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Working Group on Codes and Standards
International workshop on In-Service Inspection

11-14 April 2022

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Working Group on Codes and Standards

Workshop on In-Service Inspection

Welcome

Day 3 – Wednesday 13 April 2022

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To contact us, please use the chat function.

To ask a question, please use the Q&A section.

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Opening remarks

Dr David RUDLAND

Senior Technical Advisor for Nuclear Power Plant Materials

US Nuclear Regulatory Commission

Ms Carol NOVE

Senior Materials Engineering

US Nuclear Regulatory Commission

Agenda of Day 3

Time*	Topic
2:00PM – 2:15PM	Opening remarks
2:15PM - 3:15PM Session 2	Presentations: - ***Mr Joel HARRISON*** *(PNNL, United States): The Evolution of Ultrasonic ISI for RPV Weld Exams* - ***Mr Nawal PRINJA*** *(WNA CORDEL): AI in UT inspection of welds* - ***Mr Ronan TANGUY*** *(WNA CORDEL): Harmonisation of the certification of NDE personnel*
3:15PM – 3:45PM (30min)	Panel discussion and questions from the audience
3:45PM - 3:55 PM (10min)	Break
3:55 PM – 5:15 PM Session 3	Presentation: - ***Mr David DIJAMCO*** *(NRC, United States): Use of probabilistic fracture mechanics in ISI extension - regulatory perspective* - ***Mr Nathan PALM*** *(EPRI, United States): BWR RPV circ weld inspection elimination* - ***Mr Anees UDAYAWAR*** *(Westinghouse): PWR ISI Interval Extension* - ***Mr Gary LOFTHUS*** *(Southern Nuclear, United States): Licensee perspective*
5:15 PM – 5:45 PM(30min)	Panel discussion and questions from the audience
5:45 PM – 6:00 PM (15 min)	Conclusion of day 3

*Paris time

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Session 2
Evolution of NDE techniques

Mr Joel HARRISON

PNNL, United States

Pacific Northwest
NATIONAL LABORATORY

PNNL-SA-171878

The Evolution of ISI of Nuclear Reactor Pressure Vessels

Joel Harrison
Research Analyst/ASNT NDE Level III

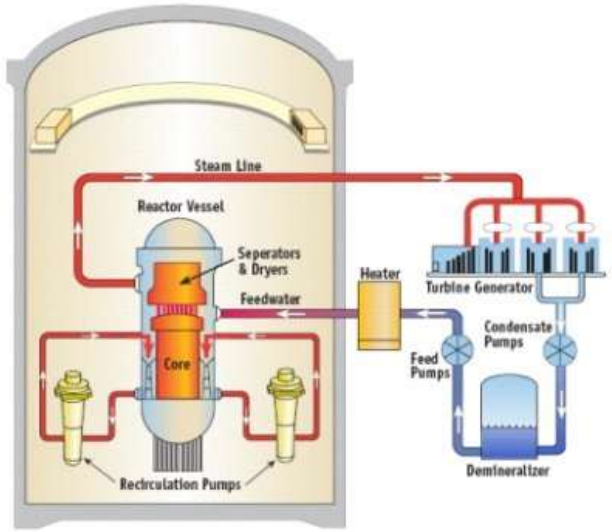
Aaron Diaz
Chief Scientist – NDE & Ultrasonics

WGCS International Workshop on Mechanical Codes and Standards: In-Service Inspection
April 11-14, 2022

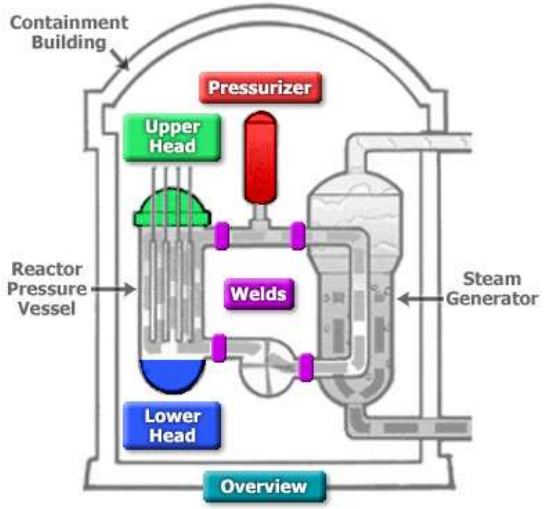
Presentation developed under U.S. NRC Office of Research Agreement NRC-HQ-60-17-D-0010.
Carol Nove, NRC COR

Reactor Vessel

- Inspection & maintenance of nuclear power plant components has evolved over the past 40 years
  - Plant life extension programs
  - Replacement of critical components
    - Steam Generators
    - Pressurizers
    - Recirculation & reactor coolant pumps
- Reactor pressure vessels (RPV) pose challenges
  - Cannot replace the reactor vessel
  - Examinations performed from inside or outside depending on access



BWR



PWR

Inservice Inspection (ISI) of Nuclear Power Reactors

- U.S. Reactor Designs
  - BWR – GE
  - PWR – Westinghouse, B&W, Combustion Engineering (CE)
- Reactor Vessel ISI Divided into 4 categories
  - Reactor Vessel Shell Welds
  - Upper Head Shell Welds & Penetrations
  - Lower Head Bottom Mounted Instrumentation Penetration Nozzles
  - Reactor Vessel Internals
- Examinations defined in American Society of Mechanical Engineers (ASME) Code Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” and EPRI Materials Reliability Program (MRP)

ISI of Nuclear Power Reactors
RPV Nozzle & Shell Welds

Prior to Mid-1990's	After Appendix VIII * Implementation
Exams based on utility or vendor procedures – performance demonstration not mandatory. Included both manual and encoded examinations	Procedures must be demonstrated by performance demonstration. Manual & encoded procedures have been qualified
Personnel certified by employer	Personnel qualified by performance demonstration
Exams based on ASME Section XI, Mandatory Appendix I, Ultrasonic Examinations	Exams Segregated into supplements Supplement 4 – Inner 15% of clad vessels Supplement 5 – Nozzle inside radius Supplement 6 – Unclad vessels & outer 85% of clad vessels
Shell welds examined with standard 45° & 60° Shear wave Transducers	Supplement 4 & 6 welds examined with dual element 60° refracted longitudinal wave mode focused transducers for flaw detection.
Inner radius transducer wedge angles determined from mathematical calculation from design drawings or physical measurements from nozzle	Transducer wedge angles must be determined from site specific modeling to determine scan area, transducer nominal angle, probe skew, misorientation angle, and anticipated maximum metal path (EPRI "Conventional Nozzle Inner Radius Generic Procedure and Modeling Process")

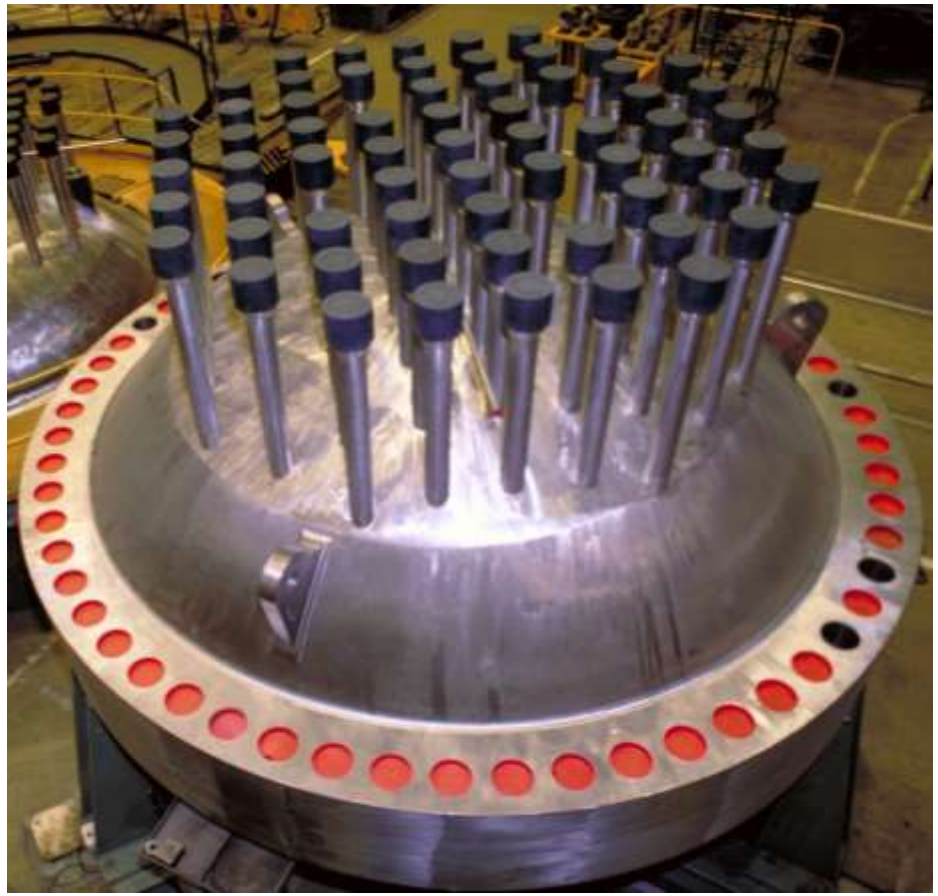
* Section XI, Mandatory Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems"

ISI of Nuclear Power Reactors
Upper Head Examinations - BWR

- BWR RPV heads are examined in accordance with Section XI, Appendix VIII
- Supplement 4, “Qualification Requirements for the Clad/Base Metal Interface of Reactor Vessel”
- Supplement 6, “Qualification Requirement for Reactor Vessel Welds Other Than Clad/Base Metal Interface”

ISI of Nuclear Power Reactors
Upper Head Examinations - PWR

Examination requirements for RPV head penetration welds were originally identified ASME Section XI, Table IWB-2500-1, Examination Category B-P in the 1980 through 2004 Edition and Examination Category B-E in the 1980 through 1992 Edition.



ISI of Nuclear Power Reactors
Upper Head Examinations - PWR

Discovery of Control Rod Drive Mechanism (CRDM) cracking in 2000 led to NRC Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles”

- Licensees directed to evaluate susceptibility of head penetration nozzles to PWSCC

- NRC Order EA-03-009 “Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors”

- ASME Section XI Code Case N-729-1 “Alternative Examination Requirements for Pressurized-Water Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial Penetration Welds”
  - Superseded EA-03-009 by rulemaking in Title 10 of the Code of Federal Regulations (10 CFR) Part 50.55(a)(g)(6)(ii)(D)(1).
  - Licensees were directed to perform demonstrated surface or volumetric leak path assessment of all J-groove welds in the RPV head.

ISI of Nuclear Power Reactors
Upper Head Examinations - PWR

- The Code Case and the NRC 10 CFR 50.55a rule of September 10, 2008, provided requirements to qualify ultrasonic procedures and personnel
  - required demonstrated volumetric leak path assessment procedures
- Since August 29, 2017, as mandated by 10 CFR 50.55a, the reactor vessel upper head inspection and qualification requirements in ASME Code Case N-729-4 have been implemented.
- After the issuance of the ASME Code Case, the EPRI Materials Reliability Program (MRP) directed the MRP Inspection Technical Advisory Committee to develop a performance demonstration program to qualify the nondestructive examination procedures and personnel for upper head penetrations.

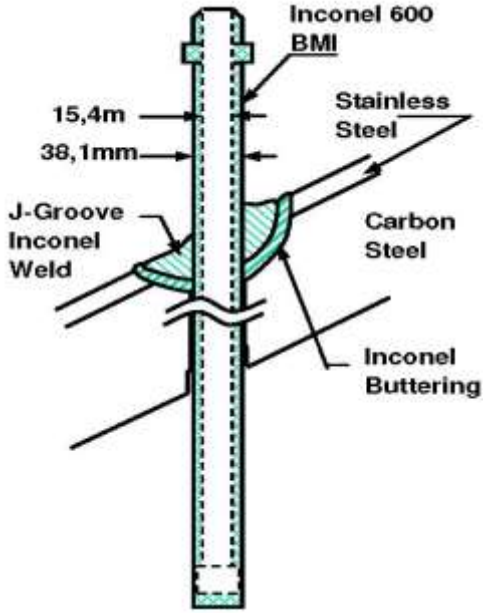
ISI of Nuclear Power Reactors
Upper Head Examinations - PWR

- As a result, in July 2018, an ASME Code action proposed a new Supplement 15, “Qualification Requirements for PWR Reactor Vessel Upper Head Penetrations,” to ASME Section XI, Appendix VIII, to specifically identify performance demonstration requirements for upper head penetrations in Section XI.
  - Supplement 15 approved by the ASME Board on May 28, 2020.

- ASME is now considering a revision to Code Case N-729, now at Revision 9, to remove the upper head penetration examination procedure and qualification requirements and simply reference the requirements of Appendix VIII, Supplement 15.

ISI of Nuclear Power Reactors
Bottom Head Examinations

- Bottom Mounted Instrumentation Nozzles
  - Significant issue is access
    - Physical restriction
    - Radioactive environment
    - J-groove weld surface condition can impact ET exams
  - Examination access must be via remote devices



ISI of Nuclear Power Reactors
Reactor Vessel Internals

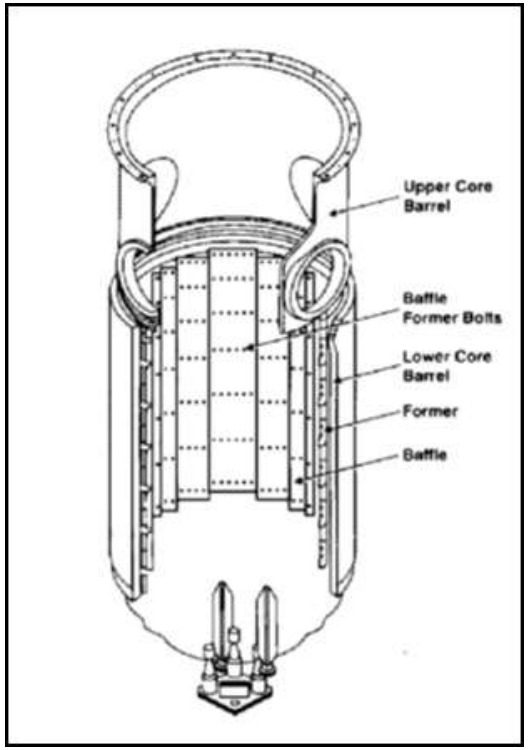
- Examination of reactor vessel internals poses significant challenges
  - Remote examinations in a highly radioactive environment
- PWR Internals provide:
  - Support, guidance and protection of the reactor core
  - A passageway for the distribution of reactor coolant flow to the reactor core
  - A passageway for support, guidance, and protection for control elements in-vessel/core instrumentation
  - Gamma and neutron shielding for the reactor core.

ISI of Nuclear Power Reactors
Reactor Vessel Internals

- RPV internals are subjected to:
  - Thermal & irradiation embrittlement
  - Flow-induced vibration and wear
  - Irradiation Assisted Stress Corrosion Cracking (IASCC)
  - Neutron embrittlement of nozzle forgings & upper shell course
  - General material aging
- As part of the license renewal process, licensees have committed to periodically inspect the reactor vessel internals

ISI of Nuclear Power Reactors
Reactor Vessel Internals

- Baffle-former Bolts
  - hold together a structure inside the reactor vessel of Westinghouse pressurized water reactors (PWRs).
  - In spring 2016 two PWRs in the USA identified a large number of degraded baffle-former bolts during refueling outage ultrasonic (UT) inspections
  - During normal operation, the separation of a bolt head or a bolt lock tab can introduce loose parts or foreign material to the reactor coolant system.
  - The MRP convened a special focus group on May 16, 2016, to support an integrated approach among industry organizations and address the recent operating experience with baffle-former bolts.



ISI of Nuclear Power Reactors
Reactor Vessel Internals

- Materials Reliability Program addressed these issues in MRP-227 "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"
  - MRP-227 Revision 1-A issued in 2019 following an NRC Safety Evaluation (contained in Revision 1-A).
  - Provides Inspection and evaluation guidelines
  - Guidelines for use by industry to develop engineering programs to manage aging in PWR internals
  - Guidelines to monitor the condition of the internals to maintain appropriate levels of plant safety and reliability
  - Systematic approach to monitor potential degradation of internals components, support power up-rate activities, and meet license renewal requirements.

ISI of Nuclear Power Reactors
Reactor Vessel Internals

- MRP-228 “Inspection Standard for Pressurized Water Reactor Internals”
  - developed to implement the applicable reactor internals inspection program described in MRP-227
- Contains requirements specific to the examination methodologies and requirements for qualification of the nondestructive evaluation (NDE) systems used to perform those inspections
  - Visual VT-1, Visual VT-3, Enhanced Visual EVT-1, Eddy Current, Ultrasound
- MRP-227 and MRP-228 do NOT supersede ASME Code Section XI requirements

Summary

- Operating experience drives changes to codes & standards and regulatory requirements.
- The most significant improvement to ASME Section XI inservice inspection of reactor pressure vessel shell welds has been the implementation of Appendix VIII and mandatory performance demonstration for procedures & personnel
- Performance demonstration requirements for upper head exams have been incorporated into ASME Code Section XI, Appendix VIII
- EPRI Materials Reliability Program addresses emergent issues with aging plant components, for instance, through the implementation of MRP-227 & MRP-228 for inspection of PWR RPV internals

Dr Nawal PRINJA

WNA Cordel

AI for UT of Welds

NEA/CNRA/USNRC VIRTUAL WORKSHOP ON ISI (APRIL 11-14, 2022)

Organised by the NEA Working Group on Codes and Standards (WGCS) in co-operation with the United States Nuclear Regulatory Commission (US NRC)

Prof. Nawal K Prinja BSc, MSc, PhD, DEng, CEng, FIMechE

Contents

- Brief Introduction – What is AI
- Why we need AI for weld inspection
- AI Models
  - Type of AI Models for UT
  - Training of AI Models
  - Testing and accuracy of AI models for UT
- Future Tasks

My Introduction

Current Position

- Technology Director, Jacobs
- Honorary Professor in the School of Engineering at Aberdeen University.
- Honorary Professor in the College of Engineering, Brunel University London.
- Honorary Professor, School of Engineering, Bolton University.
- Honorary Professor, School of Computer Science and Electronic Engineering, the International Centre of Nuclear Engineering, Bangor University.

