

Life extension programme for Kori-1 nuclear power plant

- Day-4 -

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Day-4 : Life extension programme for Kori-1 nuclear power plant

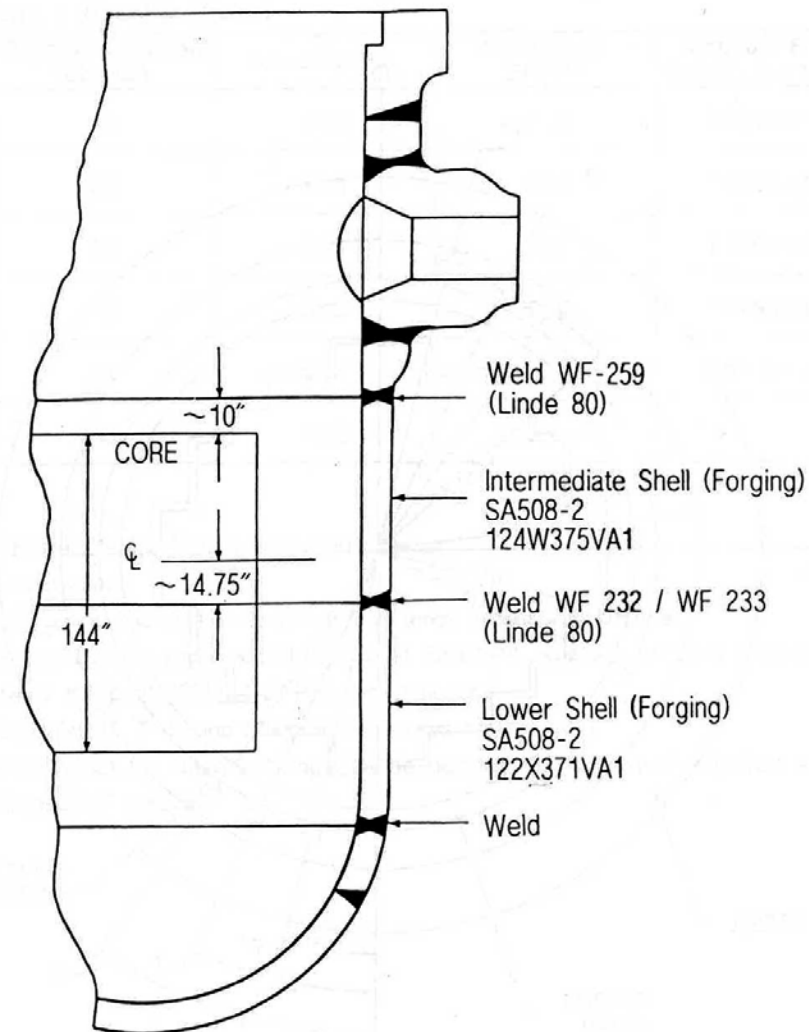
- Integrity assessment of Kori-1 RPV low toughness weld
- PSR (Periodic Safety Review) & LTO (Long-Term Operation)
- Further consideration for Kori-1 LTO application

20 operating nuclear power plants in Korea

Plant (design life, yr)	Reactor Type	Capacity (MWe)	NSSS Supplier	Plant A/E	Commercial Operation	RPV Material	
Kori (30 for #1 / 40 for #2,3,4)	#1	PWR	587	W/H	Gilbert	'78.4	SA508-Gr.2
	#2	PWR	650	W/H	Gilbert	'83.7	SA533-B1
	#3	PWR	950	W/H	Bechtel/KOPEC	'85.9	SA533-B1
	#4	PWR	950	W/H	Bechtel/KOPEC	'86.4	SA533-B1
Wolsong (30)	#1	PHWR	679	AECL	AECL	'83.4	-
	#2	PHWR	700	AECL/DOOSAN	AECL/KOPEC	'97.6	-
	#3	PHWR	700	AECL/DOOSAN	AECL/KOPEC	'98.6	-
	#4	PHWR	700	AECL/DOOSAN	AECL/KOPEC	'99.9	-
Yonggwang (40)	#1	PWR	950	W/H	Bechtel/KOPEC	'86.8	SA533-B1
	#2	PWR	950	W/H	Bechtel/KOPEC	'87.6	SA533-B1
	#3	PWR	1000	DOOSAN/CE	KOPEC	'95.3	SA508-Gr.3
	#4	PWR	1000	DOOSAN/CE	KOPEC	'96.1	SA508-Gr.3
	#5	PWR	1000	DOOSAN	KOPEC	'02.5	SA508-Gr.3
	#6	PWR	1000	DOOSAN	KOPEC	'02.12	SA508-Gr.3
Ulchin (40)	#1	PWR	950	Framatome	Framatome	'88.9	16MND5
	#2	PWR	950	Framatome	Framatome	'89.9	16MND5
	#3	PWR	1000	DOOSAN/CE	KOPEC	'98.8	SA508-Gr.3
	#4	PWR	1000	DOOSAN/CE	KOPEC	'99.12	SA508-Gr.3
	#5	PWR	1000	DOOSAN	KOPEC	'04.7	SA508-Gr.3
	#6	PWR	1000	DOOSAN	KOPEC	'05.4	SA508-Gr.3

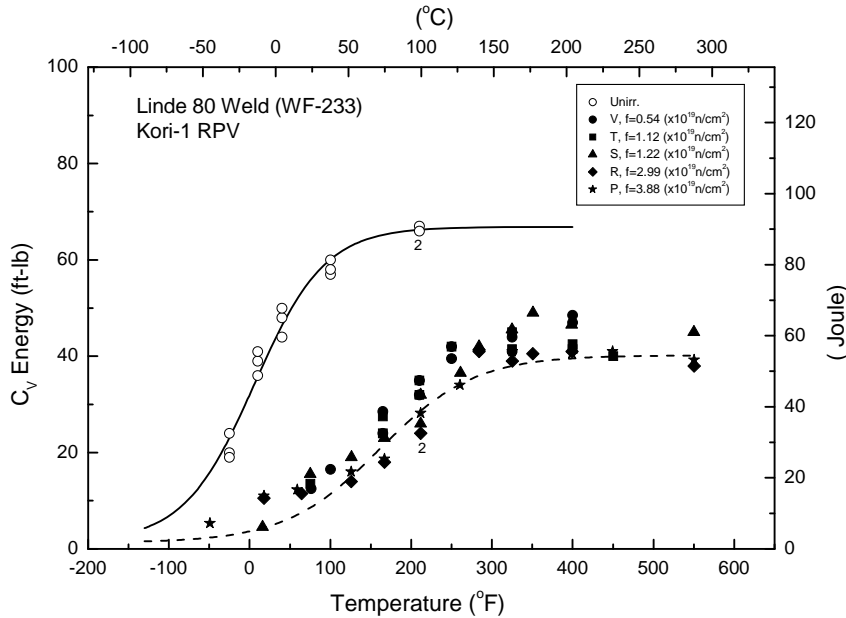
Kori-1 PWR (launched in 1978)

- ❑ Westinghouse Design (2-loop PWR)
Bobcock & Wilcox RPV & Weld
- ❑ Base Metal: SA508-cl.2 steel forging
- ❑ Beltline circumferential weld
Linde 80 flux weld (WF-233)
Weld wire heat no. = T27944
Flux lot no. = 8790
- ❑ Size: 600 MW capacity
ID 132 inch (3.35m)
Thickness 6.5 inch (165 mm)
- ❑ Design/Operating Pressure:
2485 / 2235 psig (170 / 152 atm)
- ❑ Design Life : 30 yr (FSAR),
40 yr (Equipment Spec. WH-676413, 1969)

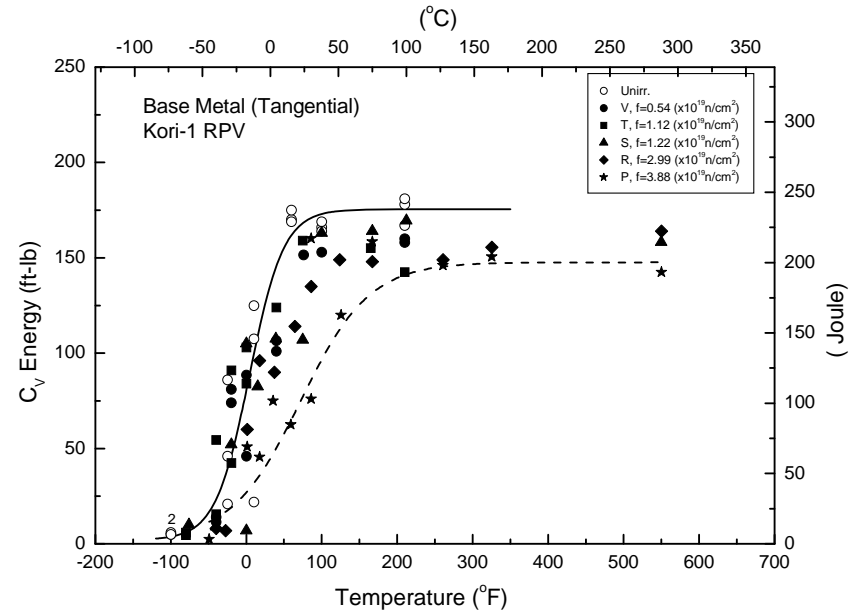


Surveillance tests summary

Weld (Linde 80, WF-233)



Base Forging (SA508-CI.2)



High copper Linde 80 weld is the limiting material.

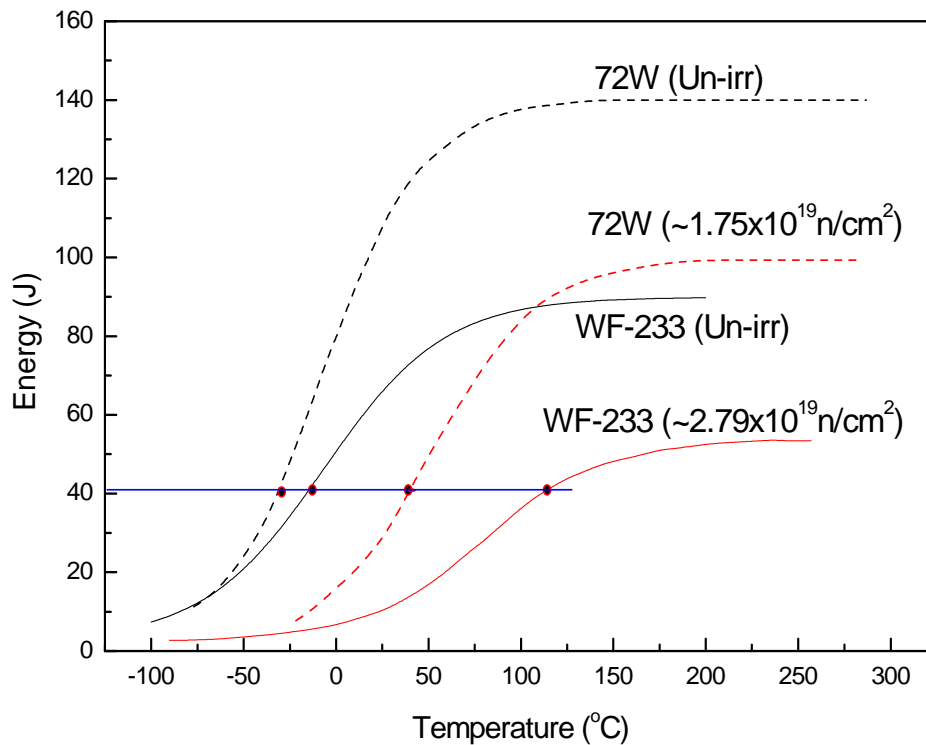
- USE is lower than 50 ft-lb (after 6 EFPY)
- RT_{PTS} exceeds 300°F (after 27 EFPY)



Requires a plant-specific safety analysis based on F/M tests.

Linde 80 (Linde 1092) flux welds issues

Several NPP's constructed in USA in early 70's (including Kori-1, Korea) are expected to exceed the screening criteria during the life extension period.



USE is low.

RT_{NDT} is high.



Plant-specific safety analysis was necessary to conform to the regulatory rules.

Many investigations have been focused on these materials.

