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IASCC of stainless steels in a PWR environment

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1. Introduction

Reactor vessels of a pressurized water reactor (PWR) are designed for a 40 years usage. Under the consideration of a life extension to 60 years or longer, utilities need to set up a degradation management strategy of the reactor internals like the upper core structure, core baffle/former/barrel, thermal shield, and the lower core support structure.[1] It is required to extend the life of PWRs with a knowledge based degradation management related to the mechanical and corrosion properties of stainless steels in a high dose exposure environment. Degradation mechanisms of the reactor internals are an irradiation embrittlement, void swelling, and an irradiation-assisted SCC(IASCC). These mechanisms need to be understood in order to assure the long term integrity of the internals.

This paper aims to review the IASCC experience around the world and in Korea, to introduce a feature of an IASCC of stainless steel mainly used for baffle former bolts in PWR reactor internals.

2. Methods and Results

2.1 Description of Reactor Internals

The reactor internals are mainly composed of an upper core structure, core baffle/former/barrel, thermal shield, lower core support structure as shown in Fig. 1. Austenitic stainless steels are used for the internal structure because they have a relatively high strength and ductility and corrosion resistance. IASCC is the degradation mode which will be studied in this work.

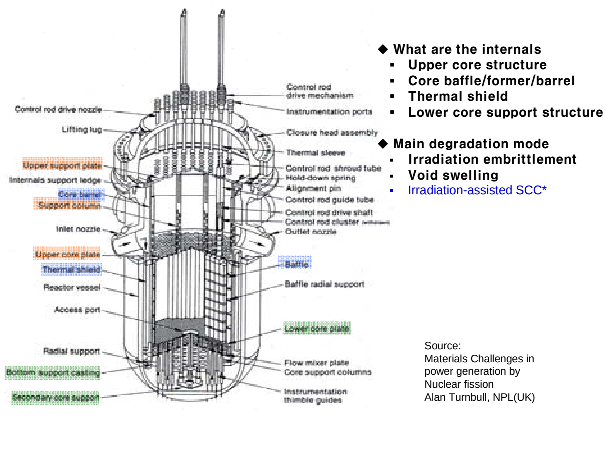

Fig. 1 Schematic of the reactor internals of PWRs

2.2 IASCC

Stress corrosion cracking(SCC) is a phenomenon caused by a combination of a material, a stress and a corrosion environment. Irradiation by neutron is another accelerating factor for a SCC, which is called an IASCC shown in Fig. 2

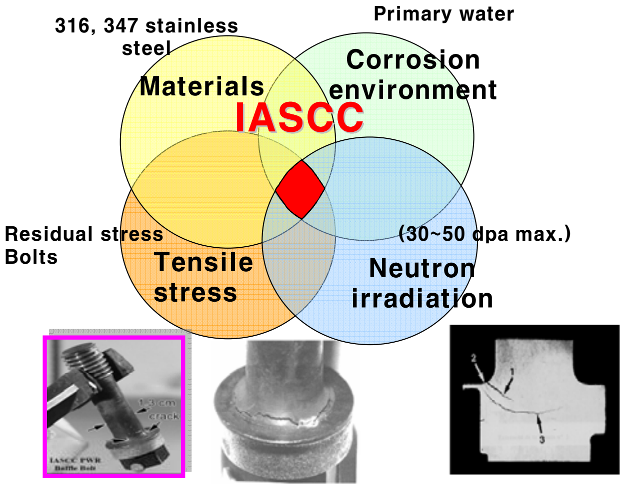

Fig. 2 IASCC under the combination of complex environments

The IASCC does not occur under the environment except one of these four contributors. Though the materials of the primary coolant contact side are very corrosion resistant, cracking of the materials is consistently has been reported.

2.3 IASCC experience in PWR internals

Table 1 shows the cracking experience of the PWR internals. Since a cracking in Bugey-2 was reported first, some plants in USA have shown these defects. The concerns with an irradiation assisted stress corrosion cracking (IASCC) in the Korea power plants started in 1999 with the inspection of Kori-1 baffle former bolts. The baffle former bolts were evaluated to exceed an IASCC threshold (5 x 10^{21} n/cm^2, E>1.0) by calculations on the basis of USA research results. The UT inspection was performed in accordance with ASME Code Section XI. The inspected parts were 728 baffle former bolts and 176 baffle edge bolts. All bolts were made of CW 316 stainless steels. The UT inspections in 1999 indicated 2 defective baffle former bolts and 6 uninterpretable baffle former bolts.

