1700

Fourth Unit to TVO

PLANT	SUPPLIER	TYPE	THERMAL POWER [MWt]	ELECTRICAL OUTPUT [MWe]
ABWR	Toshiba-Westinghouse	Boiling water reactor	4300	ca. 1600
ESBWR	GE- Hitachi	Boiling water reactor	4500	ca. 1600
APR1400	Korean Hydro & NP	Pressurised water react.	4000	ca. 1400
EU-APWR	Mitsubishi Heavy Ind.	Pressurised water react.	4450	ca. 1700
EPR	AREVA	Pressurised water react.	4590	ca. 1700

FIN6 and FIN7: Civil issues under study

- Geological and seismic conditions in northern part of Finland
  - Fennovoima apply seismic design criteria for Pyhäjoki (0,2 g)
  - Further information requested for understanding design margins
- Sensitivity studies of seismic hazards of Finnish NPP sites
  - Large co-operation under search
- Acceptability of large diameter rebars beyond 32 mm (1,26 in) in Containment
- Composite construction technology for massive concrete structures
  - YVL guide under development
  - Applicability of Japanese standard JEAC 4618-2008 for designing composite steel and concrete walls
  - Research for corresponding inspection protocols started in STUK
  - VTT taking part in European research project

Thank you

IRSN
INSTITUT
DE RADIOPROTECTION
ET DE SÛRETÉ NUCLÉAIRE

Faire avancer la sûreté nucléaire

CSNI - IAGE
Concrete sub-group
8 April 2013 meeting

IRSN status

F. Tarallo and J.M. Rambach

CONTENT

- Post Fukushima Complementary Safety Assessments
- Current R&D activities related to civil works

Complementary Safety Assessments 1/5

The Post Fukushima Complementary Safety Assessments (CSA) constitute the first step in the multi-year experience feedback process following the Fukushima accident.

They cover not only the reactors, but also the research facilities and the fuel cycle facilities.

2011: For 79 priority facilities, the CSA were submitted by the operators and reviewed by IRSN

IRSN proposed the creation of a “hardened safety core” of material and organisational measures designed to ensure control of the basic safety functions in extreme situations.

Extreme situations: external hazards and long-lasting situations of loss of ultimate heat sink and electric supply

Complementary Safety Assessments 2/5

ASN decisions on 3 January 2012: Provisions to improve the robustness of the facilities to extreme situations

1
- Creation of a "hard-core" of material and organisational measures designed to ensure control of the basic safety functions in extreme situations; the licensees will propose ASN the content and specifications of this "hard-core" for each facility before 30^{th} June 2012;

2
- For the nuclear power plants: as of this year, gradual creation of the "Nuclear rapid response force (FARN)" proposed by EDF, a national response system comprising specialist crews and equipment, able to take over from the personnel of a site affected by an accident and deploy additional emergency response resources in less than 24 hours. The system will be fully operational by the end of 2014;

3
- For the spent fuel storage pools on the various nuclear facilities: implementation of complementary strengthened measures to reduce the risk of dewatering of the fuel;

4
- For the nuclear power plants and silos at La Hague: feasibility studies with a view to the implementation of technical measures, such as a geotechnical containment or system with equivalent effect, designed to protect the groundwater and surface water in the event of a severe accident.

Complementary Safety Assessments 3/5

1 **HARDENED SAFETY CORE**

June 2012: the licensees proposed the content and the specifications of the hardened safety core >>> review by IRSN >> report December 2012

2013: more work on hardened safety core with the licensees, to reach satisfactory proposals on : content, definition of design hazards levels, safety requirements, sizing or verification principles

2 **NUCLEAR RAPID RESPONSE FORCE**

Currently under implementation by EDF

Complementary Safety Assessments 4/5

3 **SPENT FUEL STORAGE POOLS**

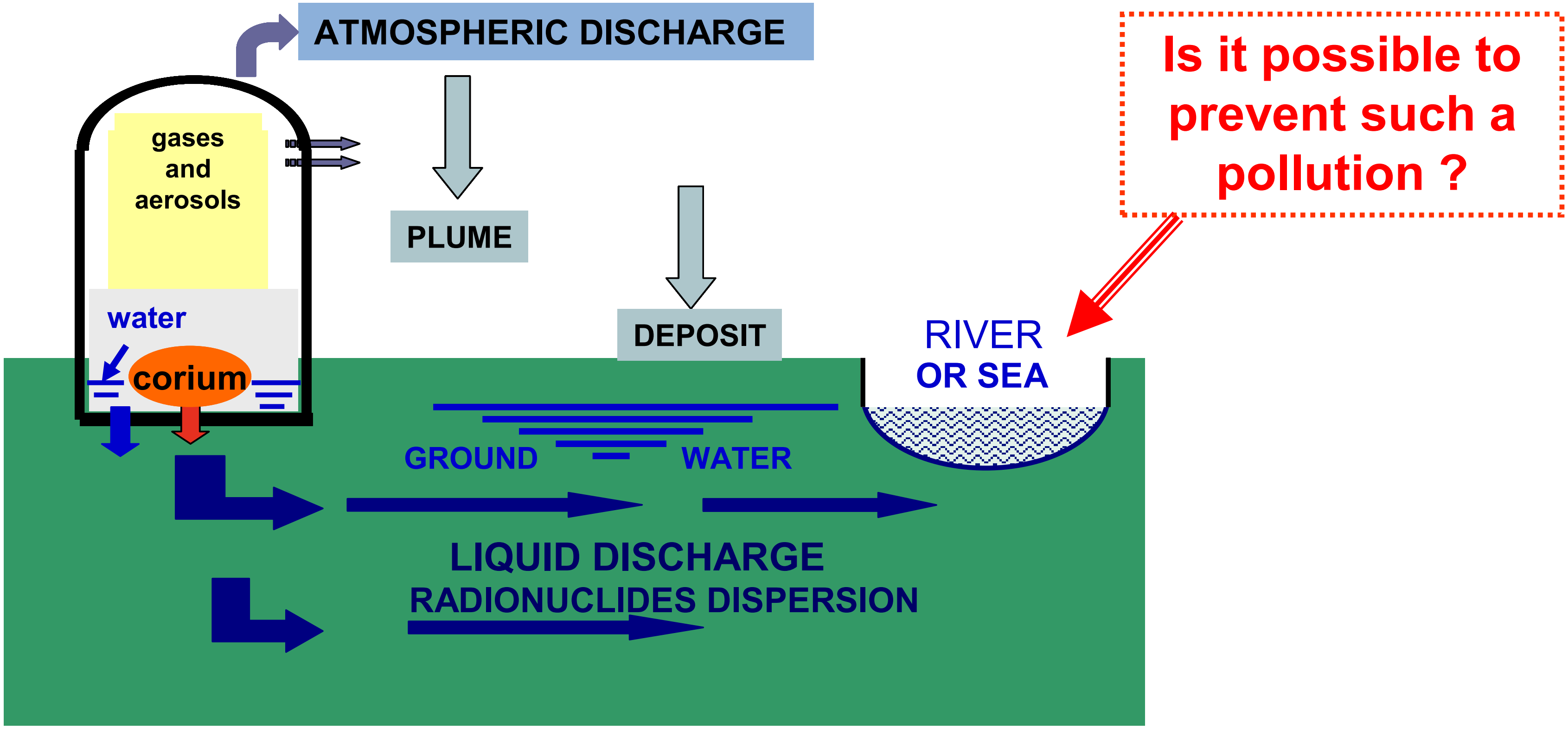
2012: the licensees proposed strengthened water supplies to reduce the risk of dewatering of the fuel >> globally meet the requirements

4 **PROTECTION OF GROUNDWATER AND SURFACE WATERS**

End 2012: Feasibility studies provided by EDF and AREVA, now under review

Complementary Safety Assessments 5/5

4 | **Recall: PROTECTION OF GROUNDWATER AND SURFACE WATERS**



- IRSN feasibility study began in 2010
- last year presentation to Concrete sub-group
- SMIRT 21 paper # 823 in New Delhi

Current R&D activities related to civil works

- Ageing and pathologies of concrete structures
  - Creep, early age cracking, creep in traction
  - DEF (Delayed Ettringite Formation), ASR (Alcali Silica Reaction)
  - We intend to join the ASCET CAPS
- Confinement of concrete: cracking and leakage of containments
- Behaviour of concrete structures submitted to impacts, fast dynamics:
  - Experimental research: involvement in IMPACT VTT program in Finland
  - Involvement in the Organizing Committee of IRIS_2012
  - Numerical simulation: IRIS_2012 benchmark in OECD

Contact: Georges NAHAS   georges.nahas@irsn.fr

Association Française pour les règles
de conception, de construction
et de surveillance en exploitation
des matériels des Chaudières Electro-Nucléaires

AFCEN Proposal
to deal with extreme hazards for new plants

2013 OECD , April the 8 & 9th

PRINCIPLE

2 domains are defined:

DESIGN BASIS DOMAIN: DBD

DESIGN EXTENTION DOMAIN: DED

DED (design extension domain is itself divided into 2 sub-domains:

