Ongoing Researches in Age-Related Degradation of Reactor Materials in Korea

<Int. Workshop on age-related degradation of reactor vessels and internals >

23~24, May 2019, USNRC Rockville, MD, USA

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CONTENTS

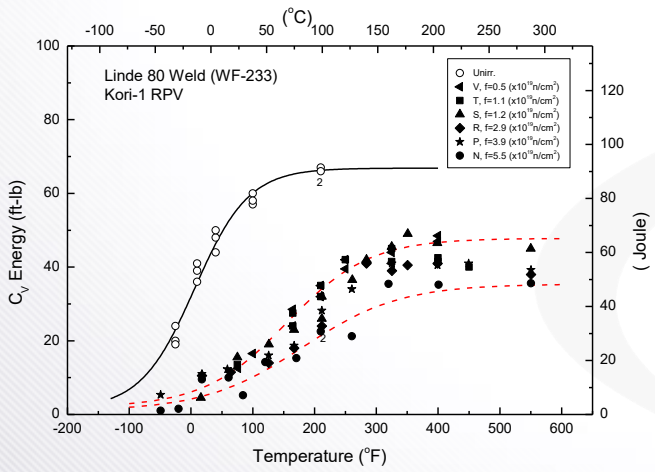
- Surveillance Tests of High Copper Weld; Linde 80, WF-233
- Irradiation Experiments of RPV steels by using Research Reactor, HANARO
- Examination Plan of RPV Internal of Kori-1 after 40yr operation
  - ✓ Flaw Indications in the Baffle Former Bolts at the Lowest Position
- Plan of Material Harvesting Projects for the Retired Kori-1 Components

- Radiation Damage in CANDU Fuel Channel Components
- SCC (Stress Corrosion Cracking) & Corrosion Related Projects
- Advanced Technology Development for Diagnosis and NDE

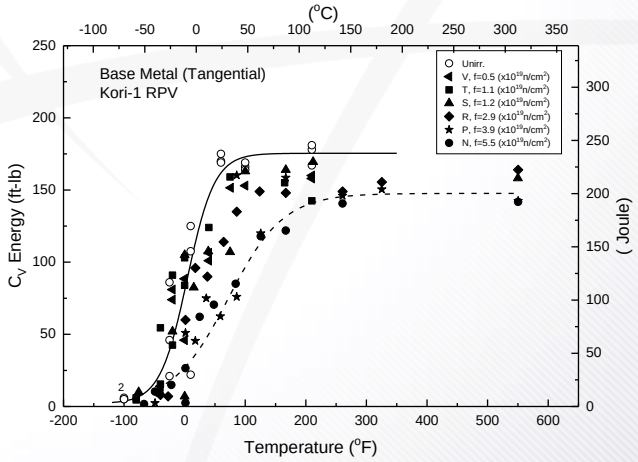
1 Surveillance Test Data of a High Copper Linde 80 Weld

- Kori-1 : 590MWe PWR, W/H 2-loop (1977~2017, 40 yr operation)
- Beltline circumferential weld: 0.23% Cu, Linde 80, WF-233
- The six surveillance capsules were all tested.
- Low USE (<50 ft-lb) & High ΔRT_{NDT} (RT_{PTS}>300°F)
- J-R tests by using modified 1X-WOL specimens
- M/C (T_o) tests by using reconstituted precracked Charpy specimens

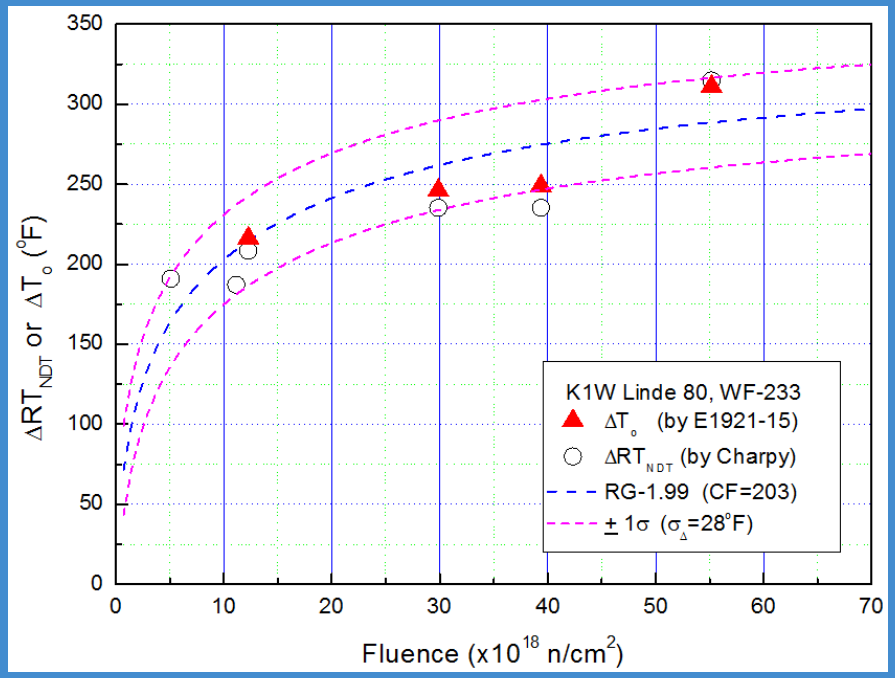
Weld (Linde 80, WF-233)



Base Forging (SA508-Cl.2)