Experience

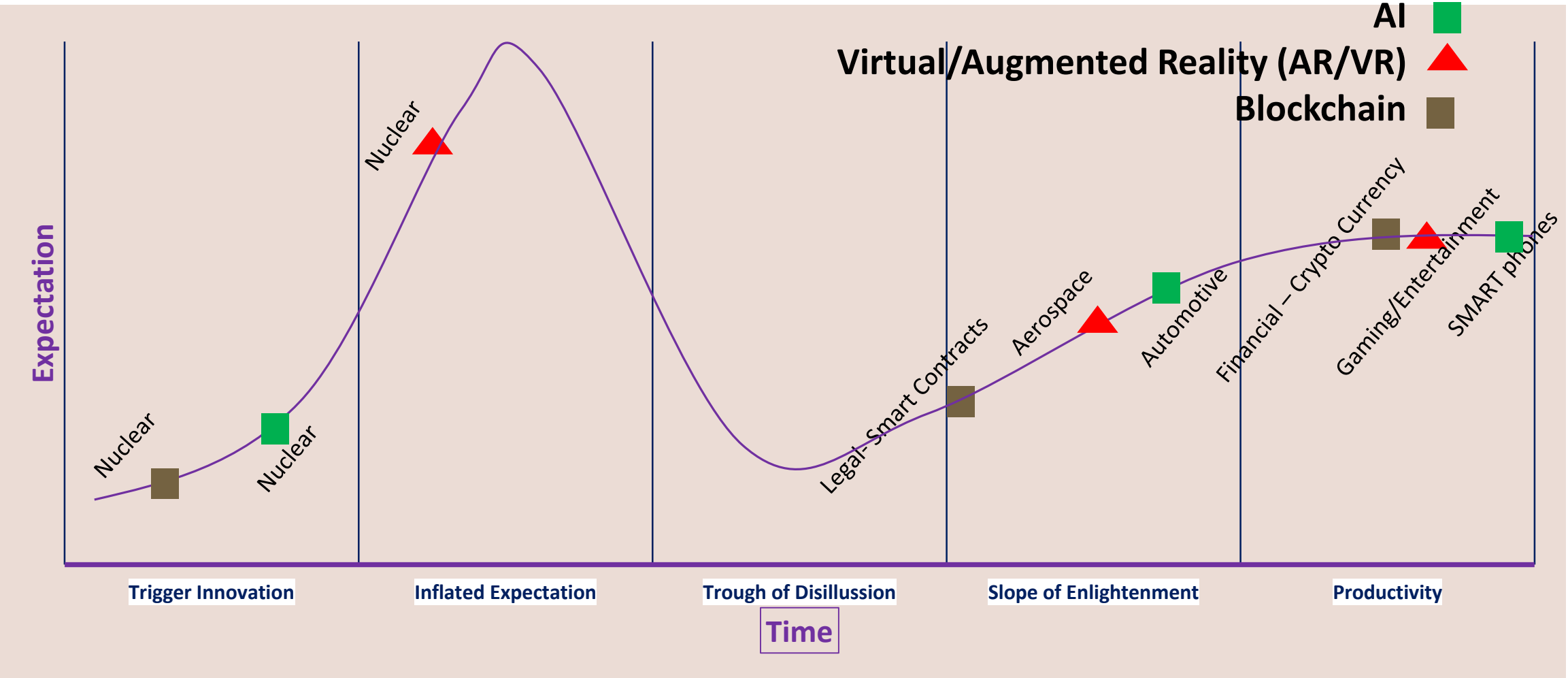
- 40 years of engineering and technology experience in aerospace, automotive, oil & gas and nuclear power.
- Over 50 Technical publications including 3 books.

Appointments

- Advisor to the Ministry of Defence (MOD) on the Nuclear Propulsion Research & Technology programme for the nuclear submarines and chair of AI Technology Focus Group for nuclear propulsion.
- Member of the Nuclear R&D Advisory Board to the UK Government.
- Chairman of CORDEL at WNA.
- Technical Expert invited by the IAEA (United Nations) to chair expert meetings on safety classification and Technology Readiness Levels and to participate in Nuclear Knowledge Management and Seismic expert missions to UAE, S Africa, China, Spain and Poland.
- Independent assessor appointed by the Innovate UK of UKRI.
- Member of the EC funded FENET and EASIT2 projects aimed at developing computer based simulation competencies.
- Member of the Board of Directors for the Professional Simulation Engineer (PSE) certification scheme.
- Chair of Industry Advisory Committee for the National Structural Integrity Research Centre at Cambridge.
- Ex-Member of Technical Assessment Panel of Fusion for Energy (F4E)
- Member of the Fusion Advisory Board, EPSRC of UKRI.
- UK representative and Vice Chair of the Senior Industry Advisory Panel of Gen IV International Forum (GIF)
- Member of Plant Systems Design code committee of ASME.

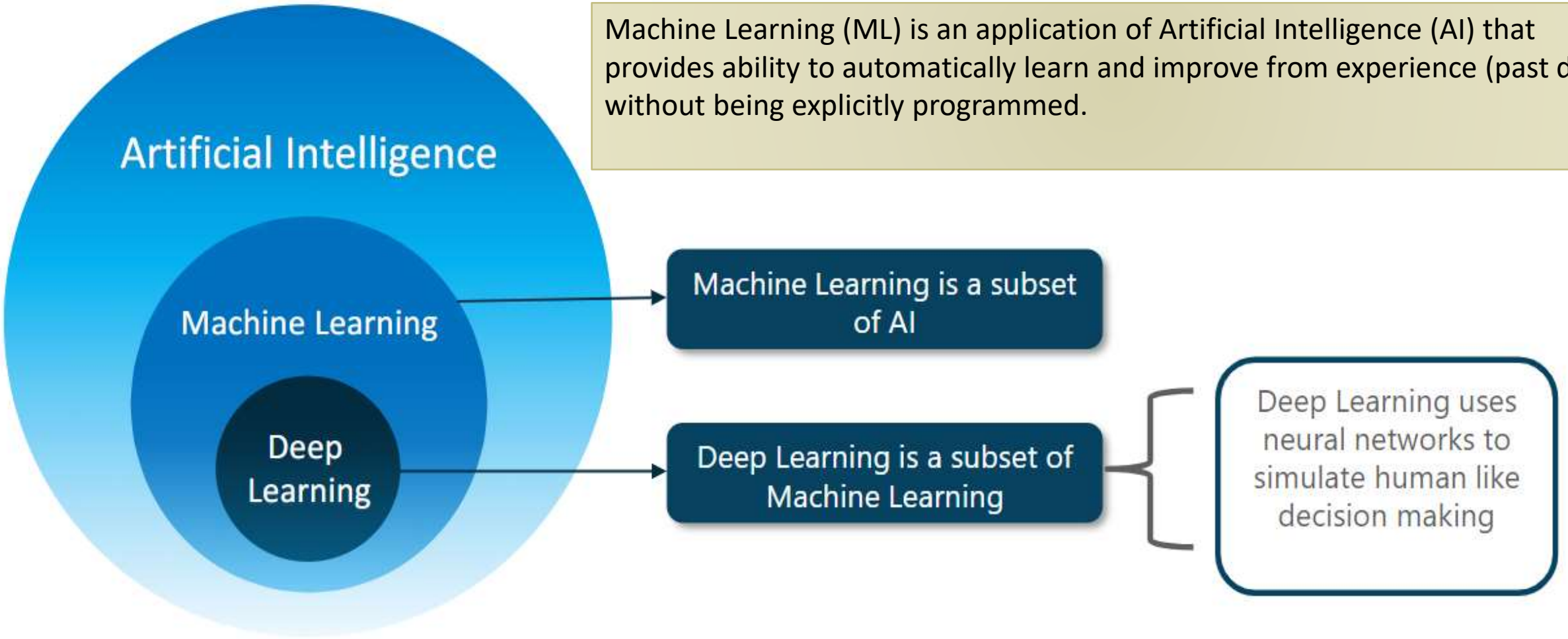
New Technology : Expectation vs Time

Is Nuclear Sector Lagging Behind Others?

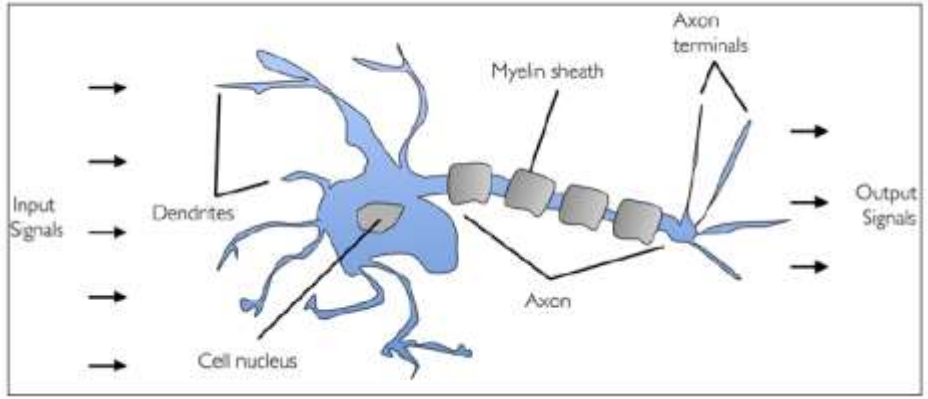


Artificial Intelligence - Subsets

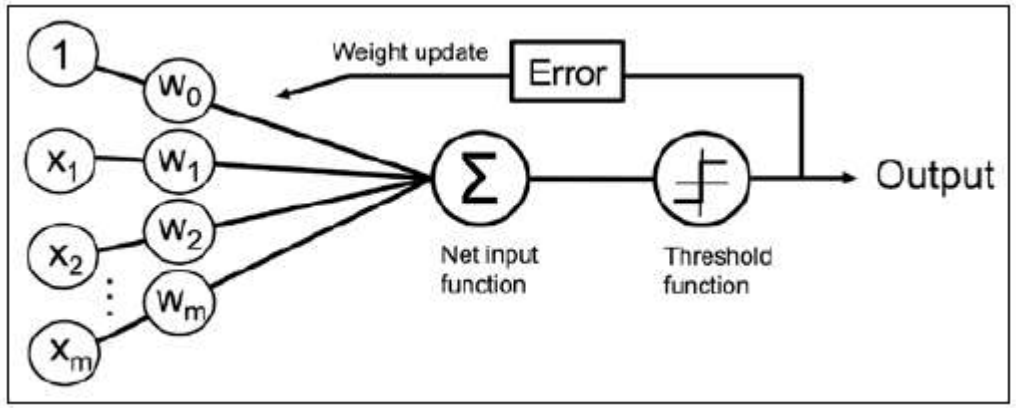
Machine Learning (ML) is an application of Artificial Intelligence (AI) that provides ability to automatically learn and improve from experience (past data) without being explicitly programmed.



Deep Learning : Artificial Neural Network

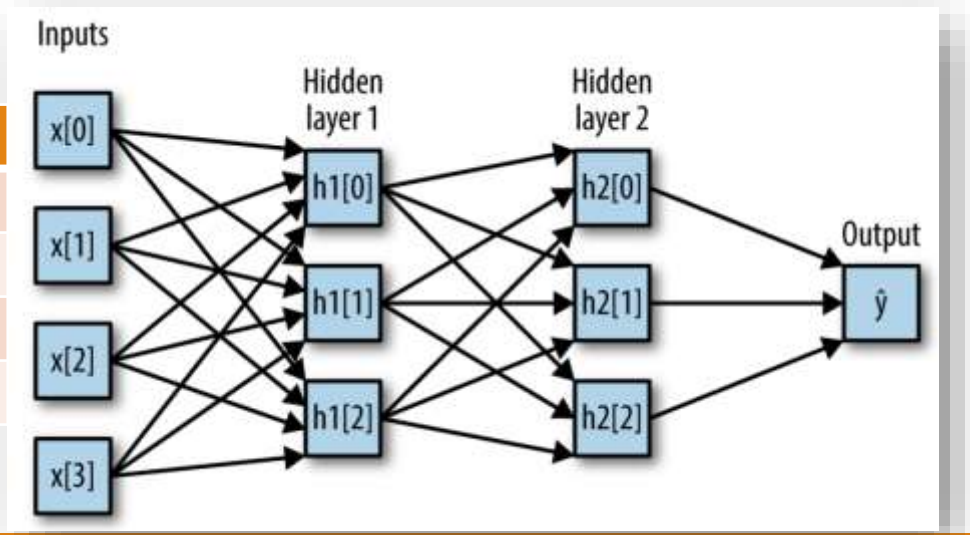


Neurons are interconnected nerve cells in the brain that process and transmit chemical and electrical signals.

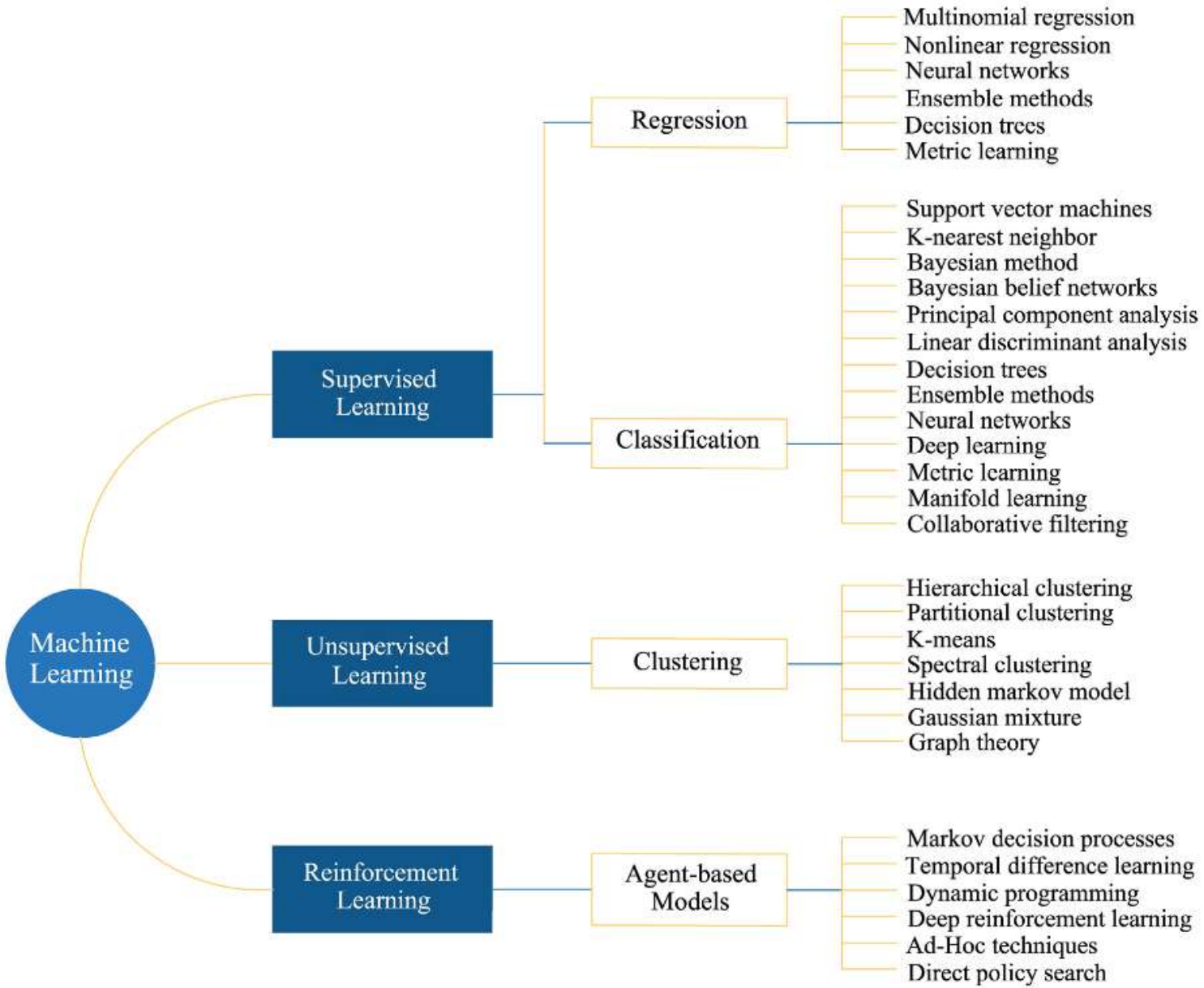


Perceptron receives the inputs **x** and combines them with the weights **w** to compute the net input. The net input is then passed on to the threshold function, which generates a binary output +1 (Go) or -1 (No-go).

	Brain	Computer
No. of units	~10^{11}	~10^{9}
Type	Neurons	Transistors
Switch time	~10^{-3} s	~10^{-9} s
Model	Parallel	Serial



- Large neural networks made up of many hidden layers of computation inspired the term “deep learning.”
- No need to explicitly program.
- They learn from training samples.



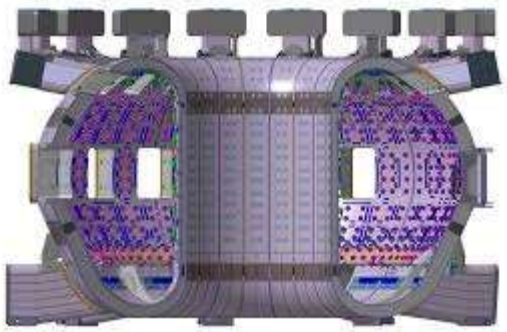
ML Algorithms used for AI4UT

How to select an algorithm ?