- **DEC: design extension conditions -> accidental conditions inside containment due to major failure (…..severe accident)**

- **DEH: design extension hazards -> extreme level of hazards**

  ……. such as earthquake ("beyond design basis eartquake")

For DEC (accident inside containment)

-> « level II» design criteria according to IAEA NS-G 1.10

-> (limited permanent deformation)

For DEH:

- resistance: design criteria is **HCLPF capacity > DEH**

- tightness: strain limited to criteria depending on material properties and construction control level

+ additional detailing for liner/concrete interface especially for pools

For DED: only structures belonging to HSKE * have to be justified

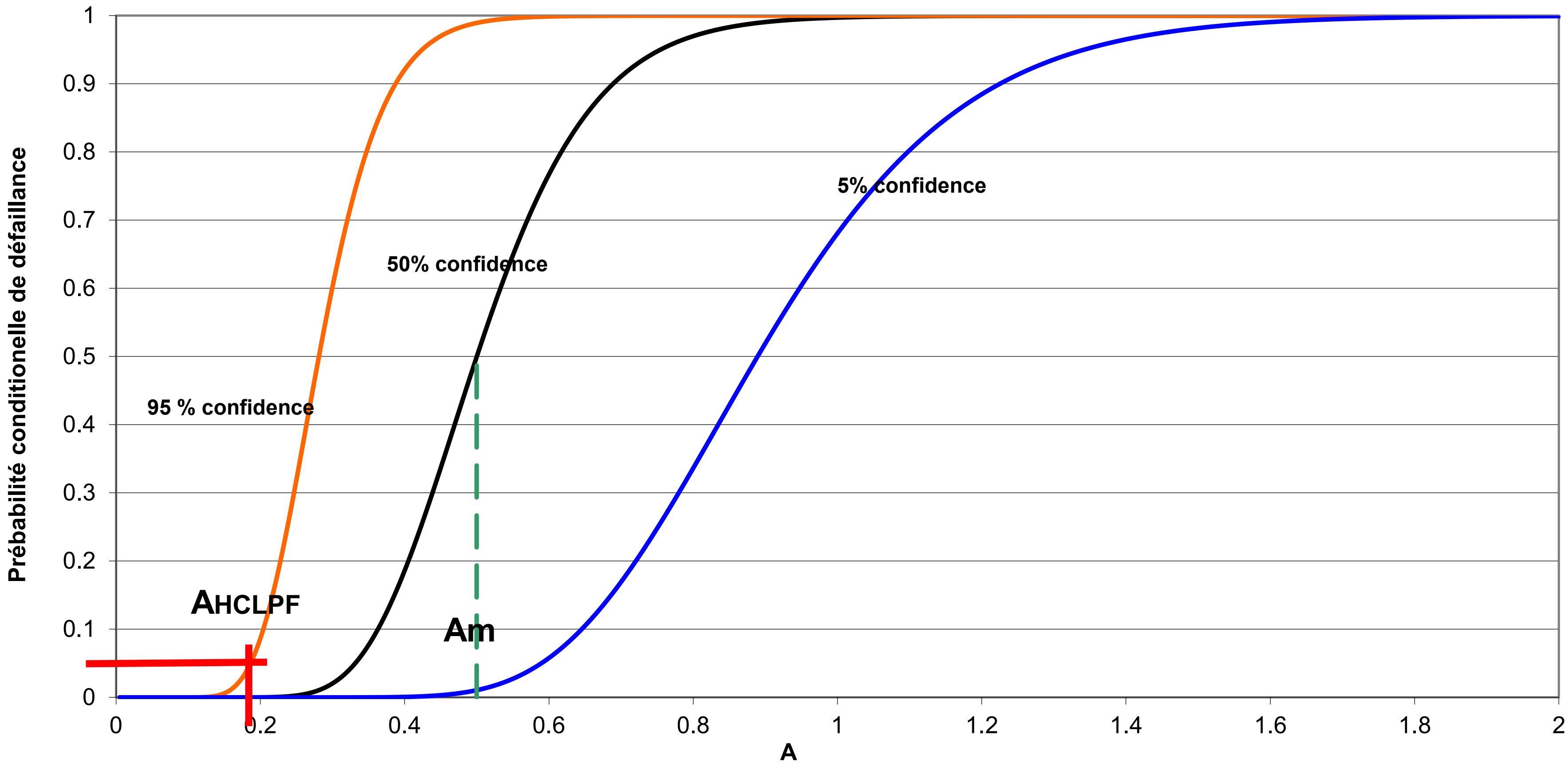
(*) Hardened Safety Kernel (that is to say only a reduced list of components)

IAEA DEC definition (extract from SSR-2/1 [NPP design])

Requirement 20: Design extension conditions

A set of design extension conditions shall be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments for the purpose of further improving the safety of the nuclear power plant by enhancing the plant's capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures. These design extension conditions shall be used to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences if they do occur.

HCLPF definition



HCLPF METHOD <=> conditional probability = 0.01

6 steps of the methodology:

- **Analysis of the structure and the different possible failure modes**
- **Identification of the failure modes which has to be considered for HCLPF assessment**
- **Assessment of the median parameters and standard deviation of the material identified in the failure modes considered**
- **Assessment of the median capacity of every identified failure modes considered**
- **Assessment of the logarithmic standard deviation due to epistemic uncertainty and of the logarithmic standard deviation due to randomness**
- **HCLPF calculation = $Am\ e^{-1,65\ (\beta u+\beta r)}$**

Thank you for your attention

German Research Activities* Related to Concrete

Christian Heckötter, Jürgen Sievers GRS

08-09.04. 2013

Concrete Subgroup Meeting – CSNI /NEA Working Group IAGE

* sponsored by the German Federal Ministry of Economics and Technology (BMWi)

OUTLINE

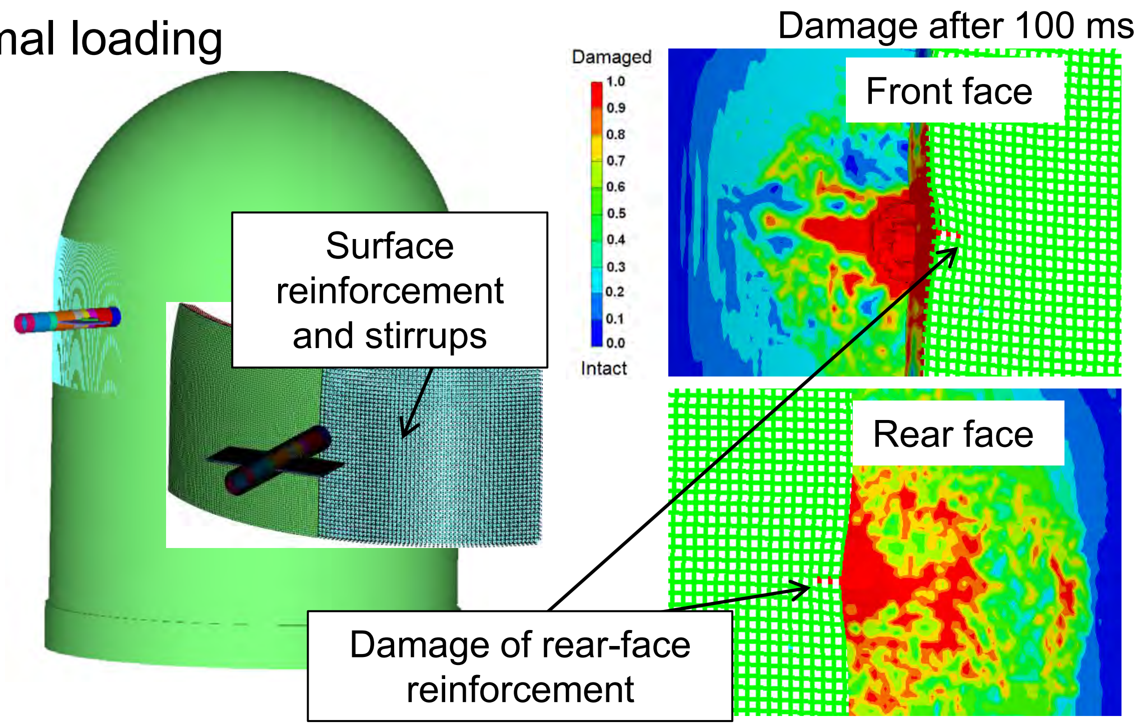
German Research Activities* Related to Concrete on

- Structural Dynamic Behaviour of Containment Structures
- Complex Loading Conditions at High-Dynamic Impact
- Joint-Project: Behaviour of Concrete at High Strain-Rates
    - Biaxial Strength
    - Simulation of Concrete Behaviour Under Impact Loading
    - Modelling of Concrete at the Meso-Level
- Undercut Anchors in Nuclear Facilities
- Material Modelling of Reinforced Concrete

* sponsored by the German Federal Ministry of Economics and Technology (BMWi)