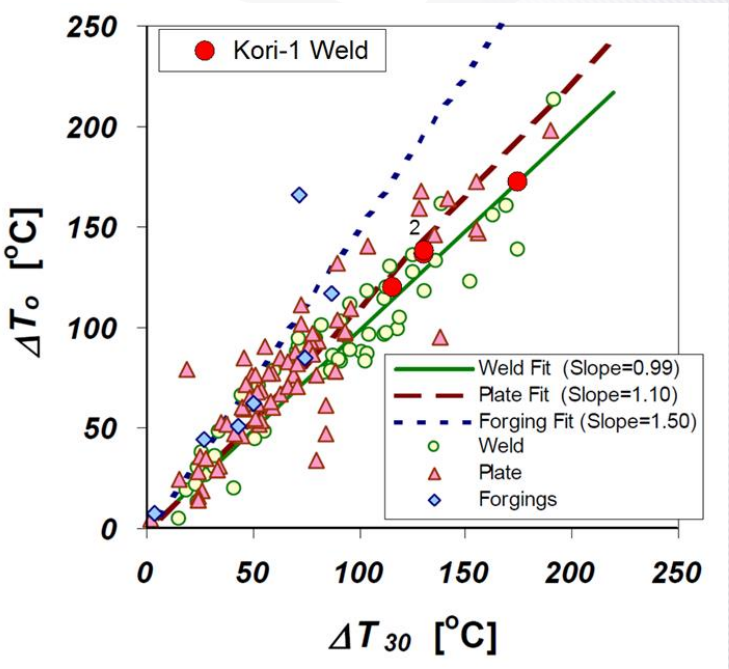
1 Transition temperature shifts by neutron irradiation



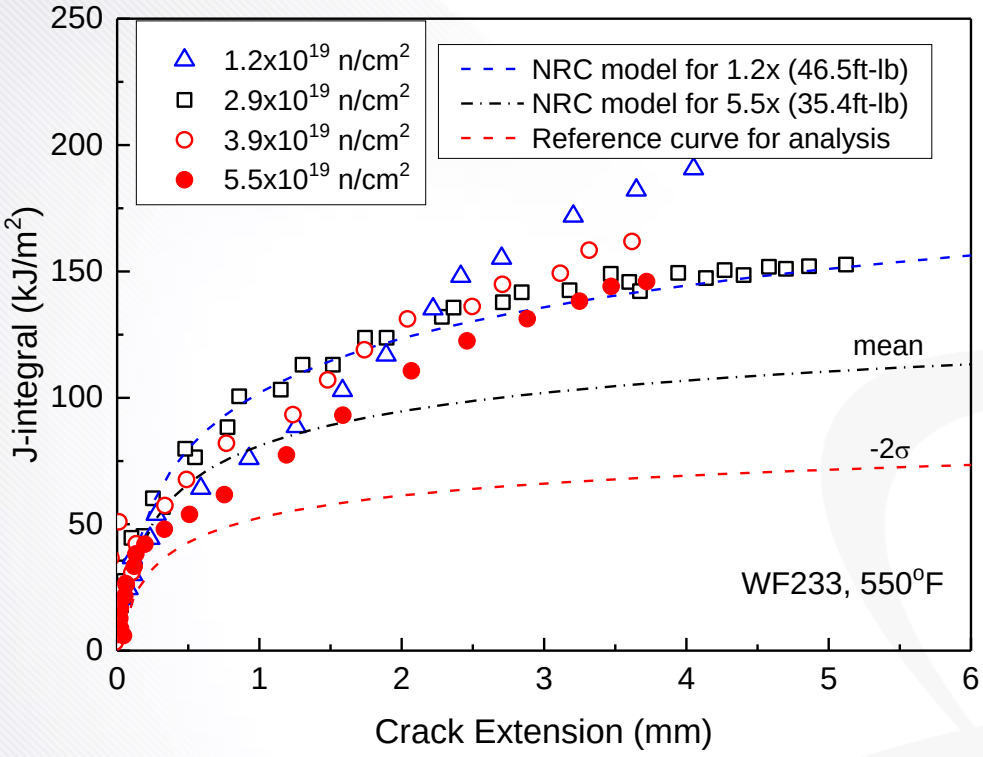
$\Delta T_{To} \approx \Delta T_{cv,30ft\text{-}lb}$ for the Linde 80 weld

Scatter was smaller in T_o values than Charpy.

by NUREG-1807

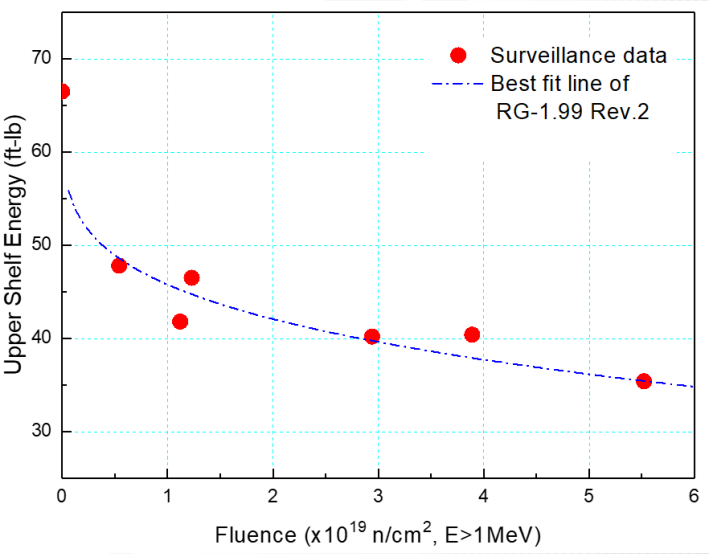


1 J-R curves from modified 1X-WOL specimens at 550°F



USNRC RG-1.161 database models could bound the K1W test data.