1. Training data – quantity with no bias
2. Level of accuracy
3. Complexity – Tuning of parameters
4. Nonlinearity
5. Number of features
6. Scalability

Example : AI model to accelerate construction of Vacuum Vessel for Fusion

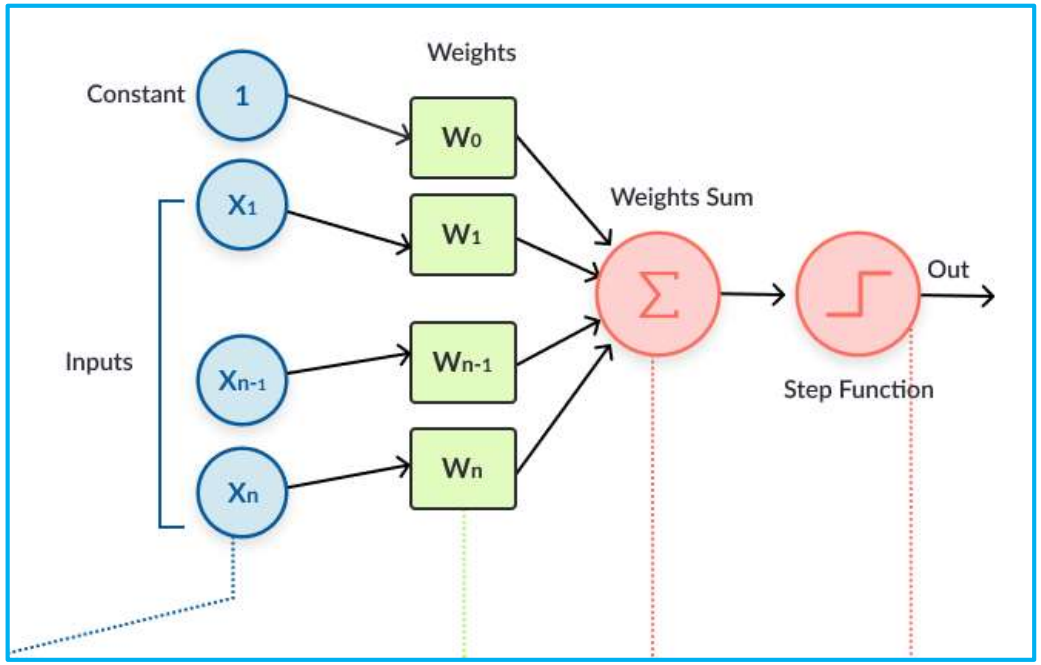
Based on input of parameters of planned welding it can use previous experience to predict the outcome and help with decision to proceed or not leading to better planning, de-risking and accelerating the programme.



Input: EB welding parameters.

EB Parameters	EB Conditions
Length	Sector/ Segment
Current	Orientation
Welding Velocity	Supplier
Focussing System Current	Type of weld
	Position

AI Model



Machine Learning from previous weld data

Trained AI using 1802 welds data. Predicted outcome of the 70 new welds to do.

There are 9 sectors. Each sector requires over 10000 different types of welds with full 100% inspection.

Output: Probability of non-conformance

Result	Conformance
NC_EBW_R	Not conform - Defect related to EB Process - Weld to be repaired
NC_OTH_R	Not conform - Defect not related to EB Process - Weld to be repaired
NC_OTH_NR	Not conform - Defect not related to EB Process - No repairing need
NC_DOC_NR	NC related to documentation - No repairing need
C	Compliant- Weld accepted

Prediction	Actual
One weld with 90% chance of failure	Failed
16 others with 56% to 60% chance of failure.	7 out of 16 failed

Quality Control vs Quality Assurance

Analysis and evaluation of test results, Learning from Experience and implementing a plan to reduce risk can be improved.

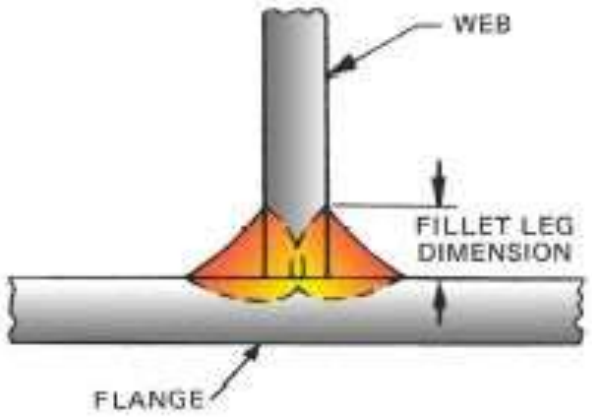
Overall QA is very good.
Processes and responsibilities are well defined.

AI can de-risk and accelerate projects

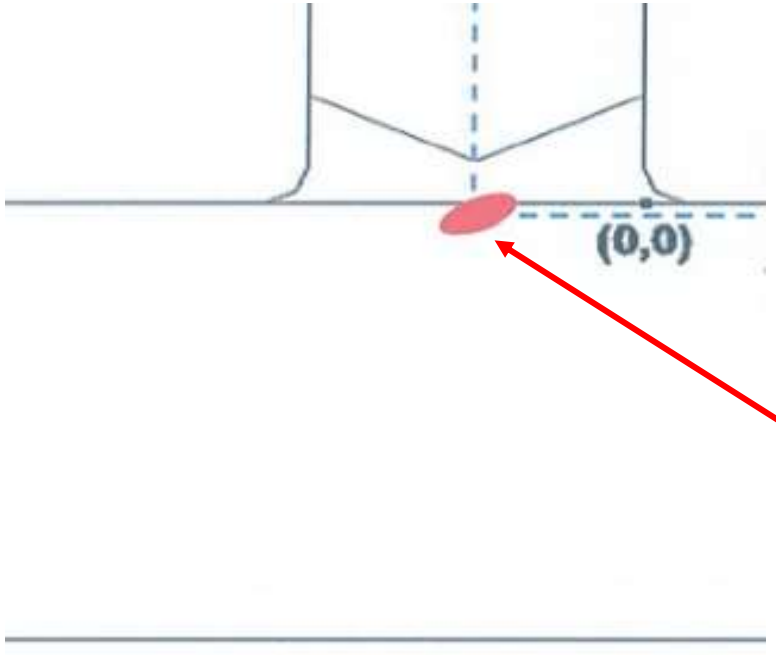
Why we need AI for UT

- As a digital aid to assist inspectors and help increase probability of detection
- Accelerate the UT inspection process
- Introduce automation
- Keep humans out of harms way and increase safety
- UT is being preferred over RT but its speed and accuracy needs to increase

Example : AI for UT Inspection of welds



Double or single fillet welds considered

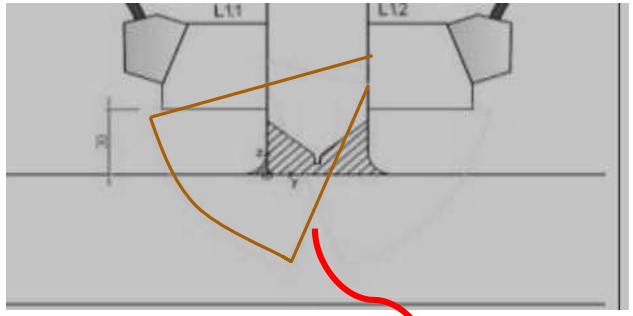


Flaw in full penetration T-joint

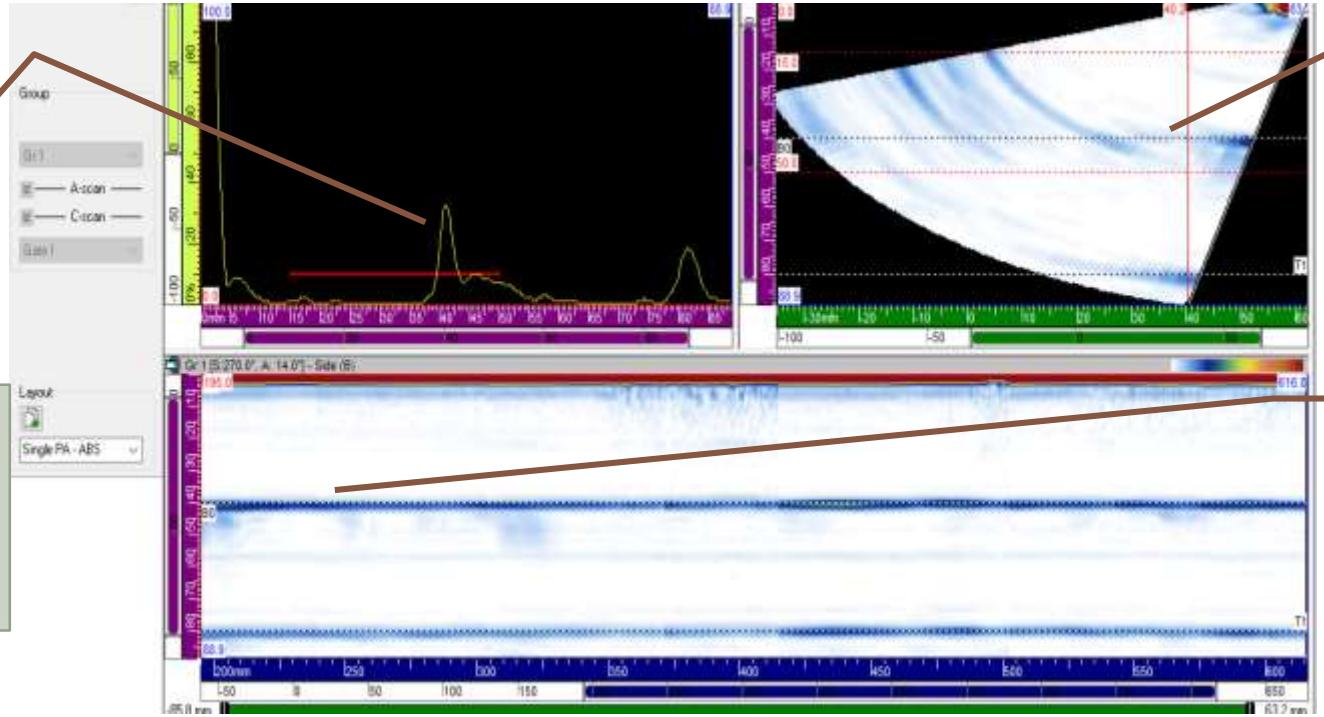
- UT is preferred over radiographs
- It can cost 1$m to 5$m to train and qualify an inspector
- UT acquisitions and analysis can take days

Typical UT Acquisitions

Recently, for a particular weld form, AI model identified defects with accuracy of 98%.



A-Scan
Beam indicating presence of defect through sound echo
320 X-Y data points



S-Scan Sector

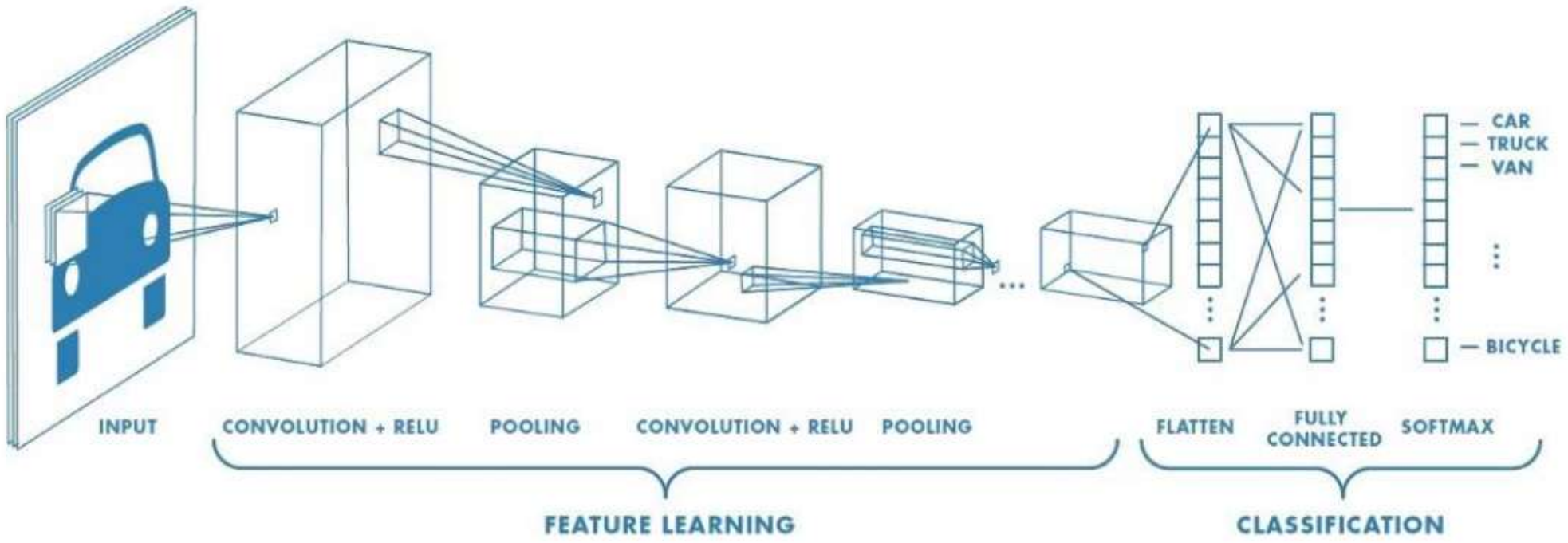
Built from number of beams (29 or greater) swept across the sector. 29x320 data points.

AI models will use A and S Scans which are selected by moving along the B-scan and recorded every 1mm interval along the length of the weld.

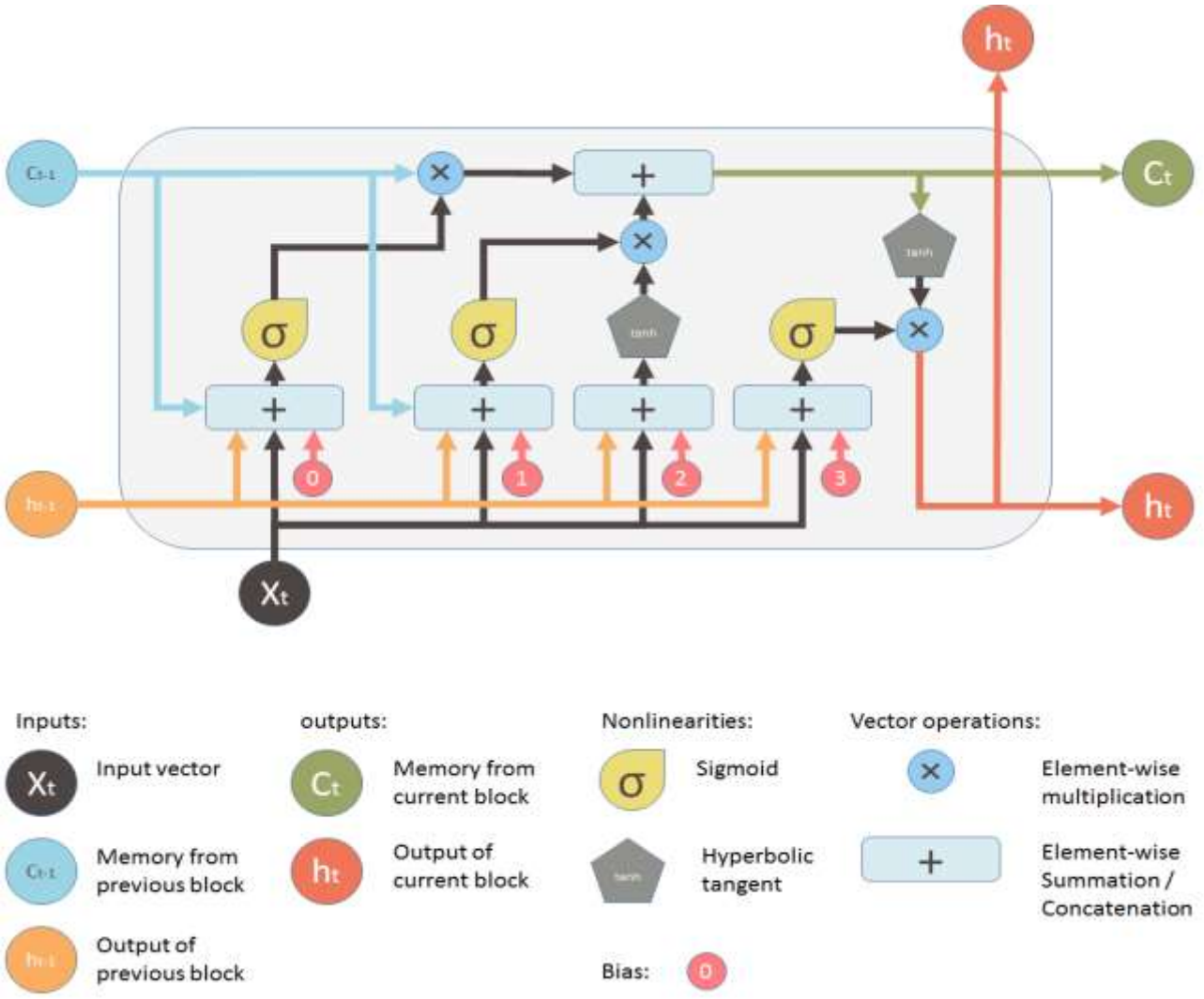
B-Scan (Side)
A scan over 400mm will generate 400 x 29 x 320 = 3.71 million data points

CNN for Image analysis

A Convolutional Neural Network (CNN) is a Deep Learning algorithm which can take in an input image, assign importance (learnable weights and biases) to various aspects/objects in the image and be able to differentiate one from the other.



LSTM for time series (wave) analysis



Instead of having a single neural network layer, there are four layers (0 to 3) that interact in a very special way.