Other inspections were carried out by Korea Plant Service & Engineering (KPS)in 2006[2]. One bolt was uninterpretable, because this bolt was inaccessible to acquire a normal signal. All bolts with indications or uninterpretable results were identified as safe in 2006.

On the other hand, the control rod guide tubes were potentially affected by an exposure over the IASCC threshold. The control rod guide tubes were replaced with CW 316 stainless steels from Inconel X-750 Rev. B in 2007.

Table 1. Cracking experience in PWR internals

Power Plant	Facilities	Operating year	Generating power (MW)	Loop number	BFB number	BFB material	Defect number	Defect position
FRANCE	Bugey-2	20	955	3	960	316SS	87	#2,3 Former
	Bugey-3	20	955	3	960	316SS	18	-
	Bugey-4	-	937	3	960		3	-
	Fessenheim-1	-	950	-	960		28	-
	Fessenheim-2	-	950	-	960	347SS	47	-
USA	Farley-1	22	860	3	1024		No	-
	Point Beach-2	27	510	2	728	347SS	55	#1,3,8 Former
	Ginna	-	600	2	728		59	-
KOREA	Kori-1	21	600	2	728	316SS	2	#1,6 Former
	Kori-2	16	650	2	800	316SS	-	-
	Kori-3·4 YoungGwang-3·4	12 ~ 14	950	3	960	316SS	-	-
	Ulchin-1·2	10 ~ 11	950	3	1029	316SS	-	-

Most susceptible component concerning IASCC is a baffle former bolt as shown in Fig. 3. And a threshold fluence level for an IASCC to occur in stainless steels is considered to be 4 dpa(displacement per atom).

2.4 Fluence calculation

Neutron fluence was calculated for the reactor internal structures of Kori-2. The calculation using a code called DORT 3.1 was based on an operation history from 1 to 21 fuel cycles. The calculation result shows that the maximum fluence region is a baffle plate of an azimuthal angle of 15.26 and a third former plate from the bottom as presented in Fig. 4.

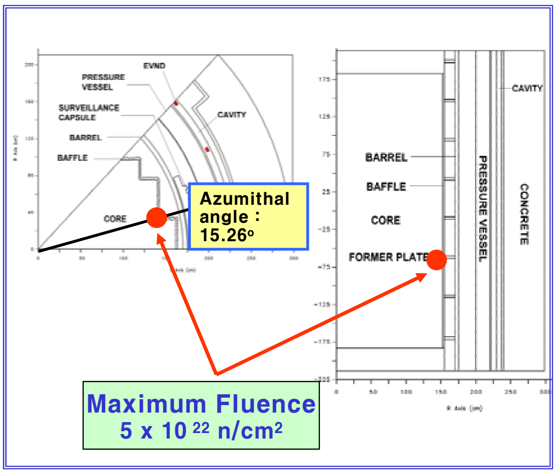

Fig. 4 Fluence calculation results on the Kori-2

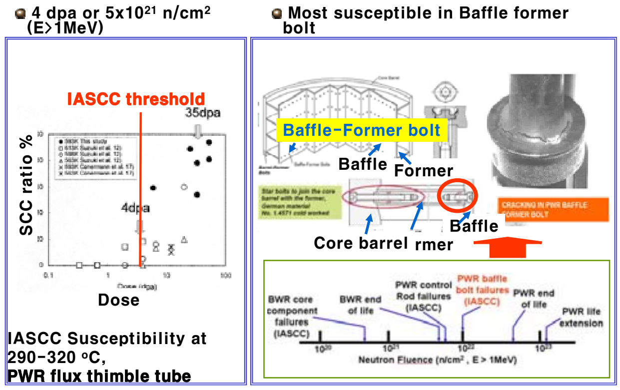

Fig. 3 IASCC threshold fluence and most susceptible area

3. Conclusions

IASCC of stainless steels have been reported in PWRs. Korea nuclear power plants have not shown any crack indication as yet. In order to set up a preventive management technology, IASCC management project was started in Korea in 2007. A threshold fluence level for an IASCC occur in stainless steels is considered to be 4 dpa. The calculation result shows that the maximum fluence region is a baffle plate of an azimuthal angle of 15.26 and a third former plate from the bottom.

REFERENCES

[1] H.T. Tang, 'Aging management of reactor internals and license renewal of US PER plants' Fontevraud 6, Sept. 18-22, 2006.

[2] Final report for the Integrity inspection of the baffle former bolts of Kori 1, KPS/NTSC-RAFR 034/06, KR1-BFB-Q, 2006.