USE decreases according to RG1.99

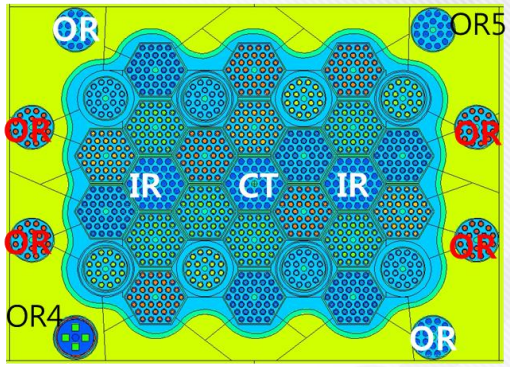


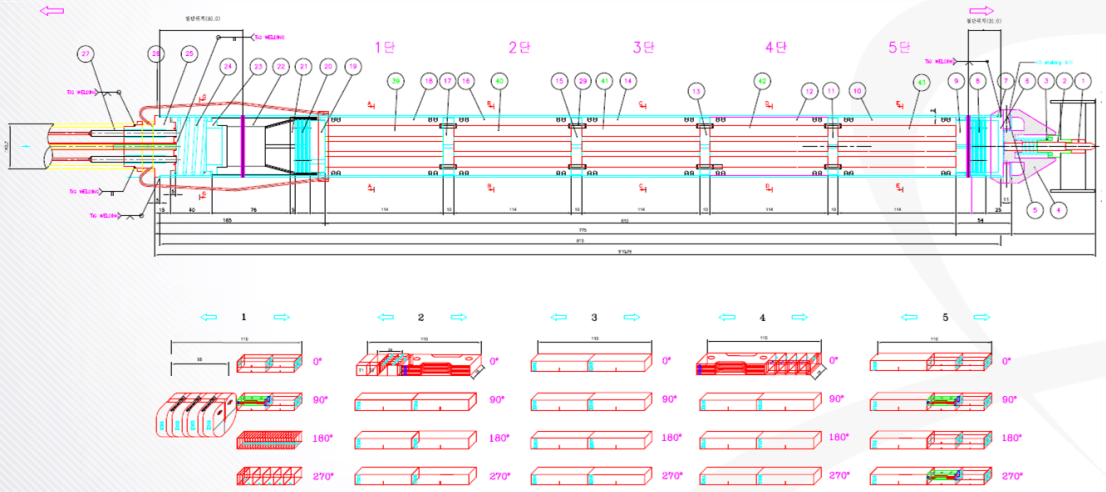
2 Irradiation Test Data of RPV steels using HANARO

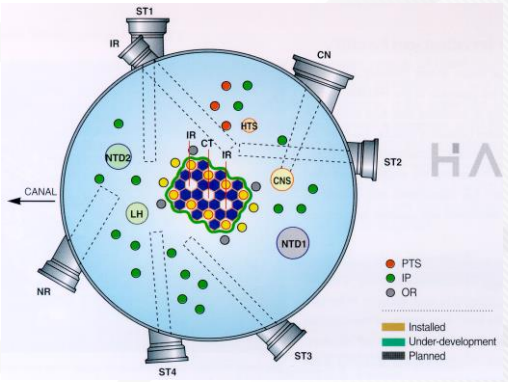
Instrumented capsule for material irradiation test

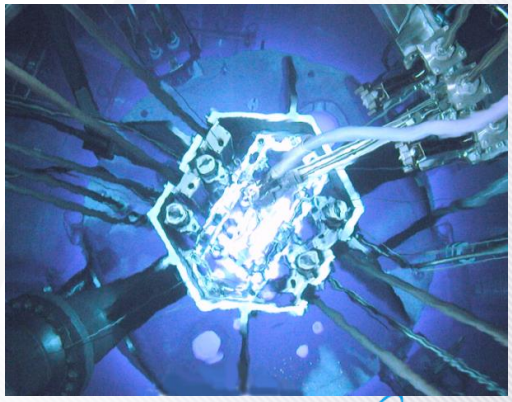
- CT hole flux: 0.7~1.5x10^{14} n/cm^2 · sec
- OR hole flux: 0.6~1.7x10^{13} n/cm^2 · sec

A capsule accommodates about 40 Cv specimens.