- 0 Forget Layer with Forget Valve
- 1 Input gate Layer to Update
- 2 New Values and New memory Valve
- 3 Output Valve

There is a memory unit and three information gates or valves.

The gates are different neural networks that decide which information is allowed in the cell state. The gates can learn what information is relevant to keep or forget during training.

The key to LSTMs is the cell state, the horizontal line running through the top of the diagram. It is a kind of conveyor belt running straight down the entire chain, with only some minor linear interactions (x or +).

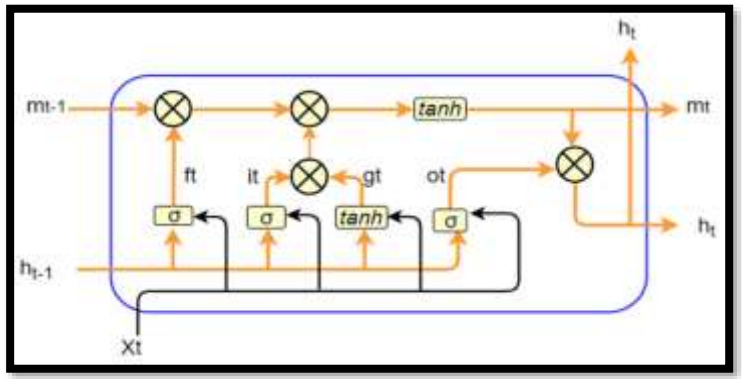
You can train LSTM to predict share values in stock market but do it at your own risk.

AI4UT Models

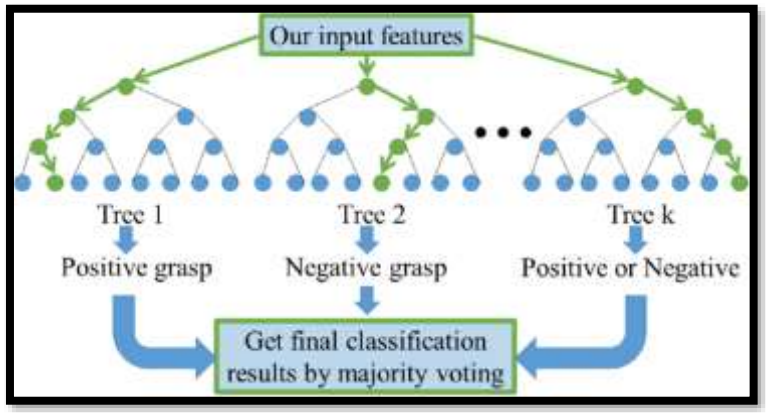
Three types of AI models have been used for classification (defect / No defect):

i. Convolutional Neural Network (CNN) models for image processing (S-scans).
ii. Recurrent Neural Networks (RNN) or more advanced LSTM for processing of the acoustic signals (A-scans).
iii. Random Forest classifier of A-scans.

Example of CNN for image processing



Example of LSTM model for wave analysis



Random Forest Classifier

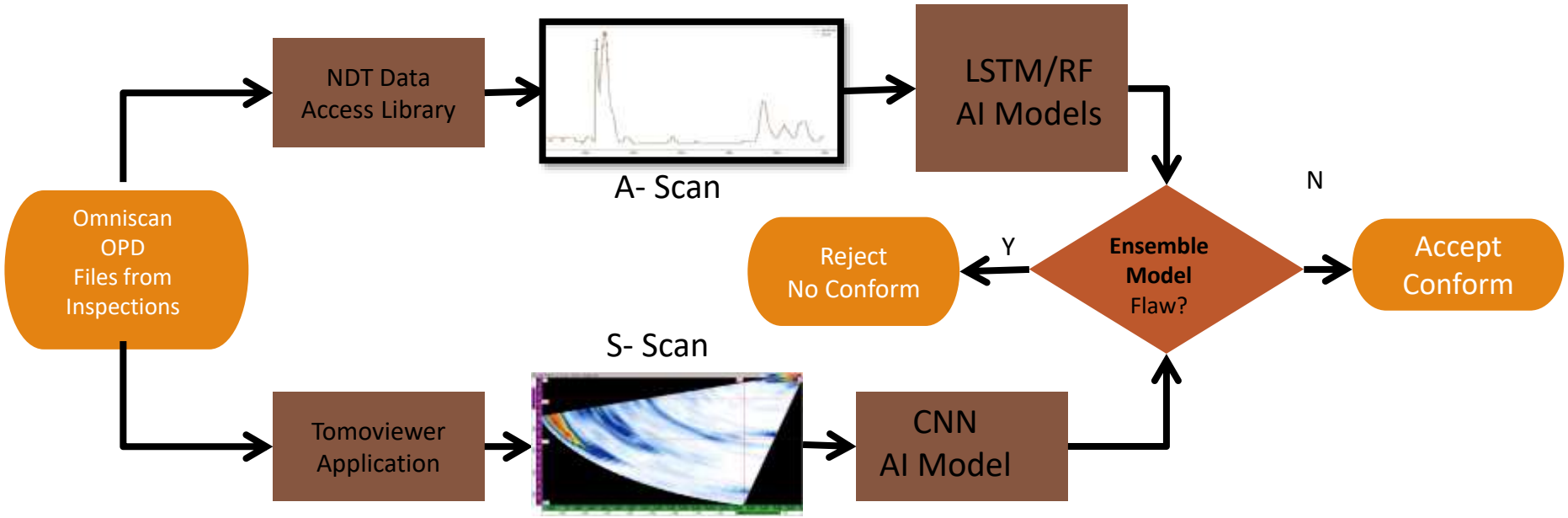
Proof of Concept Study to Apply AI for Weld UT Inspections

Problem Statement

Full UT examination of welded joint is an essential part of construction of a safety Class 1 component. There are several issues which prevent acquisition of a good valid image for UT inspection and it can take several hours to analyse. AI can help select valid images and help make rapid decision on presence of a flaw.

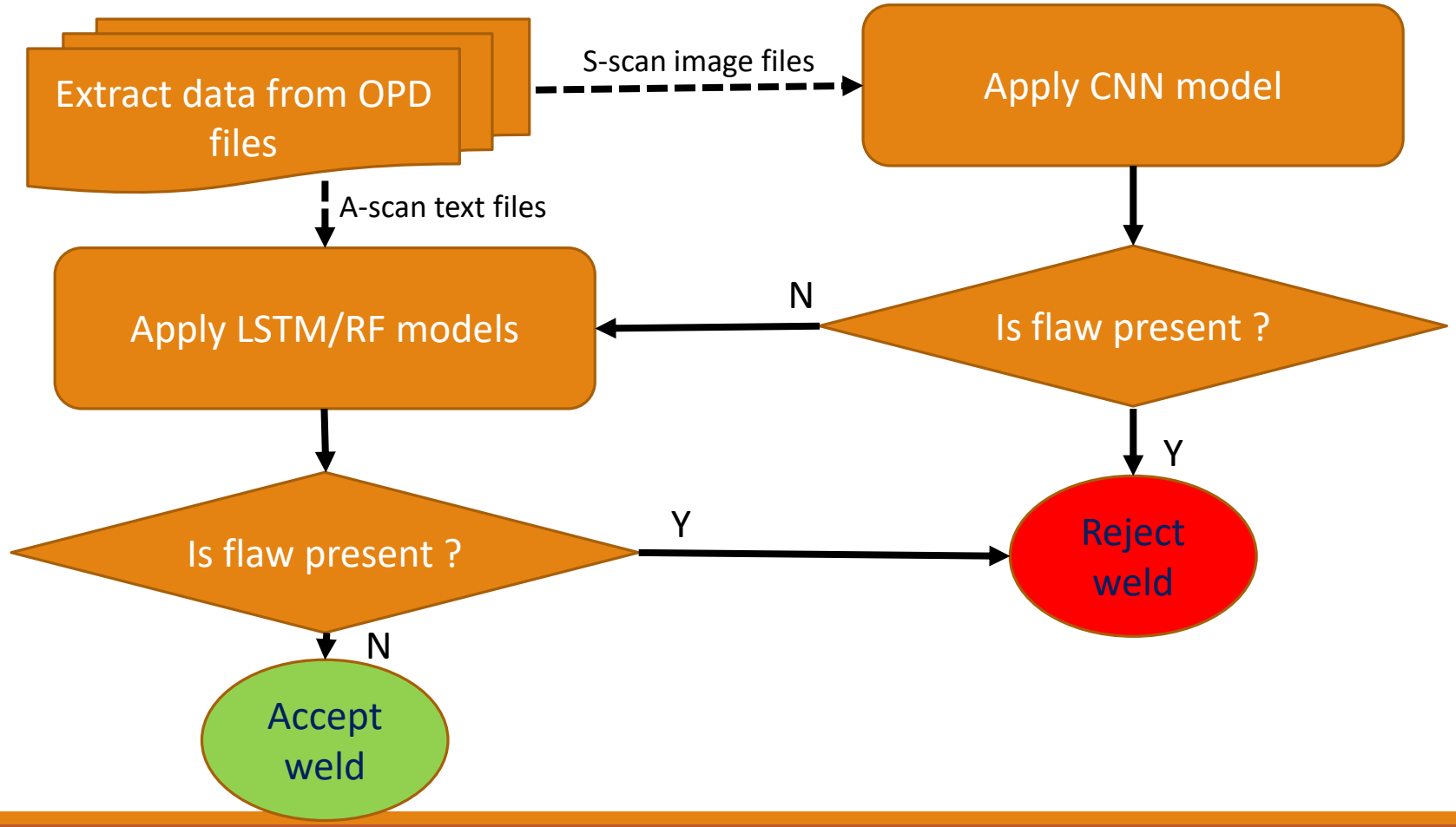
Solution

- Image processing AI models (CNN) to analyse S-scans
- AI models (LSTM/RF) to analyse A-scans
- Ensemble from A and S scans to classify if UT image contains defect



End

Ensemble to decide presence of flaw



1. Total length of weld in mm covered by the scan
2. Number of S-scan images. Should be one for every mm of length.
3. If S-scan indicates flaw. Reject weld.
4. If S-scan shows no flaw then do A-scan checks by LSTM models. Can be 29 beams (A-scans) in each S-scan.
5. If A-scan indicates flaw. Plot the A-Scan and reject weld.
6. If A-scan shows no flaw, accept the weld.

Training and Testing of AI Models

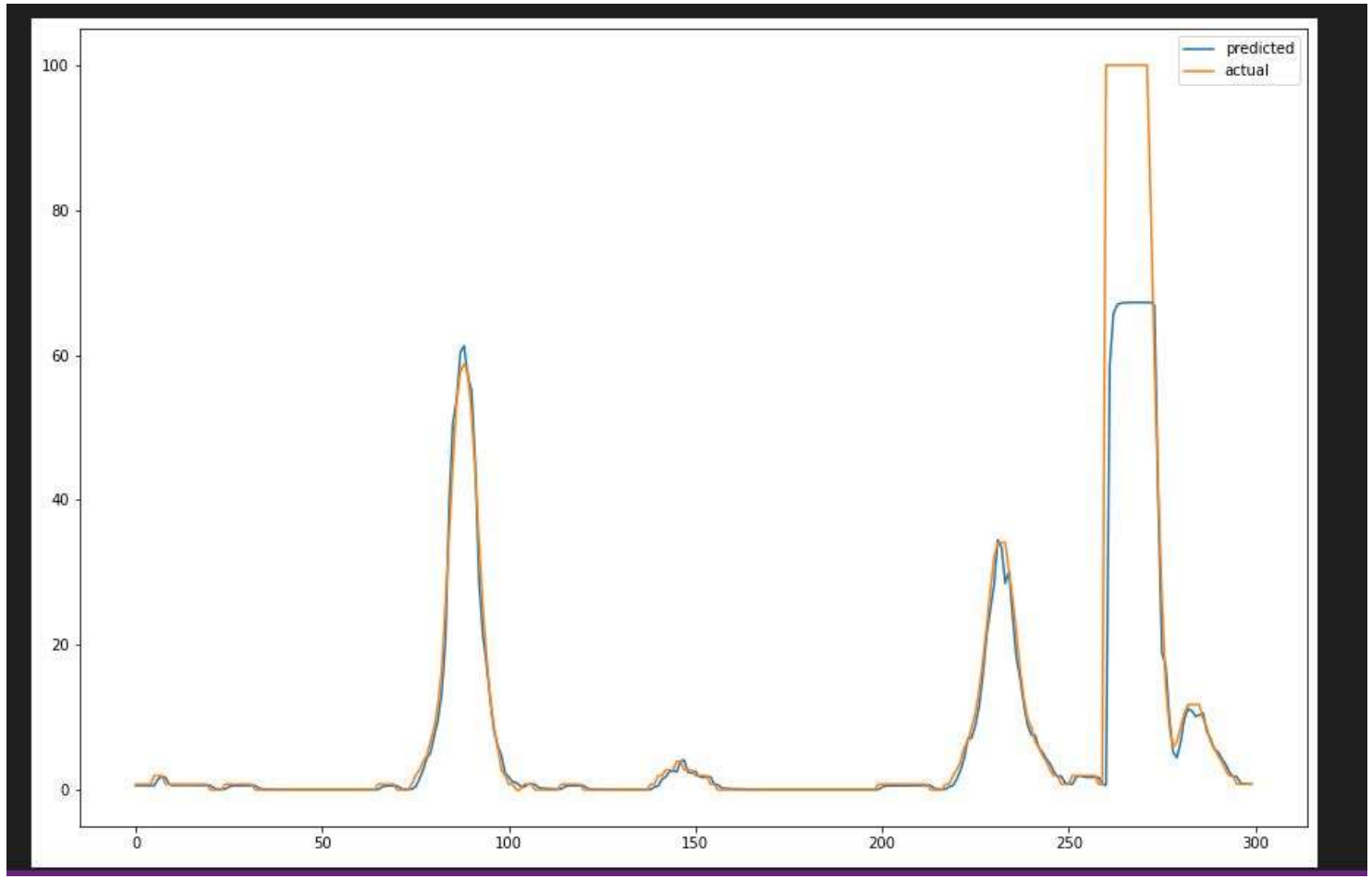
In total 5 AI models are trained and tested.
One CNN model for S-scan images for single and double fillet welds.
Four LSTM models for A-scans with and without flaw for single and double fillet welds.

AI Model	Scan type	Number of scans used for training	
		Without flaw	With flaw
LSTM (Double fillet)	A-scan	995	995
LSTM (Single fillet)	A-scan	1800	900
CNN (both double & single fillet)	S-scan	2668	2668

Testing on unseen data

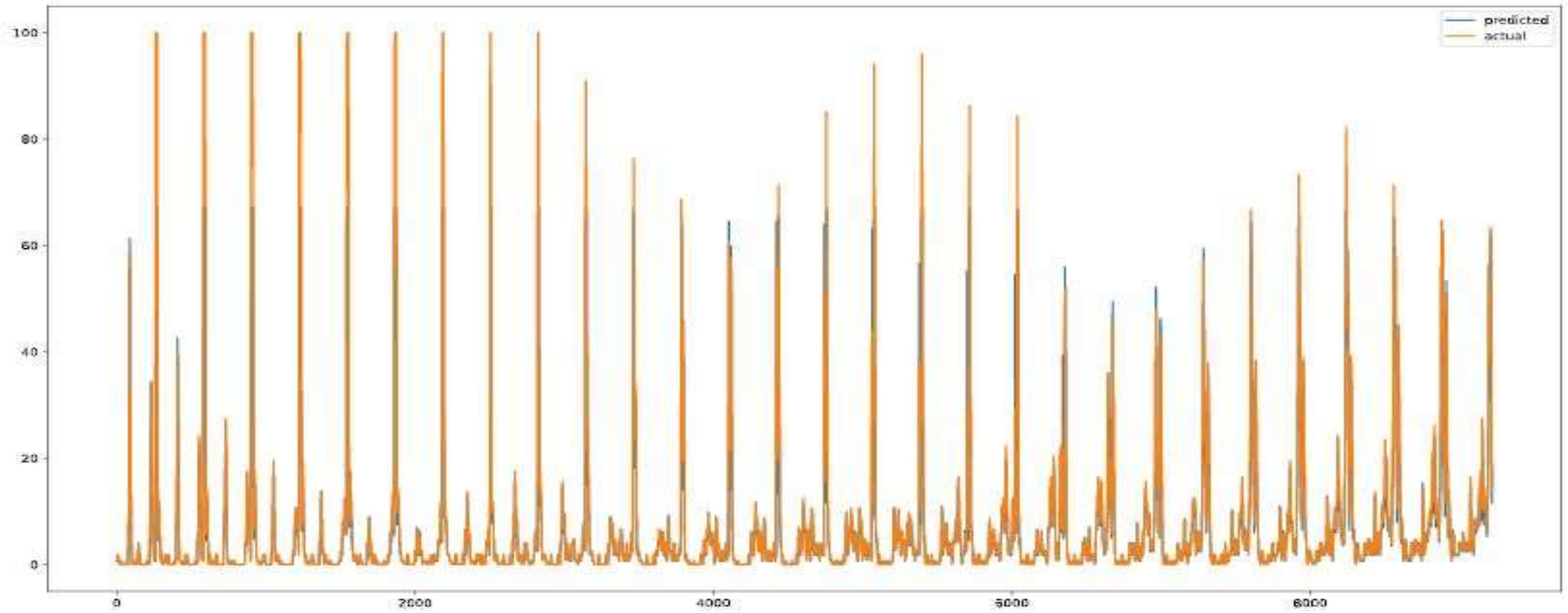
AI Model	Accuracy
LSTM for A-scan (with defect)	23/29 (79%)
LSTM for A-scan (without defect)	157/186 (84%)
CNN for S-scan (with defect)	29/29 (100%)
CNN for S-scan (without defect)	16/16 (100%)

