Short Review on

Irradiation Assisted Stress Corrosion Cracking

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Introduction

Irradiation-assisted stress-corrosion cracking (IASCC) has become potentially a critical phenomenon for core internals in light water reactors (LWR). Most internal component of nuclear power constructed of austenitic stainless steels which choosen initially due to their good resistance to corrosion in the reactor coolant and their favourable mechanical properties. During the course of the operational life of the reactor, these components are subject to intense irradiations. These irradiations bring about a number of modifications in stainless steels, which alter the initial properties of the material and may degrade both the mechanical properties as the corrosion resistance of the material. IASCC is the premature cracking of material in an aggressive environment system exposed to ionizing radiation. It is a result of the interaction of irradiation, material, environment, temperature and stress. The complexity of IASCC arises of the fact that irradiation has an impact on all the variables listed above so that the knowledge available on SCC of materials in non-irradiated environmental conditions is not sufficient to solve te IASCC problem.

IASCC Service History

Instance of IASCC were first reported in the early 1960's. A summary of field service history is as follow :

1. All cracking was intergranular and initiated at contact points with the water environment, and crack branching was observed. Post-irradiation test in a dry environment found only ductile, transgranular cracking.
2. No grain boundary chromium-carbide precipitates were found.
3. There existed a definite correspondence between time to failure and stress level. Failure was first noted to occur in fuel rods with thin cladding where swelling strains were the highest.
4. A higher incidence of cracking existed in the areas of peak heat flux which also correspond to the areas of greatest fuel-cladding interaction and stress and strain.
5. Fewer reports of intergranular cracking in PWR's occurred, and at the time these incidents were believed to be the results of off-chemistry conditions or stress rupture. In restropect, however, IASCC is the likely mechanism by which these failures occurred. Hydrogen over-pressure and the resulting lower corrosion potential could possibly be off-set by the higher temperatures.

Some IASCC service experience shown in the Table 1 below

Component	Material	Reactor	Sources of Stress
Fuel Cladding	304 SS	BWR	Fuel Swelling
Fuel Cladding	304 SS	PWR	Fuel Swelling
Fuel Cladding	20%Cr/25% Ni/Nb	AGR	Fuel Swelling
Fuel Cladding Ferrules	20%Cr/25% Ni/Nb	SGHWR	Fabrication
Neutron Source Holders	304 SS	BWR	Welding & Be Swelling
Instrument Dry Tubes	304 SS	BWR	Fabrication
Control Rod Absorber Tubes	304 SS	BWR	B_4C Swelling
Fuel Bundle Cap Screws	304 SS	BWR	Fabrication
Control Rod Follower Rivets	304 SS	BWR	Fabrication
Control Blade Handle	304 SS	BWR	Low Stress
Control Blade Sheath	304 SS	BWR	Low Stress
Plate Type Control Blade	304 SS	BWR	Low Stress
Various Bolts*	A-286	PWR & BWR	Service
Steam Separator Dryer Bolts*	A-286	BWR	Service
Shroud Head Bolts*	600	BWR	Service
Various Bolts	X-750	BWR & PWR	Service
Guide Tube Support Pins	X-750	PWR	Service
Jet Pump Beams*	X-750	BWR	Service
Various Springs	X-750	BWR & PWR	Service
Various Springs	718	PWR	Service

* Cracking of Core Internal Occurs Away from High Neutron and Gamma Fluxes

Table.1. IASCC Service Experience

The above summary and Table.1 show that Type 304 stainless steels suffered extensive cracking, especially in its use as a fuel cladding which is subject to high stresses. Because of this it was subsequently replaced with Zircaloy-2. The cracking suffered by the 304 stainless steel was also the first sign that annealed type stainless steel could suffer from IASCC in a BWR environment.

In more recent instances of IASCC, failure of lower-stress components has been noticed and are noted in Table.2 and Table.3. This leads to the conclusion that cracks may occur at lower stresses for higher fluencies.

Component	Fluence (N/cm^2)	Source of Stress
Fuel Cladding	$5x10^{20}$-$2x10^{21}$	Fabrication & Fuel Cladding Interaction
Neutron Source Holders	10^{21}-10^{22}	Welding & Beryllium Swelling after Initial Crevice Attack
Control Rod Absorbers Tubes	$5x10^{20}$-$3x10^{21}$	B_4C Swelling
Fuel Bundle Cap Screws	10^{21}-10^{22} (estimated)	Fabrication and / or Assembly
Rivets in Control Rod Follower	$5x10^{20}$	Unknown

Table.2. Summary of field IASCC experience up to 1980

Component	Fluence (N/cm^2)	Source of Stress
Plate Type Control Blade	$2x10^{21}$	B_4C Swelling
IRM/SRM Dry Tubes	$\sim 1x10^{22}$	Fabrication

Table.3. Summary of post 1980 post IASCC experience

IASCC occurs irrespective of reactor type. Specific BWR vs. PWR comparisons were performed using in-core swelling tubes fabricated from a variety of commercial and high purity heats of types 304, 316, and 348 stainless steel and Alloy X-750, 718, and 625. Basedon identical strings of specimens placed in fuel rod locations, there was little distinction in the IASCC response between the two reactor types. Thus, it is becoming increasingly evident that the problem is widespread without regards to environment or alloy, and that numerous core components may be susceptible to this form of degradation. We can say that the future of current light water reactors ans future water reactor concepts may well rest on the solution to the IASCC problem.