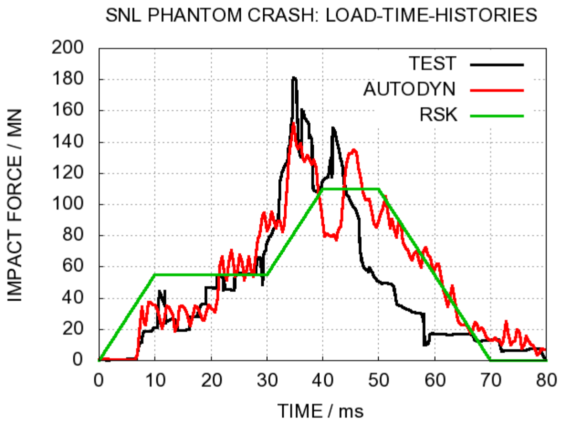
Further Development and Testing of Analysis Methods for the Determination of the Structural Dynamic Behaviour of Containment Structures

- Contractor: Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH
- Initiated: 01.2011 / Completed: 06.2014
- Goal is to provide methods to evaluate limit load-carrying capacity of containment structures including consideration of imperfections (e.g. locks)
- Internal pressure and thermal loading
- Aircraft crash

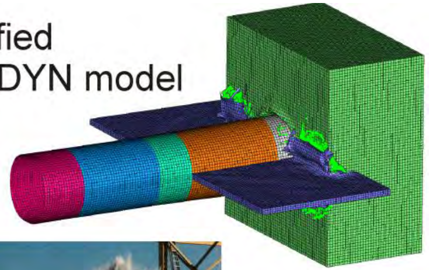


Further Development of the Analysis Method for Consideration of Complex Loading Conditions at High-Dynamic Impact on Reinforced Concrete

- Contractor: Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH
- Initiated: 05.2012 / Completed: 10.2015
- Validation of simulation models for high-dynamic complex loading of reinforced concrete structures based on selected experiments
  - Complex geometry for impactor (e.g. airplanes, engines, ...)
  - Effects of high explosives
  - Participation in VTT IMPACT and IRIS



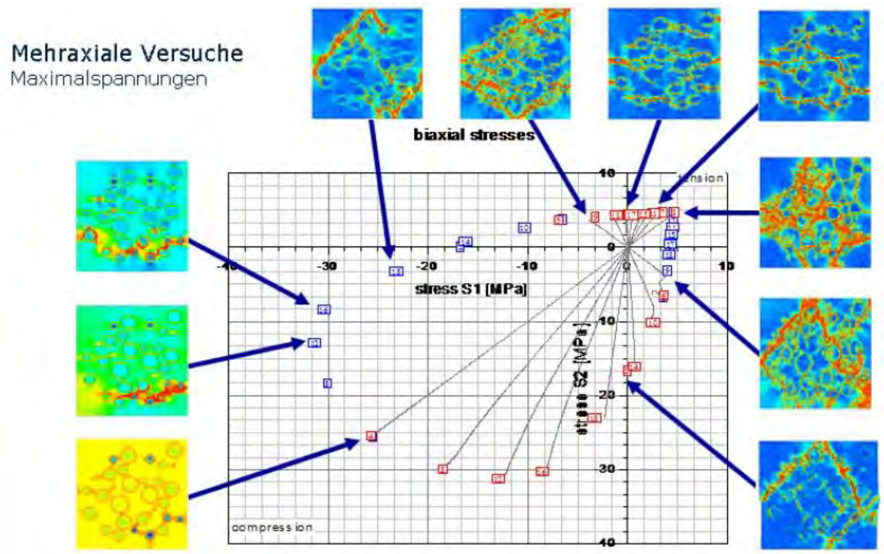
Simplified AUTODYN model



Video frame from SNL

Behaviour of Concrete at High Strain-Rates: Experimental Investigations of Biaxial Strength and Formulation of a Constitutive Description

- Contractor: TU Dresden
- Initiated: 01.2009 / Completed: 12.2012
- Goal is to develop a Split-Hopkinson-Bar facility for bi-axial testing
- Testing of different concretes under different stress conditions and strain-rates
- Development of a multi-dimensional macroscopic constitutive law

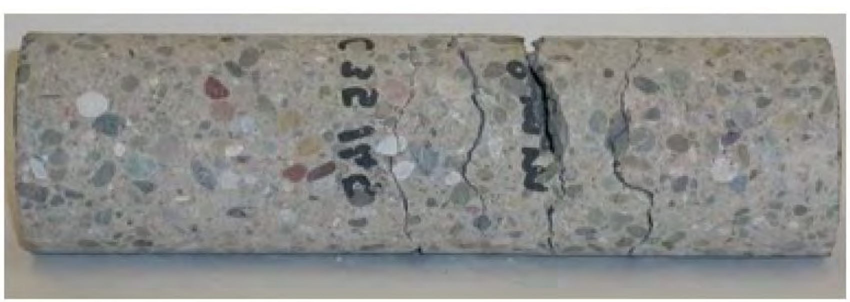


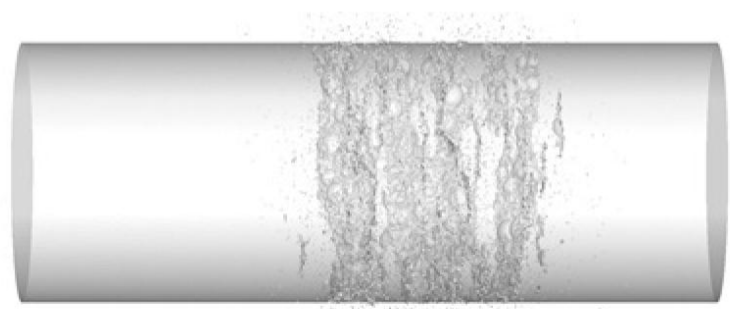
Behaviour of Concrete at High Strain-Rates: Simulation of Concrete Behaviour Under Impact Loading Using Discrete-Element Method

- Contractor: TU Dresden
- Initiated: 01.2009 / Completed: 12.2012
- Goal is to develop a DEM code for the simulation of concrete under high strain rates
- Simulation of concrete fracture due to impact loading
- Simulation of wave propagation due to impact loading
- Simulation of crack initiation and damage propagation

Behaviour of Concrete at High Strain-Rates: Constitutive Material Description and Numerical Modelling of Concrete at the Meso-Level

- Contractor: Ernst-Mach-Institute (EMI)
- Initiated: 01.2009 / Completed: 12.2012
- Investigation of mechanical properties of concrete components (cement, aggregate, interfaces, fibres) and their influence on dynamic material behaviour
- Development of a meso-mechanical material model





Undercut Anchor in Nuclear Facilities - Design and Ageing Effects

- Contractor: MPA of Karlsruhe Institute of Technology (KIT)
- Initiated: 04.2012 / Completed: 03.2015
- Goal is to provide assessment tools for existing and new anchor bolt connections under consideration of special requirements in nuclear facilities.
- Include long-term effects such as ageing of concrete
- Realistic installation and loading conditions
- Focus on earthquake loading

Material Modelling of Reinforced Concrete - Multi-Scale Approach and Model Order Reduction

- Contractor: Ruhr-Universität Bochum
- Initiated: 07.2012 / Completed: 06.2014
- Goal of this project is the development of an efficient concrete-steel bond model
- Focus on shell-structures
- Validation of models based on tests with complex geometry (e.g. Sandia containment mock-up)

Current Topics of Interest concerning Swiss Nuclear Power Plants

1. Political Situation for Nuclear Energy in Switzerland
2. Ruling of the Swiss Federal Court concerning the unlimited Permit of the Mühleberg Power Plant
3. Aircraft Impact on Swiss Nuclear Power Plant
4. PEGASOS Refinement Project
5. Various Topics

Christian Schneeberger, Structural Engineer MSc ETH,

Civil Engineering Section, Swiss Federal Nuclear Safety Inspectorate

OECD / NEA / 18TH MEETING OF THE IAGE SUB-GROUP ON THE AGEING OF CONCRETE STRUCTURES, 8-9 April 2013, Paris

1. Political Situation for Nuclear Energy in Switzerland

- Federal Council decided in 2011 to gradually phase out of nuclear energy programme and Swiss parliament [National Council and Council of States] approved this decision. Permits for new plants are no more possible. Research in the Nuclear Energy field is still possible.
- The 5 NPPs in Switzerland have unlimited operation permit. Until now no limitation of operation was decided.
- There will be in the next year 2 public votings:
  Sate of Berne concerning the NPP Mühleberg
  Swiss voting with 45 year lifetime limitation of existing plants and abandoning of nuclear energy

Anlage	Reaktortyp	Nettoleistung in MWe	Kommerzielle Inbetriebnahme	Jahresproduktion 2008
Beznau I	Druckwasser	365	1969	2.918 Mrd. kWh
Beznau II	Druckwasser	365	1971	3.052 Mrd. kWh
Mühleberg	Siedewasser	373	1972	2.956 Mrd. kWh
Gösgen	Druckwasser	970	1979	7.898 Mrd. kWh
Leibstadt	Siedewasser	1165	1984	9.308 Mrd. kWh