LSTM Model for A-scans



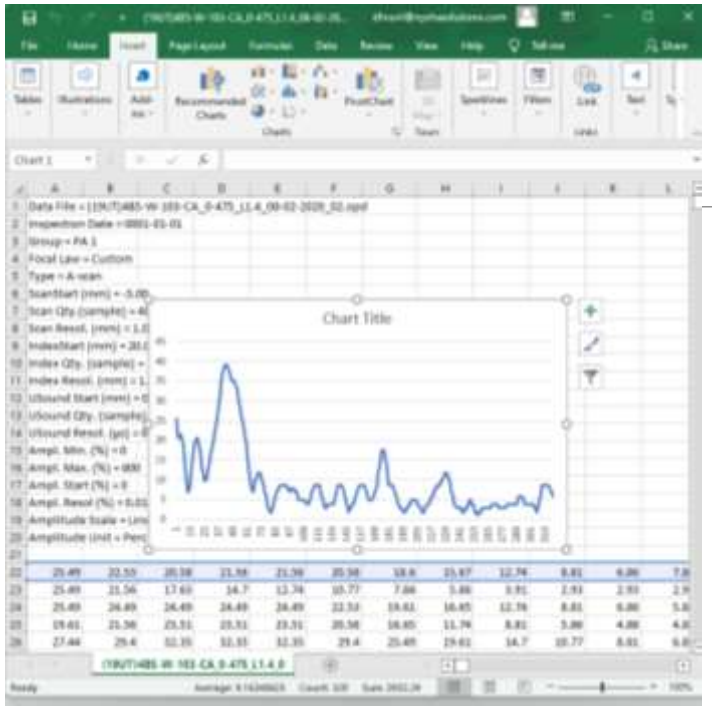
Mean Square error (MSE) is used between predicted and actual to see which LSTM model fits (with or without defect)

LSTM Output Predictions for all Beams in A-Scan

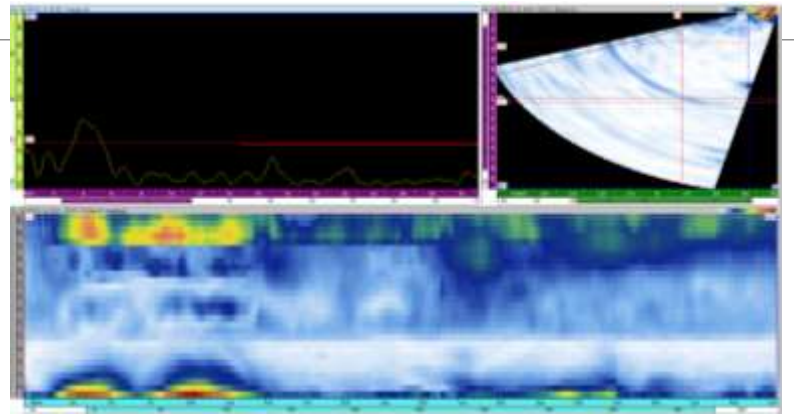


All 29 A-Scan beams. With 320 X-Y data points per beam, this plot shows 29 x 320 = 8280 points

Fast Extraction of A-Scan Data with OMNIPC



Extracted Data, each row is A-scan with 320 columns



Tomoviewer

OMNIPC

A-Scan Extracted Data Comparison with Tomoviewer & OMNIPC

Accuracy Achieved with Fast Extraction of A-scan Data

Training

Total number of A-scan beams = 1356
No flaws = 685
With flaws = 671

Testing

Total number of A-scan beams = 339
No flaws = 165
With flaws = 174

Algorithm Name	Accuracy
Random Forest Algorithm	99.70%
LSTM	81.00%
XGBoost	99.11%
SVM	99.41%

Future Tasks

1. Develop End-to-End UI.
2. Full ‘AI for weld UT’ system launch in workshops.
3. AI models to help distinguish between valid and invalid acquisitions.
4. AI for UT models that can be applied by welders
5. Integrate AI and probabilistic approaches to enhance the sparse data set

Enhancing the Sparse Data-set

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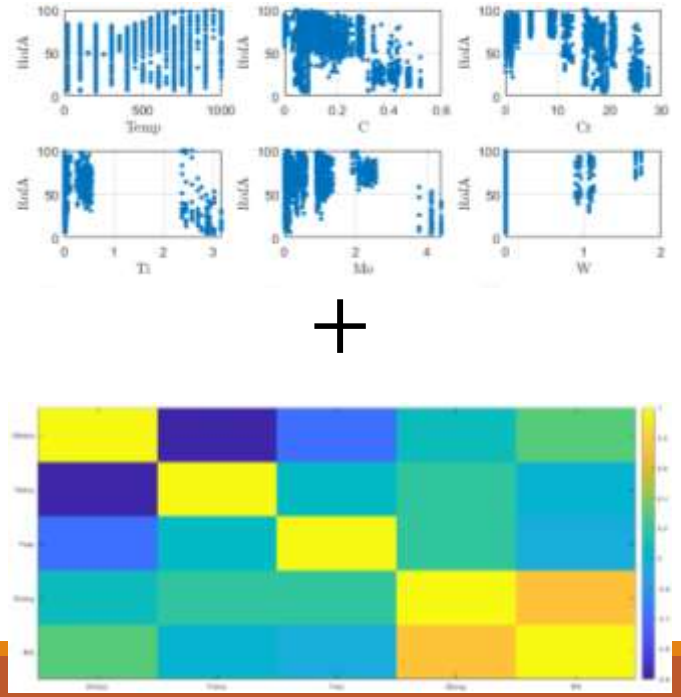
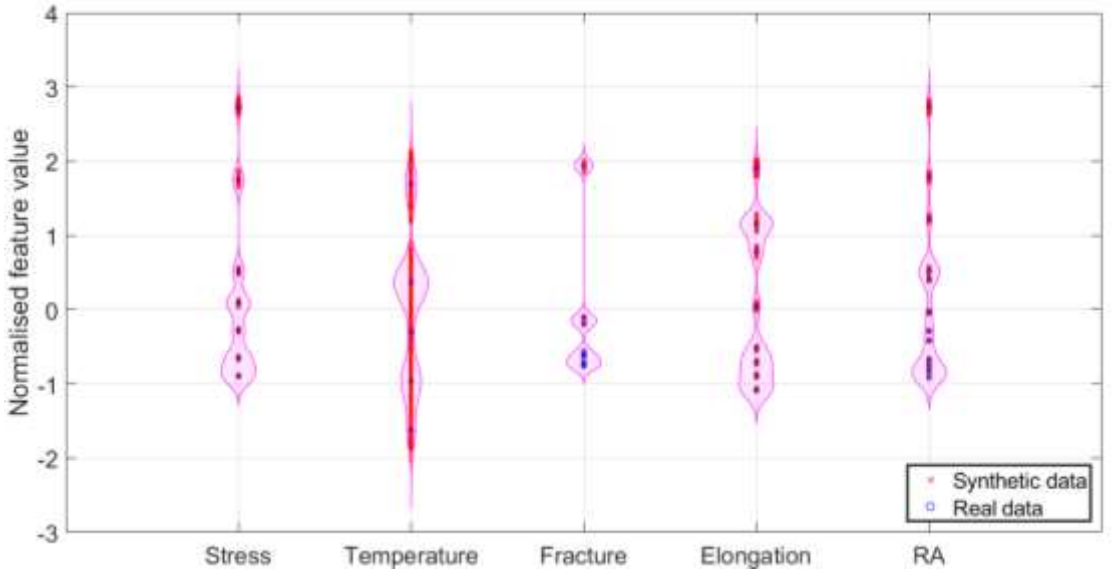
Another PoC study completed jointly with University of Strathclyde and Liverpool University in the UK

Stochastic enhancement of data-set:

- Retaining the physical relationship among variables;
- Consider the associated uncertainty of each data-point (simulated);
- From initial 13 points (1 category only), to 10000 training data points.





Conclusion : The Way We Do Engineering is Changing

Impact of AI and Data Science

Past

Empirical Understanding

Present

Mechanistic Understanding

Future

Data Based Understanding

- **Regulatory approval needed for AI for UT**
- **ISI is an ideal candidate to benefit from AI and Data Science**

Mr Ronan TANGUY

WNA Cordel

WORLD NUCLEAR ASSOCIATION

CORDEL: Certification of NDE Personnel

Harmonization of International Code Requirements

Ronan Tanguy, CORDEL Programme Lead

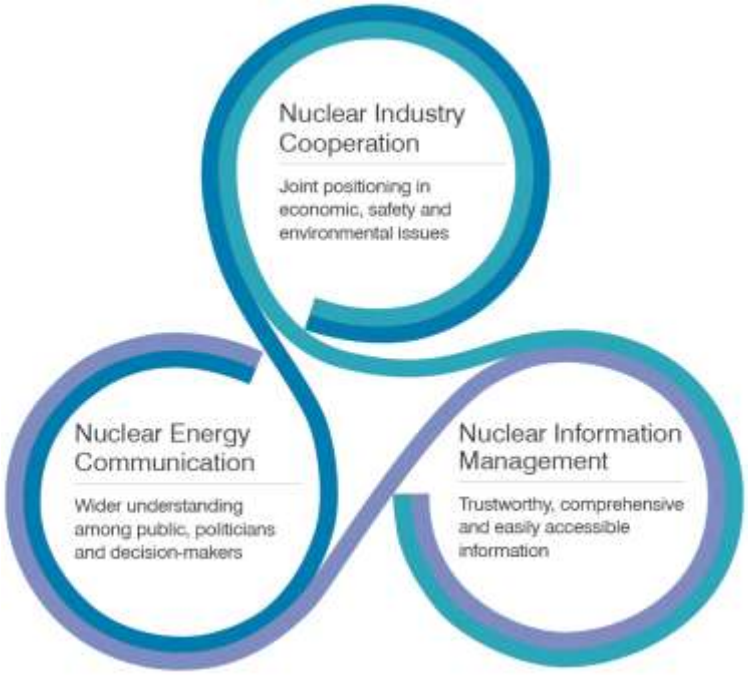
WGCS International Workshop
on Mechanical Codes and Standards:
In-Service Inspection
13th April 2022

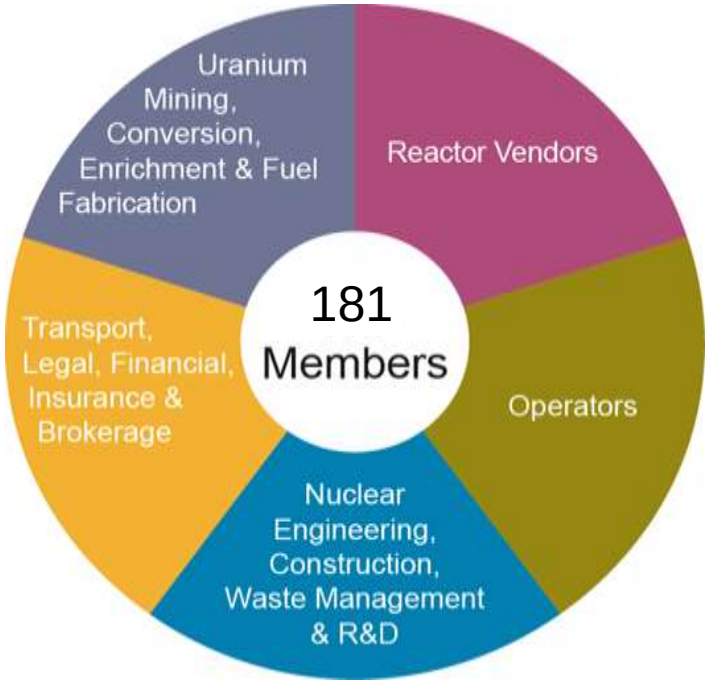
World Nuclear Association

- The World Nuclear Association is the international organization that *promotes* nuclear energy and *supports* the many companies that comprise the global nuclear industry

- World Nuclear Association membership encompasses all aspects of nuclear energy

- CORDEL working group aims to standardize reactor designs so they can be deployed internationally without major design changes due to national regulations



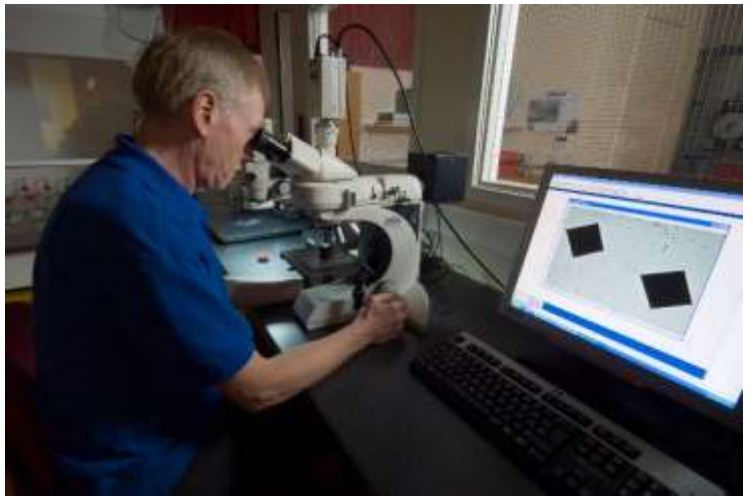


CORDEL Value Proposition

- Increased **safety** – operational feedback
- Increased **licensing** predictability - consistent approach
- Reduced construction **schedule** – nth of a kind
- Reduced **costs** – greater predictability
- Facilitate a consistent **supply chain** – codes and standards
- Reduced **risks** and increased investor confidence – consistent deployment

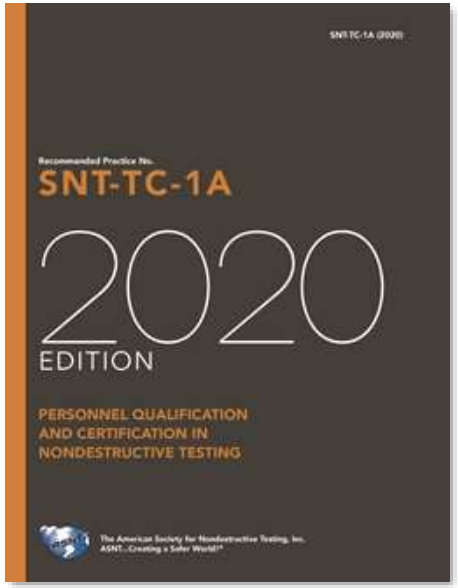
Introduction

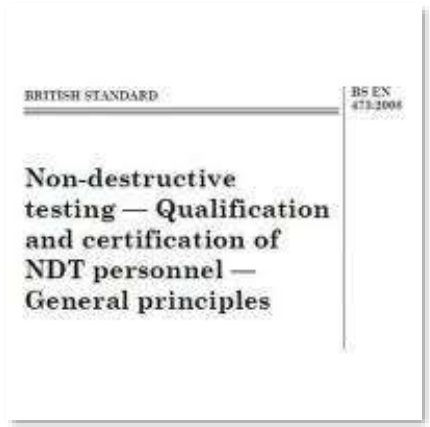
- Non-destructive examination (NDE) provides assurance of the absence of unacceptable defects.
- National regulations & their codes and standards differ from country to country → **creates barriers** to the transferability of NDE certification.
- Increased nuclear new build → **increased need for NDE** → insufficient capacity
- Harmonization of certification of NDE personnel would mitigate this risk.



Background on NDE personnel certification

- Currently….
  - Individuals required to be certified by law to perform the tests.
  - Certification, Training and Qualification are required in most countries
  - Certification is required by C&S and regulators.
  - Training is not imposed by C&S but required by employers based upon their needs.
  - Qualification requires validation by an independent 3rd party
- Differing requirements **limit the transfer of certification** between countries.





Transferable certification scheme barriers

- Mechanical nuclear code requirements and regulatory requirements
- The requirement for a 3rd party vs in-house certification responsibility of companies employing the NDE personnel
- Requirement to develop and submit a Written Practice
- Industrial application of the requirements
- Responsibilities of inspection qualification body

Codes and standards comparison findings

- ASME BPVC, AFCEN RCC-M, JSME, NIKIET PNAE, KEPIC MNB 5521, CSA
- C&S requirements divide into two main categories; those that require a 3rd party certification and those that require a 2nd party certification in combination with a Written Practice.
- International options have been added to code requirements on an ad-hoc basis.
- C&S requirements for NDE personnel certification have already converged to a certain extent.

