Thermal shock resistance of irradiated PWR claddings

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ECCS acceptance criteria

- In order to ***avoid significant cladding embrittlement and maintain coolable geometry*** of reactor core,

  maximum cladding temperature (1200ºC) and oxidation (15%) are defined in ECCS acceptance criteria.

Where is boundary for loss of coolable geometry?

Oxidation	Fracture behavior
Slight	Intact
	Partial Cracking
	Circumferential cracking
	Split fracture
	Rod degradation in short length
	Rod degradation in long distance
Severe	

Fuel pellet dispersal

Cladding fragmentation

JAERI's view on criteria for "significant cladding embrittlement" to maintain coolable geometry

Survive/fracture boundary under LOCA conditions can be the criteria with margin.

Because, if cladding survives the quench,

- **significant loss of rod geometry and fuel pellet dispersal should be avoided**
- **and guarantee coolable geometry of reactor core.**

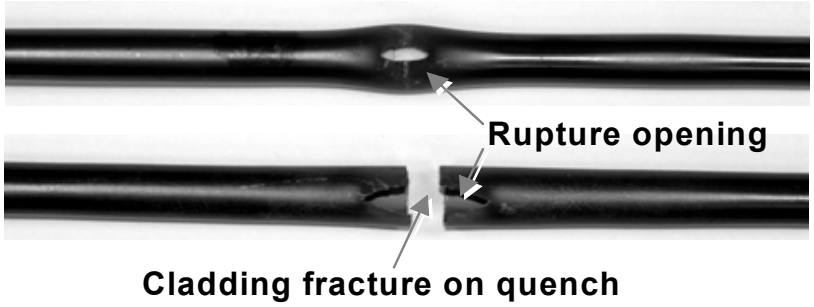
JAERI's experimental method to determine conditions for cladding fracture on quenching

Integral thermal shock test

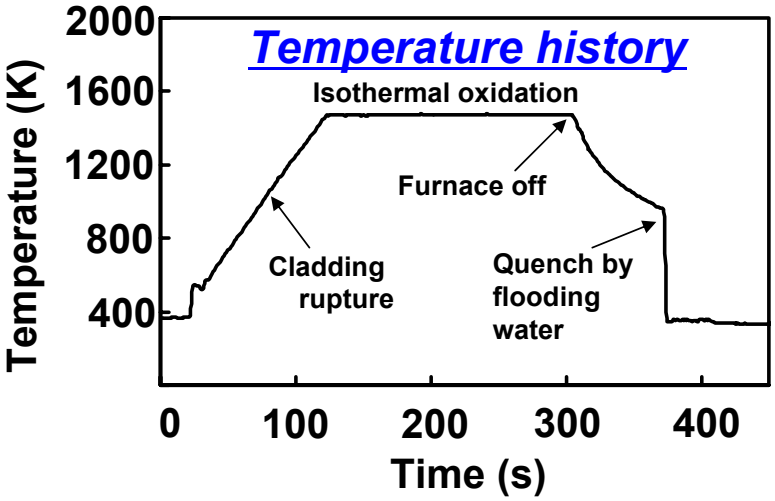
- The tests simulates LOCA conditions and fuel behaviors.

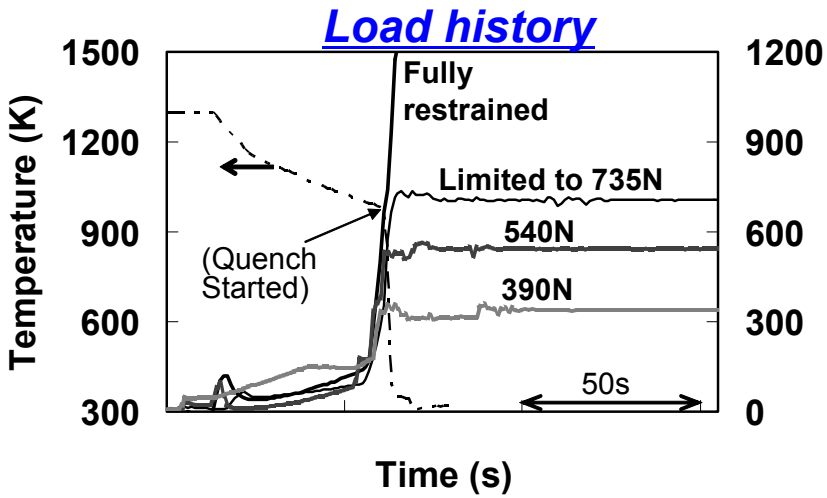
  The cladding balloons and ruptures, and is oxidized in steam and finally quenched by flooding water.