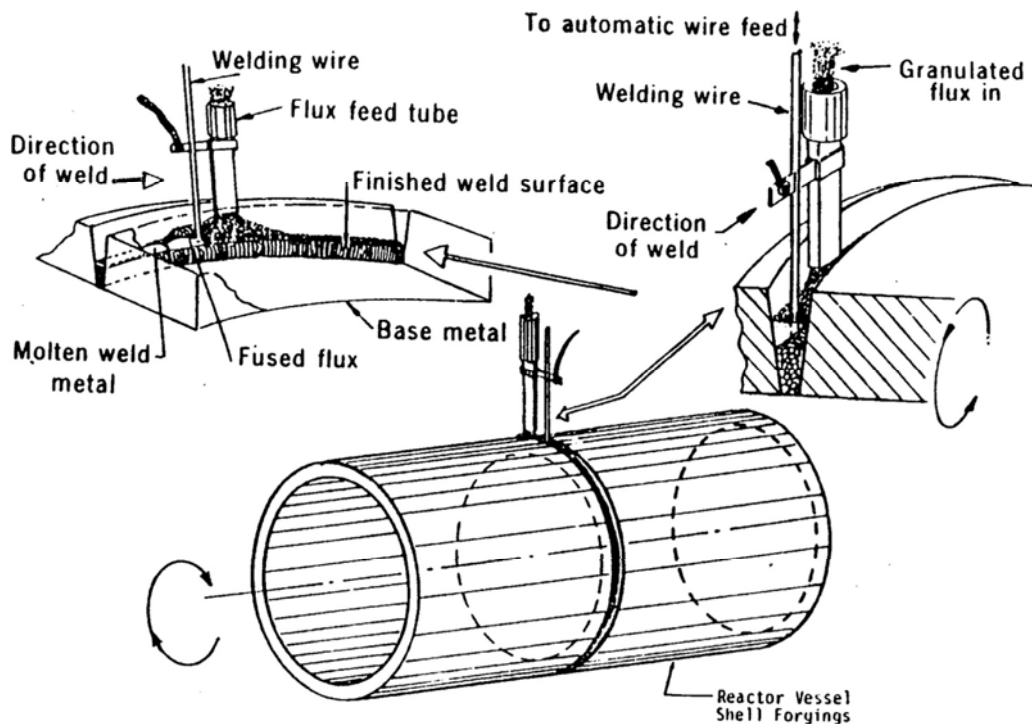
Low upper-shelf, High copper Linde 80 flux weld

Standard welding practice in USA, 1967-1971

Automatic Submerged-arc weld process

Cu plated* Mn-Mo-Ni filler wire and Linde 80 flux ($\text{SiO}_2 + 8-10\% \text{MnO}$)

* *Electrical conductance during welding*
Corrosion resistance during storage



<ex, WF-233>

Cu : ~0.23 wt%

Ni : ~0.61 wt%

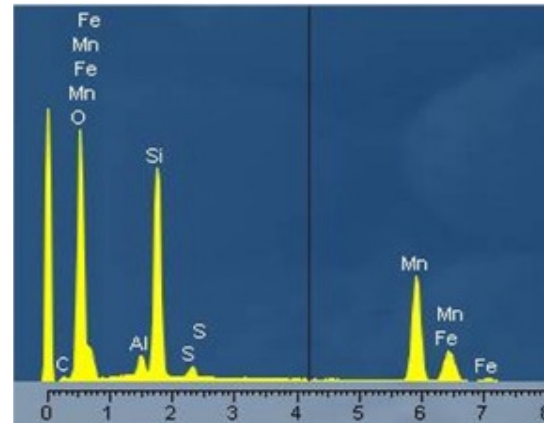
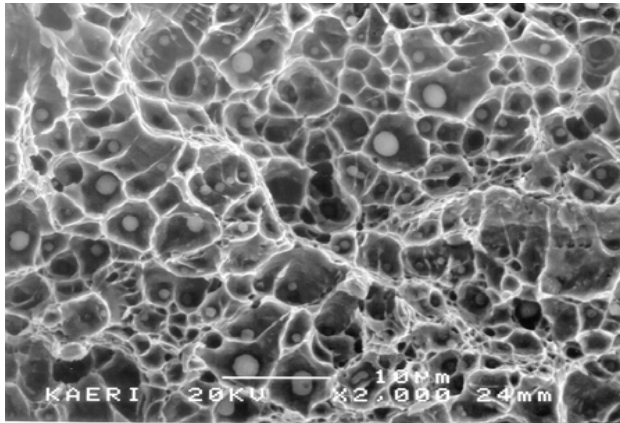
P, S : ~0.016 wt%

C : ~0.1 wt%

Mn : ~1.6 wt%

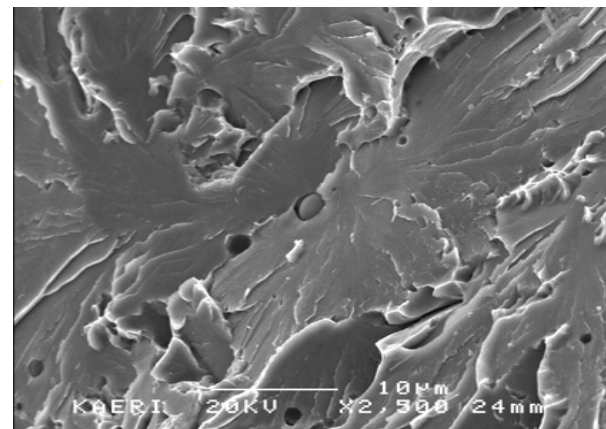
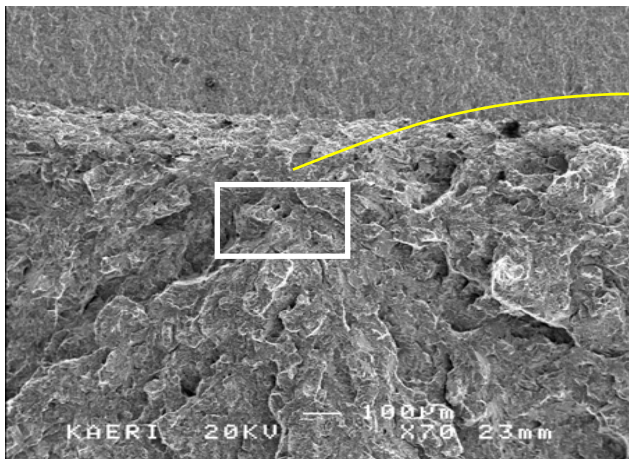
Si, Mo : ~0.5 wt%

Metallurgical investigations



Mn-Si-Al type inclusions prevail among the inside microstructure.

The inclusions act like a cleavage initiation site for brittle fracture.



Regulatory actions related to the Charpy properties

- ❑ Low upper-shelf toughness (<50 ft-lb)
 - An equivalent fracture mechanics analysis based on the J-R method.
 - USNRC RG-1.161 provides both the evaluation procedure and J-R database.

- ❑ High RT_{NDT} (RT_{PTS}) (>300°F)
 - A direct estimation of fracture toughness by using the “Master Curve” method (ASTM Standard E1921: T_0 , A reference temperature describing ductile-brittle transition of fracture toughness)
 - ASME Code Case N-629: RT_{T_0} , An alternative indexing of RT_{NDT}

$$RT_{T_0} = T_0 + 35^\circ\text{F}$$

Plant-specific fracture mechanics analysis based on the fracture toughness data rather than the Charpy impact data.

Master Curve Approach

Similar issues in WWER-440 RPV

- Some reactors (V-230) do not have surveillance programs and “boat-samples” were taken from the RPV inner surface of operated units.
- Design concept was transportable by land (by train and/or by road). => Smaller diameter and thinner wall with higher strength materials => Smaller water gap thickness between fuel and the vessel wall => Higher neutron flux on the wall => requires higher resistance against IE
- Relatively high content of phosphorus in the beltline welds (~0.03wt%)
Copper content is also relatively high (0.15~0.20wt%)
Limited information about initial transition temperature.
- Some reactor vessels were thermally annealed to recover the embrittlement.
=> Studied on re-embrittlement mechanisms, kinetics and prediction models



IAEA Coordinated Research Programs.
IAEA International Database on RPV materials (IDRPVM)



Master Curve
Approach

Irradiation embrittlement of the beltline weld (Linde 80) was the key.

Efforts to solve the Kori-1 case

S1: Low upper-shelf issue (J-R tests)

**S2: Plant specific PTS analysis
(probabilistic fracture mechanics)**

**S3: RT_{PTS} re-evaluation
(by Master curve testing)**

**S4: Pressure-temperature limit curve
(by Code case N-641)**

- ❑ 5th Surveillance tests (Charpy) (1980, 1984, 1986, 1990, 2000)
- ❑ Low upper-shelf evaluation (J-R tests by modified 1X-WOL specimens)
 - 1988, per NUREG-0744
 - 1994 & 2005, per USNRC DG-1023 & USNRC RG-1.161
- ❑ Master curve fracture toughness tests (RT_{PTS} re-evaluation)
 - Unirradiated archive weld (2000, ASTM E1921-97)
 - Irradiated tests by reconstituted specimens (2006, 3/4/5 capsules)
 - Additional unirradiated tests (2006, ASTM E1921-05)
- ✓ Plant-specific analysis of PTS, based on the probabilistic fracture mechanics, USNRC RG-1.154 (2003)
- ✓ P-T limit curve was revised by CC N-641 (circumferential flaw, 2004)

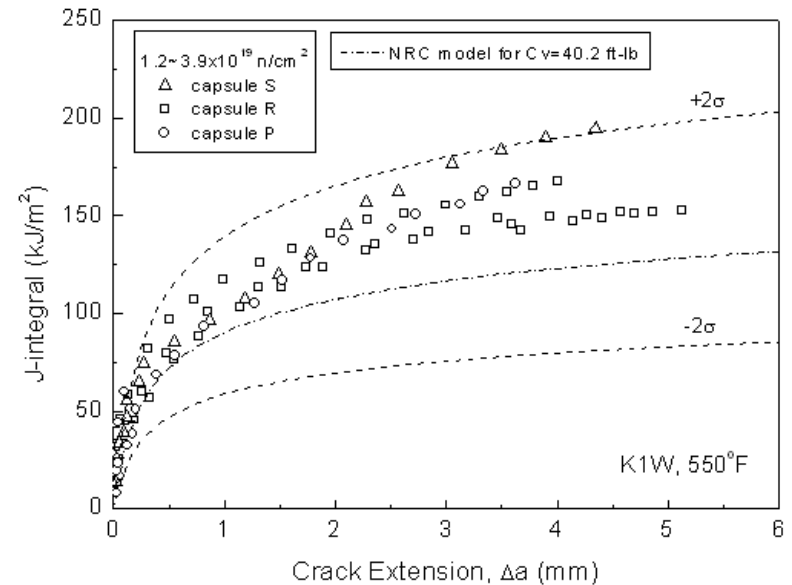
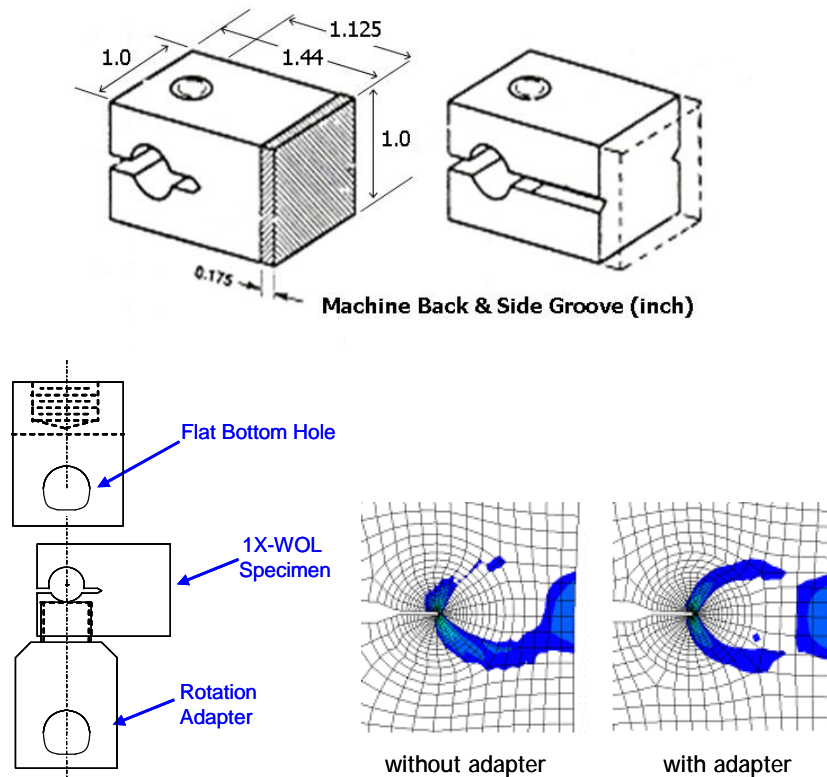
S1: Low upper-shelf energy => J-R tests & analysis

Elastic-plastic fracture mechanics methodology (J-R) :