2nd party and 3rd party certification

	2nd Party	3rd Party
Issued by	• Employer (company based)	• Independent, nationally recognized organization
Advantages	• Supported by detailed Written Practice that is tailored to a specific task	• Certification based on international standards • Nationally transferable between companies
Drawbacks	• Potential conflicts of interest • Company specific Written Practices • Experience needed to draft suitable Written Practices	• Can lack detail for specific tasks • Additional in-house training can be required • No automatically internationally transferable

Written Practice and Quality Procedure

	Written Practice	Quality Procedure
Definition	• ASME BVPC Section XI	• ICNDT
Required content	• Level of qualification • Vision test requirements • Training course content • Required training time • Experience time • Administration and grading of certification • Requirements for initial certification • Requirements for recertification • Revocation and suspension of certification • Reinstatement of certification • Limited certification	• Applicable codes and standards • General responsibilities of level 1, 2 and 3 • Certification requirements (sector, method, level) • Persons designated by the employed to be responsible for issuing the authorization to operate • Control of in-house training and examination • Responsibility for maintenance of records (training, education, experience, vision and certification)

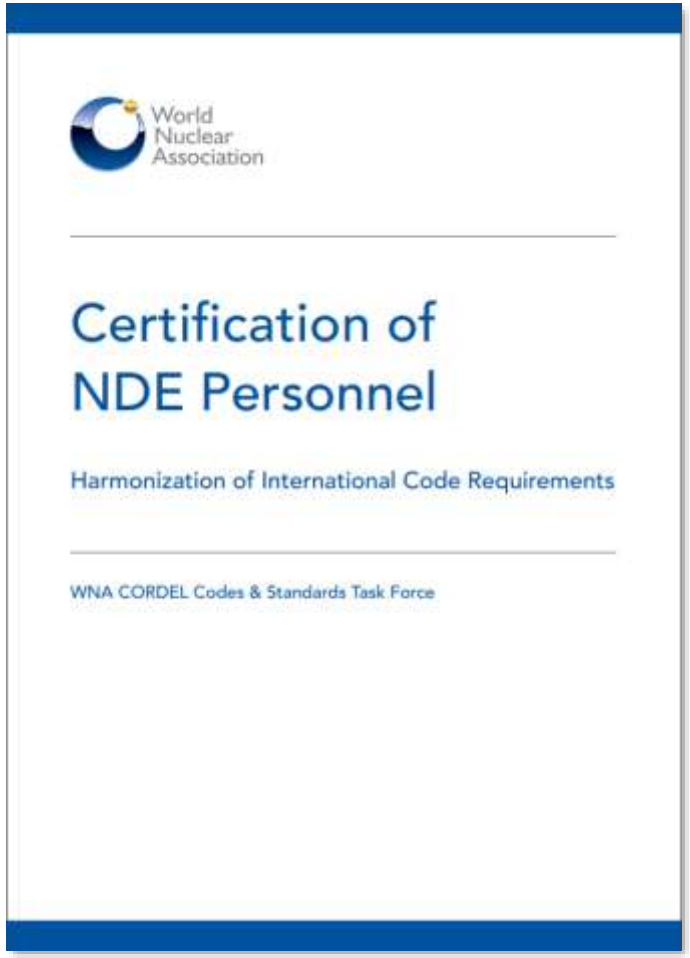
Industrial Application

- Throughout the world, industry is forced to hold multiple certifications to work internationally.
- This is a particular burden for vendors, component manufacturers and engineering service companies

Region	Certification
Asia	China: HAF602 Japan: JIS Z 2305, ASNT-TC-1A, EN 473 South Korea: MEN A1, SNT-TC-1A, EN 473
Europe	Within EU: COFREND, BINDT Outside EU: PCN/SNT-TC-1A
North America	Canada: CAN/CGSB-48.97 USA: SNT-TC 1A
Russia	PNAEG-7-010-89

Conclusions

- Harmonization of certification → greater confidence in certified personnel → better international transferability of personnel.
- 2 categories of C&S requirements:
  - 3rd party certification
  - 2nd party certification plus a Written Practice.
- 3rd party certification is the preferred method
- CORDEL recommends that SDOs provide a harmonized alternative to the certification of NDE personnel to be added to their C&S.
- This would be a 3rd party certification according to ISO 9712:2012 supported by a Written Practice.
- Full report available on WNA website



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Panel discussion and Q/A from the audience

Ms Carol NOVE

Senior Materials Engineering

US Nuclear Regulatory Commission

NEA
NUCLEAR ENERGY AGENCY

OECD
BETTER POLICIES FOR BETTER LIVES

Working Group on Codes and Standards

Workshop on In-Service Inspection

Break

Workshop will resume in 10 minutes

Day 3 – Wednesday 13 April 2022

NEA
NUCLEAR ENERGY AGENCY

OECD
BETTER POLICIES FOR BETTER LIVES

Session 3
Extending ISI intervals - approvals and basis

Mr David DIJAMCO

US Nuclear Regulatory Commission

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

Use of Probabilistic Fracture Mechanics for ISI Extension

A Regulatory Perspective

David Dijamco

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Vessels and Internals Branch

2022 International Inservice Inspection Workshop · April 13, 2022

The views in the presentation herein reflect those of the presenter and not of the U.S. NRC.

What is the risk of extending ISI for a component?

Probabilistic fracture mechanics (PFM) has been used to answer this, at least partially. (Part 1)

To what extent can PFM address the question of risk? (Part 2)

PFM in a nutshell

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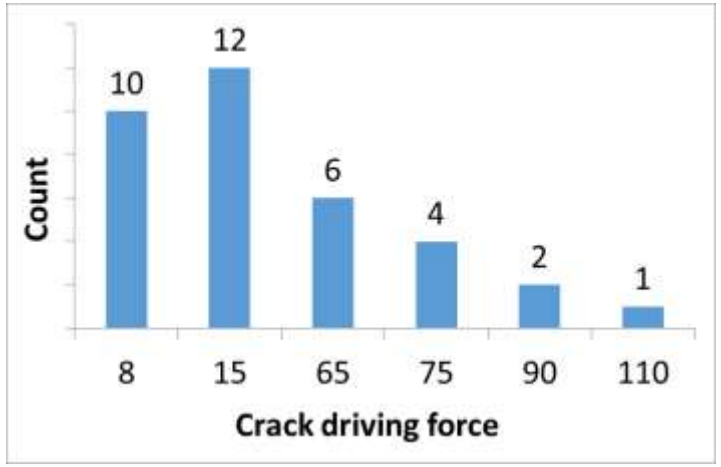
Fracture mechanics, crack driving force < material resistance

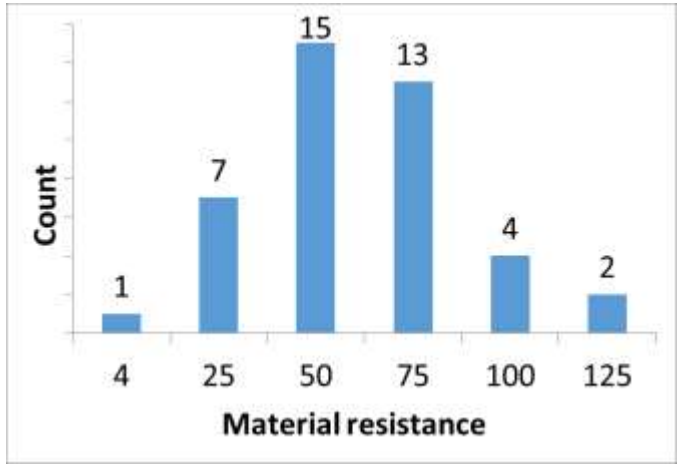
In deterministic fracture mechanics, you apply safety factors on either side (or both) and perform one calculation.

PFM is performing many fracture mechanics calculations in which the crack driving force is compared to the material resistance, then counting how many failures you get.

Many calculations are performed because the input parameters are not a single value, but a **distribution of values** that are sampled.

PFM illustration





Sample (trial or realization)	Crack driving force	Material resistance	Fail? (crack driving force > material resistance?)
1	8	75	No
2	15	50	No
3	75	25	Yes
. . . 10 million	. . . 65	. . . 100	. . . No

Probability of failure = number of failures ÷ number of samples

PFM is complex

Performed with computer codes such as FAVOR and xLPR

PFM input parameters, each with distributed values:

- Flaw size/density
- Fracture toughness (or associated parameters such as RT_{NDT})
- Probability of detection
- Weld residual stress
- Fluence
- Crack growth rate

PFM guidance: Regulatory Guide 1.245 and NUREG/CR-7278

Two points in using PFM for ISI extension

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Effect of ISI and examination coverage need to be included

If generic PFM results are used, the ISI and examination coverage in the generic analysis would need to closely align with the ISI and examination coverage that have been performed for the specific component.

Sensitivity analyses (SA) and sensitivity studies (SS) performed to address and understand uncertainties

SA helps flush out the inputs that are driving the desired analysis output (e.g., probability of failure); therefore, this is a **global** look at the analysis.

SS helps in determining the effect of a particular input in the analysis (e.g., stress); this is a **local** look at the analysis.

In a PFM analysis, probability of failure (PoF) is often queried as the output.

PoF is counting the number of failures out of all possible scenarios that arise from the uncertainties in the input parameters.

Provided that the resulting PoF is less than the specified criteria, is going by PoF alone sufficient for extending ISI? (moving into Part 2)

Underlying concept behind ISI

Monitor the condition of a component throughout its operating life. But why?

Components are designed with safety margins, and so at the start of operation (baseline condition) the component is deemed acceptable.

Monitor condition of the component to compare with the baseline condition to determine if there has been a **change** (which could mean presence of service-induced **degradation** that must be addressed). This ensures the **safety margins** in the component are maintained, so that it will continue to perform its intended function, and thereby provide reasonable assurance of overall plant safety.

Extending ISI period reduces ISI examinations, and this impacts the intent of ISI

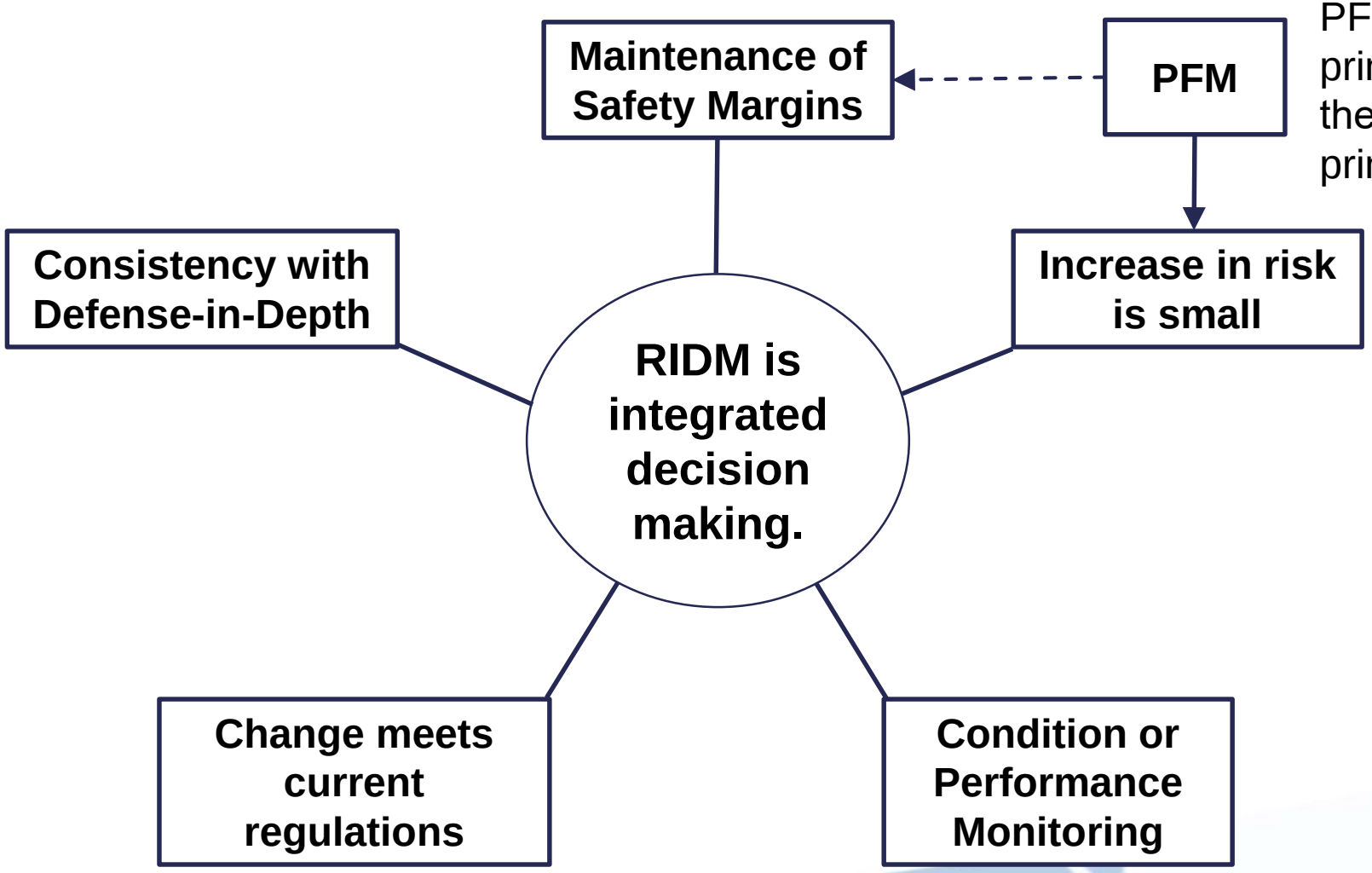
Risk of this reduction must be addressed.

Other aspects that must be addressed based on ISI concept:

- Monitoring condition
- Degradation
- Safety margins

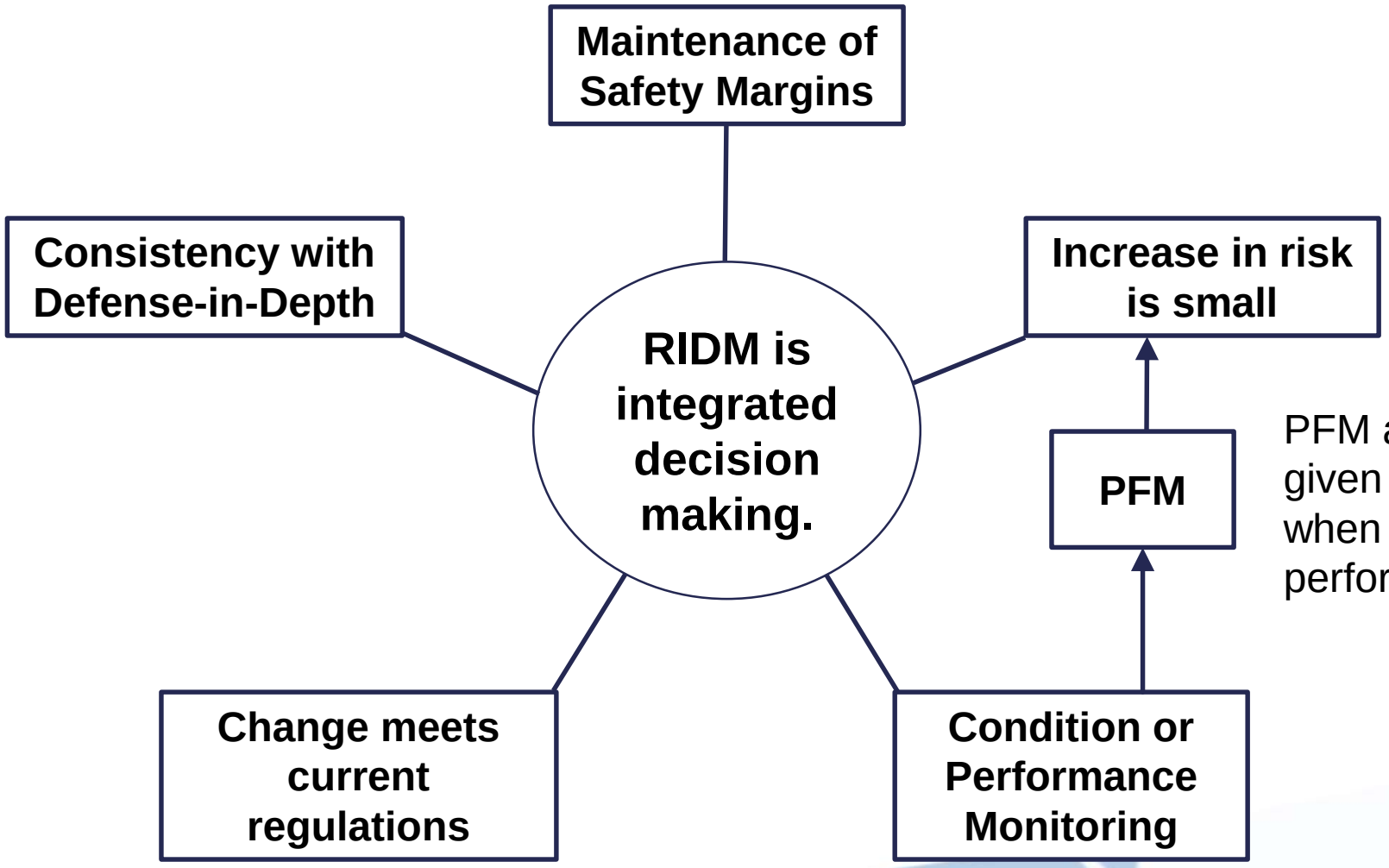
This makes extending ISI a risk-informed decision.

Risk-Informed Decision Making (RIDM)



PFM addresses the risk principle and can inform the “safety margin” principle.

Risk-Informed Decision Making (RIDM)



PFM allows weight to be given to the risk principle when combined with performance monitoring.

Aspects of performance monitoring

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Starting with keywords from ISI concept…

- Monitor condition for change in degradation
- Maintenance of safety margin

Three key aspects of performance monitoring can be derived.

- Direct evidence of presence and/or extent of degradation
- Timely method to detect novel and/or unexpected degradation
- Validation/confirmation of continued adequacy of analyses

Performance monitoring is important for ISI extension, especially extensions during long-term operation because of increased uncertainty.

Examples of PFM + performance monitoring as basis for ISI extension

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Elimination ISI of BWR vessel circumferential welds (covered in presentation tomorrow)

20-year ISI extension of PWR vessel welds

- PFM: FAVOR analyses
- Performance monitoring: coordinated ISI of the same-fleet units to ensure regular fleet ISI data; one-time ISI for subsequent extensions to validate that generic flaw-distribution used in the PFM analysis is bounding for the unit.

Note: both are implemented through the alternative request process in NRC regulations.

Summary

PFM alone is not an adequate basis for ISI extension.

For ISI extension, PFM allows weight to be given to the risk principle of RIDM when combined with performance monitoring.