1. Political Situation for Nuclear Energy in Switzerland cont.

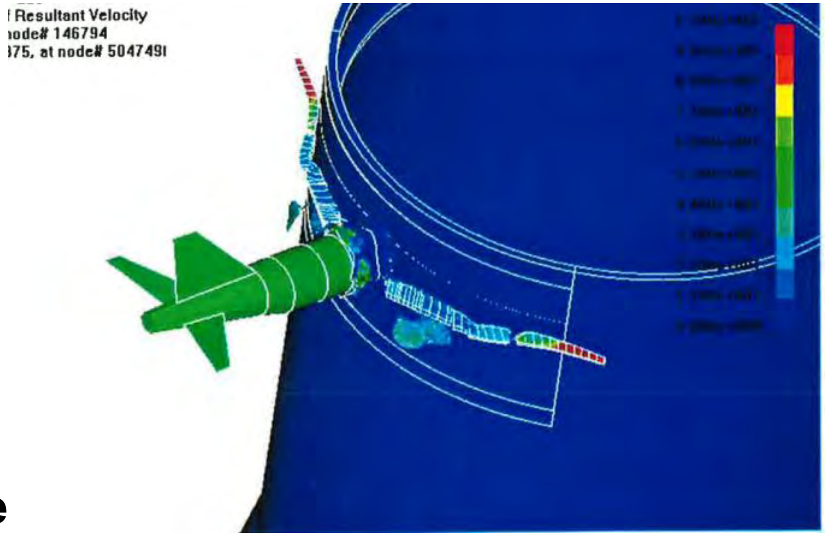
- The Federal Council started the project Energy Strategy 2050.
- Many people doubt that is possible to replace nuclear energy (currently 40% by NPPs) with expansion of hydropower a. s. o. without big drawbacks for private people and industry.

2. Ruling of the Swiss Federal Court concerning the unlimited Permit of the Mühleberg Power Plant

- The Mühleberg NPP got the unlimited operating permit from environment ministry (DETEC – Federal Department of Environment, Transport, Energy and Communications) in 2009. Several persons living near the plant made a complaint against the unlimited permit.
- Switzerland's Federal Administrative Court (FAC) ruled on 1 March that the Mühleberg plant can only operate until 28 June 2013, overturning the decision from DETEC.
- The FAC recorded that DETEC made an error in 2009, issuing the unlimited permit only based on the review reports of Swiss Federal Nuclear Safety Inspectorate ENSI. DETEC should have done own safety reviews.
- The FAC didn't care about the current organization of ENSI and DETEC
- The operator of the Mühleberg Plant (BKW FMB Energy Company) and DETEC appealed against the FACs ruling.

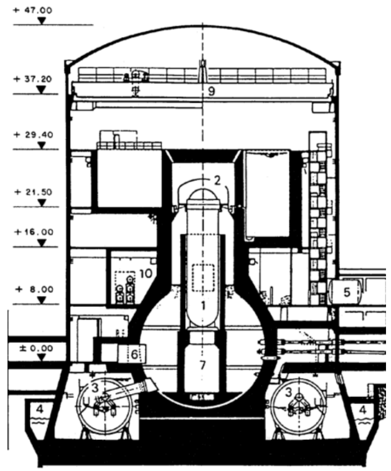
2. Ruling of the Swiss Federal Court concerning the unlimited Permit of the Mühleberg Power Plant cont.

- Before Easter the Swiss Federal Court treated the two complaints.
- Swiss Federal Court came to the decision that the ruling of the FAC was not correct and approved the two complaints. The Swiss Federal Court stated that the responsible authority to ensure nuclear safety in Switzerland is ENSI.
- In contrast to the FAC the Swiss Federal Court didn't deal with any safety aspect [Core Shroud, Earthquake, Cooling Facilities].
- With this decision the Mühleberg NPP now has got the unlimited operation permit too.
- Opponents of Mühleberg plant and some politicians are still demanding for a 'second' authority to ensure nuclear safety.
- Federal Council is satisfied by the decision of the Swiss Federal Court. The DETEC doesn't have trained staff to perform safety reviews in the nuclear field.



3. Aircraft Impact on Swiss Nuclear Powe

- The plants Mühleberg (1972) and Beznau (1969/1971) were not designed for aircraft impact.
- Gösgen (1979) and Leibstadt (1984) were designed against aircraft impact.
- After 9/11 ENSI (formerly HSK) ordered from operators large aircraft impact analyses (Boeing 707, v = 100 m/s, restfuel).
- A report of these analyses can be downloaded [in german] from ENSI webpage. One can read in this report that the older plants can fulfil the design requirements of the new plants (Boeing 707).
- The newer plants can resist Boeing 747 (v > 150 m/s)
- In my opinion ENSI was wrong to publish aircraft details in the public report.



3. Aircraft Impact on Swiss Nuclear Power Plant cont.

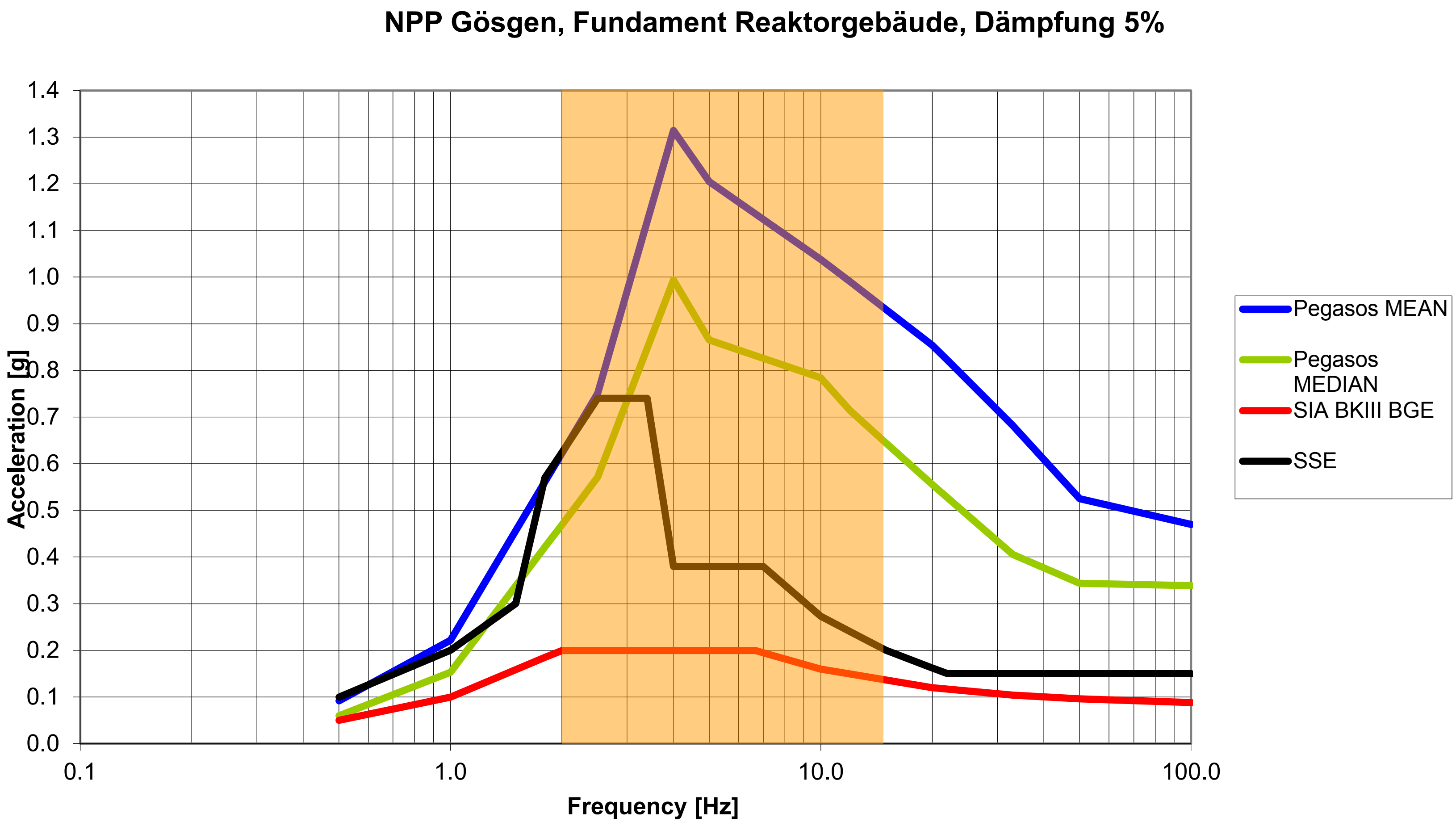
- The analyses performed after 9/11 concluded with the statement that Swiss NPPs have a good protection against aircraft impact.
- Tests with flight simulator showed that it’s difficult to hit reactor building in Mühleberg because of topography.
- A former Swiss pilot read the 9/11-report. He claims that nowadays it’s no problem to hit the reactor building of a Swiss NPP. According to his opinion it’s easier to hit the reactor building of Mühleberg plant than to land in London Heathrow.
- The Swiss pilot [with other people] submitted in March 2013 a complaint to DETEC. DETEC should order an immediate decommissioning of Mühleberg plant because of low resistance against impact of modern aeroplanes.

