Irradiation Damage Mechanism of Reactor Pressure Vessel Materials in PWR Nuclear Power Plant

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ABSTRACT

This paper discusses the irradiation damage mechanism of reactor pressure vessel materials in Pressurized water reactor nuclear power plant, and introduces the phenomenon and principle of irradiation accelerate embrittlement, irradiation accelerate corrosion, irradiation accelerate creep, irradiation accelerate swelling, it provides a certain reference significance for reactor pressure vessel irradiation damage protection.

INTRODUCTION

In the process of nuclear power plant operation service, the reactor pressure vessel materials inevitably influenced by high temperature environment, high pressure, high irradiation, vibration and other factors, which make the reactor pressure vessel material may occurs damage, cracks and other defects, it will impact the safe operation of the unit.

The irradiation damage problems of reactor pressure vessel materials caused more and more attention of research people in recent years, because the reactor pressure vessel is the second boundary of containing radioactive, once the reactor pressure vessel failure, it will lead to a large number of radioactive substances released into the containment vessel, this serious accident is not allowed in the safety design of the nuclear power plant. And the reactor pressure vessel cannot be replaced during the whole life of the nuclear power station, materials of reactor pressure vessel are under a high irradiation during services, it is easy to have the irradiation problems.

In order to study the irradiation issues of the reactor pressure vessel material, many studies has been done at home and abroad. This article summarizes the irradiation damage mechanism of reactor pressure vessel materials in pressurized water reactor (PWR) nuclear plant, and introduces the irradiation mechanism,
irradiation accelerate embrittlement, irradiation accelerate corrosion, irradiation accelerate creep, irradiation accelerate swelling phenomenon, it provides a certain reference significance for reactor pressure vessel irradiation damage protection for the researchers.

**IRRADIATION MECHANISM**

Irradiation affecting factors includes the composition of alloy, irradiation temperature, microstructure features and neutron flux energy spectrum.

The composition of alloy (especially the impurity elements copper, phosphorus and nickel elements) is sensitivity for irradiation[1]. Irradiation temperature has influence on the degree of irradiation damage. Data show that the irradiation will cause the biggest embrittlement when the irradiation temperature is less than 120℃ in the early 1960s. Recent research has proved that higher than 310 ℃, the irradiation embrittlement is abate, the reason was that the embrittlement dynamic annealing effect in service (Dynamic the in - situ annealing). The microstructure characteristics such as grain size and microstructure (low or high levels of bainite and ferrite), it will affect irradiation damage severity under a given flux. Neutron flux energy spectrum is secondary effect on the contribution of ferritic steel embrittlement.

Irradiation incident particles include the following three types, Neutral particles (neutrons, gamma rays), Charged particles (alpha particle, protons and electrons), Energetic atoms and ions (fission products, a collision recoil atoms, accelerated ions). Relative to the energy of incident particles, solid material (target) is considered to be a relatively static atoms. The interaction between incident particles and solid depends on the charge of the injected particle number and the rate of incident particles, the interaction between incident particles and atomic nucleus or extranuclear electron is relatively independent.

The irradiation process can be described as follows,

Step1, the high energy incident particles interacting with the atoms of the lattice,
Step2, the kinetic energy of incident particles transfer to the impacted atoms,
Step3, the impacted atoms move from the atomic lattice array point, and become a Primary off normal atoms (PKA),
Step4, PKA continue to impact other atoms, and form atomic displacement spike (cascade collision - below cascade),
Step5, When the cascade collision stop, leaving space (vacancy) and interstitial, as well as the vacancy and interstitial atom clusters.
Step6, Along with the particles energy loss, it will form heat peak near the path, and atoms rearrange cause element segregation, precipitate out and activation phenomenon.

The high-energy PKA can make a lot of atomic displacement, PKA can hit 104 normal atoms move away from their lattice position in one fast neutron collision, and produce the same number of lattice vacancy. Partial vacancy meet the interstitial, then composite the interstitial, and ultimately disappear. Partial vacancy composite with dislocation and grain boundary, and produce poor atomic area, and microhole, etc. Partial vacancy and interstitial can respectively be through gathering, collapsing, forming stacking fault, etc.
The research difficulties of irradiation damage are studying some trace elements and impurity elements behavior after irradiation. The influence of irradiation damage on the material mechanics performance is that the material will have higher yield strength and plasticity will reduce after irradiation.

Although the defect size of irradiation is very small, but the macro mechanical properties, chemical properties and service life of material has great changes. The purpose of research on irradiation damage is to understand the damage mechanism, predict the influence of irradiation on the properties of materials, develop new materials, and choose a more suitable equipment material.

IRRADIATION ACCELERATE EMBRITTLEMENT

Neutrons impact active atoms by the chain reaction, making a large number of atoms get away from its lattice position. The given location number of neutron bombardment of traditionally called fluence (n/cm², E > 1.0 MeV). The recent measurement of neutron exposure destroy is each atom dislocation (displacements per atom, dpa), this measurement considers more about the wider neutron energy spectrum. The fluence or the dpa provides some information for evaluation of radiation embrittlement. The radiation embrittlement influence factors include neutron irradiation, environment and materials.

The forging/rolling austenitic stainless steel do not show a sharp ductile brittle transition characteristics like the low alloy steel and carbon steel, the fracture toughness drops more with the injection quantity increases caused by irradiation, when the injection quantity is greater than 1 x 10²¹ n/cm², the fracture toughness tends to saturation. There are few information show the irradiation embrittlement of the reactors in quantitative.