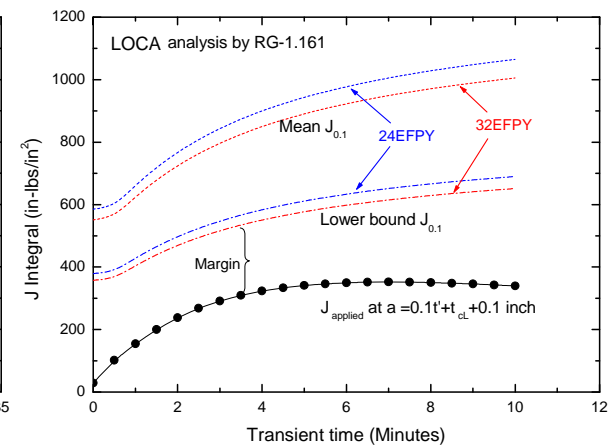
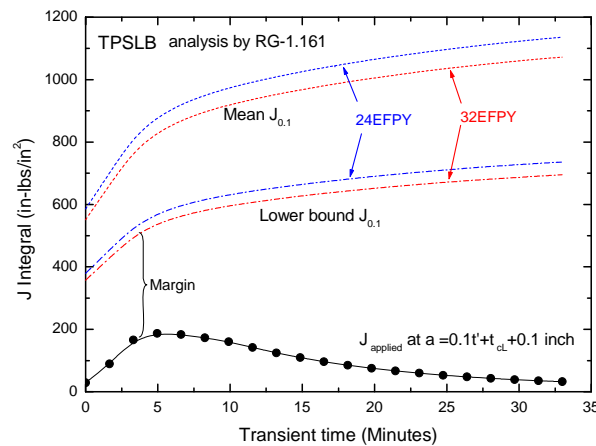
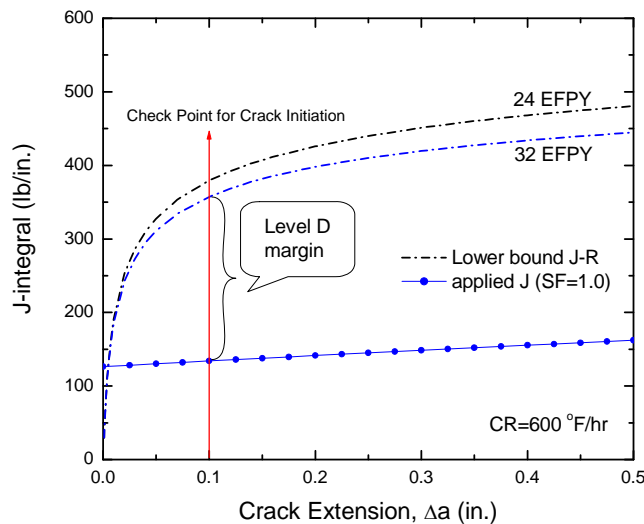
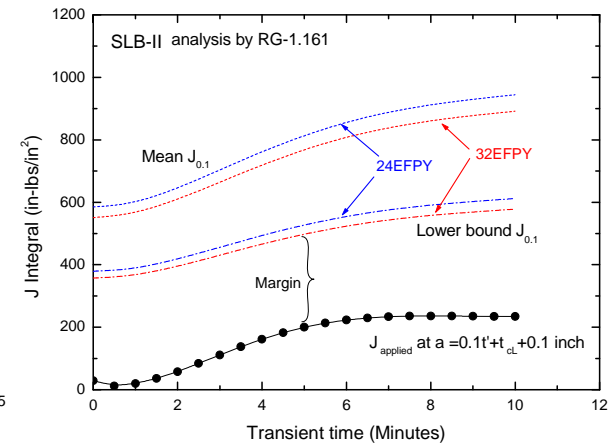
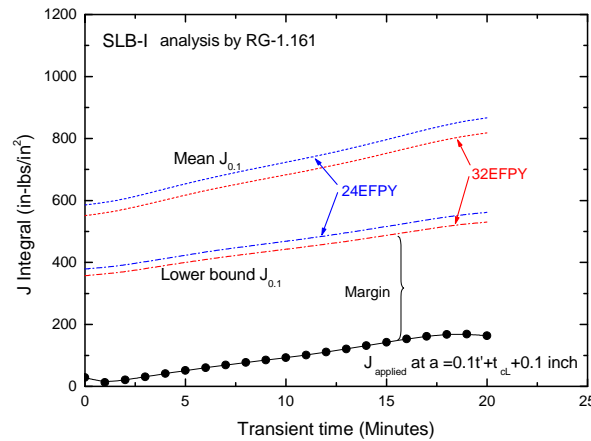
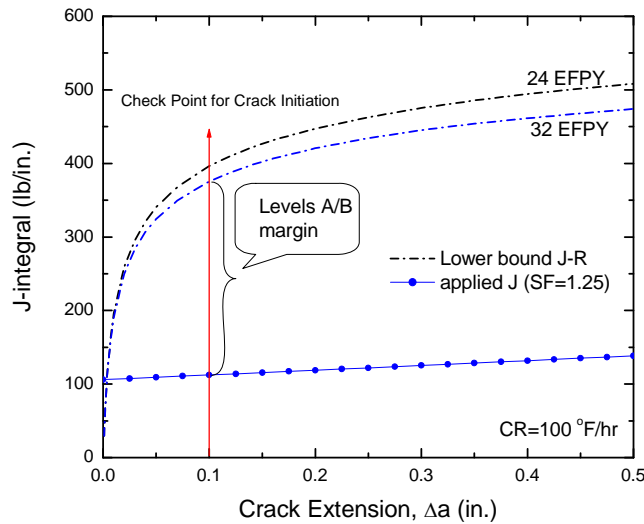
J-T method, DPFAD, J-R CDF

USNRC RG-1.161 (1995) provides the safety criteria and J-R database.

Kori-1 produced its plant specific J-R data from irradiated WOL specimens.



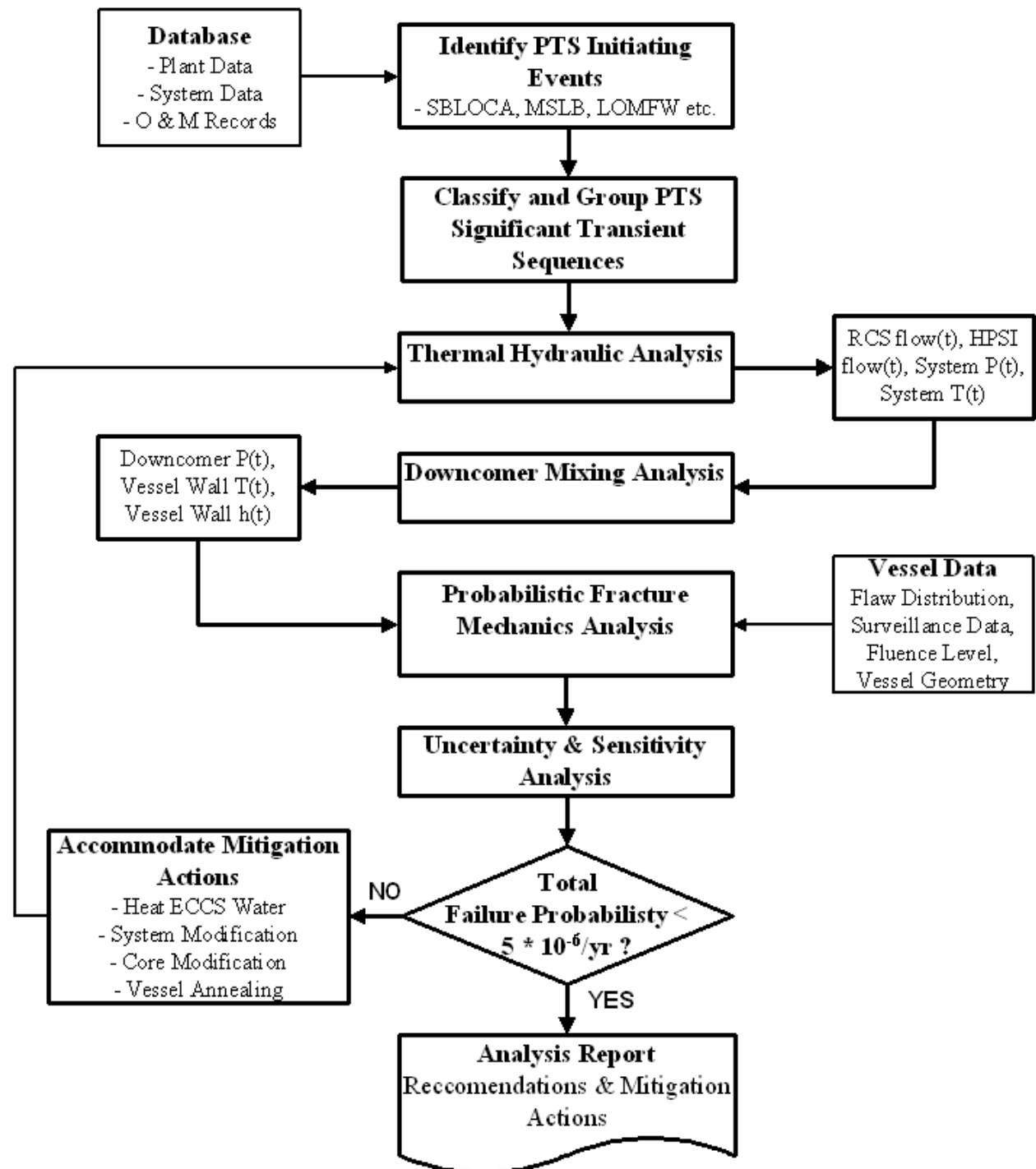
A generic analysis and the plant specific analysis demonstrated a sufficient safety margin against ductile failure at the upper-shelf region.



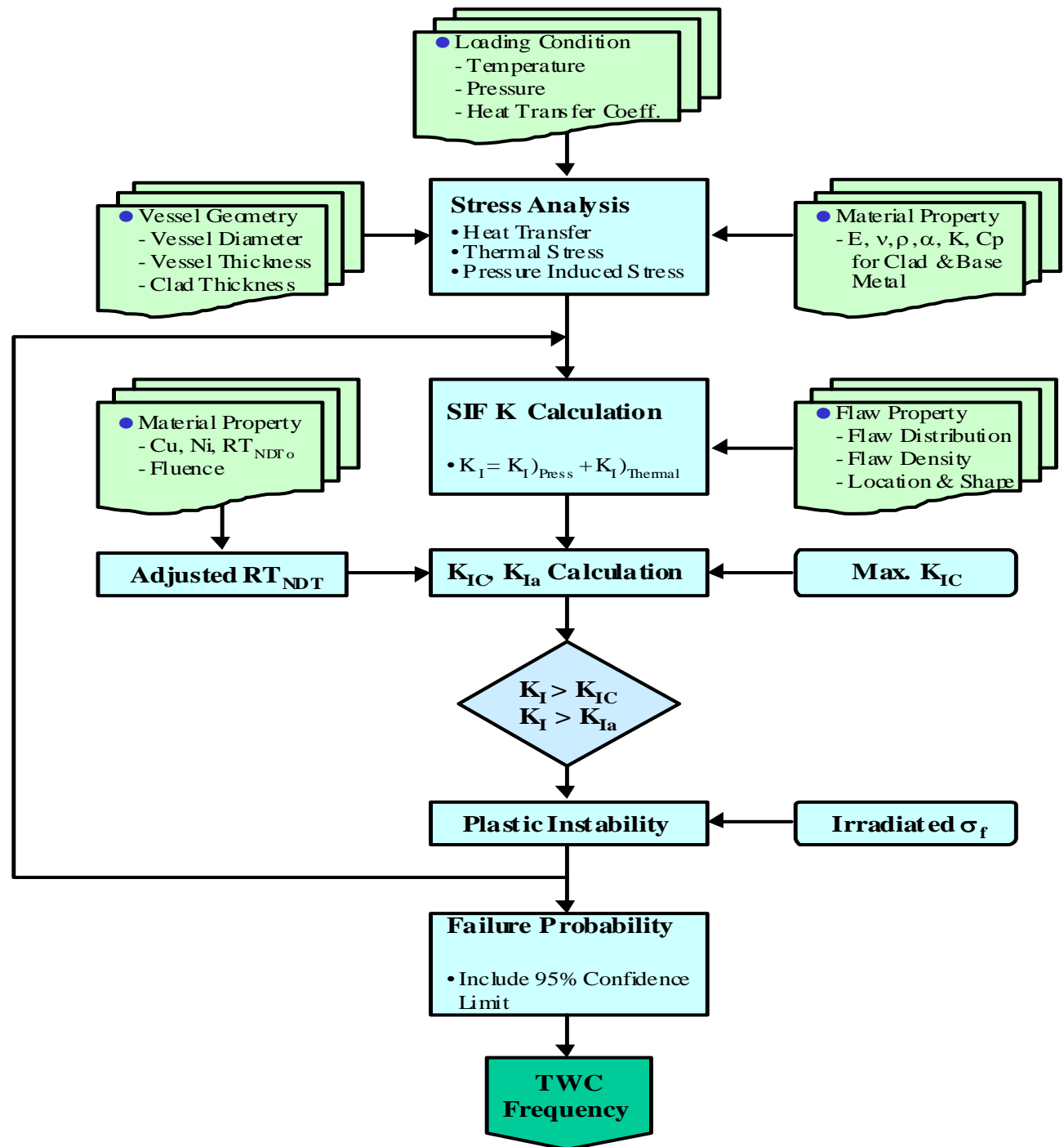
S2: Plant specific PTS analysis by R.G. 1.154

- ✓ **Initiating Events and Sequences Quantification**
- ✓ **Thermal Hydraulic Analyses of System Transients**
- ✓ **Mixing Analysis**
- ✓ **Probabilistic Fracture Mechanics Analysis**
- ✓ **Integrated PTS Risk and Sensitivity**

Flow chart of Plant specific PTS evaluation



PFM: Probabilistic fracture mechanics analysis



<Computer Codes>

RETRAN-3D

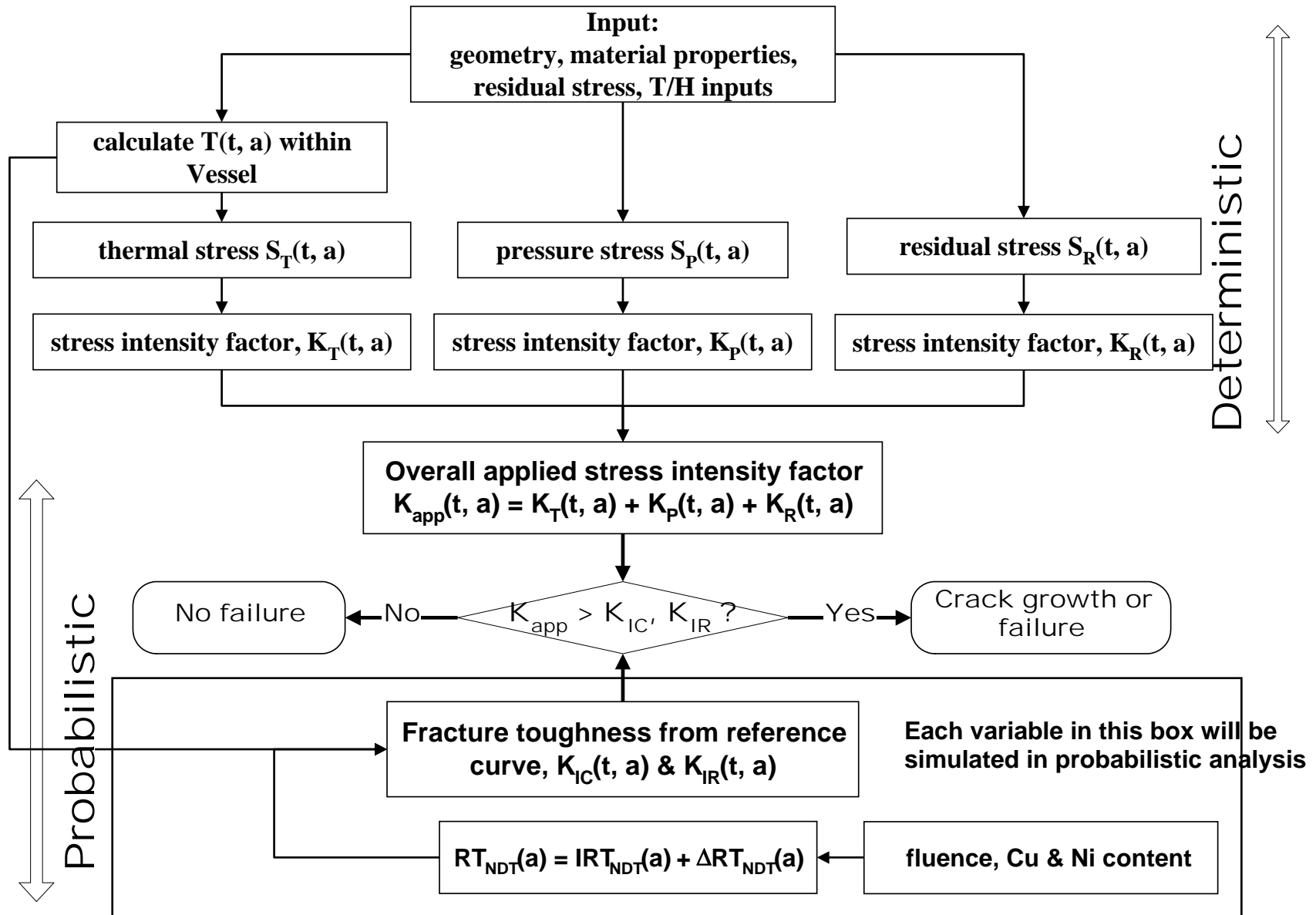
RELAP5/Mod3.2

VISA-II

Favor

...