4. PEGASOS Refinement Project

- 1999 ENSI (formerly HSK) required from the operators new Probabilistic Seismic Hazard Analysis. The requirements in the order of ENSI were like this, that operators had to use SSHAC 4 Level procedure.
- Operators launches the project ‘PEGASOS’ with the following four subprojects:
  SP1: Seismic Source Characterization
  SP2: Ground Motion Characterization
  SP3: Site Response Characterization
  SP4: Seismic Hazard Computation
- PEGASOS was finished with the Final Report in 2004
- With the objective to reduce epistemic uncertainties in 2007 PRP was started. PRP included a new Subproject: *SP5 Earthquake Scenarios*
- In May 2011 Swissnuclear published PRP Intermediate Hazards for all plants. The PRP IH were used to run Swiss Fukushima Stress Test.
- Final report of PRP was delayed just many times. Currently it’s expected in 2013.

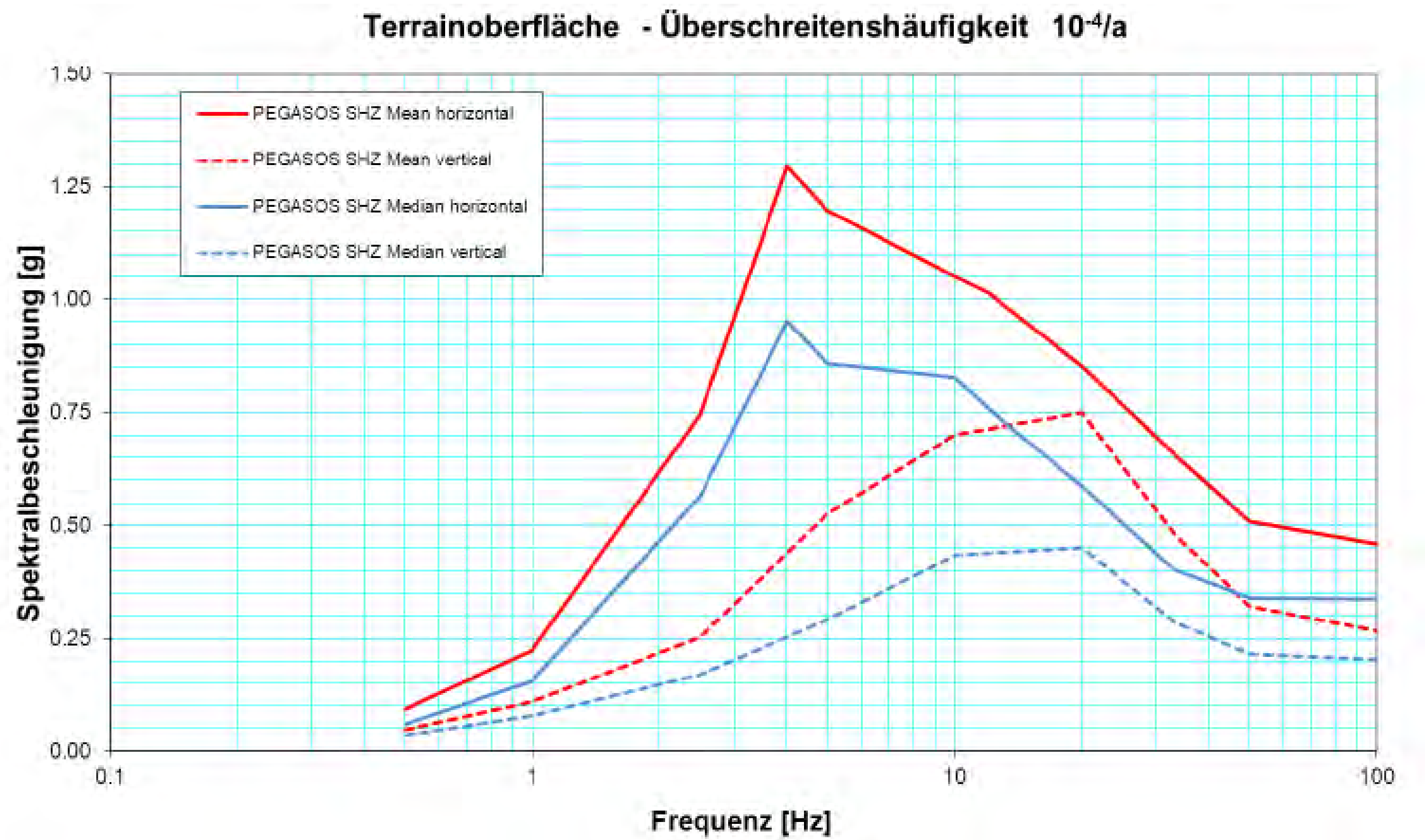
4. PEGASOS Refinement Project cont.

- Basemat of reactor building (-9 m) / 10^{-4}



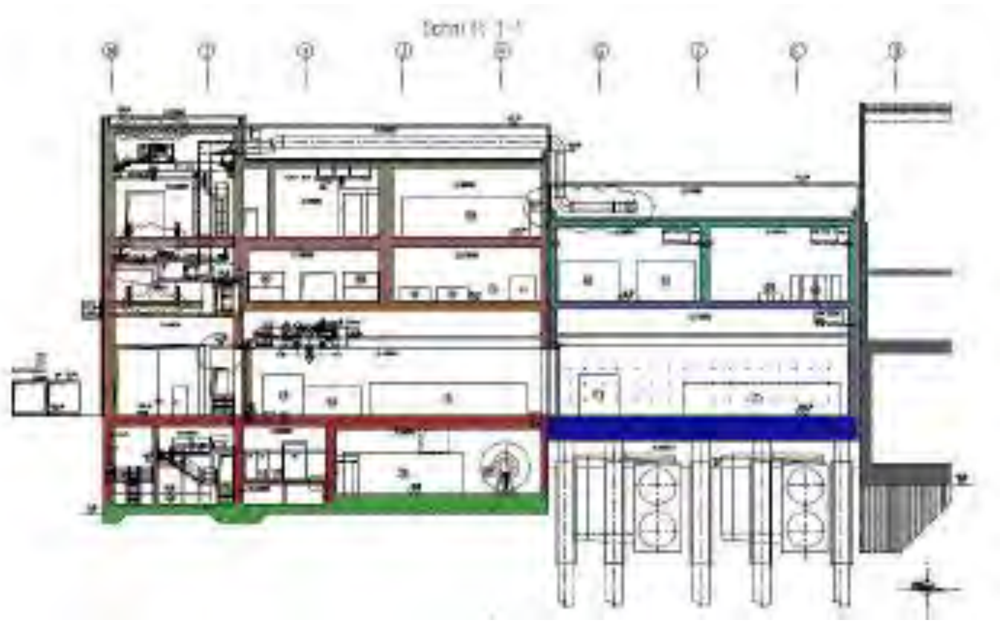
4. PEGASOS Refinement Project cont.

- PRP IH May 2011 for NPP Gösgen
- Soil Surface Spectra (Free Field)



5. Various Topics

- Civil Engineering Section of ENSI participates in the Benchmark-projects SMART2013 and is Partner in the IMPACT III Project (Impact tests performed in Finland by VTT).
- Three new buildings are still under construction in Swiss Plants. Two emergency diesel buildings in Beznau and a building to store big nuclear components in Leibstadt.



- There was a revision of the Swiss code for concrete structures.

Research Related to Concrete Technology at USNRC

Presented by
Andrew J. Murphy, Senior Technical Advisor
for Seismology & Earth Sciences
Office of Nuclear Regulatory Research, USNRC

Concrete Subgroup Meeting - CSNI-NEA Working Group IAGE
Paris, France
April, 2013

Topics

- Round Robin Study on Containment Post-tensioning Methods
- Modeling of Concrete Behavior under Impact Loads
- Degraded Containment Research
- Standards and Guidance for Design of Modular Composite Structures
- Spent Fuel Pool (SFP) Scoping Study
- LEVEL 3 PRA (PSA)
- Radiation Effects on Concrete
- Expanded Proactive Materials Degradation Analysis (EMDA)
- Research Related to Transportation Casks
- Revision of USNRC Regulatory Guides
- Future Potential Research for Concrete in NPPs

Round Robin Study on Containment Post-tensioning Methods

U.S.NRC
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Protecting People and the Environment

- Scope:
  - Investigate the structural behavior of a concrete containment vessel with grouted and un-grouted tendon systems
  - Document and compare in-service inspection requirements in peer-reviewed standards and codes for grouted and un-grouted tendons
  - Examine if corrosion protection used for grouted and un-grouted tendons are adequate for the expected life of the structure.
- Organization doing work for NRC: Sandia National Labs (SNL)

Round Robin Study on Containment Post-tensioning Methods (Cont'd)