For the 304 stainless steel in pile component of irradiated materials which serviced in the run of boiling water reactor, new study described the results of its fracture toughness within the injection scope of 1 x 10²¹ n/cm² and 6 x 10²¹ n/cm² (E > 1.0 MeV), the results confirmed that consider all amount injection, the saturation fracture toughness is the same number, it can be directly applied to the evaluation of the reactors in high irradiation. For the 304, 316CW and 347 stainless steel RVI materials in pressurized water reactor, the fracture toughness value showed a high level in 2 x 10²² n/cm² (E > 0.1 MeV) fluence, it could also be directly applied to the evaluation of the reactors in high irradiation.

As the crack resistance in pile material fall with the neutron fluence rise, it requires a detailed finite element analysis and enough relevant material properties of the data (e.g., crack extension rate, fracture toughness) to evaluate in quantitative.

The neutron irradiation embrittlement scope includes the core shell, the inlet nozzle, the outlet nozzle, bottom sealing head transition section, the ring weld located in the core parts and the ring weld located in the bottom sealing head, the above parts are easy to have neutron irradiation embrittlement problem. In order to timely assessment the RPV irradiation embrittlement, embrittlement aging management activities include evaluation and relief ways. The evaluation way is through platform on energy (USE) and brittle transition temperature (RTNDT) to assess the degree of irradiation embrittlement, the other evaluation way is using the method of main Curve (Master Curve methodology) to assess the severity of
irradiation embrittlement. The relief way is using thermal annealing to erase the RPV irradiation sensitive.

One mechanism of embrittlement is the stainless steel helium embrittlement[2,3], the generation of helium is through neutron reaction with boron and nickel element, helium yield which through neutron reaction with boron is high, and a small amount of B segregation in the grain boundary which has a big harm. Because helium is inert gas, it can not be dissolved in the metal substrate, it is easily gathered together in the place such as space, dislocation and grain boundary, and grow up to form bubbles, the accumulation of the helium bubble on the substrate and the grain boundary caused the attenuation of the material, the helium also exists in the form of interstitial, and it cause the lattice distortion and brittle the materials, the above factors leads to helium embrittlement.

IRRADIATION ACCELERATE CORROSION

The common irradiation accelerating corrosion phenomenon is irradiation promoting stress corrosion cracking (IASCC), IASCC has the same characteristics of crack initiation and propagation with intergranular stress corrosion cracking , but it has some difference, the austenitic stainless steel can also occur IASCC without thermal, IASCC is also influenced by neutron injection

When neutron injection meets or exceeds the neutron flux threshold under a certain stress level, annealing and austenitic stainless steel after irradiation are sensitive for the IASCC. Whether it has stable processing, stainless steel has IASCC sensitivity.

IASCC Influence factors include sensitive materials, neutron irradiation, environment and tensile stress. Based on the field and laboratory data, for high stress components annealing 304, 347, and 348 stainless steel material, the IASCC aging mechanism of neutron flux (E > 1 Mev) "threshold" is about 5 x 1020 n/cm² (about 0.8 dpa), "threshold" is about 2 x 1021 n/cm² (about 3.1 dpa) for low stress components. Although PWR reactor internals has not widely appeared phenomenon that IASCC lead to failure so far. But as the increase of service time, it should be paid more attention on this mechanism. There have been reports that IASCC caused coaming bolt appear cracks, the control rods have also discovered IASCC in the cladding. This approved that the IASCC will become the important role in pile component aging mechanism.

IRRADIATION ACCELERATE CREEP

Through fission process, a large number of cracks and pores are produced in dislocations, grain boundaries by the neutron irradiation.

Its Influence factors include neutron irradiation, environment temperature, and size (such as part thickness, etc.). If materials are under some strong stress, voids can migrate in a direction perpendicular to the applied stress, it creates the creep relaxation or creep under irradiation[4]. There are cases show that the annealing 304L tested under 390 °C temperature after irradiation, its diameter increases, the cause of strain due to irradiation and creep. The creep caused by irradiation is a linear change with the neutron flux.
IRRADIATION ACCELERATES SWELLING

If excessive gap combines in the metal hollow or cavity, it will cause swelling in the structure.

Because operating temperature is low enough of the pressurized water reactor, the swelling is light in RPV materials. However, in local thick parts caused by irradiation will cause swelling and local strain. The geometric shape change is the most important change caused by Swelling. This brings potential influence, it will affect the control rod movement and coolant flow.

The common mechanism of irradiation accelerate swelling is the stainless steel irradiation accelerate swelling[5]. With the increase of irradiation dose, there are two ways that makes the Ni in the stainless steel substrate depleted, irradiation induced precipitation of second phase, and it cause a lot of Ni and Si deprived from the matrix, the low Ni content in matrix will lead to the increase of empty and swelling. When the empty and swelling increases, the Ni element through the uphill diffusion (Kirkendall effect) and segregate in the hole surface.

The material performance changes caused by the depletion of Ni in matrix, it can reduce the fault energy, and the low-level fault energy can make it easy for material to cause martensite transformation. The above situation is most likely to occur in the low temperature, and under different reactor temperature, swelling of the stainless steel is different, for the 316 stainless steel pipe in 400°C, 130 dpa irradiation will cause 14% of the swelling, swelling embrittlement is very serious, and for cold deformation of the 316 stainless steel in PWR environment (300-400 °C), 100 dpa irradiation, the swelling volume is expected to less than 3%.

In general, the swelling caused by irradiation in the world has got more and more attention, and there have been some research results which try to explain this mechanism.

SUMMARY

This article summarizes the irradiation damage mechanism of reactor pressure vessel materials in PWR nuclear plant, and introduces irradiation accelerate embrittlement, irradiation accelerate corrosion, irradiation accelerate creep, irradiation accelerate swelling phenomenon. It provides a certain reference significance for reactor pressure vessel irradiation damage protection.

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