Probabilistic fracture mechanics analysis



Sequence Quantification and Grouping

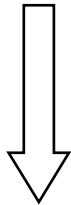
Initiating event	Power	Group name	Grouping Criteria	Representative Sequence	Sequence	Frequency
Small MSLB	Full Power	A1	All actuation & oprtor act'n succ.	A001	A001	8.089E-04
		A2	Auxiliary feedwater overfeed	A014	A002, A004, A005, A007A008, A010, A011, 013, A014, A015	3.070E-04
		A3	<2 stm dmp w fail to close	Ab25 ⁽¹⁾	A016~A024, A027~A031	1.389E-05
		A4	>2 stm dmp w fail to close	C1, C3, C4	A033, A035, A036, A038A039	2.698E-07
		A5	Frequency < E-10/yr	Residual	A003, A006, A009, A012, A025, A026, A032, A034, A037, A040, A041	3.517E-09
	Hot Zero Power	B1	All actuation & oprtor act'n succ	B001	B001	2.718E-04
		B2	Auxiliary feedwater & charging flow overfeed	B014	B002, B004, B005, B007,B008 B010, B011, B013, B014	1.032E-04
		B3	Residual(Frequency < E-10/yr)	Residual	B003, B006, B009, B012,B015	7.908E-10
Large MSLB	Full Power	C1	All actuation & oprtor act'n succ	C005	C001, C002, C004, C005	7.277E-04
		C2	1 steam generator blowdown	C007	C007	5.895E-04
		C3	1 SG BD & aux. fw overfeed	C011	C008, C010, C011	6.919E-05
		C4	1 SG BD & chg flow overfeed	C014	C013, C014	7.319E-05
		C5	2 steam generators blowdown	Cb17 ⁽²⁾	C016, C017, C018	4.380E-07
		C6	Frequency < E-10/yr	Residual	C003, C006, C009, C012, C015, C019	1.813E-09
	Hot Zero Power	D1	All actuation & oprtor act'n succ	D005	D001, D002, D004,D005	2.243E-05
		D2	1 steam generator blowdown	D006	D006	1.817E-05
		D3	1 SG BD & aux. fw overfeed	D012	D007, D009~D012	4.388E-06
		D4	2 steam generators blowdown	D013	D013	1.350E-08
D5		Frequency < E-10/yr	Residual	D003, D008, D014	4.856E-11	
Total			29	24	134	

Calculated through-wall-cracking frequency of Kori-1 RPV

Integrated PTS Risk

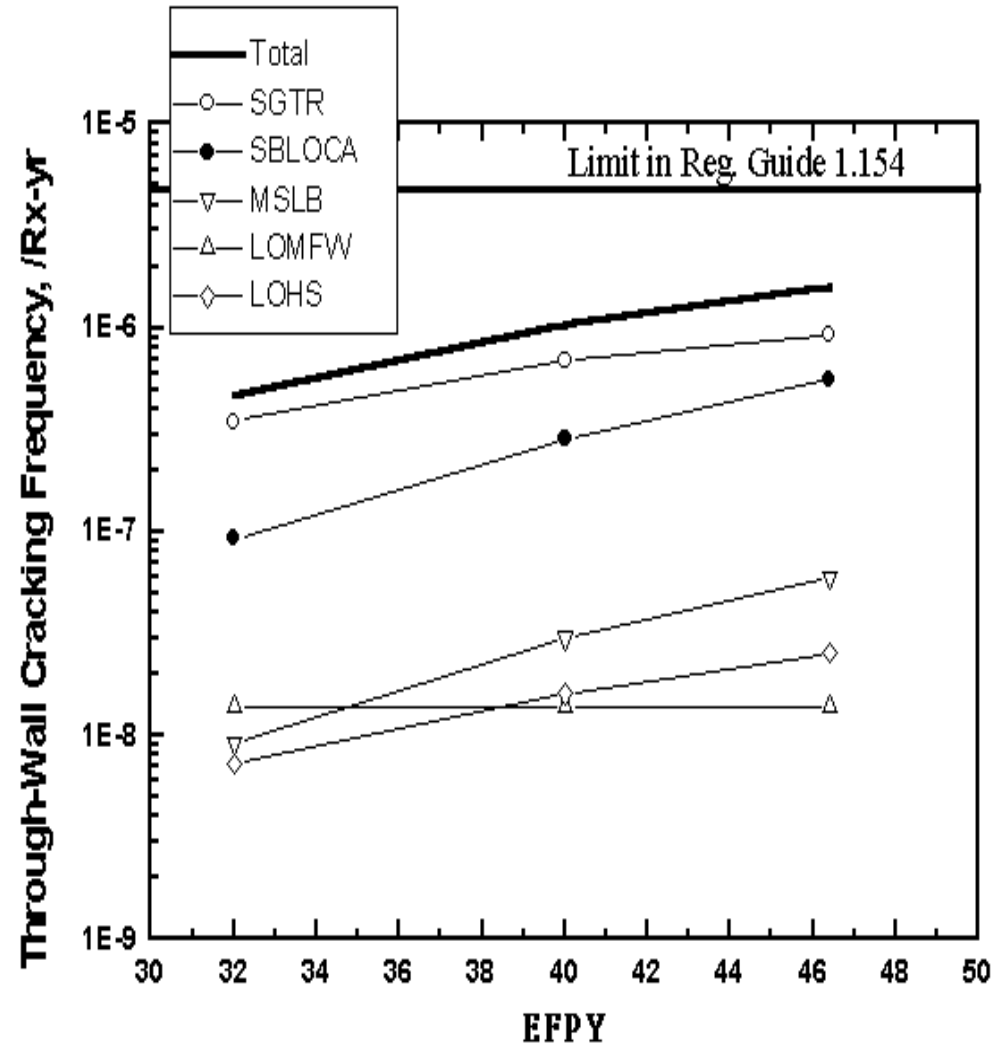
Total TWC, $\sum(f(i) \times P(F/E)i)$

Less than $5.0 \times 10^{-6} / \text{Rx-yr}$



$4.7 \times 10^{-7} / \text{Rx-yr}$

At 32EFPY



SGTR & SBLOCA Represent >90% of Total Risk

Sensitivity study on PFM analysis

To see

- Priorities of the inputs
- What plant conditions modified to reduce the probability of through wall crack (TWC)
- Change in TWC when a parameter changes $\pm\sigma$

Parameter	Sensitivity (TWC_I / TWC_o)
Flaw density	74.37
Downcomer fluid temperature	13.15
Event frequency of SGTR at HZP	3.06
Copper contents	2.29
Initial RT_{NDT}	2.25

Flaw Density is the Most Sensitive Parameter

PFM Analysis (2)

- Calculate Conditional TWC Probability, $P(F/E)$ during Overcooling PTS Transient
- Use VISA-II and FAVOR Codes
 - More conservative VISA-II than FAVOR
 - VISA-II for licensing and FAVOR best estimation
- PFM Analysis
 - Failure risk of RPV caused by the PTS event with Monte Carlo technique
 - Low USE input by lowering the max. fracture toughness of the weld material
 - Marshal flaw distribution assumed
 - Consider analysis conservatism

Careful selection of analysis codes is important.

(T/H code, Mixing code, PFM code)

Current screening criteria might be over-conservative.

NUREG-1806,
10CFR50.61

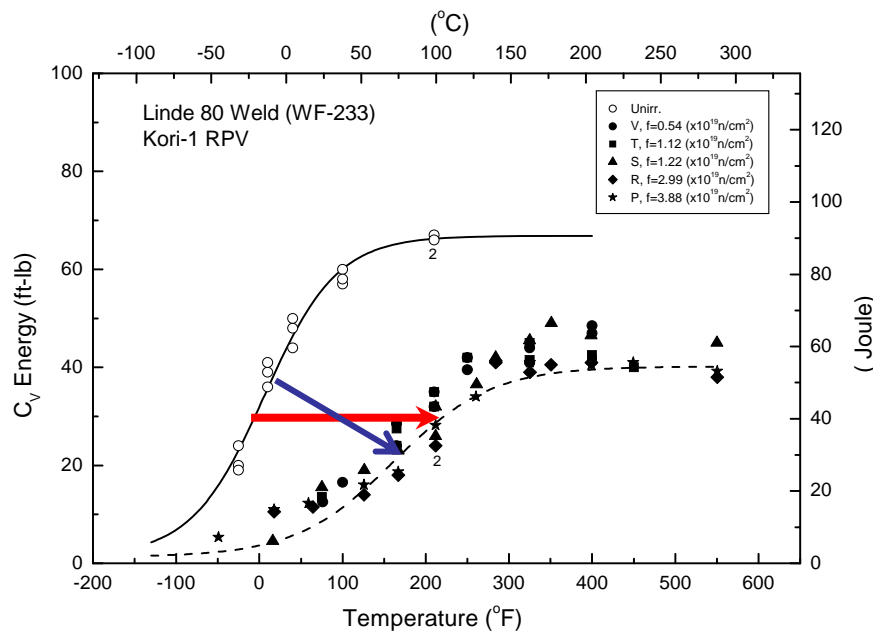
S3: RT_{PTS} re-evaluation by Master curve tests

◆ initial RT_{NDT} :

Conservative value from NDTT and Charpy $T_{50ft-lb} -60^{\circ}F$

◆ Irradiated RT_{NDT} : $RT_{NDT} = initialT_{NDT} + \Delta RT_{NDT} + Margin$

ΔRT_{NDT} : Increase of 30ft-lb temperature by irradiation



RT_{NDT} is determined too high by Charpy v-notch tests for low toughness materials.

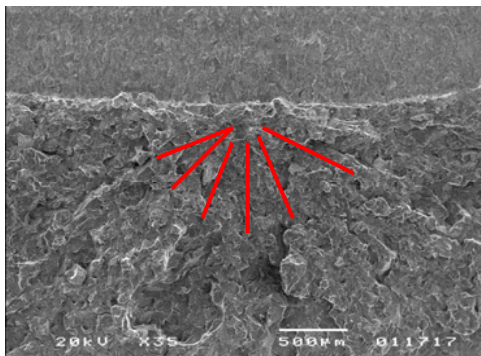


Direct evaluation of fracture toughness based on the Master curve method by using Weld Reconstitution

Master Curve Test Method (ASTM E1921) (1)

Standard Test Method for Determination of Reference Temperature, T_0 , for Ferritic Steels in the Transition Range

- ◆ Conventional concept: Fracture toughness decreases with specimen size and the standard requires relatively large specimens for validity.
- ◆ Master Curve Concept: Dependence of cleavage fracture toughness on specimen size (thickness) can be quantified by a statistical weakest-link model.



$$K_{JC(IT)} = K_{\min} + \left[K_{JC(X)} - K_{\min} \right] \left(\frac{B_X}{B_{1T}} \right)^{1/m}$$

(Empirically, $K_{\min} = 20 \text{MPa}\sqrt{\text{m}}$ for RPV steels)

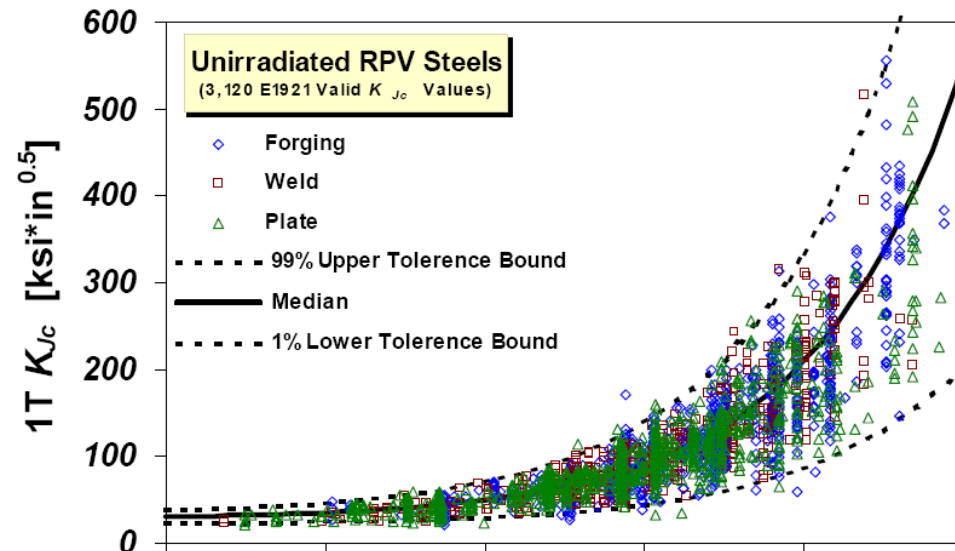
- Cleavage fracture is initiated at the weakest site ahead of crack tip.
- Large specimens have a bigger probability to get the weakest site.