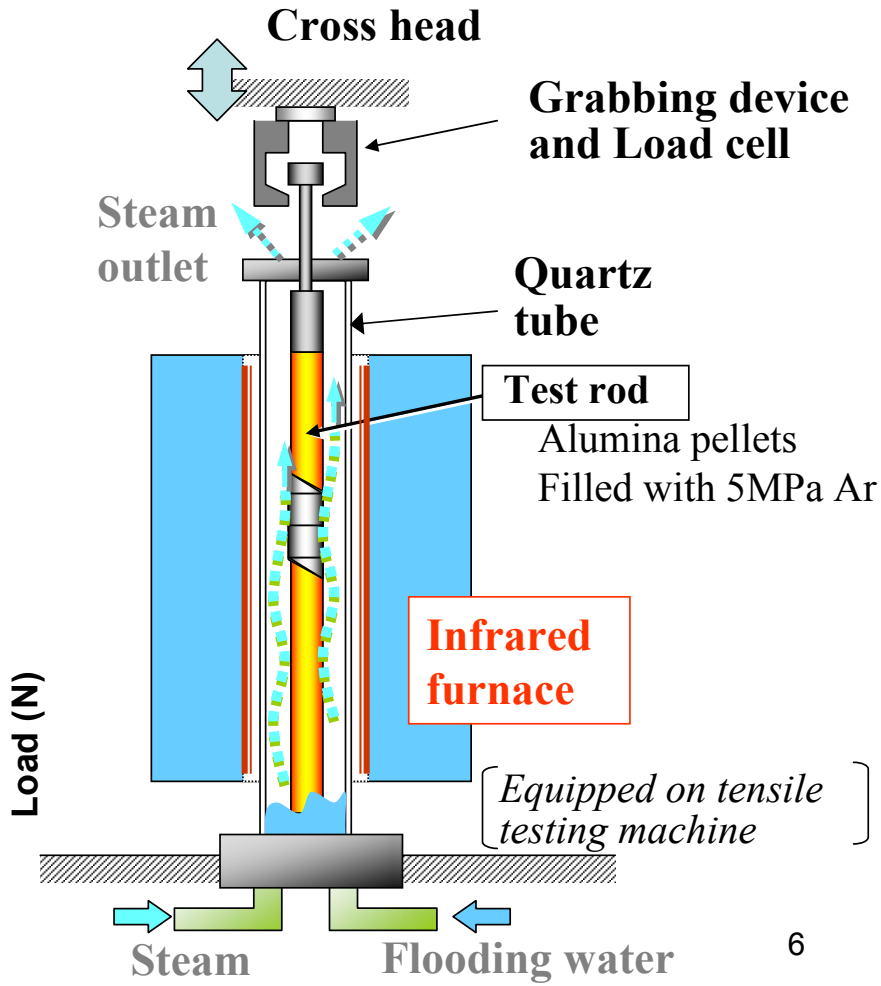
- It is best for determining ***boundary condition for cladding fracture on quenching***.
- It is historically adopted for confirming the safety.
- It is also best for understanding the whole behaviors under LOCA conditions.