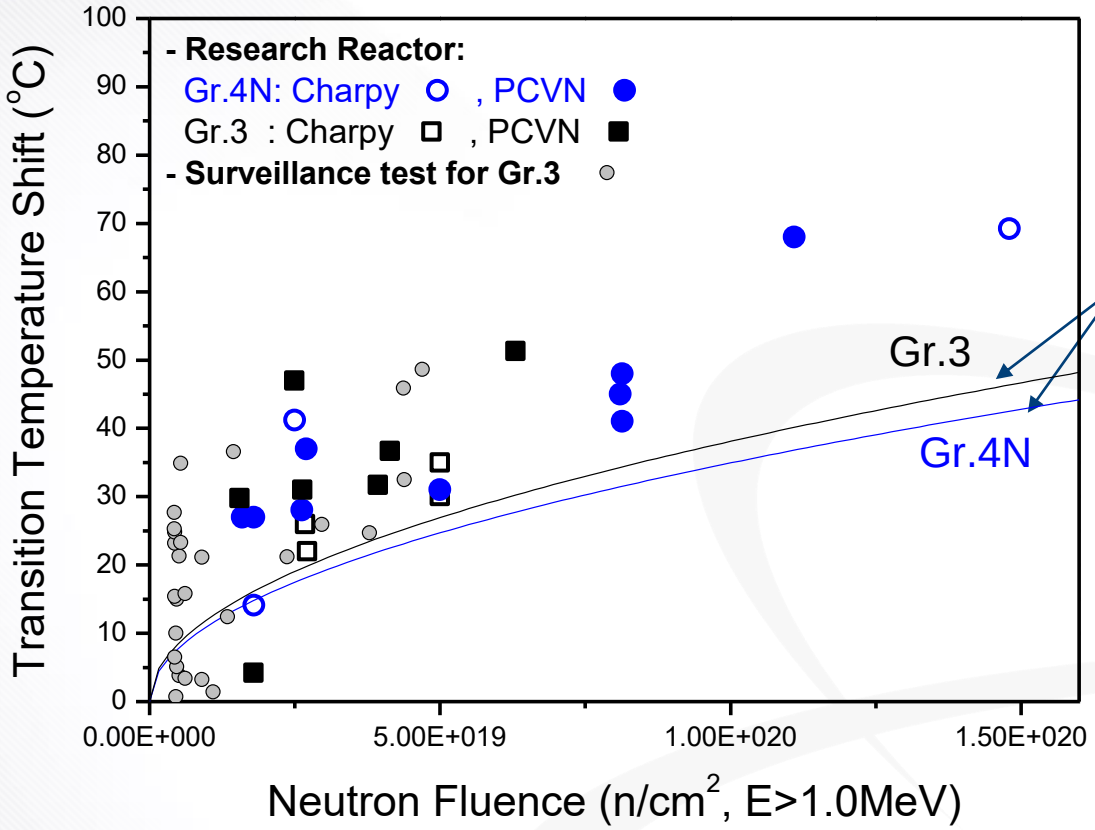
2 Summary of RPV Steels Irradiation Tests at HANARO

Capsule	Period	Fluence : $x10^{19}$ n/cm^2 (>1MeV)	Irr. Temp.: ℃	Materials	Irr. Hole	Remark
98M-02K	99.7.9~7.11	1.1~2.3	271~297	SA508 Gr.3	CT	RPV steel for OPR1000
99M-01K	00.6.14~6.17	1.3~2.9	272~305	SA508 Gr.3	IR2	
99M-02H	00.5.31~6.3	1.4~2.9	281~309	SA508 Gr.3	IR2	
00M-02K	01.5.2~5.6	1.2~3.3	282~309	SA508 Gr.3	IR2	
02M-02K	03.8.15~8.21	3.0~6.4	271~297	SA508 Gr.3	CT	
08M-01K	08.10.8~10.15	3.9~8.3	255~317	SA508 Gr.4N	CT	Model Alloy
08M-02K	09.4.7~4.28, 4.30~5.3	1.3~4.4	269~310	SA508 Gr.4N	OR5	
11M-25K	12.2.24~3.7	7.3~15.3	265~320	SA508 Gr.4N	CT	Model Alloy
13M-02K	13.5.27~6.24, 7.1~7.28	2.4~8.9	275~301	SA508 Gr.3	OR5	RPV steel for OPR1000 (High fluence effects)
16M-01K*	18.5.25~6.24, 7.9~7.30	1.9~7.1x10^{19}	290	SA508 Gr.3	OR4	RPV steel for APR1400
17M-01K*	18.11.20~11.28	4.6~9.1x10^{19}	194~249	SA533 B1	CT	RPV steel for WH RPV in Korea

Specimens : Charpy, PCVN, Tensile, CT, mini-PCVN, Small Punch, etc.

2 Irradiation Embrittlement Trend of Korean Forgings

◆ SA508 Gr.3 steels & Gr.4N model alloys



Trend curves
(EONY model)