Mr Nathan PALM

EPRI, United States

BWR RPV Circumferential Weld Inspection Elimination

Nathan A. Palm, PE
BWRVIP Program Manager

OECD/NEA/US NRC WGCS International Workshop on Mechanical Codes and Standards: In-Service Inspection
April 13, 2022

Presentation Outline

- Background on ASME and NRC Requirements
- Development and Implementation of BWRVIP-05
- Extension of BWRVIP-05 to address Long-Term Operation
- Summary and Conclusions

History of Section XI RV Examination Requirements and Changes

Applicable Code Edition or Addenda	Extent of Examination for First Inspection Interval	Extent of Examination for Successive Inspection Intervals
1970 Edition to 1975 Winter Addenda	• 10% of the length of longitudinal (axial) welds • 5% of the length of circumferential welds • 50% of the length of welds in the beltline region that receive a neutron fluence exceeding $1x10^{19}$ n/cm^2 (E>1MeV)	"A similar percentage of components not previously inspected shall be required in each successive inspection interval"
Winter 1975 Addenda to Winter 1982 Addenda	100% of the length of all welds • BWRs were designed prior to the requirement being in place and did not have access to perform examinations • BWRs had to submit "relief requests" to NRC asking to deviate from fully complying with Section XI requirements	• 100% of one circumferential weld in the beltline region • 100% of one longitudinal weld in the beltline region
Winter 1982 Addenda to Winter 1988 Addenda	100% of the length of all welds	100% of one weld (circumferential or longitudinal) in the beltline region
Winter 1988 Addenda to 2021 Edition	100% of the length of all welds	100% of the length of all welds

U.S. NRC Endorsement of the 1988 Addenda of Section XI

- In August 1992, a revision to 10CFR50.55a endorsed the 1988 Addenda of Section XI with the requirement to inspect 100% of RV shell welds in each interval
- This new rule revoked previously granted licensee relief requests regarding the extent of volumetric examination on reactor vessel shell welds
- All licensees were required to perform once, as part of the current inspection interval, an augmented examination of "essentially 100%" of the circumferential and longitudinal reactor vessel shell welds
  - "Essentially 100%" was defined as greater than 90% of the examination volume of each weld

Basis for NRC Augmented Examination Requirements

- The NRC position with regard to the augmented examination of reactor vessel shell welds, as described in their August 1992 rulemaking, was based on the following three major concerns
  - An embrittlement concern stemming from irradiation surveillance material test results which show that certain reactor vessel materials undergo greater radiation damage than previously expected
  - A stress corrosion cracking (SCC) concern resulting from operational data which indicates that SCC of BWR reactor vessels is more probable than previously thought
  - A concern regarding significant service induced cracking that has occurred in large vessels (e.g., pressurizer, steam generators) designed and fabricated to the ASME Code

BWRVIP Response to Augmented Examination Requirements

- Examination of 100% of all BWR RPV shell welds was not possible in U.S. BWRs due to access limitations and component interferences
  - Examination of circumferential welds is most challenging
- Appeal made to NRC management that the requirements could not be met with current NDE technologies
- Effort initiated to develop a technical basis to define a more appropriate goal for examination volume that would still achieve safety goals and satisfy NRC ➔ BWRVIP-05

BWRVIP-05

- In September 1995, the BWRVIP published report BWRVIP-05, “BWR Reactor Pressure Vessel Shell Weld Inspection Recommendation”
- The purpose of this report was to provide a technical basis to justify a reduction in the required number of ASME Section XI examinations of longitudinal and circumferential welds in the reactor pressure vessel (RPV).
- The results from probabilistic fracture mechanics (PFM) analyses were used to provide justification for a proposed reduced examination scope
  - Inspect 50% of axial welds
  - Inspect 0% of circumferential welds
- The PFM evaluation was performed for a postulated, low temperature isothermal over-pressure transient event, which had been determined to dominate the estimated failure frequency for normal operating and postulated transient events in RPVs in the U.S. BWR fleet.

BWRVIP-05 PFM Analyses – Key Features

- Probabilistic treatment of vessel fracture toughness and radiation embrittlement
- Assumed fabrication defects – Marshall distribution with all defects moved to the vessel ID surface
- Multiple random variable, Monte Carlo simulations used to compute vessel failure probabilities
- SCC initiation in cladding
- SCC growth in low alloy steel
- Effects of periodic inservice inspection
- "Importance sampling" technique to improve the efficiency of Monte Carlo calculations
- Two types of "failures" addressed in analysis
  - Crack growth to > 80% through-wall crack (leak)
  - K_I exceeding K_{IC} under worst case low temperature stress condition, with irradiation embrittlement effects (break)

BWRVIP-05 PFM Results, Conclusions, and NRC Review

- Results and Conclusions
  - Calculated vessel failure probability is two orders of magnitude lower than the NRC safety goal
  - Realistic failure probabilities would be even smaller if conservatisms are removed from the analysis
  - Proposed reduction in the inspection scope has insignificant change on failure probabilities that are already small
- NRC Review
  - BWRVIP-05 was submitted for NRC review and approval on September 28, 1995
  - During the course of NRC review, the BWRVIP changed the proposed inspection scope for axial welds from 50% to essentially 100% to satisfy NRC
  - The NRC's safety evaluation (SE) report was issued on July 28, 1998

Implementation of BWRVIP-05

- The SE report identified that BWR licensees may request relief from the ASME Section XI inservice inspection requirements by demonstrating that:
  - At the expiration of their license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the SE, and
  - They have implemented operator training and established procedures that limit the frequency of cold over pressure events to the amount specified in the SE (1×10^{-3}/yr)
- The NRC also issued a Generic Letter (GL 98-05) that allowed plants to request permanent (i.e., for the remaining term of operation under the existing, initial license) relief from the circumferential weld inservice inspection requirements
  - Licensees were required to still perform inspections of "essentially 100 percent" of all axial welds

All U.S. BWRs have been approved to implement BWRVIP-05 and have not performed the 100% examination of RV circumferential shell welds required by ASME Section XI

Considerations for Axial Welds for Operation Beyond 40 Years

- The SE for BWRVIP-05 stated that the NRC had identified a need to further evaluate the high conditional failure probability levels for axial welds in BWR RPVs
- On March 7, 2000, the NRC issued a supplemental SE for BWRVIP-05
  - The failure frequency of the limiting axial welds in the BWR fleet satisfies the NRC safety goals for the original 40-year license term
  - Plant specific treatment will be required to address axial welds for license renewal
- On October 18, 2001, the NRC issued a SE for BWRVIP-74, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines for License Renewal"
  - Includes an extension of applicability for BWRVIP-05 for 60 years
  - Identifies a limit for axial weld mean RT_{NDT} of 114°F
  - Axial weld mean RT_{NDT} limit was incorporated into the NRC's Standard Review Plan for Review of License Renewal Applications (SRP-LR), NUREG-1800, Revision 2

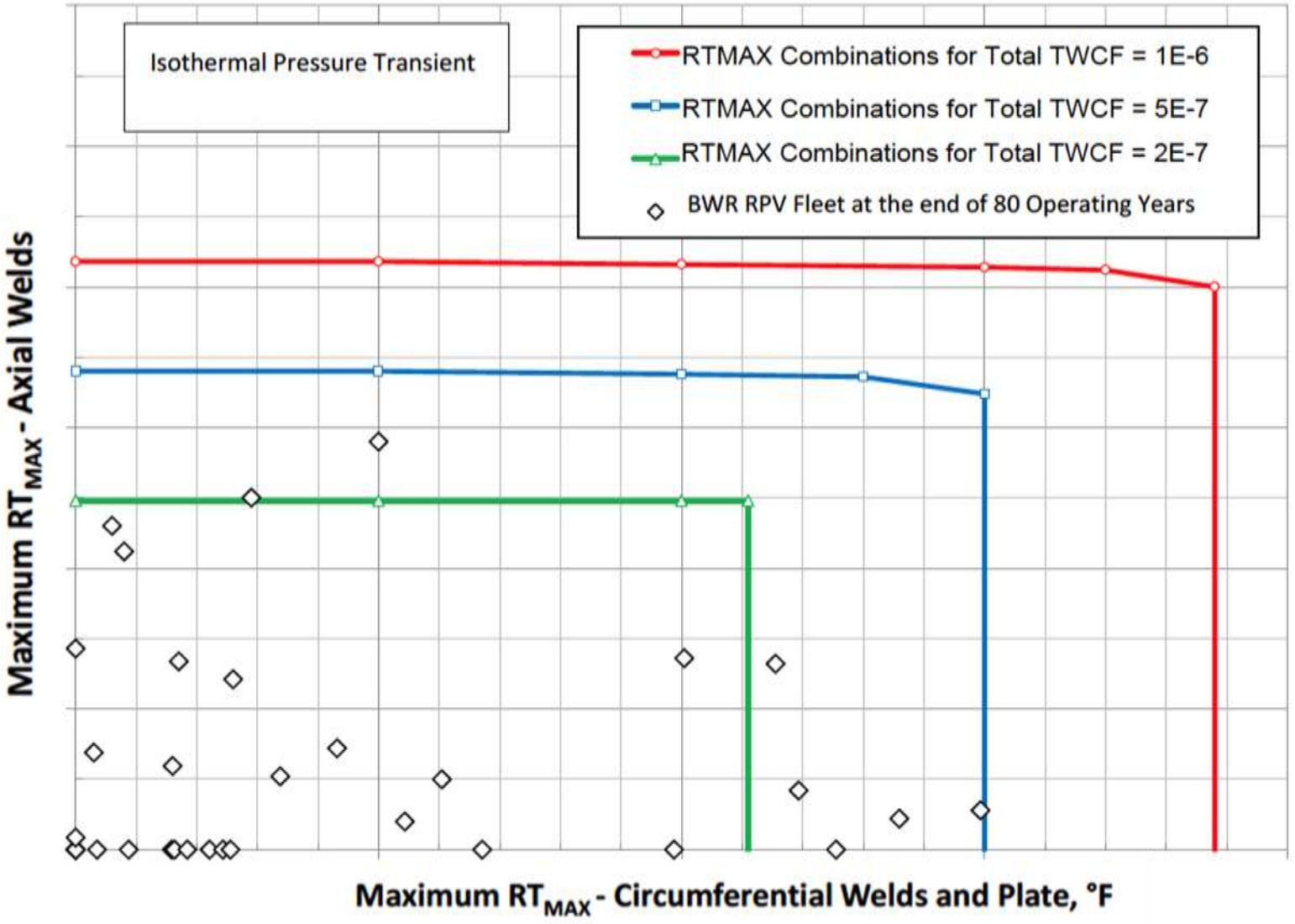
Extending BWRVIP-05 to Address Long-Term Operation

- In 2016, a project was initiated by BWRVIP to develop a new PFM analysis that provides additional margin and can be used to replace BWRVIP-05:
  - Use NRC safety goals and PFM analysis procedures that have been developed since the publication of BWRVIP-05 to update the evaluation procedure and acceptance criteria specified in BWRVIP-74
  - Provide an evaluation procedure that can be used to remove the SRP-LR limitation of mean RT_{NDT} < 114°F for axial beltline welds
  - Provide a technical basis for continuing relief from the examination of circumferential welds in RPVs for operation beyond 60 years
- Results published in new report, BWRVIP-329

BWRVIP-329 – Approach and Tasks

- PFM evaluation procedures and acceptance criteria applied consistent with those currently deemed acceptable by NRC
  - Use of FAVOR Code for PFM analyses
  - Safety goal of Through-Wall Cracking Frequency (TWCF) < 1E-6 / yr.
- Sensitivity studies performed to address uncertainty in cladding thickness, flaw distributions and vessel geometry.
- Set of limiting vessel geometry, cladding, and flaw conditions defined
- Where applicable, surveillance data evaluation results were used to assess the change in material toughness due to irradiation (ΔT_{30}).
- PFM analyses performed to determine the combinations of mean RT_{NDT} (referred to as RT_{MAX}) for plate and axial and circumferential welds that will ensure the regulatory risk goals are maintained at the end of any specified operating interval.

BWRVIP-329 – Results



- The results from the PFM analyses were used to define the combinations of RT_{MAX} for plate and axial and circumferential welds that ensure total TWCF ≤ 1E-6 yr^{-1} for the RPV beltline and $TWCF_{CW}$ ≤ 1E-7 yr^{-1} for circumferential beltline welds in BWR RPVs during a postulated low temperature isothermal pressure transient.
- Data points represent estimated end-of-life RT_{MAX} values for U.S. BWRs where data were available. Results indicate that large margins exist.

Note that X and Y axis values are EPRI Proprietary Information and have been removed from this figure

BWRVIP-329 – Summary and Conclusions

- BWRVIP-329 provides BWR owners with a continuing basis to justify relief from examination of RPV circumferential welds, and the capability to demonstrate acceptable integrity for axial welds
- U.S. BWR RPVs were evaluated to assess the suitability of the results to support operation through 80 years. All RPVs have predicted TWCF ≤ 1E-6 / yr. with substantial margin at the end of an 80-year operating interval.
- NRC Safety evaluation report received April 12, 2021, and incorporated into final "-A" report
  - *BWRVIP-329-A: BWR Vessel and Internals Program: Updated Probabilistic Fracture Mechanics Analyses for BWR RPV Welds to Address Extended Operations*. EPRI, Palo Alto, CA: 2021. 3002022650.

Summary and Conclusions

- Although ASME Section XI requires 100% inspection of RPV shell welds, no U.S. BWRs have inspected 100% of their circumferential welds
- BWRVIP-05 provides a technical basis for BWR circumferential weld inspection elimination that has been utilized in relief requests implemented by the entire U.S. BWR fleet
- BWRVIP-329 provides an updated technical basis for BWR circumferential weld inspection elimination that addresses long-term operation and has been recently approved by the U.S. NRC

Mr Anees UDAYAWAR

Westinghouse, United States

PWROG

PWR Owners Group

Global Expertise • One Voice

Pressurized Water Reactor (PWR) Vessel In-Service Examination Extension

Anees Udyawar (Westinghouse) – April 13, 2022

OECD/NEA/US NRC WGCS International Workshop on Mechanical Codes and Standards: In-service Inspection

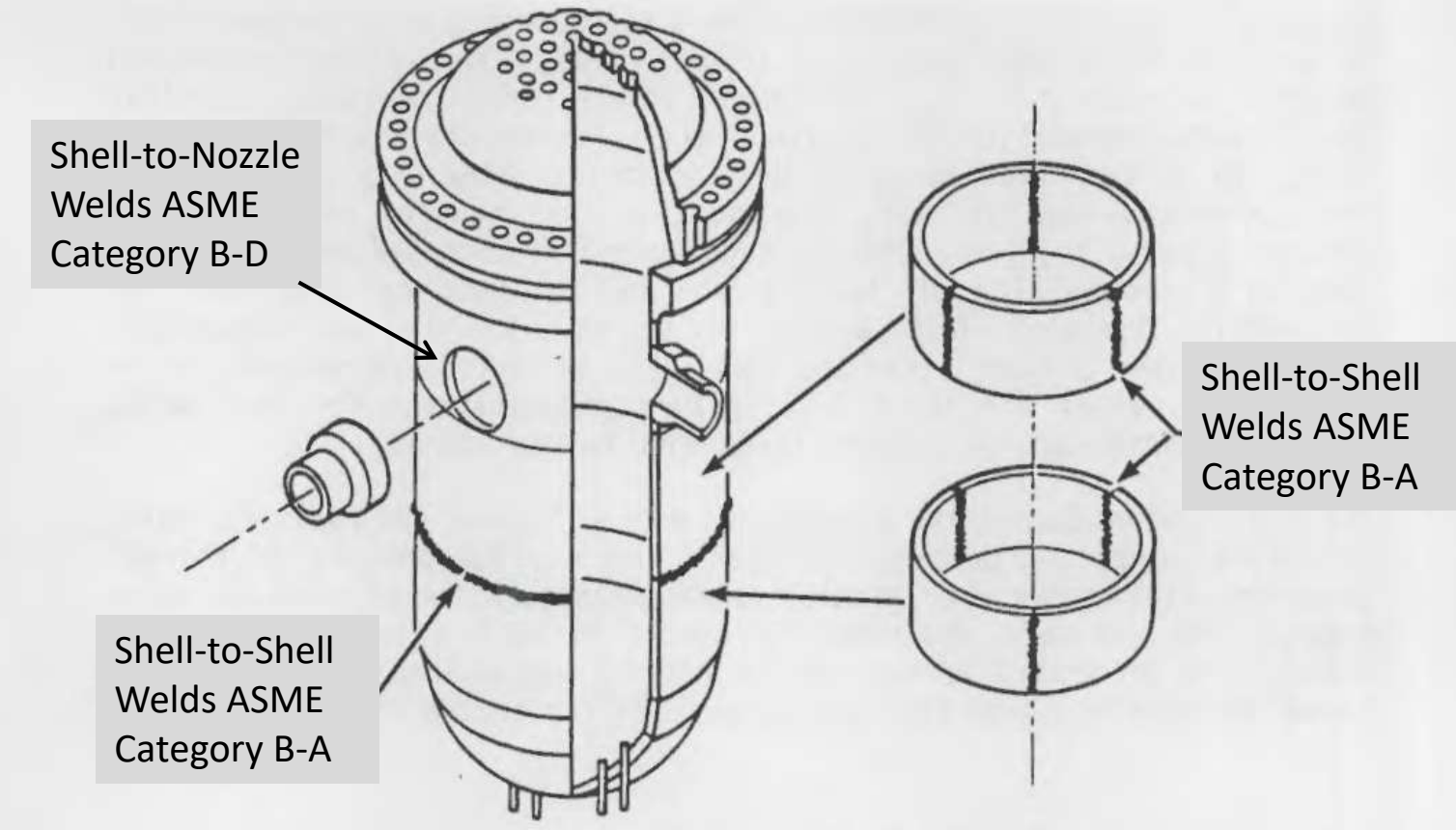
Outline

- Introduction
- Background
- Methodology
- Pilot Plants
- Bounding Flaw Location for RV Failure
- Basis for Risk Determination (i.e. Accident and operational transients, flaw size, fluence, material toughness, crack growth, stresses, inspection)
- Pilot Plant Results
- Plant Specific Application
- Summary

Background

- ASME Boiler and Pressure Vessel Code, Section XI requires inspections on a 10-year interval
- NRC grants exemptions or relief based on experience, evaluations, man-rem reductions, and hardships
- PWROG methodology developed for extending reactor vessel (RV) in-service inspection interval from 10 to 20 years
  - Interval extension applicable to all RV shell-to-shell and shell-to-nozzle (ASME Section XI Inspection Item B-A and B-D welds)
  - Significant reduction in outage costs (~$7 million per plant)
  - Reduction in man-rem exposure costs (~$37,000 per plant)