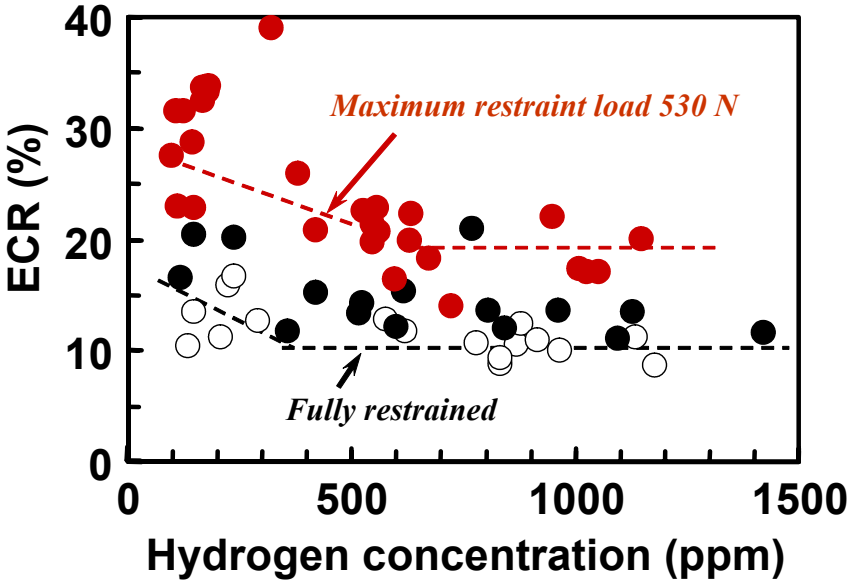
Integral thermal shock test

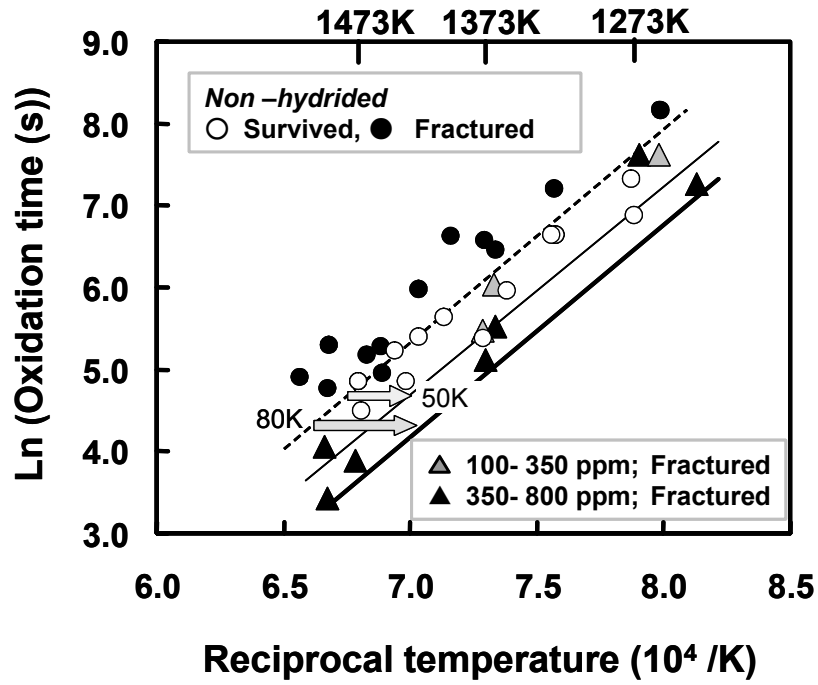






Hydrogen effect on fracture condition
(Results with non-irradiated Zircaloy-4 cladding)