Master Curve Test Method (ASTM E1921) (2)

- K_{IC} Large specimens should be used to get a valid fracture toughness.
- ↓ Specimen size-dependence on the measured (cleavage) fracture toughness can be compensated by a weakest-link theory.
- K_{JC}

- ✓ Cleavage fracture toughness data from the same (1 inch) size specimens show a single master curve shape for ferritic RPV steels.
- ✓ T_0 (reference temperature) : at 100 MPa√m median value

Indexing Fracture Toughness For Many Different RPV Steels



$T - T_0$ (°F)
(NUREG-1807, by Mark Kirk)

Charpy-size specimens (PCVN) can be valid for direct indexing of fracture toughness.

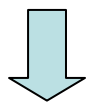
ASME K_{IC} curve : CLB (Current Licensing Basis)

How to get a fracture toughness data ?

ASME K_{IC} Curve:

WRC Bulletin-175 (1972)

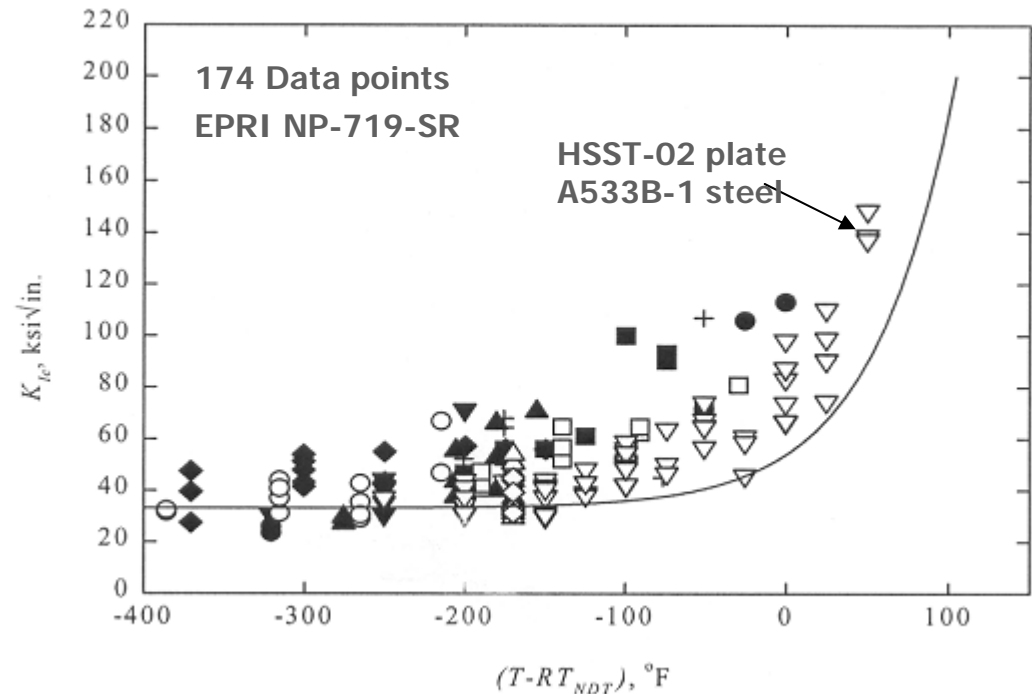
Fracture toughness data were normalized by RT_{NDT} parameter from Charpy tests, conservatively.



assuming

Irradiated K_{IC} curves :

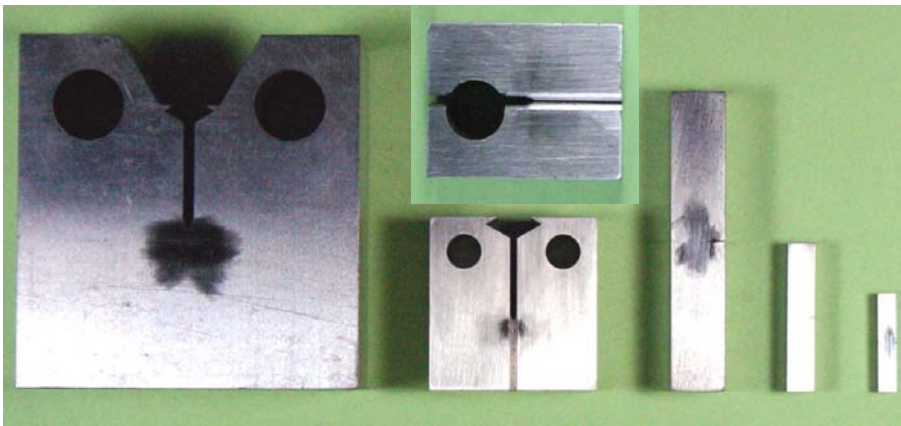
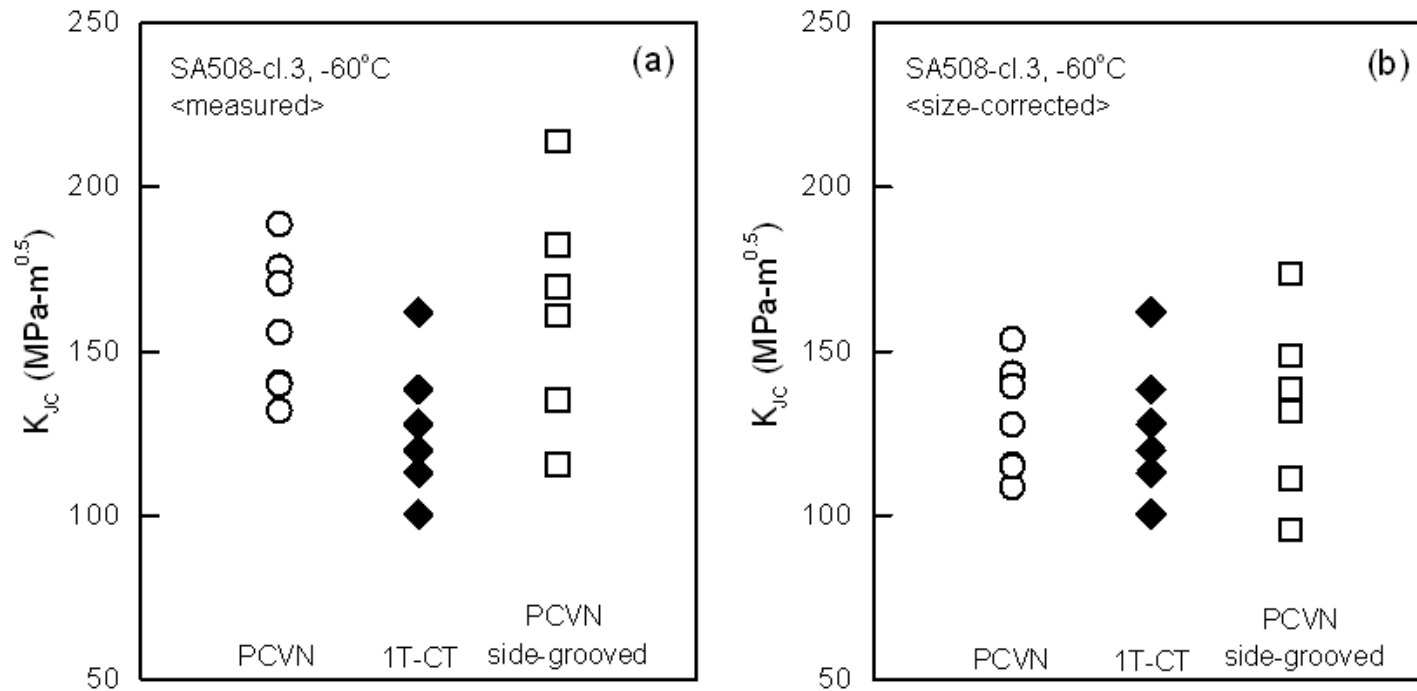
Those are shifted by the same amount of $\Delta T_{30ft-lb}$ from Charpy tests.



ASME K_{IC} Curve (lower bound)
and its original database (1972)

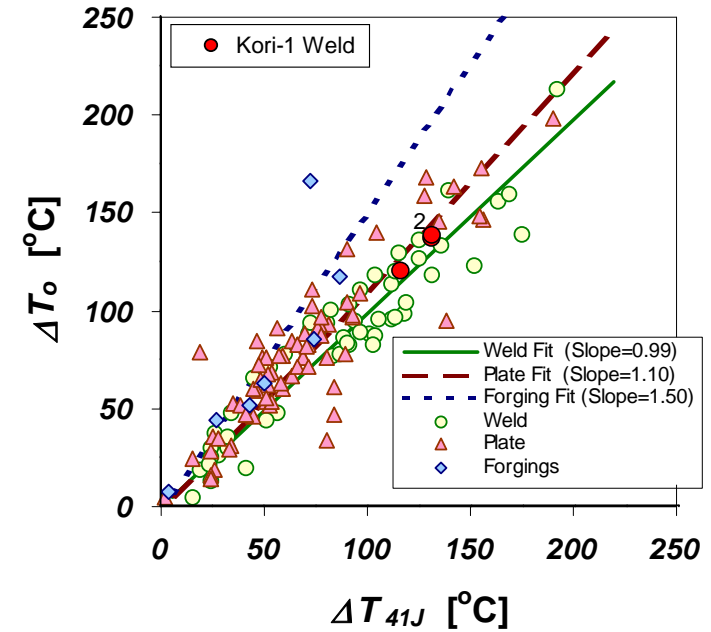
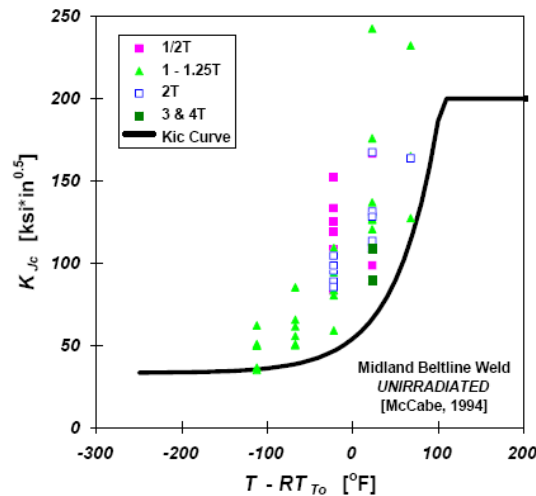
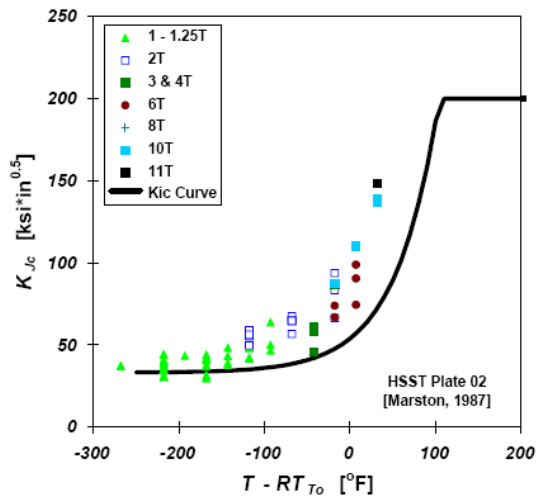
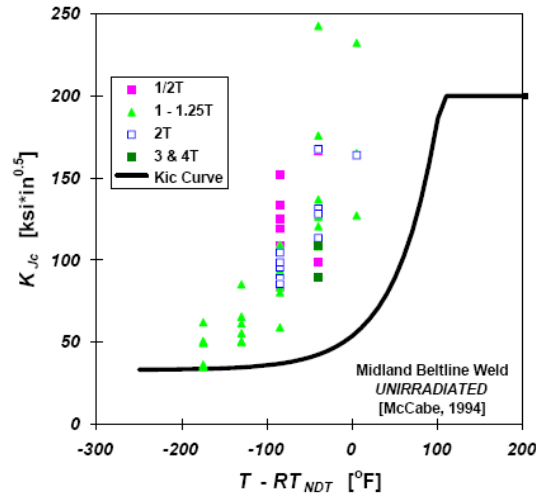
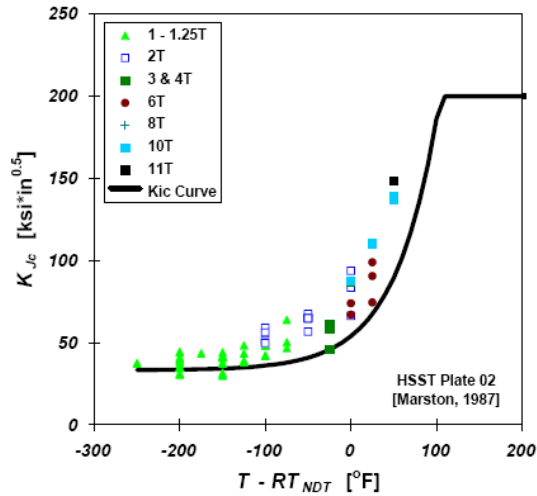
$$K_{IC} = 36.5 + 22.783 * \exp[0.036(T - RT_{NDT})] \quad (\text{in MPa}\sqrt{\text{m}}, \text{ } ^\circ\text{C})$$

Size correction based on the Weibull fracture probability



Master curve testing works properly even for smaller specimens as a half sized Charpy (precracked).

Equivalence to the current RT_{NDT} rules



Depends on the material.