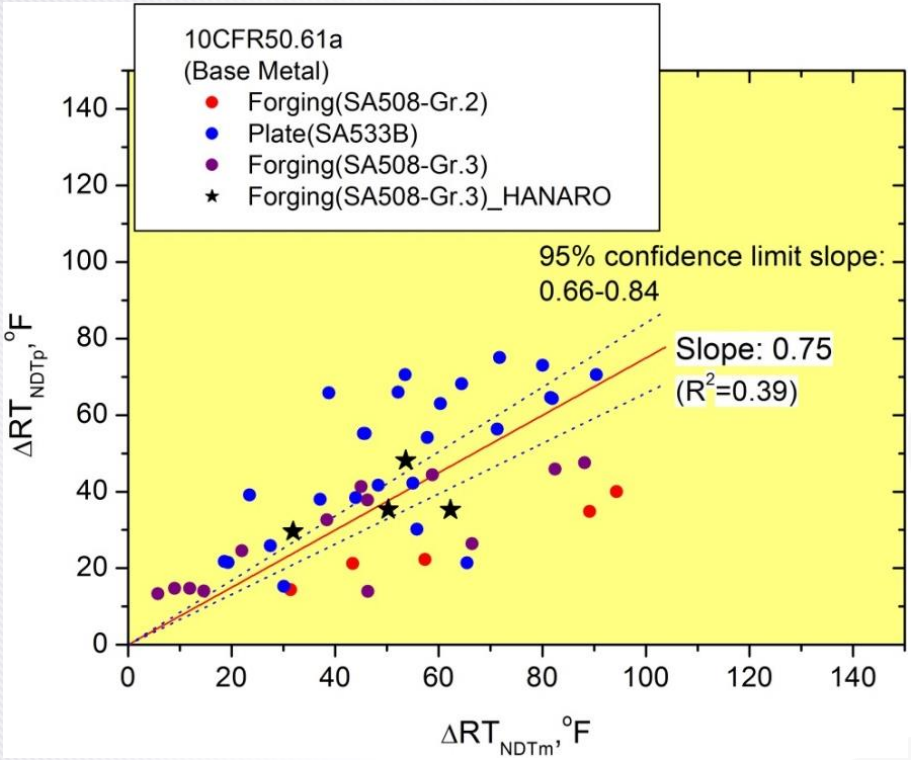
ΔT_{41J} = MD + CRP

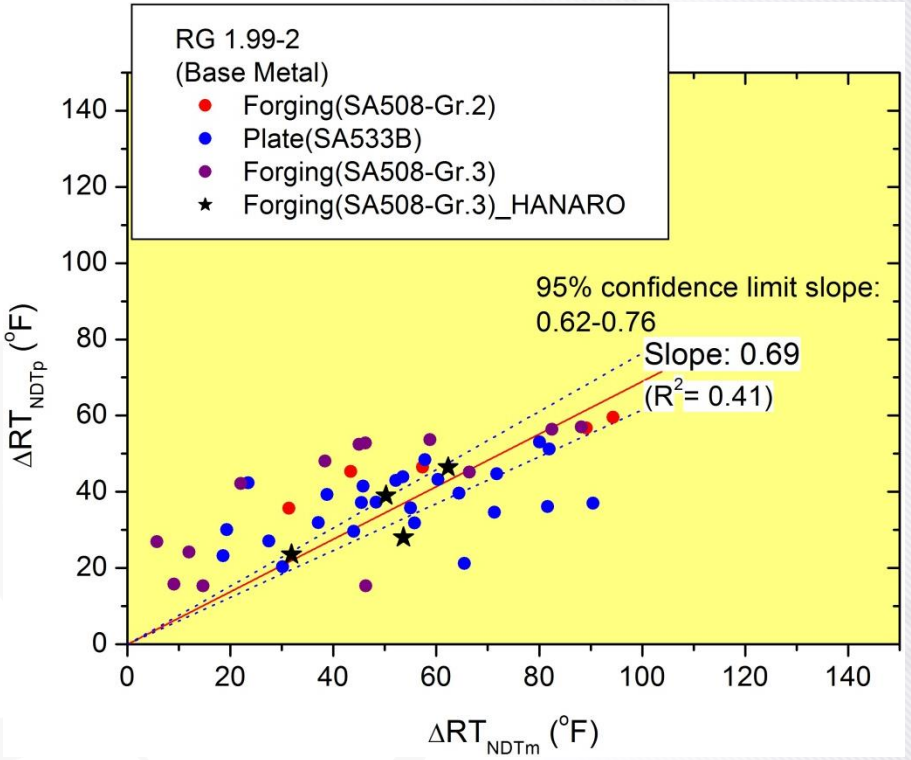
- Low Cu content
- Only MD was considered

⇒ Overall irradiation embrittlement trend of SA508 Gr.4N model alloys was similar to that of SA508 Gr.3 steels
Similar result was reported previously (*G. Wire et al., ASTM STP 1447, 2004)

⇒ Ni content in SA508 Gr.4N is higher, but P & Mn contents are low

2 Embrittlement Prediction of Surveillance Data





- Measured shifts are bigger than the model predictions
- HANARO research reactor data are comparable to plant data.

  (even though the flux range is about three orders higher, ~10^{14}n/cm^2.sec)

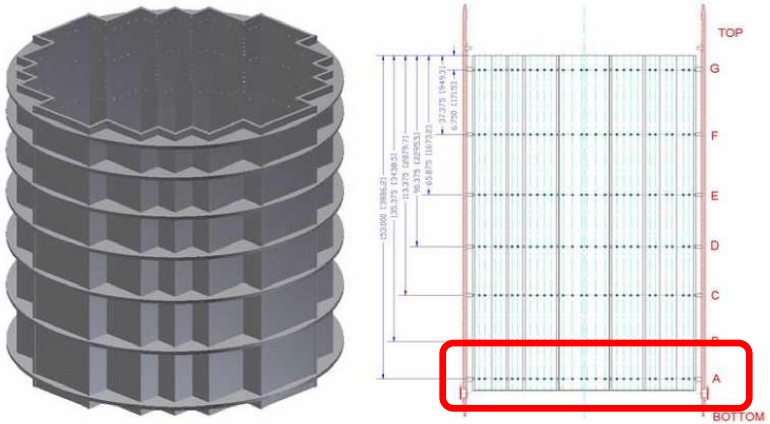
3 Examination Plan of RPV Internal of Kori-1 after 40 yr

- Defect indications on 8 baffle-former-bolts at Kori-1 RV internal (May 2015)
  - Defect signals were found at a lower dose level position.
- Kori-1 was permanently shutdown in 2017 after 40 yr operation.
- The bolts of defect signals will be removed for destructive analysis.
  - To identify the actual defect morphology of the signal indications
  - For better understanding on degradation(IASCC) mechanism of BFB materials
  - For better modelling of IASCC parameters, such as dose and stress, etc.
  - To improve the aging management of RV internals
- Will be extended to the materials harvesting project for the retired Kori-1 components for evaluation of irradiation effect and materials degradation on the actual components(RPV, Internals, Nozzles, SG)