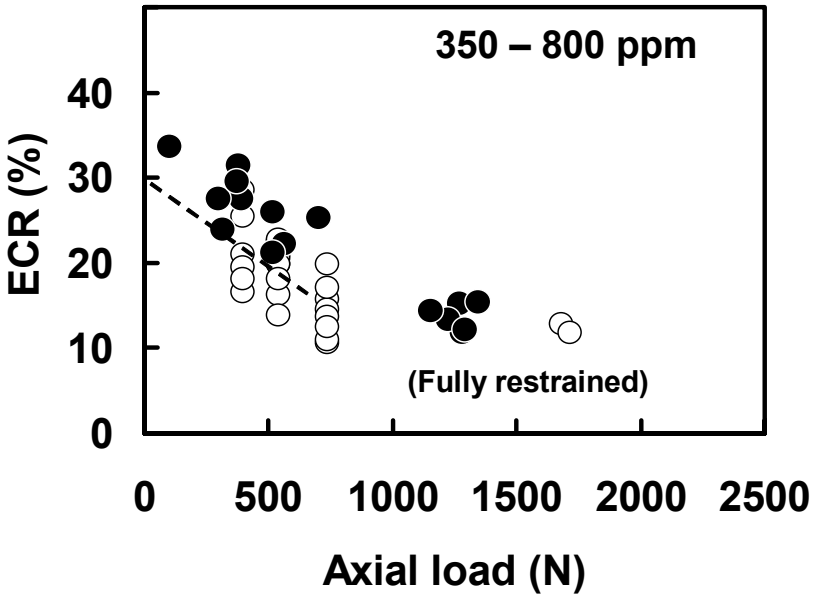


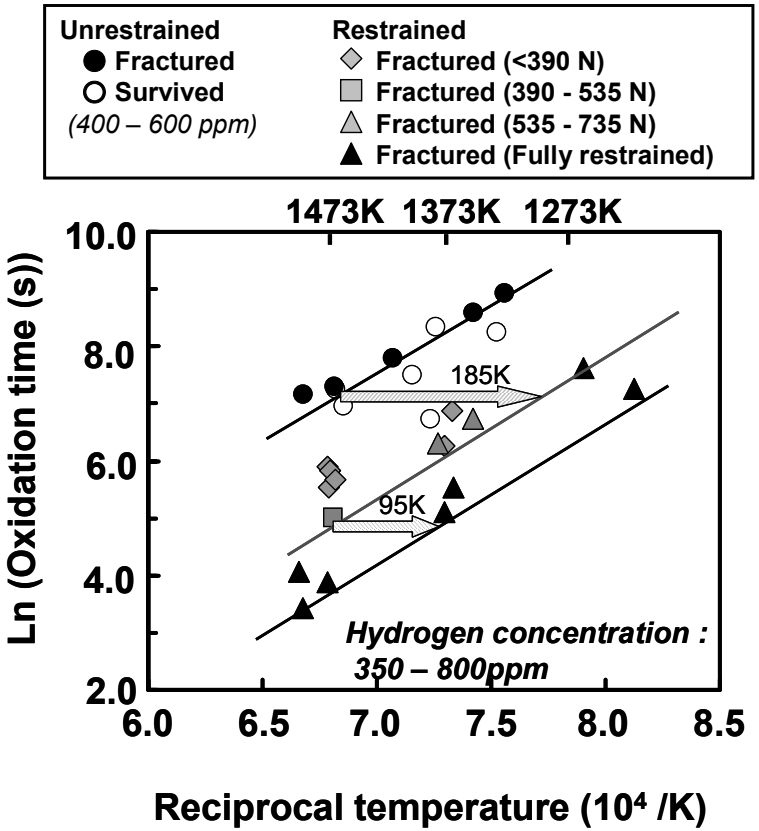


ECR: Equivalent Cladding Reacted (Fraction of cladding thickness oxidized assuming that all absorbed oxygen forms stoichiometric ZrO2) calculated by Baker-Just equation.

(Cooperative research with Japanese PWR utilities)

Influence of restraint load on Fracture conditions
(Results with non-irradiated Zircaloy-4 cladding)





(Cooperative research with Japanese PWR utilities)

Sample and test conditions of integral thermal shock tests with irradiated PWR claddings

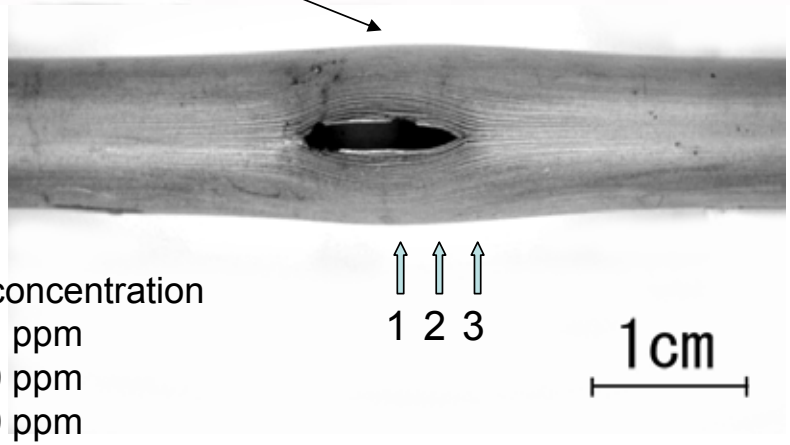
(Cooperative research with Japanese PWR utilities)

Test No.	1	2	3
Sample No.	A 3-1	A1-2	B L-3
Rod Average Burn up (GWd/t)	43.9		39.1
Corrosion Layer Thickness (μ m)	20	25	18
Fuel Cladding	Low tin Zircaloy-4		
Oxidation Temperature (K) [(ºC)]	1453 [1180]	1465 [1192]	1430 [1157]
Oxidation Time (sec)	486	120	200
ECR	30.0	17.9	16.4*
Failed / Survived	Failed	Survived	Survived
Load at Failure (N) [(kgf)]	498 [50.8]	--	--
Maximum restraint load (N)	--	540	540

*:Estimeted values by Baker-Just equation taking account of double sided oxidation and wall thinning by ballooning.