Equivalent for the current limiting material HHST-02.

$$RT_{T_0} = T_0 + 35^\circ\text{F} \quad (\text{from CC N-629})$$

Benefits of Master Curve approach

- 10 Charpy specimens are sufficient to get a standard fracture toughness curve if pre-cracked and tested by the master curve method.
- Why should we be tied down by the indirect Charpy rules for surveillance tests ?
- During the last decade, international collaborative researches have been focused on this subject for enlargement of database (incl. irradiation) and codification of the application.
- Surveillance test standards (ASTM E185, etc.) permit the use of master curve testing.
- ASME Code provides an alternative indexing of RT_{NDT} by $RT_{T_0}(=T_0+35^\circ\text{F})$.
 RT_{T_0} is very useful for the case of lack of the initial RT_{NDT} information.

PCVN (precracked Charpy) specimens from conventional surveillance programs can be used for master curve testing to give highly reliable results.

Application of Master Curve for PLEX (plant life extension)³⁰

- Nuclear power plants constructed in 60~70's, many of which have low upper-shelf high-copper beltline welds, are the focus of the life extension.
- Fracture toughness should be evaluated by using an advanced method.
- Master curve method is the most promising tool for the demonstration of margins.

How can we obtain specimens for the M/C tests if all surveillance tests were already finished ?

=> Weld reconstitution of broken Charpy specimens

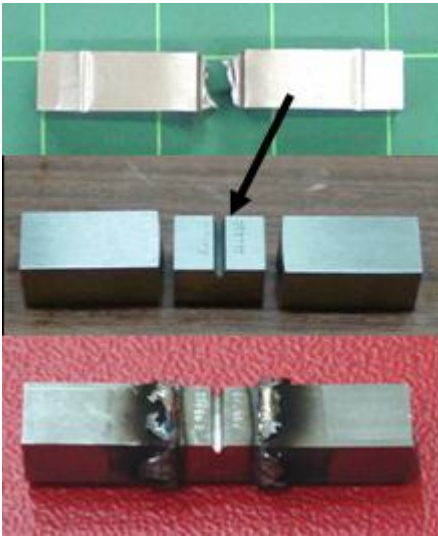
(ASTM E1253, Standard Guide for
Reconstitution of Irradiated Charpy-sized Specimens)

Weld reconstitution of broken Charpy specimens

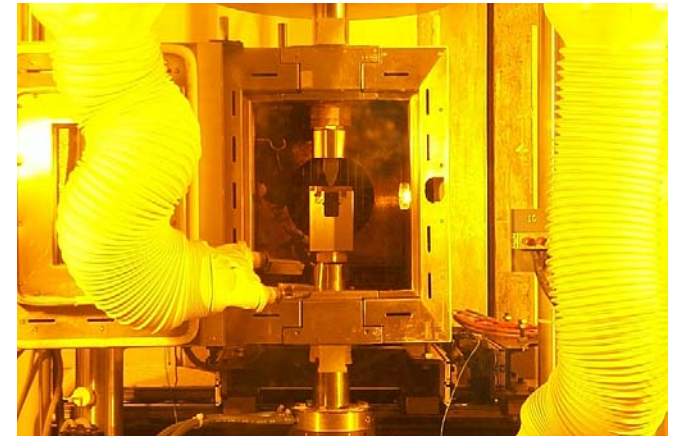
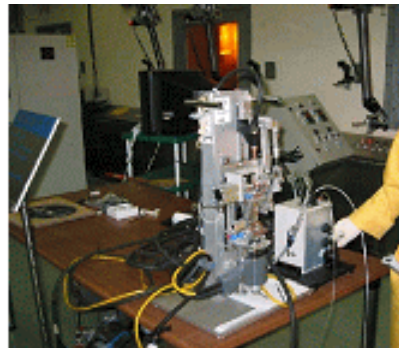
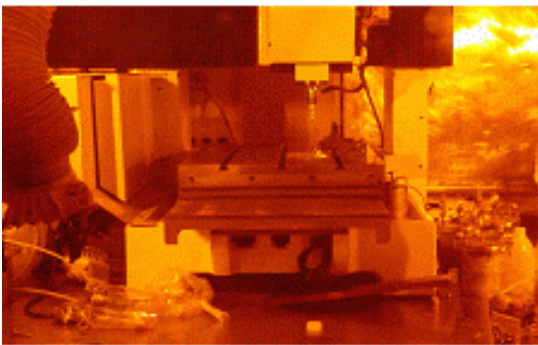
A broken half of Charpy specimens from surveillance programs



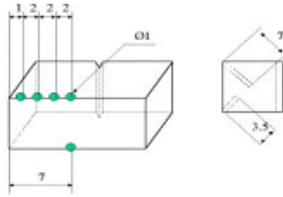
A new PCVN specimen for the master curve testing



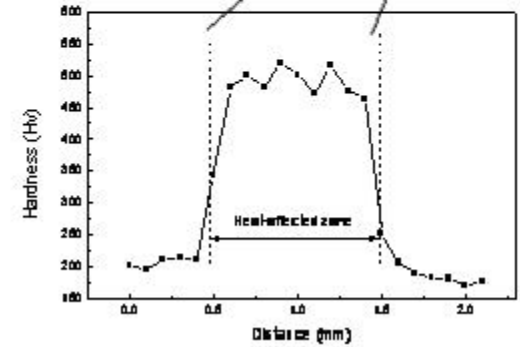
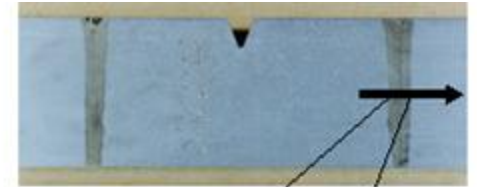
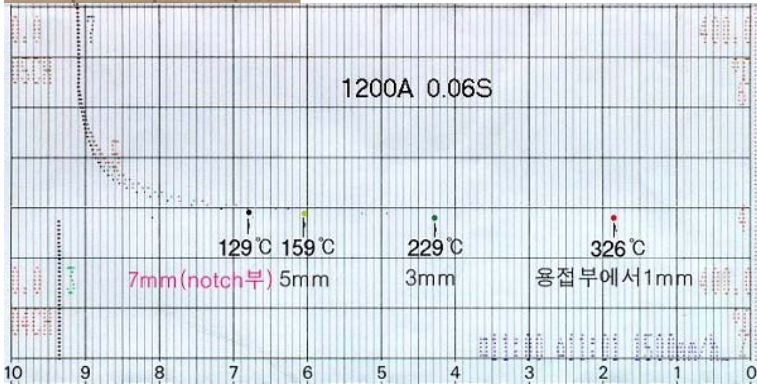
- ✓ Insert machining of a broken piece.
- ✓ Arc stud welding in a short period (~ 0.2 sec)
- ✓ Final machining and pre-cracking



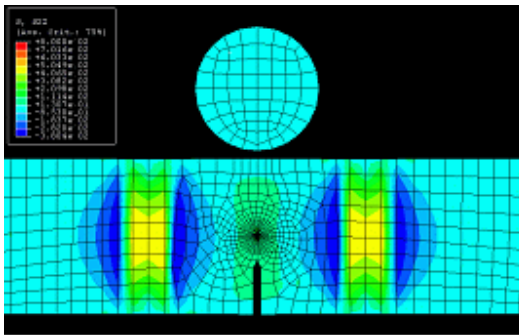
Validation of weld reconstitution techniques



Temperature Measurements

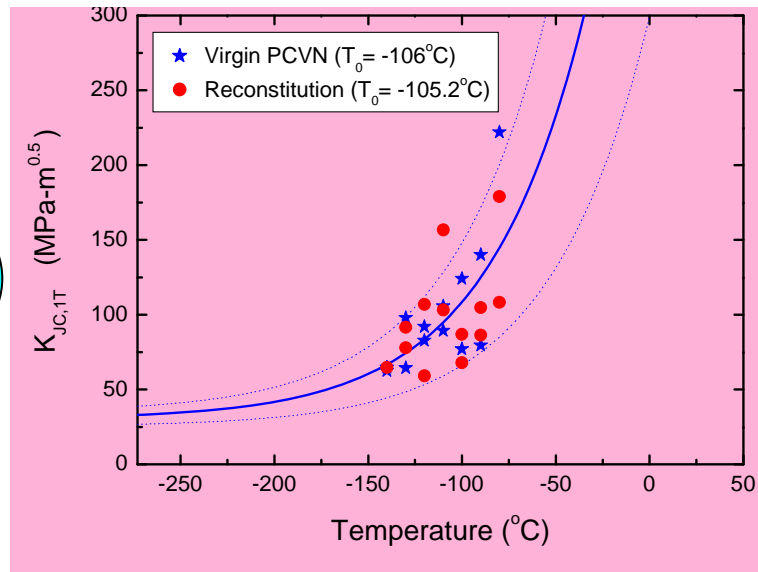


Hardness Profiles



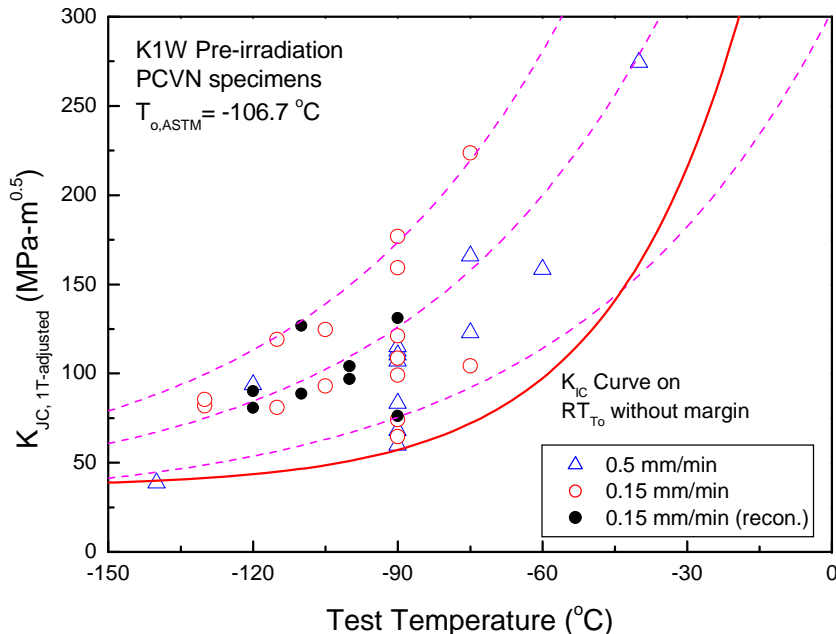
FEM stress calculation

Practically, the same.

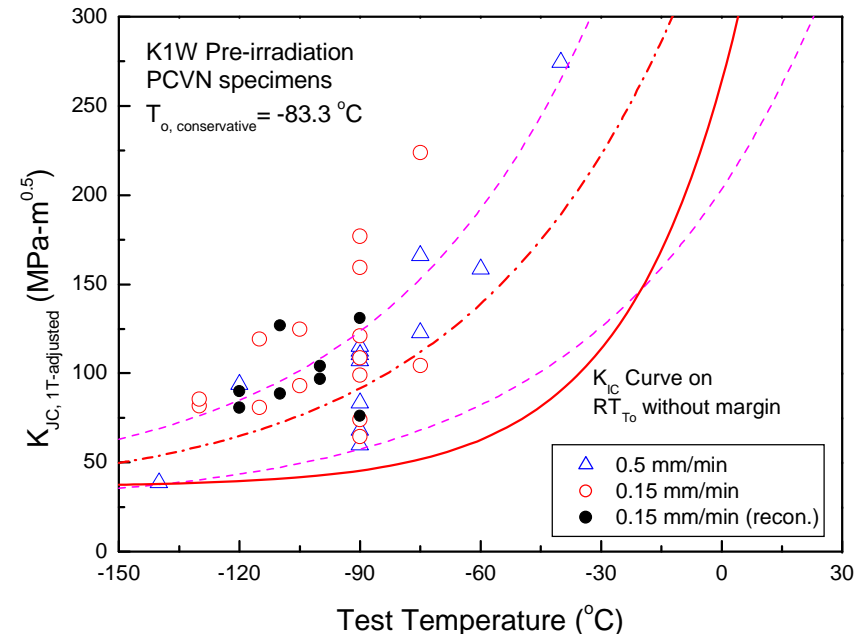


Un-irradiated material data (WF-233)

Standard T_0 value from all data
(E1921-05)



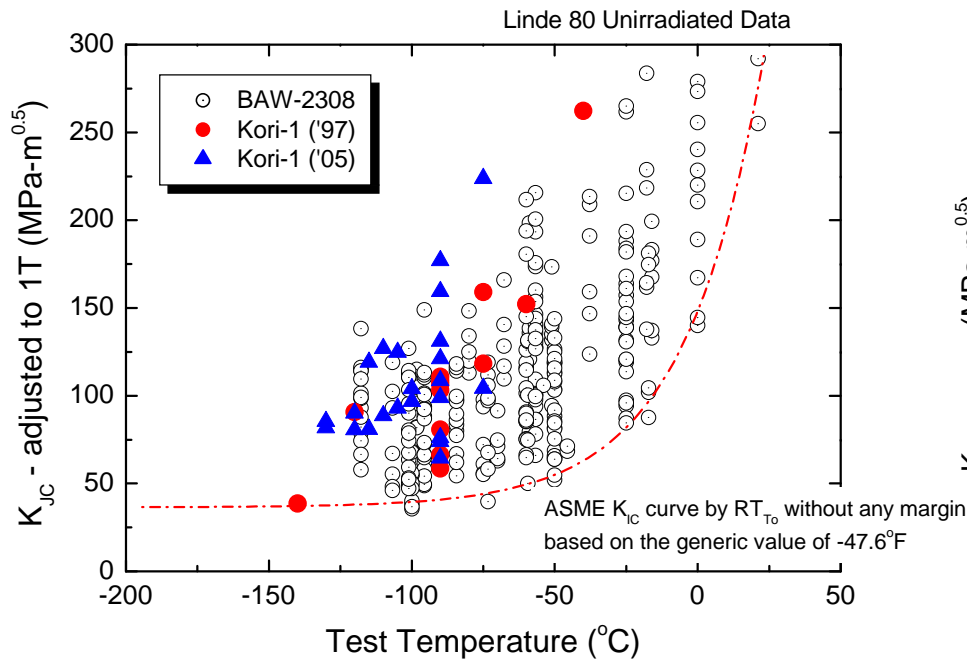
Conservative T_0 value from a data set
at a single temperature (E1921-97)