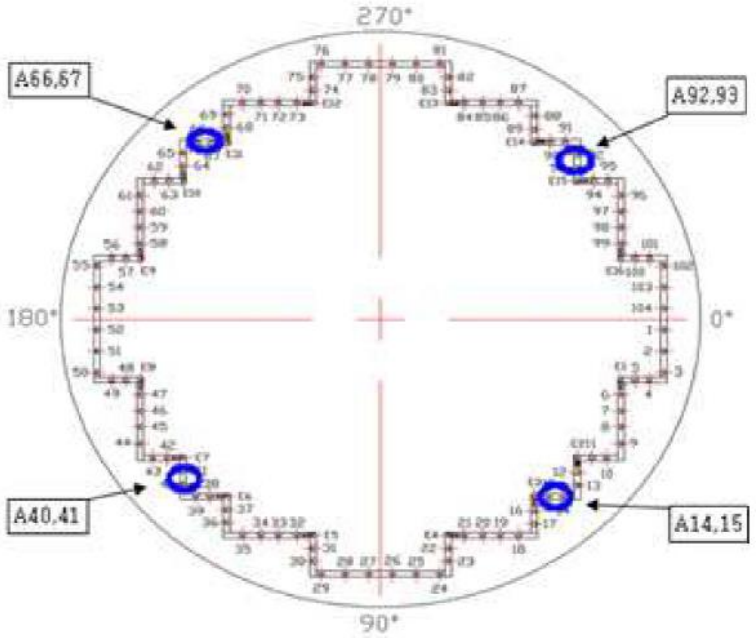
3 Defect Indications on BFBs in Kori-1*

- Kori-1: W/H 2 loop PWR(WH-60), commercial start from April 1978
- Baffle Former Bolts 728 ea, Baffle Plate Edge Bolt 176 ea
- Bolt material : Cold worked 316 stainless steel
- NDE signal at 8 bolts : UDL(Unknown Defect Location) May 2015
- Location: Head shank of the bolt (Lowest former A)

Baffle Former Assembly of Kori-1



8 Positions of BFBs with defect indications



*Source: KINS/RR-1355

3 Destructive Analysis Plan

- To confirm the defect signals on Kori-1 BFBs
  - Chemical analysis, Metallographic examinations, Micro hardness
  - TEM microstructure, Radiation induced segregation
  - Map of defect morphologies (Comparison UT signals with metallography)
- Benchmark Zorita projects
- Collaboration with other countries

- EPRI is very much interested in the IASCC mechanism on this matter.
  - While there are some studies on the removed BFBs with UT indications, cases from the lower fluence position are rare to date.
  - Data on these bolts would fill a significant gap in our understanding of the quantitative effects of radiation dose and stress on the irradiation assisted stress corrosion cracking of baffle bolts
  - It would be also interesting to obtain a barrel weld material and to assess potential differences in behavior (fracture toughness) as a function of local dose.

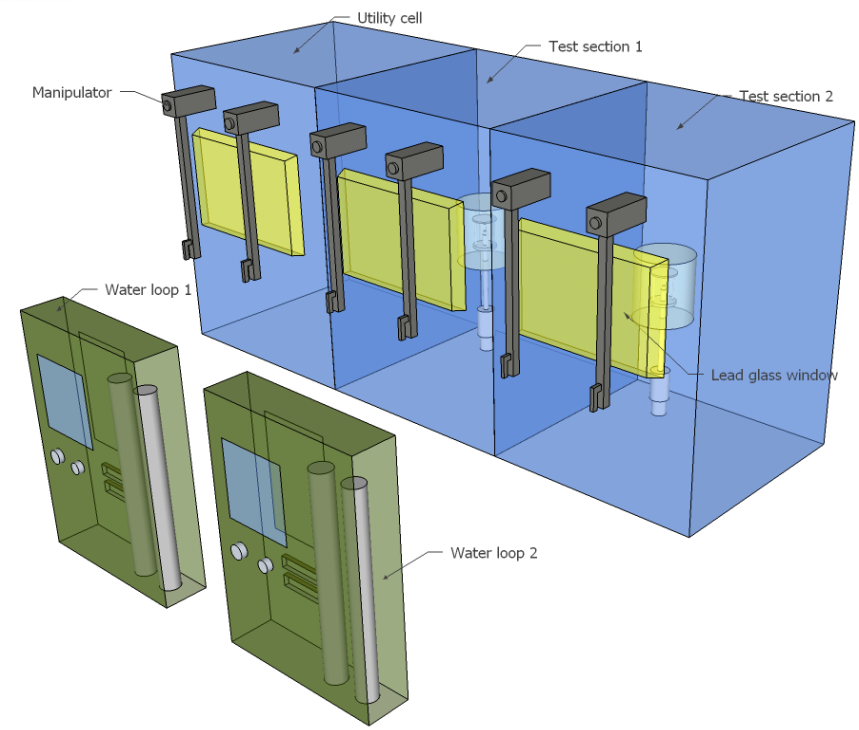
4 Plan of National Project on the Kori-1 Harvested Materials