U.S.NRC
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- Progress made by NRC in last 2.5 years:
  - Project initiated in September 2010
  - Participated in Expert Meeting of the OECD Post Tensioning Methodologies for Containment Building (PTM) hosted by EdF April 20-21, 2011.
  - Sandia began with NASTRAN model , and converted to Abaqus. Final full 3-D models for prestressed concrete containment with grouted and greased tendons were developed.
  - Report on study of structural behavior, strength, and expected failure modes is complete.
  - Draft report completed on comparison of in-service inspection requirements in peer-reviewed standards and codes for grouted and un-grouted tendons.
  - Draft report completed on corrosion protection used for grouted and un-grouted tendons.
  - Participated in 2nd Expert Meeting of the OECD PTM hosted by EdF, Nov. 26-27, 2012
  - Guidance from this OECD activity is to be completed by the end of 2013

Modeling of Concrete for Impact Loads

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- NRC participates in a multiple partner, international experimental research program, called IMPACT, carried out by the Technical Research Center of Finland (VTT).
- The IMPACT program uses small and medium scale tests to collect data on the behavior of reinforced and pre-stressed concrete slabs subjected to missile impacts.
- IMPACT also collects data on the behavior of various types of missiles (soft and hard) and their loads on structural components.
- NRC uses the test data to benchmark codes that the staff and its contractors utilize to analyze concrete components of nuclear power plants subject to impact loads.

Modeling of Concrete for Impact Loads (Cont'd)

- Current NRC analyses use the LSDYNA code
- Parameters and effects addressed include:
  - Soft and hard missiles, and liquid-filled missiles
  - Impact speed
  - Transverse reinforcement (stirrups and T-headed bars)
  - Global and local failure/damage modes of the slabs such as flexure and punching shear
  - Hydrodynamic impact loads, and
  - Concrete models

Degraded Containment Research

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- NRC sponsors research at SNL to study the effects of degradation on the performance of several types of containments in a risk-informed manner.
- The NRC anticipates that the results of this research will be used to inform on a case-by-case basis regulatory processes such as:
  - license renewal reviews if containment degradation is detected
  - decisions on inspection, monitoring and repair
  - review of containment repairs
- It is anticipated that results of the program will support identifying the relative risk significance of potential degradation processes.
- Study has integrated structural analysis results with pre-existing probabilistic risk assessment (PRA) models (NUREG-1150) for severe accident conditions
  - It used structural analysis to develop fragility curves for concrete containments with and without cases of hypothetical liner degradation

Degraded Containment Research (Cont'd)

U.S.NRC
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- Early phases of the study indicated that risk metrics such as Large Early Release Frequency (LERF) used in conjunction with Regulatory Guide (RG) 1.174 guidance for risk-informed acceptability may not fully capture effects of liner corrosion.
- Study progressed to investigate other potential metrics, e.g., consequences.
  - Results in NUREG/CR-7149 (not yet published)
- Current possible developments
  - Investigation of other metrics such as Small Early Release Frequency (SERF), not addressed in RG 1.174, and the binning of LERF and SERF used in the PRA models.
  - Assessment of potential effects of containment repair operations on containment performance

Standards and Guidance for Design of Modular Composite Structures

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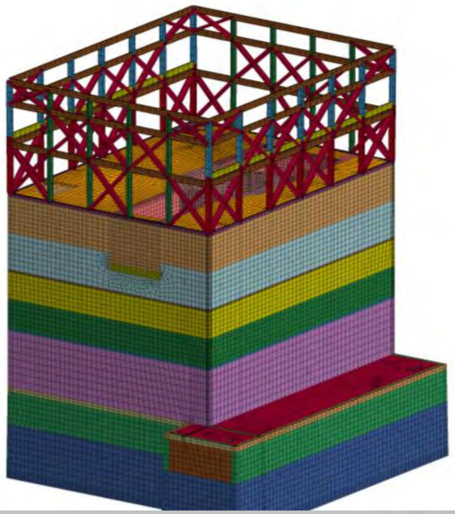
- Modular composite construction consisting of concrete reinforced with steel faceplates (SC) is being proposed for safety-related structures of new reactor designs and possibly for small modular reactors
  - This construction technique is outside the scope of current U.S. design codes and standards such as ACI 349 and AISC N690
- International codes for this type of construction already exist
  - Japan Electric Association JEAC-4618 (2009)
  - Korea's KEPIC-SNG Steel Plate Concrete Structures (2010)
- U.S. consensus standard planned as an appendix to AISC N690 is in advanced stages of preparation
  - Being prepared by a subcommittee to AISC Task Committee 12 (N690)

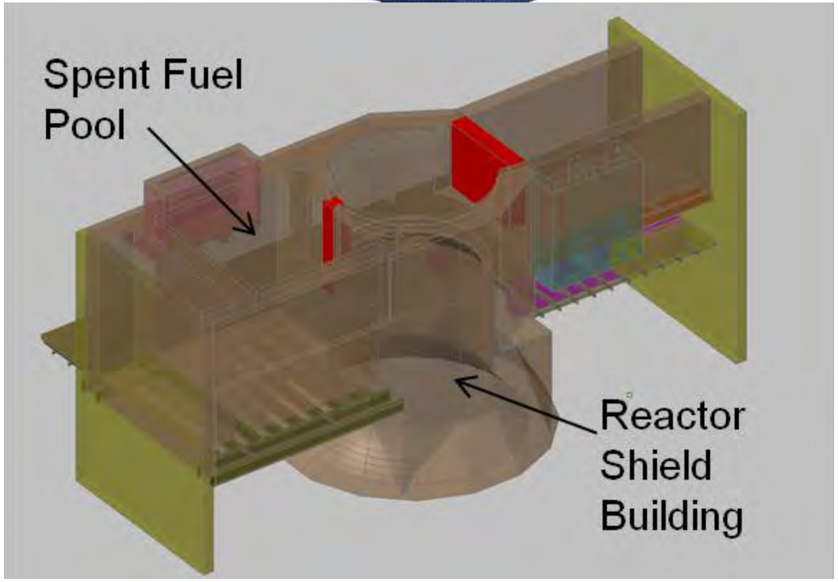
Standards and Guidance for Design of Modular Composite Structures (Cont'd)

- To support development of guidance for the design of SC structures, NRC is reviewing existing international standards and their technical bases
  - NRC is assessing the adequacy and sufficiency of the technical bases for the development of design specifications. Outside experts informed the staff on this effort.
  - Staff participates in the activities of AISC's TC 12 subcommittee developing US standard for safety-related SC structures
- NRC is sponsoring research for the evaluation of the technical bases for the SC structures
  - Research grant to the University of Toronto on modeling tools that can capture load redistributions and be used, for example, to assess seismic safety margins implicit in code provisions

Spent Fuel Pool (SFP) Scoping Study

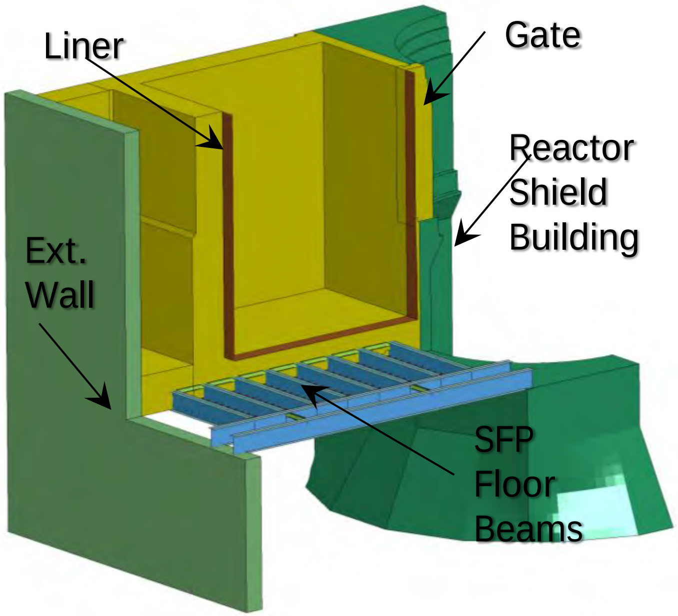
- Focus: reexamination of the effects on SFP consequences of moving older fuel to dry cask in an expedited manner
- Two conditions to be considered
  - high-density loading
  - low-density loading
- Elements of the study include
  - Seismic and structural assessment
  - SCALE analysis (dose rates)
  - MELCOR accident progression analysis
  - Emergency planning assessment
  - MACCS2 offsite consequence analysis
  - Probabilistic considerations