Background


Shell-to-Nozzle Welds ASME Category B-D
Shell-to-Shell Welds ASME Category B-A
Shell-to-Shell Welds ASME Category B-A

Methodology

- PFM analysis to evaluate increased time between interval inspection on the frequency of RV failure due to postulated PTS transient
- RV failure is based on through-wall cracking of the RV wall
  - Likelihood of failure increase with increasing time of operation
  - Due to growth of pre-existing flaws by fatigue
  - Decrease in RV fracture resistance due to irradiation
- PFM analysis considered distribution and uncertainty
  - Flaw size and number
  - Fluence and Material Fracture Toughness
  - Crack Growth Correlations
  - Transients and Stresses
  - Effectiveness of Inspections

Methodology

- PWROG Methodology utilizes work performed for NRC PTS Risk Study
  - NUREG-1806, *Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)*
  - NUREG-1874, *Recommended Screening Limits for Pressurized Thermal Shock (PTS)*
  - Change in risk evaluated against Reg. Guide 1.174, *Approach Using Probabilistic Risk Assessment*
    - $\Delta CDF < 1E\text{-}6$
    - $\Delta LERF < 1E\text{-}7$
  - Performed PFM analysis to evaluate effects of different inspection intervals on risk
  - Led to conditional RV failure frequency due to given loading condition and inspection interval.
  - Considered different pilot plants to represent PWR designs

Pilot Plant Summary

- Risk evaluations conducted based on geometry and embrittlement for Westinghouse, Combustion Engineering (CE), and Babcock and Wilcox (B&W) PWR Designs
  - Represents the three primary manufacturers in the U.S.
    - Westinghouse: Beaver Valley Unit 1 (BV1)
    - CE: Palisades
    - B&W: Oconee Unit 1 (OC1)
- Same plants were used in NRC PTS Risk Study
- Used fleet-specific design transient data for the Westinghouse and CE designs
- Generic Heatup/Cooldown transients used for the B&W Study
- Bounding location among applicable welds locations determined

Bounding Location for RV Failure

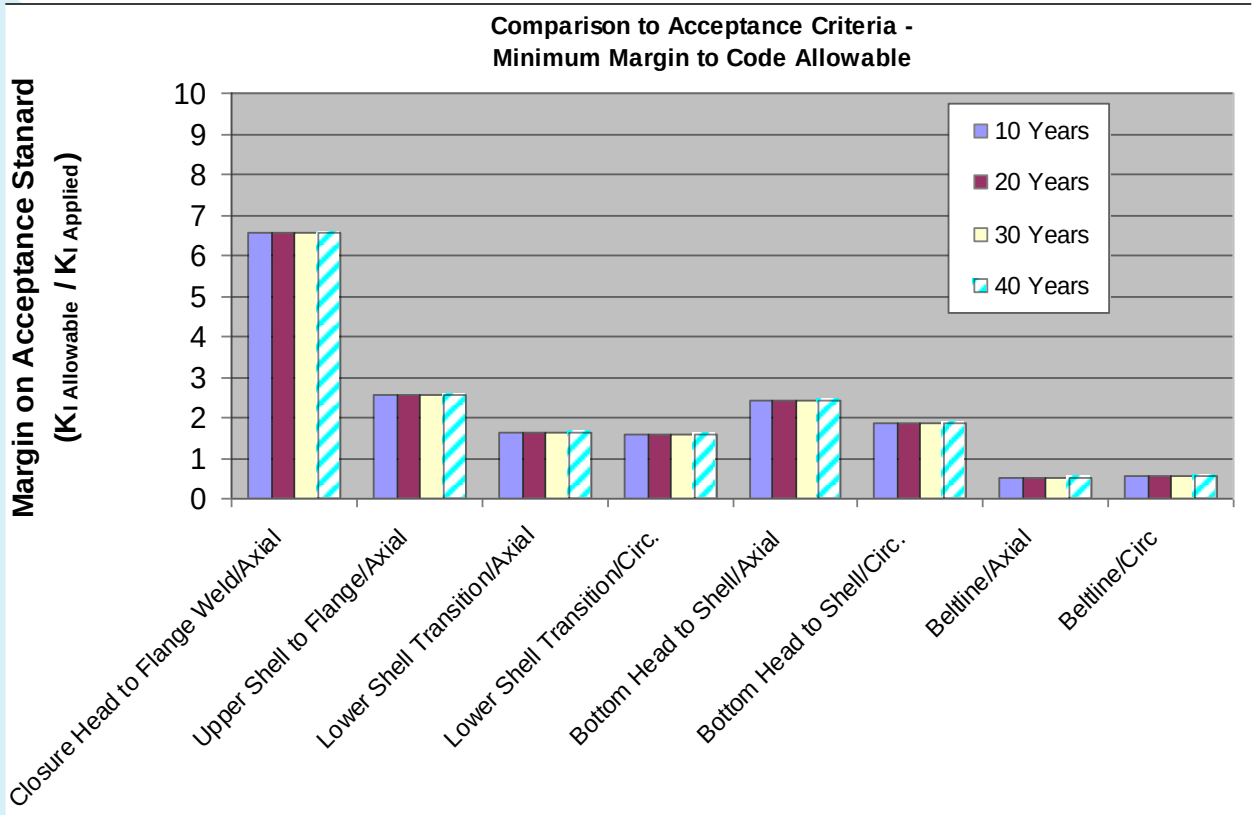
- Evaluation focused on beltline of the RV
- Needed to confirm that the beltline is the bounding location of the RV
  - All locations required for examination in the RPV were identified and considered
  - Evaluated impact of flaws in each of the RV regions
  - Beltline locations were found to be bounding due to irradiation induced change in the fracture toughness
- Consistent with the NRC PTS Risk Study location assumptions
- Assessment performed using deterministic fracture mechanics considering the following
  - Postulated 10-percent through-wall initial flaw
  - End-of-life 40 EFPY embrittlement
  - Peak reactor vessel ID fluence

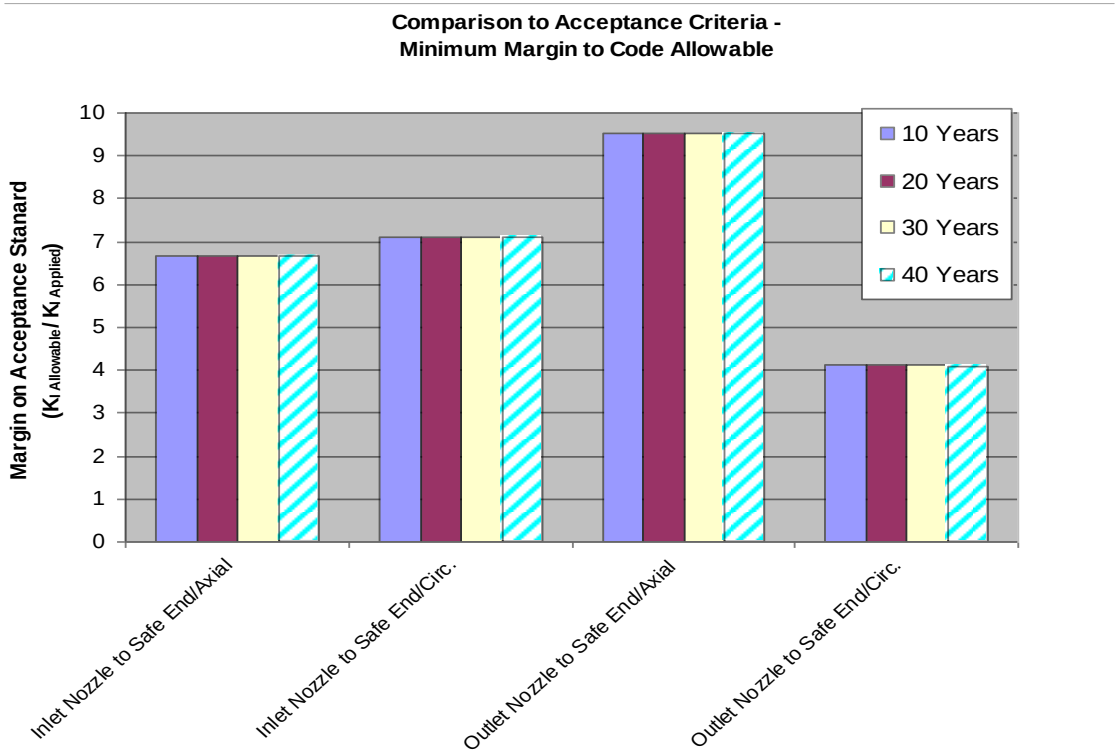
Bounding Location for RV Failure

- Inservice fatigue crack growth of flaw due to normal transients
- Design basis heatup and cooldown transients
- 8 weld locations considered
  - Closure head to flange, upper shell to flange, lower shell transition, bottom head to shell, beltline, nozzle to shell, inlet nozzle to safe end, and outlet nozzle to safe end

- Analysis evaluated effect of various ISI intervals (i.e. 10, 20, 30, 40 years)
  - Compared the change in margins on ASME Code Allowable Flaw sizes
- Each location evaluated
  - Calculated amount of crack growth over 10 year period
  - Crack length evaluated for maximum Stress Intensity Factor (K_I)
  - Used ratio of Max allowable K_I (per ASME Section XI App A) to applied K_I as a measure of margins of a flaw has to the acceptance criteria
- Based on the results beltline is the limiting region, same conclusion as the NRC Risk Study

Bounding Flaw Location for RV Failure





Basis for Risk Determination

- Calculate change in RV failure frequency due to a change in inspection interval (RG 1.174 guidance used)
- Probabilistic Fracture Mechanics Computer Tool and Methodology
- Following inputs included in the RV PFM and risk analysis
  - Accident Transients and Frequency
  - Operational Transients and Cycles
  - Initial Flaw Distribution
  - Fluence Distribution
  - Material Fracture Toughness
  - Crack Growth Rate Correlations
  - Cladding and Residual Stresses
  - Effectiveness of ISI

Pilot Plant Results

- Insignificant difference between “ISI Every 10 Years” and “10-year ISI Only cases”
  - Difference between the mean values is less than the standard error for each case
- Change in failure frequency was acceptable per RG 1.174 for each pilot plant (i.e. below LERF of 1E-7/year in RG 1.174)

Large Early Release Frequencies			
	BV1 (per year)	Palisades (per year)	OC1 (per year)
10-Year ISI Only	5.04E-09	7.62E-08	3.11E-08
Upper Bound	5.55E-09	8.44E-08	3.62E-08
ISI Every 10 Years	5.23E-09	7.39E-08	2.62E-08
Lower Bound	4.61E-09	6.63E-08	2.36E-08
Bounding Change in Large Early Release Frequency	9.37E-10	1.81E-08	1.26E-08

Plant Specific Application

- Demonstrate following parameters are bounded by acceptance criteria from pilot plant
  - Dominant PTS Transients in the NRC PTS Risk Study are applicable
    - Per “PTS Generalization Study”, all PWRs currently operating in US inherently meet this requirement
  - Plant Specific 95th percentile TWCF (using NUREG-1874) must be less than the pilot plant 95th percentile TWCF. TWCF is a measure of embrittlement of the RV components weighted by their contribution to PTS failure
    - Westinghouse: 1.76 E-08 Events per year
    - CE: 3.16 E-07 Events per year
    - B&W: 4.42E-07 Events per year
  - Frequency of design basis transients (heatup and cooldown) must be bounded by pilot plants:
    - Westinghouse: 7 cooldowns per year
    - CE: 13 cooldowns per year
    - B&W: 12 cooldowns per year
  - Latest RV ISI report needs to be reviewed for recordable indications beltline materials
    - Acceptable limits defined in 10CFR50.61a “Alternate PTS Rule”

Summary

- RPV Weld ISI interval extension PFM methodology developed to reduce inspection costs and man-rem exposure
- Beltline is the most limiting region for evaluation of risk
- RV inspections performed to date have not detected any significant flaws
- Crack extension due to FCG during service is small
- Performed PFM analysis for pilot-plant application using risk evaluation guidance from NRC PTS Risk study and RG 1.174 to determine change in failure frequency and it is impact on risk
  - Failure frequencies for PWR RV due to dominant PTS transients are below 10^{-7} per year
  - The increase in RV ISI interval from 10 to 20 year satisfies all the RG 1.174 criteria
- RPV Weld ISI interval extension methodology has been approved by NRC in WCAP-16168-NP-A, Revision 3 (NRC Accession No. ML11306A084)
- Approximately 70% of the USA PWR fleet have been granted approval for ISI extension, and these plants have used the extension for their RV weld inspections

Questions

Mr Gary LOFTHUS

Southern Nuclear, United States

Southern Company

ISI Programs updates from a Utility Point of View

Gary Lofthus
Southern Nuclear
NDE Manager
April 2022

ISI Program Update Considerations

Southern Nuclear has 6 operating units and 2 units under construction

1. Plant Hatch Units 1 and 2 (BWR)
2. Plant Farley Units 1 and 2 (PWR)
3. Plant Vogtle Units 1 and 2 (PWR)
4. Plant Vogtle Units 3 and 4 (PWR – under construction)

We currently have a corporate office that supports all sites that includes Engineering, Licensing and NDE
We also have staff at each site for Engineering, Licensing and NDE.

What are the current issues related to the ISI program updates for US utilities.

1. We are currently meeting ASME Section XI and 10 CFR50.55a which has the requirements for an Inservice Inspection program.
2. Current intervals are 10 years , with 3 periods to schedule Inservice Inspection to exam requirements for our Nuclear Plants.
3. We commit to a code year and addenda, ensure all requirements from ASME code and 10CFR50.55a are meet.

ISI Program Update Considerations

- We are currently meeting ASME Section XI and 10 CFR50.55a which has the requirements for an Inservice Inspection program.
- Current intervals are 10 years , with 3 periods to schedule Inservice Inspection to exam requirements for our Nuclear Plants.
- We committed to a code year and addenda, to ensure all requirements from ASME code and 10CFR50.55a are meet.

ISI Program Update Considerations

- SNC is currently using 2007 edition with 2008 addenda.
- We have 6 operating units and 2 units that are under construction.
- From a utility standpoint we want to use the same code of record so that our NDE procedures and implementation processes (Repair / replacement , etc.) are the same.

ISI Program Update Considerations

- When changes are made to ASME XI, they are in the code case and later editions of the approved ASME code.
- Most new code cases and changes are good and include optimization, alternative rules, alternative methods, or provide a reason such that we want to use those changes.
- We can use those changes to the code such as a code case, if it approved for use by the NRC

ISI Program Update Considerations

- We can also ask for a relief request, provided we submit the code case and the technical justification. This process involves our licensee group and typically takes a year for approval.

- We also have to track 10CFR50.55a rulemaking. AS it is frequently changed and we have to ensure that all exceptions, requirements and changes are implement into our ISI program.

- Another Issue is the changing / retiring personnel in our company. SNC went a number of years in the 1980's and 1990's with the same personnel in ISI positions, NDE position and Licenses positions. This kept the knowledge base up to date.

ISI Program Update Considerations

- Many of these personnel were part of the code updates for the ISI programs.

- A code update requires reviewing and reconciling all of the changes from one year and edition, to the last required code year and edition. We also have to review relief request to see if they need to be re-submitted or changed.

- Limited exams need to be submitted with a coordinated effort of NDE, Engineering and Licensing personnel.

ISI Program Update Considerations

- When updating the ISI program, an entire review of each code item and category is required.

- Frequency of exams, how the exam is to be performed, are we using Risk Informed, code cases, and what changes are needed to our plant procedures.

- Currently there are actions in place to look at extending the ISI interval from 10 to 12 years.

ISI Program Update Considerations

- When updating the ISI program, an entire review of each code item and category is required.

- Frequency of exams, how the exam is to be performed, are we using Risk Informed, code cases, and what changes are needed to our plant procedures.

- Currently there are actions in place to look at extending the ISI interval from 10 to 12 years.

ISI Program Update Considerations

- There are also actions in place where they are talking about not having to change the code of record for 20 years.
- Do we as a utility want to implement either of these changes?
- We have to factor in the process the change and knowledge of the personnel that would perform these exams.

ISI Program Update Considerations

- For our Engineering group, we have had new ISI site engineers change jobs within the last 2 years. We have also had a number of changes to the corporate ISI Engineering group.

- We have also had changes in our NDE and Licensing personnel.

- We participate in the changes to ASME Section XI. These changes are based on Risk Informed, Technical projections such as NDE optimization, NDE personnel qualification, N-722, N-729, etc. AS the code cases have changed, we also would like to use the latest changes

ISI Program Update Considerations

- Much work goes into an ISI interval code change, much knowledge and time is also needed.

- At SNC, we are evaluating the 10 to 12 years interval change and also the 20-year use of ASME XI, staying with the same edition.

- We have 6 operating units and 2 new units we would like to keep on the same year and edition.

ISI Program Update Considerations

- Conclusion: SNC is evaluating the options for our 8 units and our ISI programs for each of the 8 units. Our engineering group is responsible for the direction we go as a fleet. New personnel, favorable code changes, the work to change 8 programs, the different ISI implementation dates all play into this decision. SNC does participate in industry efforts such as ASME XI, MRP, BWRVIP, EPRI and SGMP. This participation helps to ensure we can make the best decision for our 8 units.
- Many items to consider, and SNC will make the decision based on what we think is best for our fleet.

Questions

- What questions / issues are on your mind?

Panel discussion and Q/A from the audience

Ms Carol NOVE

Senior Materials Engineering

US Nuclear Regulatory Commission

Conclusion of day 3

NEA
NUCLEAR ENERGY AGENCY

OECD
BETTER POLICIES FOR BETTER LIVES

Working Group on Codes and Standards

Workshop on In-Service Inspection

Thank you for your participation today and see you all tomorrow @11AM (CET)!