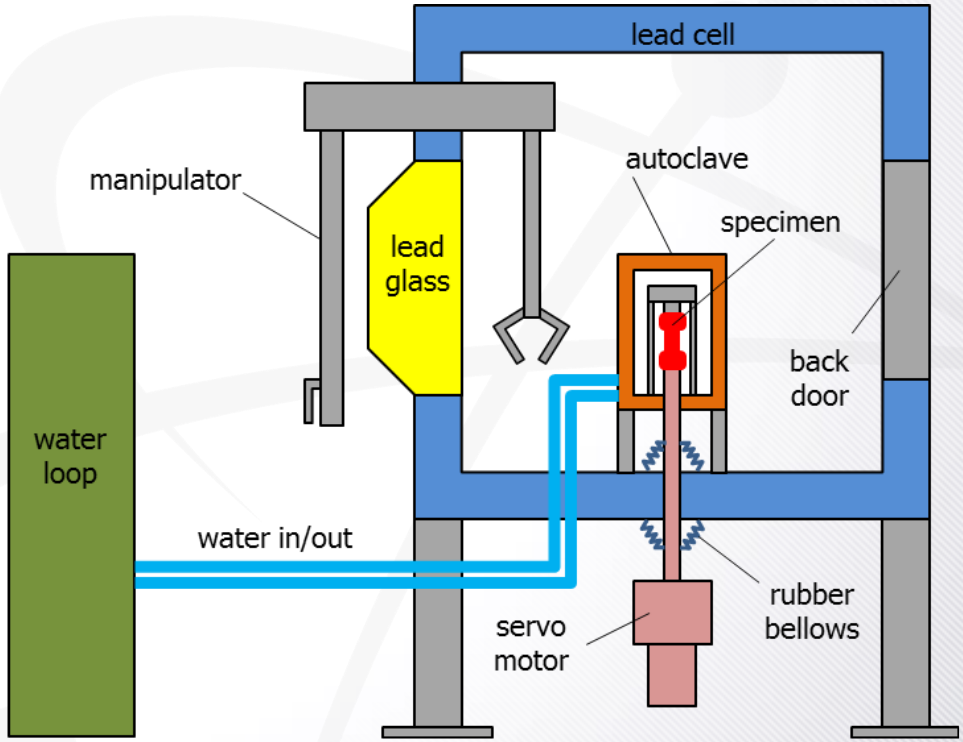
- Project title: ‘Technology development for IASCC on reactor internals’ under the program of ‘Nuclear core technology development’
- Sep. 2019 through Aug. 2024, hopefully (Stage-I)
- Participation: KAERI, KHNP-CRI and other companies
- Deliverables
  - Specimen machining technology for highly radioactive materials
  - Test facility demonstration for IASCC experiments
  - Identification and characterization of the defect signals on Kori-1 BFBs

- Stage-II will include the other topics (after 2022)
  - Embrittlement of RPV thick wall & actual welds
  - SG tube integrity & SCC resistance of Alloy 690
  - Demonstration of the NDT capability : SG tube, Pipe Welds

4 Facility Set Up for IASCC Experiments

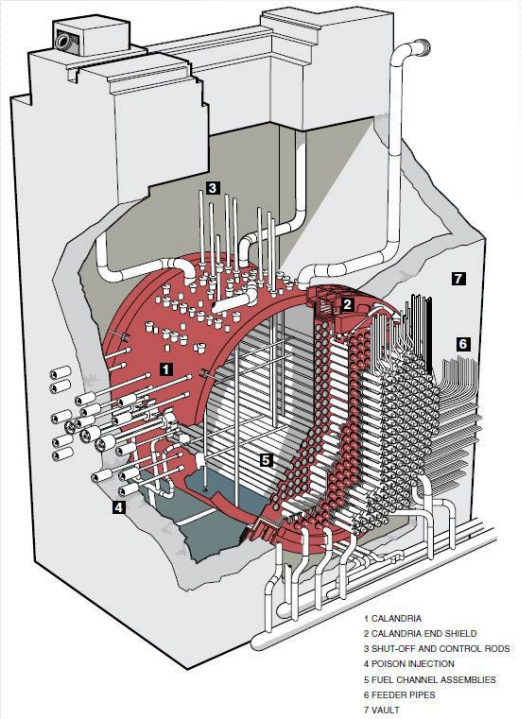
- Lead-shielded hot cells are going to be constructed for IASCC tests.
- Design of the building and the hot cell facilities are going on.
- International cooperation on high dose materials testing experiences is necessary.