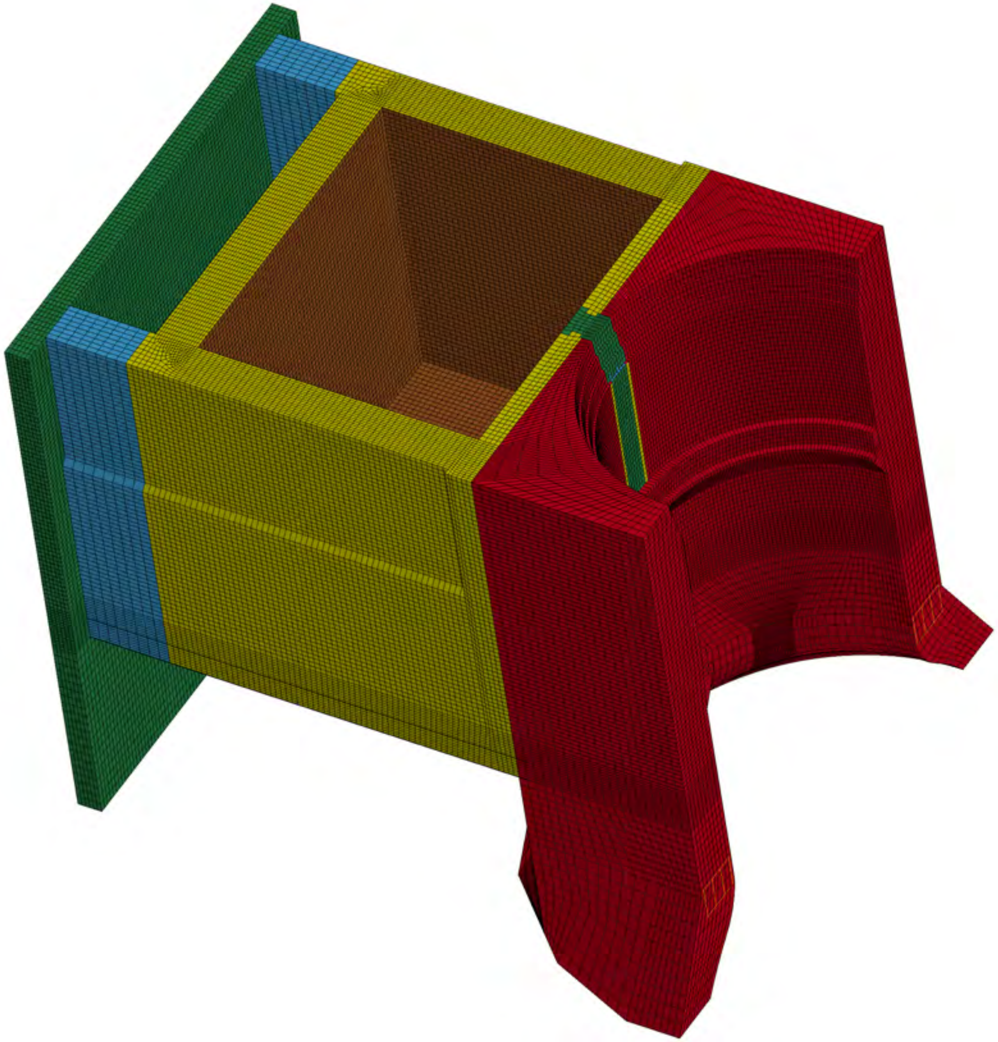


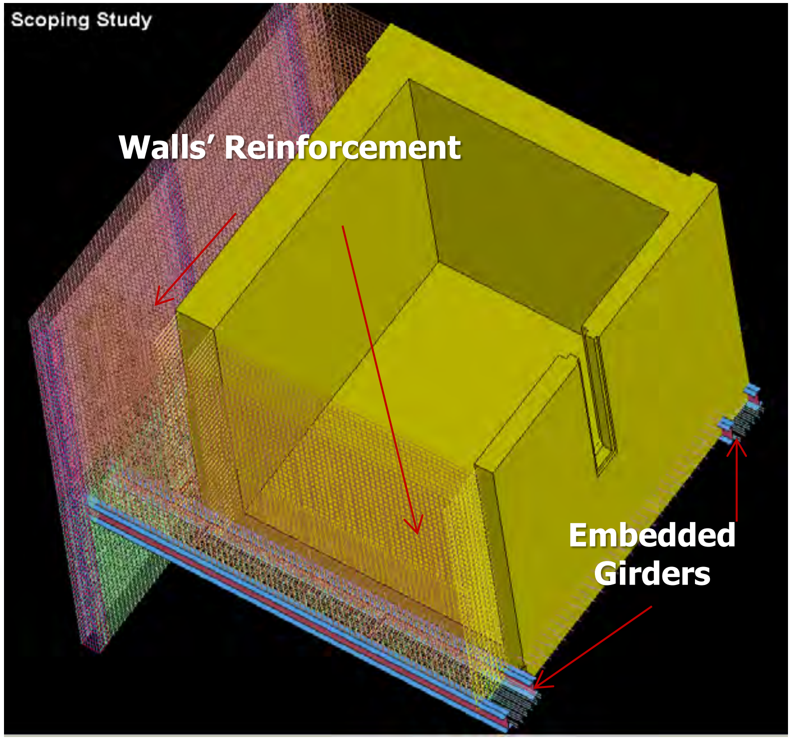
Seismic and Structural Input

- Seismic Event
  - Challenging but very low frequency of occurrence event (greater than the design basis for Central and Eastern US plants)
  - Updated ground motion characterization models (United States Geological Survey, 2008)
- Structural Assessment
  - To determine starting point for accident progression analysis
  - Assesses performance of SFP structure and liner, SFP penetrations, reactor building structure above the SFP, racks and fuel, relevant reactor shutdown systems and other relevant structures
  - Informed by past studies and new analyses
  - Nonlinear finite element analysis of the SFP structure (pseudo-dynamic)