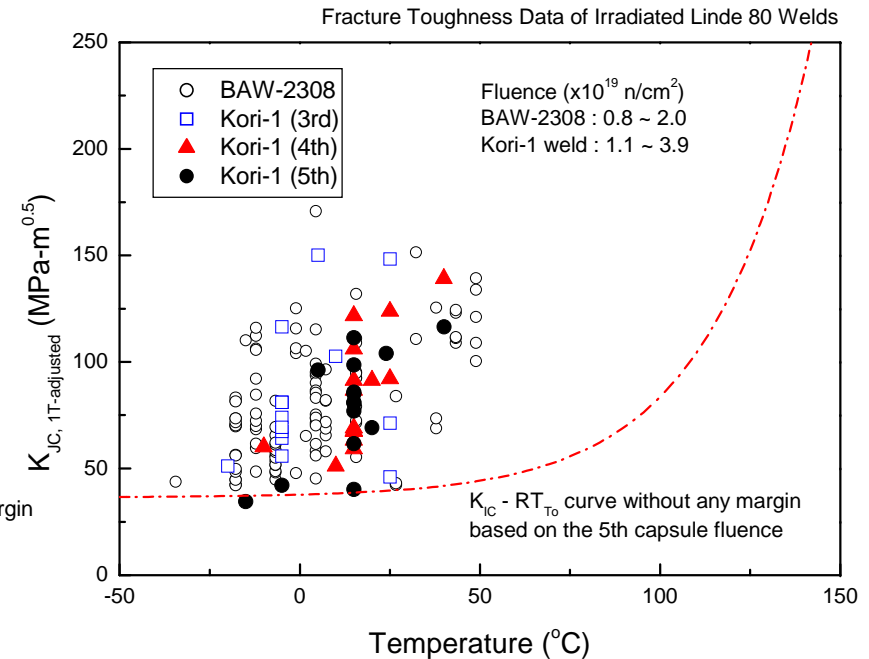
- Best-estimation T_0 (-106.7°C) value (E1921-05) envelopes the all data by the ASME code case N-629, RT_{T_0} curve.
- A conservative T_0 (-83.3°C) value was calculated from a [minimum property data set at a single temperature](#) for the initial RT_{T_0} value determination .
(Implicit margin of about 23.4°C was added to Kori-1 safety analysis.)

Comparison with US data for Linde 80 welds

Un-irradiated



Irradiated



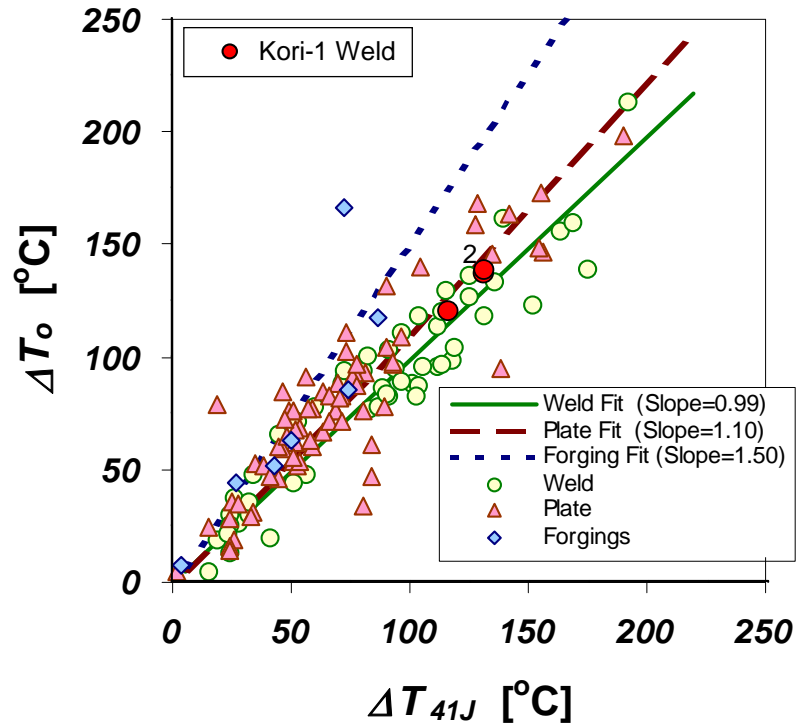
$$(T_{o,ASTM} = 13.4, 30.0, 31.5 \text{ } ^{\circ}\text{C})$$

Kori-1 RPV weld showed a better property than the generic Linde 80 data.

- $2\sigma = 65.8^{\circ}\text{F}$
- PCVN bias = 18°F
- Explicit margin terms will be added for integrity evaluation.

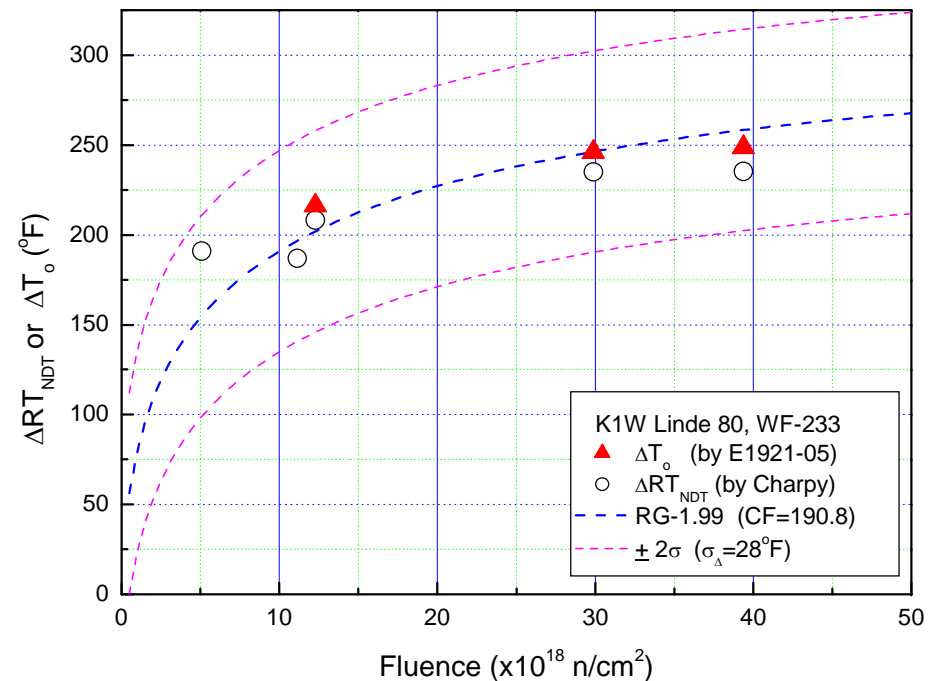
Transition temperature shifts after irradiation

(NUREG-1807 database)



<RPV steels & welds>

(USNRC RG-1.99, Rev.2)



CF = 190.8 $^{\circ}\text{F}$ (from Charpy, $\Delta T_{30\text{ft-lb}}$)

CF = 191.6 $^{\circ}\text{F}$ (Master Curve, ΔT_0)

Similar to the Charpy data for Kori-1 weld (Linde 80 / WF-233)

RT_{PTS} re-evaluation (from conservative initial data)

BAW-2308 approach

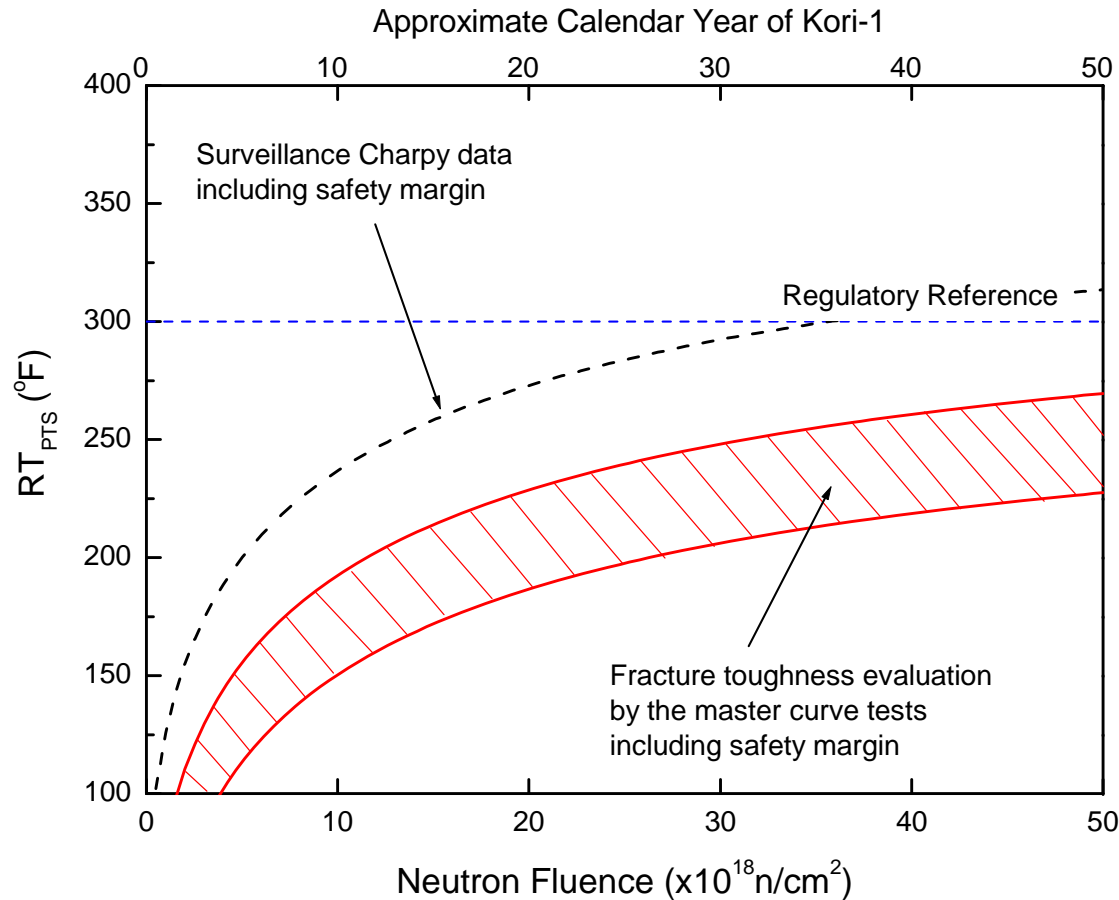
	Master curve	Charpy based
Un-irradiated T _{o,ASTM} (°C)	-106.7	-
Initial (conservative) T _o (°C)	-83.3	-
PCVN Bias (°C)	10	-
RT _{T_o} correction (°C)	19.4	-
Initial RT _{T_o} (°C)	-53.9	-
Initial RT _{T_o} (°F)	-65	-10 (RT _{NDT})
CF (°F)	191.6	190.8
Margin (°F)	65.8	56
Fluence at EOLE (40 yr)	3.94	3.94
RT _{PTS} (°F) at 40 yr	260.0	304

RT_{PTS} re-evaluation (from irradiated data)

	Kori-1 surveillance capsule		
	S	R	P
$T_{o,unirradiated}$ (PCVN)	-160 °F		
$T_{o,surveillance}$	54.1	81.9	88.9
$\Delta T_{o,surveillance}$	214.2	241.9	248.9
Fluence ($\times 10^{19}n/cm^2$)	1.222	2.989	3.883
Vessel Fluence at EOLE	3.938 (40 yr)		
$\Delta T_{o,vessel}$ (°F)	279.0	258.0	253.7
$T_{o,vessel}$	119.0	97.9	93.6
RT_{To} ($= T_o + \alpha$)	35 °F		
Margin	65.8 °F		
PCVN Bias factor	18 °F		
$ART_{To,vessel}$	237.8	216.7	212.4
RT _{PTS} (averaged)	222.3 °F		

Kewaunee
approach

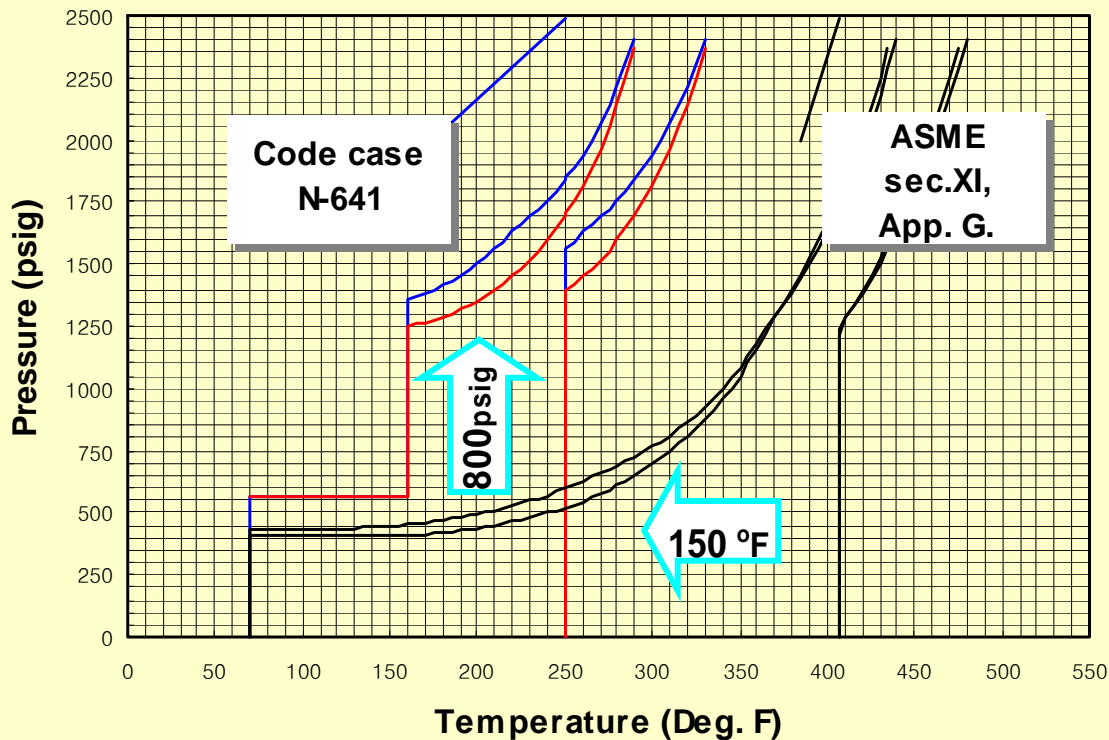
RT_{PTS} evolution curve revised for Kori-1 weld



Master curve fracture toughness data (36 un-irradiated & 45 irradiated) showed a sufficient RT_{PTS} margin for another 10 years and more operation.