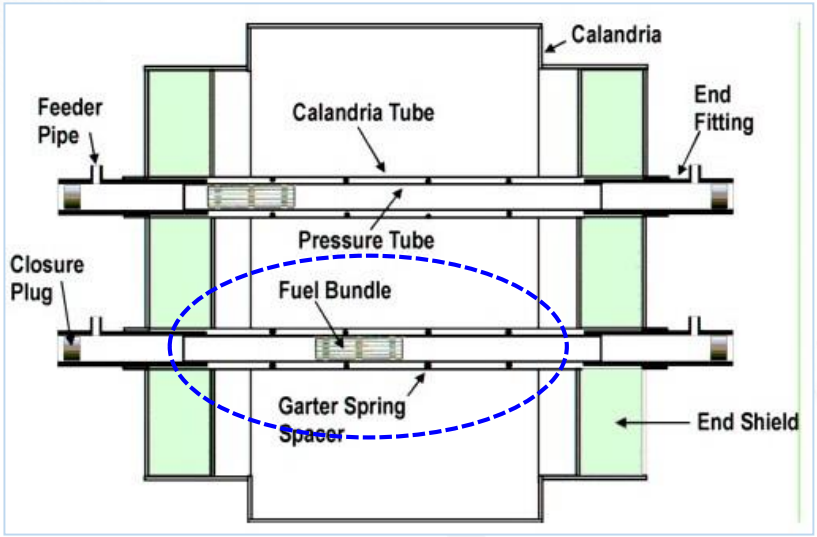


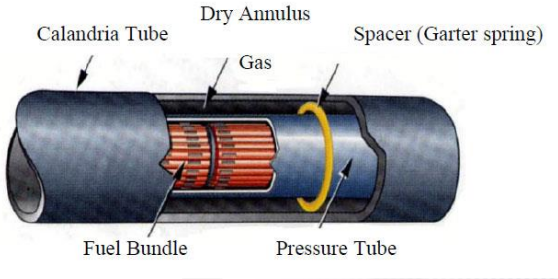


5 Radiation Damage in CANDU Fuel Channel Components

4 CANDU units of the 30 yr original design life







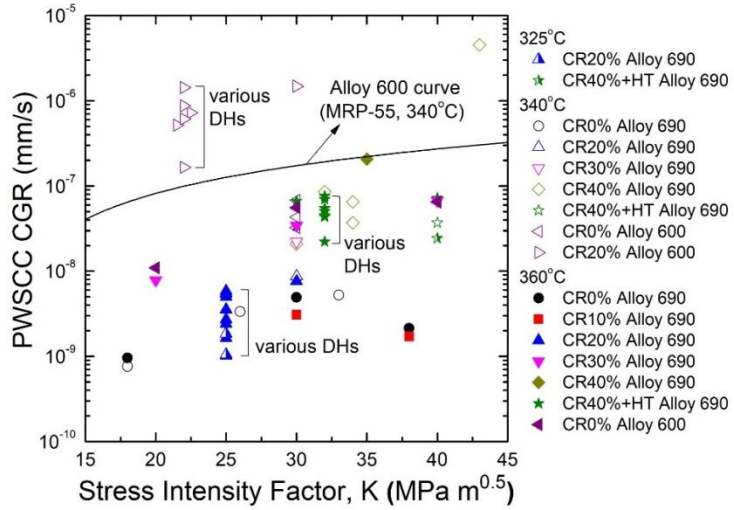
❖ CANDU 6 Pressurized heavy water reactor

	WS-1	WS-2	WS-3	WS-4
Commercial operation	1983. 4.	1997. 7.	1998. 7.	1999. 10.
Capacity (MW)	678		700	
Total Capacity	2,778MW (7.5% of total nuclear power generation)			

- Life time is controlled by Irradiation of Pressure Tube
  - Axial/Diametral Growth
  - ST (by using Removed PT) & Prediction Model
  - He Embrittlement of Spacer

6 SCC (Stress Corrosion Cracking) & Corrosion Related Projects

- PWSCC CGR model for Alloy 690 Nozzles & Tubes
- Mechanistic Studies on Crack Initiation of Alloy 690
- IASCC Studies of Proton Irradiated Stainless Steels
- FAC (flow accelerated corrosion) tests with a large test loop
- Mitigation of sludge & crud by mechanistic understanding & water chemistry



$$CGR = f_{SH}\, f_T\, f_{DH}\, K^n$$

경화도 모듈 : $f_{SH} = 1 + Q_{SH}{}^m$

온도 모듈 : $f_T = \exp\left[-\frac{E_{act}}{R}\left(\frac{1}{T} - \frac{1}{T_{ref}}\right)\right]$

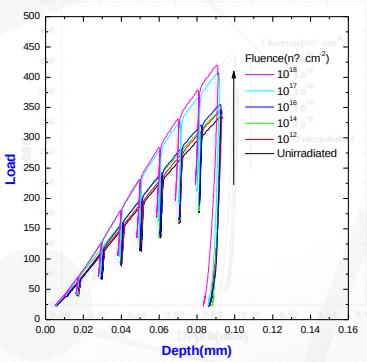
DH 모듈 : $f_{DH} = 1 + (-1)\exp\left\{-T^2\left[\ln(DH) + \frac{8440}{T} + \right]^2\right\}$

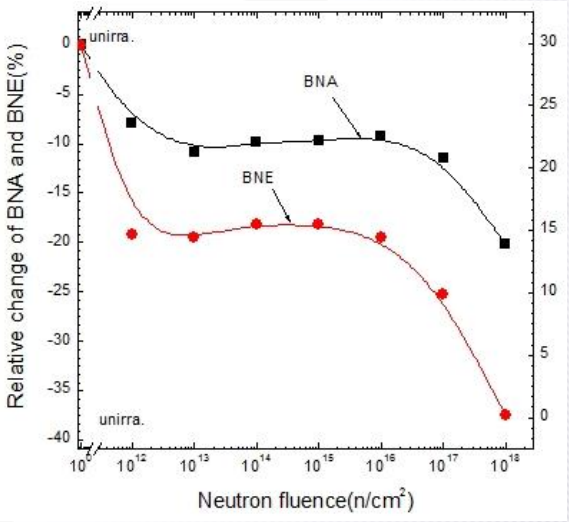
(*) ISG-TIP6 (Steam generator Tube Integrity Program) meeting at ANL this week.

7 Advanced Technology for Diagnosis and NDE

- Advanced ECT probe for diagnosis of steam generator U-bend
  - Evaluation of the effects of sludge & foreign object noise
- Non-linear UT techniques for cracks in pipe welds
- Long range guided wave technique for buried pipe inspection
- PEC (Pulsed Eddy Current) with Hall sensor array for diagnosis of pipe under insulation

- Basic Studies on Non-Destructive Characterization of Irradiation Embrittlement of RPV steels
  - Magnetic Barkhausen Noise
  - Maybe a possible tool
  - Needs samples of a series of actual neutron exposure





Data from TRIGA at 70°C

SUMMARY

- Irradiation embrittlement data from surveillance tests and research reactor tests were summarized for Korean RPV steels.
- Some baffle-former-bolts of defect signals will be removed from Kori-1 internal and investigated in detail soon.
- Materials harvesting projects for the retired Kori-1 components are being prepared under the national nuclear safety research programs.
- Other important researches are also going on;
  - Radiation Damage in CANDU Fuel Channel Components
  - SCC (Stress Corrosion Cracking) & Corrosion Related Projects
  - Advanced Technology for Diagnosis and NDE