Finite Element Analysis of the SFP



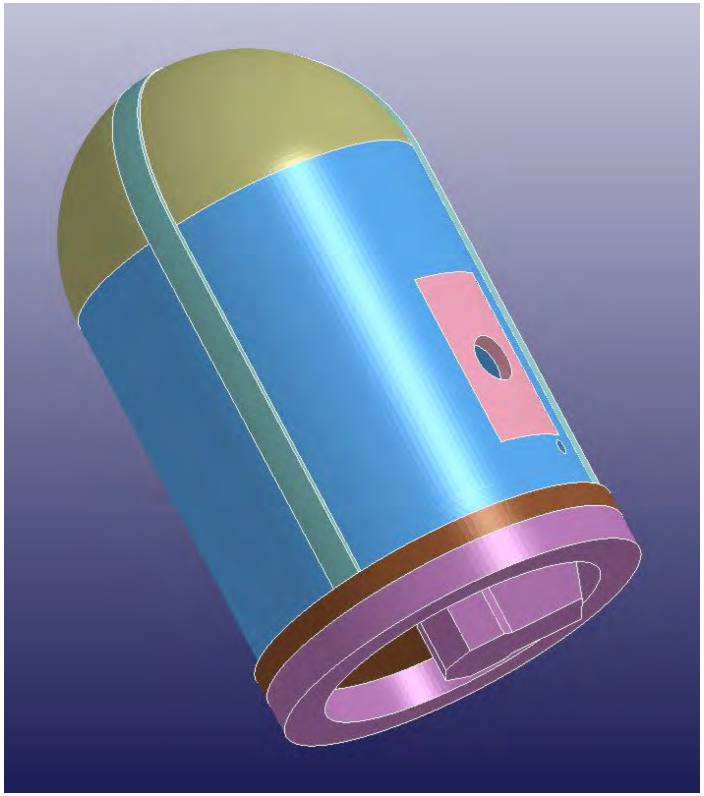


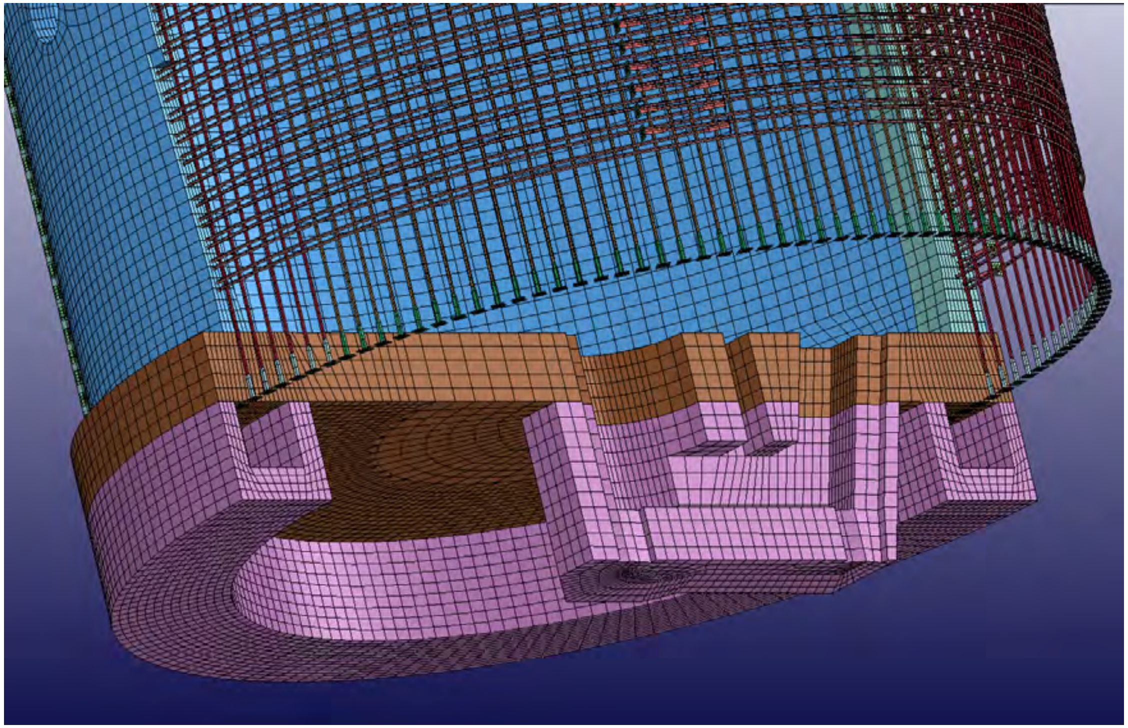
Level 3 PRA Project

- The Level 3 PRA project is a PRA study that includes Level 1, Level 2, and Level 3 analysis (i.e., the onset of core damage, the release of radiological material to the environment, and offsite radiological consequences to people and property)
  - Includes all major site radiological sources (all reactor cores, spent fuel pools, and dry storage casks)
  - Includes all internal and external safety hazards, and all modes of plant operation
  - Uses state-of-practice methods and approaches
- Structural aspects
  - SSC fragilities for external hazards (seismic and high-winds)
  - Performance of pre-stressed concrete containment (Level 2 PRA)
    - Containment fragility for beyond design basis internal pressurization

Level 3 PRA Project

- Structural aspects (Cont'd)
  - Containment finite element modeling





Radiation Effects on Concrete

- Scope:
  - Determine if current methods used to evaluate radiation fluencies on concrete structures are sufficient to determine concrete properties for new reactor designs
- Organization doing Work for NRC: Oak Ridge National Labs (ORNL)
- Progress in last 2 years:
  - Project initiated October 2010.
  - Participated in meetings of the Concrete Irradiation Damage Working Group (EPRI; NRC; Savannah River Lab)
- Expected completion: August 2013

Expanded Proactive Materials Degradation Analysis (EMDA)

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

- In collaboration (and co-funding) with the U.S. Department of Energy's Office of Nuclear Energy's (DOE:NE) Light Water Reactor Sustainability Program (LWRSP), the NRC initiated an expert panel to utilize a modified Phenomena Identification and Ranking Technique (PIRT) process to evaluate aging degradation phenomena on the Core Internals and Primary Systems; Balance of Plant Systems and Components; Reactor Pressure Vessel (RPV); **Concrete Structures**; and Electric Cabling.
- This work expands on the original NUREG 6923, Proactive Materials Degradation Analysis, and looks forward 40 years to at least 80 years of operating life.
- The time line for completing and publishing the EMDA reports (e.g., RPVs, concrete, and cables) is December 2013.

Research Related to Transportation Casks

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

- NRC is analyzing full-scale and scaled drop tests conducted by:
  - Germany's Federal Institute for Materials Research and Testing (BAM)
- An objective is to analyze full-scale and scaled test data from BAM tests to assess sensitivity of results to modeling approaches in order to support identification of best practices for the analysis of SNFT casks under impact loads
- Results would support potential safety assessments that would rely on strain-based criteria (related to future ASME code developments)

Research Related to Transportation Casks (Cont'd)

- Available test data from BAM
- Full scale 30-foot side drop test of CONSTOR® cask manufactured by Germany's GNS
  - The cylindrical container of this cask consists of two co-axial steel cylinders with a specially designed concrete infill for radiation shielding (high density aggregate)
  - Results of analysis with LSDYNA are generally consistent with the test data
- Full-scale and 1:2.5 scale drop tests of the MSF type cask of Mitsubishi Heavy Industries (MHI)
  - 30-foot side and end drop tests for both full-scale and scaled casks

Revision of USNRC Regulatory Guides (Cont'd)

- Update of RG 1.90 [revised and reissued in November 2012]:
  - Guide describes an acceptable approach in developing an appropriate surveillance program for prestressed concrete containment structures with grouted tendons
- Update of RG 1.107 [revised and reissued in June, 2011]:
  - Guide describes quality standards for Portland cement grout as the corrosion inhibitor for pre-stressing tendons in prestressed concrete containment structures

Revision of USNRC Regulatory Guides (Cont'd)

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

- A new RG on implementation of PRA-based seismic margin assessments is under preparation. This RG will provide guidance on a technically acceptable approach that satisfies regulations.

Future Potential Research For Concrete In Nuclear Installations

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

- Update design approach from prescriptive to performance-based concrete materials design
- Establish concrete performance criteria
- Develop standard tests and procedures to verify compliance with criteria
- Modeling of concrete degradation
- Aging and degradation of concrete
  - Structural implications of alkaline-aggregate reactions (AAR) in nuclear power plants
  - Degradation mechanisms for concrete in long term storage of spent fuel in dry casks
- Research supporting review of consensus standards for the design of safety-related SC structures
  - Harmonization of international standards
- Monitoring and surveillance of nuclear power plant structures to assess degradation, monitoring of repairs and condition of new construction

Future Potential Research For Concrete In Nuclear Installations (Cont'd)

U.S.NRC
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Protecting People and the Environment

- Research supporting review of consensus standards for the design of safety-related SC structures
  - Harmonization of international standards
- Monitoring and surveillance of nuclear power plant structures to assess degradation, monitoring of repairs and condition of new construction
- Research on Alkali-Silica Reaction (ASR) degradation of NPP Concrete Structures
  - Completed Scoping Study that summarizes the current state-of-knowledge on ASR effects on long-term mechanical properties of concrete
  - Will initiate research based on filling knowledge gaps identified in scoping study and develop tools to evaluate potentially affected structural capacity and repair techniques

Future Potential Research For Concrete In NPPs (Cont'd)

- Integration with Seismic Research
  - Base-Isolation for NPP Structures and Components
  - Soil-structure interaction modeling with degraded structural elements
  - Seismic assessment of degraded conditions
  - Post-earthquake "as-is" condition assessment
  - Performance of concrete structures in Kashiwazaki-Kariwa Nuclear Power Plant Facilities during July 16, 2007 earthquake
  - Performance of concrete structures following the March 11, 2011 Miyagiken-Oki earthquake

Canadian Nuclear Safety Commission
Commission canadienne de sûreté nucléaire

OECD/NEA/CSNI CAPS ASCET - Assessment of Structures subject to Concrete Pathologies

Neb Orbovic, P. Eng., Technical Specialist
Andrei Blahoianu, P. Eng., Director

OECD/NEA WG IAGE Meeting
Paris, France

April 8th- 12th , 2013

nuclearsafety.gc.ca

ASCET- INTRODUCTION

- Public acceptance of existing nuclear facilities depends on demonstrating adequate structural performance of these facilities during their entire lifetime.
- Taking into account life extensions, long- term operation becomes one of the main concerns of the nuclear community.
- Concrete pathologies [e.g. Alkali-Silica (Aggregate) Reaction] have been detected in concrete nuclear facilities in several member countries.
- it is worthwhile to organize and implement a research activity that can be publicly vetted as a means of establishing and validating evaluation techniques.

ASCET - OBJECTIVES

- The objective of this CAP is to form the foundation of general recommendations for aging management of concrete nuclear facilities subject to different concrete pathologies.
- The types of investigations (and presentation of these investigations) to understand and evaluate concrete pathologies will be decides by each member country based on its needs and sensitivities.

ASCET – Main Research Axes

1. **Material aspect**

The material aspects of concrete pathology which involves detecting concrete degradation, understanding its chemistry, establishing the link between the chemistry and material properties as well as defining material constitutive laws which take into account the pathology.

ASCET - Main Research Axes

2. **Structural aspect, including testing and numerical simulation**

   These studies involve evaluating the performance and integrity of structures subject to pathology, structural acceptance criteria, monitoring the evolution of degradation, and establishing numerical tools for simulation of the structural behaviour, as well as prediction of future behaviour, taking into account the evolution of the concrete degradation.

ASCET - Main Research Axes

3. **Repair techniques , including testing and numerical simulation**

Some member countries can conduct studies on repair techniques and perform structural testing to assess the effectiveness of these repairs techniques as well as to perform numerical simulations to predict the effectiveness of these repair techniques.

ASCET - OUTCOME

- A Workshop that discusses the programs conducted by the various countries participating in this CAPS will be conducted 18 months following agreement on the CAPS.

- This NEA/CSNI CAPS effort will result in a report summarizing collected input from member countries related to the most frequent concrete pathologies and concrete degradation as their consequence.

ASCET – Organization and Schedule

- Lead organisation in ASCET CAPS are:
  - CNSC
  - US NRC
- Establish task group. +1 months,
- Finalize list of research topics to be addressed at the Workshop. +3 months,
- The Workshop is tentatively scheduled for October 2014,
- Venue of the Workshop TBD

Thank you

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