S4: P-T (pressure-temperature) operation limit curve

- ASME Code Section XI, App.G assumes an axial surface crack of 1/4t depth.
 - The operating window was too narrow for Kori-1, which has no axial weld.
 - ASME Code Case N-641 permits a postulate flaw with a circumferential orientation, if there are only circumferential welds.
- => This could reduce the stress intensity and the P-T curve constraint.



Operating window was increased by the code case N-641 application.



Master curve data is not applied to the P-T curve generation for Kori-1.

Code & Rules for Life Extension (Continued Operation)

PSR (Periodic Safety Review)

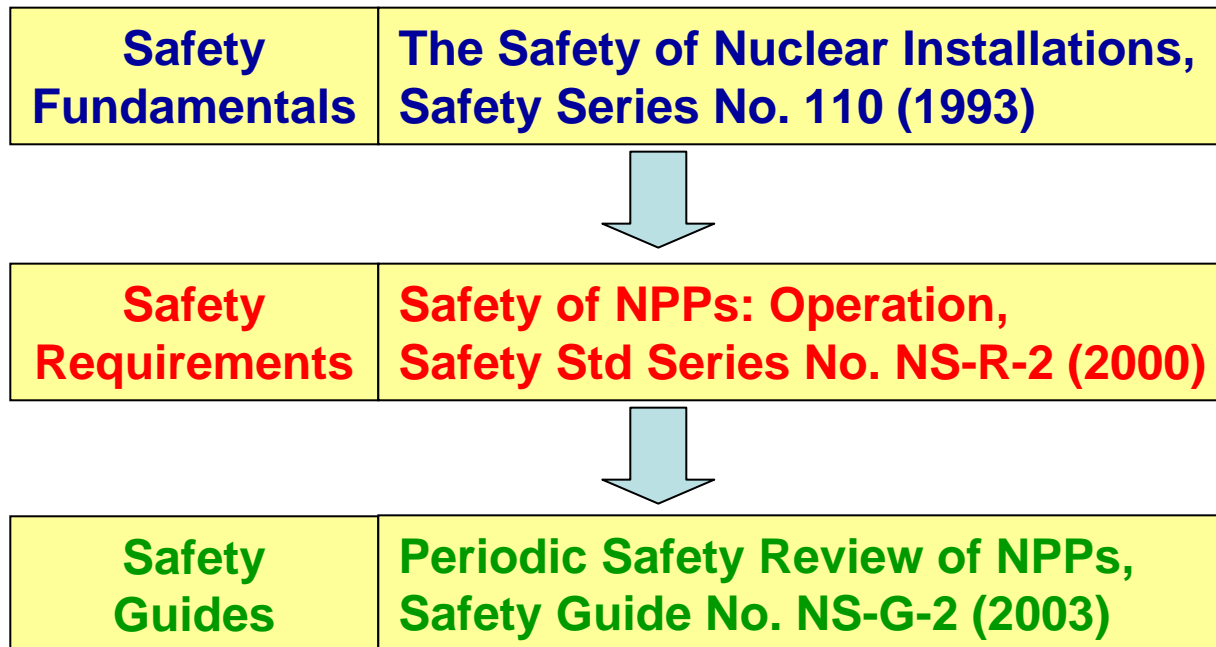
**PLiM (Plant Lifetime Management)
or AMP (Aging Management Program)**

**LTO (Long Term Operation)
or CO (Continued Operation)
or LR (License Renewal)**

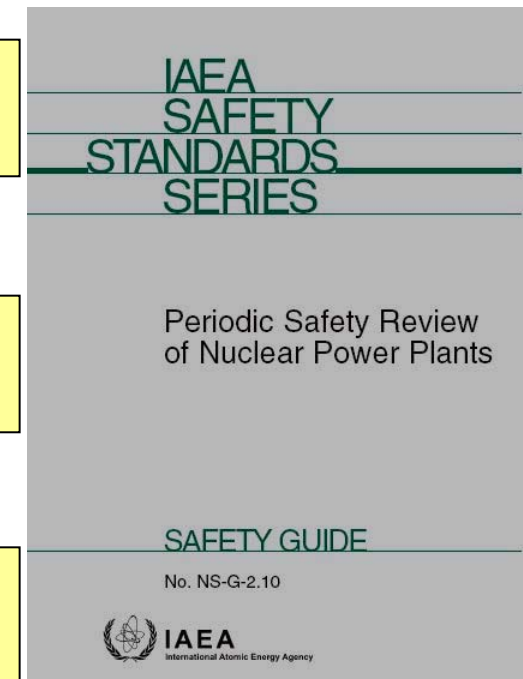
<Presentation on Day-3>

Regulatory Requirement of PSR

- **IAEA Convention on Nuclear Safety(1996.10) recommends Member States to adopt PSR**

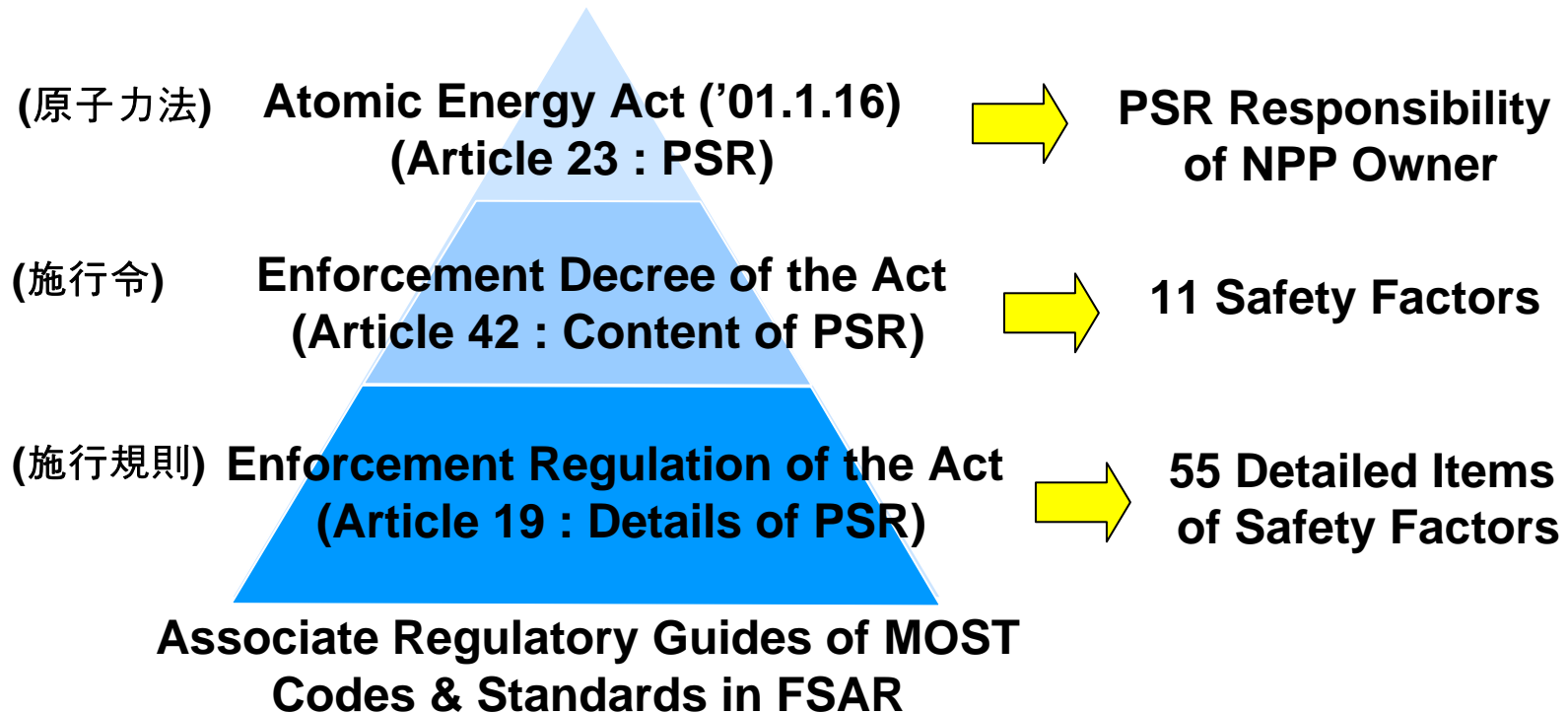


PSR of Operational NPPs, Safety Guide No. 50-SG-O12 (1994)



Legal Basis of PSR in Korea

- **Governmental policy on PSR issued(1999.12)**



PSR of Operational NPPs, IAEA Safety Guide No. 50-SG-O12 (1994)

History of PSR on Kori Unit 1

- '99.12: Governmental policy on PSR issued
- '00.5 ~ '02.11(30 months): PSR on Kori Unit 1 done
 - 706.8 MM involved /Budget of \$ 3million
- '03.11: Regulator(MOST) finished the review of PSR report
- '05.6~'06.6: Extended PSR done for continued operation
- '07.7: IAEA SALTO peer review done
- '07.12: Continued operation to be decided

SALTO: Safety Aspects of Long Term Operation

Result of PSR on Kori Unit 1

- Safety factor (aging management)
 - ◆ Determine whether aging is being effectively managed and an effective AMP is in place
 - ◆ Safety-related SSC maintained in good condition
 - ◆ Some AMP should be upgraded
- Some aged SSC's replaced before and during PSR
 - ◆ LP turbine rotor('97.6), Steam generators ('98.9)
 - ◆ Main transformer('01.10), Generator/exciter('05.5)
 - ◆ ATWS mitigation system installed('01.12)
- Total 40 safety improvements recommended
 - ◆ Installation of fatigue monitoring system
 - ◆ Preparation of alloy 600 management program
 - ◆ Performance of equipment qualification, etc
- Conclusion of PSR on Kori Unit 1
 - ◆ Maintained good physical and operational condition
 - ◆ Sufficient safety margin ensured for the safe operation

PSR on Kori Unit 1 for CO/LTO



- '05.6~'06.6: Extended PSR done for continued operation beyond design life (2007.6.9)
 - ◆ PSR report
 - ◆ Aging management program including TLAA
 - ◆ Statement of radiological impact on environments
- '07.7.23~8.3: IAEA SALTO peer review done

Kori-1 received the CO permission and public agreement with residents in Dec. 2007 and re-started a 10 year operation in Jan. 2008.



Further consideration of Kori-1 LTO (I)

◆ **Fatigue (thermal/environmental) was another TLAA issue.**

- ⇒ Very high CUF when considering PWR environmental effects.
- ⇒ Guidelines for environmental fatigue evaluation are on-going issue in USNRC, ASME, worldwide.
- ⇒ For Kori-1, 10 year life extension (from 30 to 40 years) was given at this time.
- ⇒ For another 10 or 20 year operation (60 years), environmental fatigue issue should be resolved as well as irradiation embrittlement.

Further consideration of Kori-1 LTO (II)

- ◆ **USNRC published an alternative guide for integrity assessment of irradiation embrittlement for radiation sensitive RPV.**
 - ⇒ Risk-informed integrity assessment with a revised embrittlement model
 - ⇒ Based on the probabilistic fracture mechanics evaluation
 - ⇒ NUREG-1874: Recommended Screening Limits for Pressurized Thermal Shock (PTS)
 - ⇒ Proposed Rules (10CFR50.61a) : Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, Federal Register, Vol.72, No.191, Oct. 3, 2007

Summary (I)

- ◆ An application for a 10 year extension of the operating license for the first PWR in Korea, Kori-1, has been submitted to the government after a stable operation of its initial licensing period of 30 years.

- ◆ Safety evaluation and aging management actions on irradiation embrittlement of Kori-1 RPV were successfully performed and approved.
 - RT_{PTS} was re-evaluated through the master curve test method using a total of 45 irradiated and 36 un-irradiated PCVN specimens for the limiting material, the beltline circumferential weld, Linde 80 WF-233.
 - Low upper-shelf toughness issues were resolved by J-R testing of the irradiated WOL specimens and using the USNRC RG-1.161 procedure.
 - P-T limit curve was revised by using the Code Case N-641.
 - The plant specific PTS analysis by RG 1.154 showed that the total failure risk is lower than 5×10^{-7} through probabilistic fracture mechanics analysis.

Summary (II)

- ◆ An ex-vessel dosimetry system was installed for a continuous monitoring and improvement of the calculation accuracy of neutron fluence during a continued operation of Kori-1.

- ◆ PSR, PLiM, AMP and CO/LTO are successfully implemented in operating NPPs.
 - PSR is performed every 10 years.
 - An intensive PSR is carried out for LTO (license renewal)
 - Aging management programs are implemented.
 - Some aged components like SG were replaced.
 - PWSCC sensitive parts will be replaced or repaired by weld overlay with more corrosion resistant materials.
 - Plant life management database was developed for the effective maintenance of each NPP.