Stress Corrosion Cracking

Although initially viewed as a completely independent phenomena, IASCC is now seen as an accelerant of the environmental cracking process, Stress Corrosion Cracking (SCC). To understand the complexity of which IASCC basic understanding of Stress Corrosion Cracking (SCC) is needed.

SCC is a term used to described failures in engineering materials that occur by environmentally induced crack initiation and propagation. For SCC to occur, the system must meet three basic requirements which are illustrated in Figure.1 and described as follows :

1. Susceptible Material : factors such as grain boundary chemistry and microstructure.
2. Tensile Stress : The surfaces of the components in the given environment have to be loaded in tension.
3. Aggressive Environment : A very specific environment is required fo SCC to occur for any given material. This environment provides for an electrochemical process resulting in the release of metal ions, the result being the localized dissolution of the metal.

As clearly depicted in Figure.1, SCC is not a result of any of these factors acting independently, but rather, conjointly. SCC is a result of a combined mechanical and chemical crack propagation process which ha been termed “synergistic”.

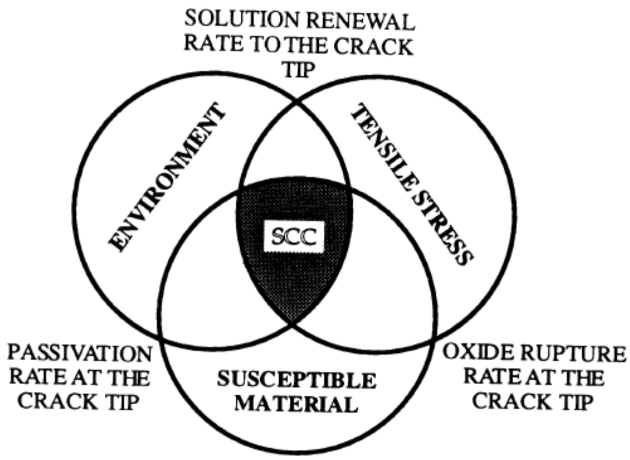

Figure.1. Venn Diagram showing the phenomenological factos required for SCC

How irradiation accelerate SCC ?

The presence of fast neutrons and gamma radiation act to enhance the process of stress corrosion cracking by affecting two legs of the SCC triad, the material and the environment. Irradiation can accelerate SCC as follow :

a. Change the material
Irradiation can change the microstructure and microchemistry of the material. Collision of an energetic particle with a lattice atom generating radiation damage. If the energy transfer of the elastic collisions is greater than the displacement threshold (E_d), a primary knock on atom (PKA) is generated and this PKA can displaced additional atoms through the lattice creating frenkel pairs (pair of vacancy and self interstitial atom) and others. The radiation damage also can result on a grain boundary segregation and radiation induced second phase precipitation.
b. Change the mechanical or stresses of the material
With increases in irradiation dose, the yield strength of the material increases, the ultimate tensile stress also increase. Formation of higher densities of vacancy and interstitials is attributed as the cause for this increase. Loss of work hardening and uniform elongation is observed after irradiation. Irradiation also enhanced creep by producing excess vacancies and interstitial and thus facilitating the easier dislocation movements.
c. Affect the aggressiveness of the environment by water radiolysis
Radiation causes decomposition of water into many species which affect the corrosion potential. The concentration of the species is proportional to the square root of the radiation flux. Fast neutron radiation has a stronger effect on water chemistry that other types of radiation such as thermal neutrons, beta particles and gamma radiation because the LET

Radiation Type	Radiation Mean LET, eV/nm	e_{aq}	H^+	OH	H_2	H_2O_2	H	HO_2
Fast n	40	0.93	0.93	1.09	0.88	0.99	0.5	0.04
γ	~0.01	2.7	2.7	2.86	0.43	0.61	0.61	0.03
mixed	n & γ	1.26	1.26	1.42	0.80	0.92	0.52	0.04
10 MeV H^+	13.5	1.46	1.46	1.52	0.70	0.90	0.64	0.04

Typical BWR Peak Fluxes: ~300 MRad/hr neutron, ~60 Mrad/hr gamma

Table.4. Linear Energy Transfer (LET) and G-values for different radiation species

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