Post-test appearance of irradiated PWR cladding

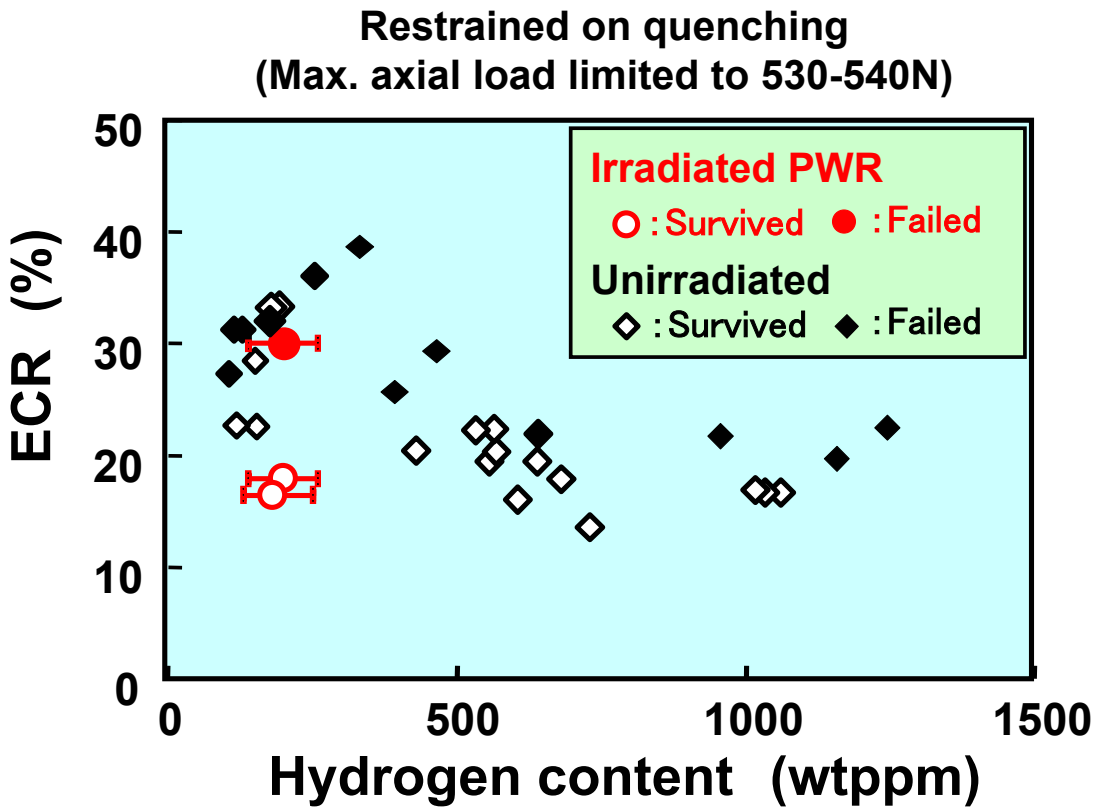
Circumferential increase 27%



Hydrogen concentration
1. 781 ppm
2. 780 ppm
3. 850 ppm

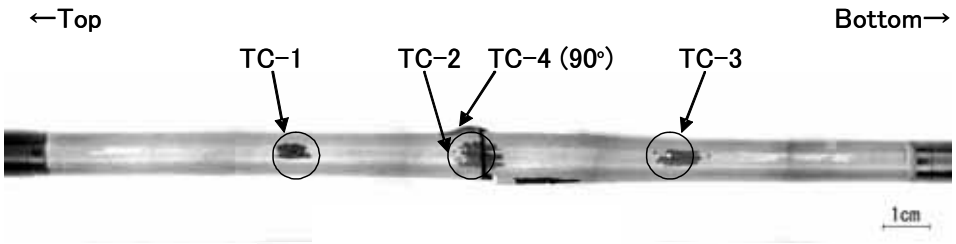
- Typical appearance after ballooning and rupture as observed in un-irradiated claddings
- Circumferential increase ranged 14 to 28 %
- Fine axial cracks in corrosion layer
- Increase in hydrogen concentration even at rupture position

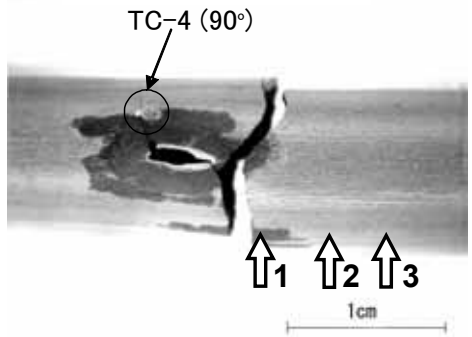
Fracture conditions of irradiated PWR cladding



- Fracture boundary is not remarkably reduced by irradiation for the examined range.
- Fractured condition agrees with results of non-irradiated claddings.

Post-test appearance of irradiated PWR cladding





Hydrogen concentration
1. 1450 ppm
2. 1850 ppm
3. 840 ppm

The cladding fractured near the rupture position with circumferential cracking.
The cladding was embrittled by both oxidation and secondary hydriding.

Conclusion

- **Coolable geometry of a core should be maintained if the fuel rods survive the quench under LOCA conditions.**
- **JAERI performs integral thermal shock tests simulating the whole LOCA sequence to determine survive/fracture boundary condition.**
- **Fracture boundary of Zircaloy-4 fuel claddings, irradiated to 39 and 44GWd/t at a PWR was higher than the limit in the Japanese ECCS acceptance criterion .**