

Trend of PWSCC in Nuclear Reactor Vessel Head Penetrations

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1. Introduction

Because Alloy 600 is susceptible to primary water stress corrosion cracking (PWSCC) in pressurized water reactors (PWRs), development of repairing and preventive maintenance and arrangement of relevant standards are considered as an issue to be addressed for PWRs. In recent years, there have been some overseas incidents where PWR vessel heads using Alloy 600 were damaged. Therefore, establishment of repairing methods to cope with emergency cases has been studied as an issue of high priority.

Repairing of reactor vessel heads requires an advanced technology to form a proper weld on cracked fusion faces by three-dimensional remote control. The applicable standards cover a wide range including evaluation of crack growth at boundaries of complicated shape, evaluation of ductile fracture, evaluation of brittle fracture, welding on cracked areas, and inspection. To apply repairing methods to actual reactors, it is necessary to establish both technologies and standards.

This technical report mentions the following issues: 1) present status in Japan and abroad of working on standards and maintenance of reactor vessel heads, 2) information on the incident of vessel head leakage in unit 3 of the Ohi nuclear power plant on May 4, 2004 and its repair, and 3) future prospects based on the above 1) and 2).

2. Measures taken for vessel head repair

2.1 Overseas leakage incidents and responses in Japan

As shown in Figure 1, a reactor vessel head is made of low carbon alloy steel of about 180 mm in thickness and has vertical penetrations of about 100 mm in outside diameter. Each nozzle made of

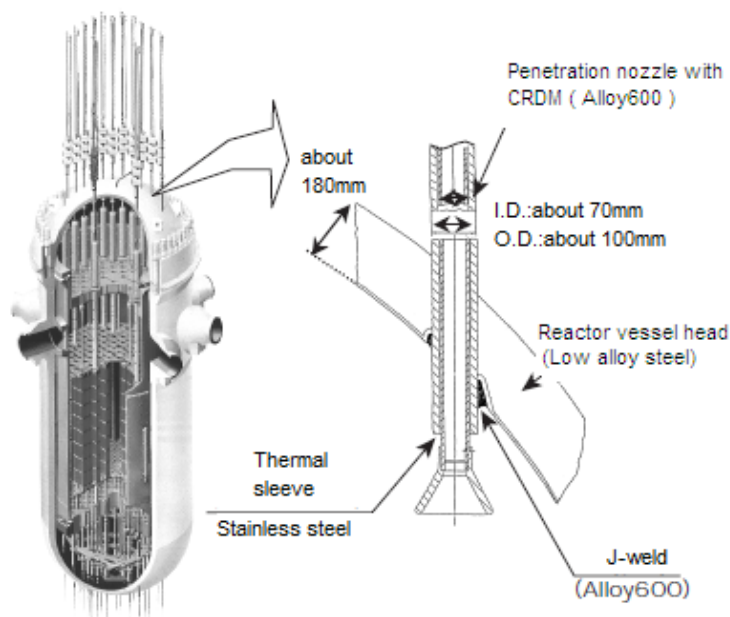


Fig.1 Structure of PWR vessel head penetration

Alloy 600 is welded into inside the vessel head (J-weld). Because a control rod drive mechanism (CRDM) is set inside of the penetrations, the accuracy of their position and inclination are important in view of control rod insertion performance. So the weld is done with a small amount of weld material and the nozzle is installed with high accuracy.

Leakage caused by SCC occurred in the vessel head of unit 3 of the Bugey nuclear power plant in France in 1991, followed by similar incidents in the United States. At the time in domestic plants, countermeasures were taken in Japan by conducting an eddy current test (ECT) and other inspections and replacing the vessel heads of plants that had been operated for relatively long periods of time with new heads using Alloy 690, which excels in corrosion resistance. In the other domestic plants, SCC was suppressed by increasing the amount of coolant diverted to the vessel head so as to reduce its temperature [1] (see Fig. 2).

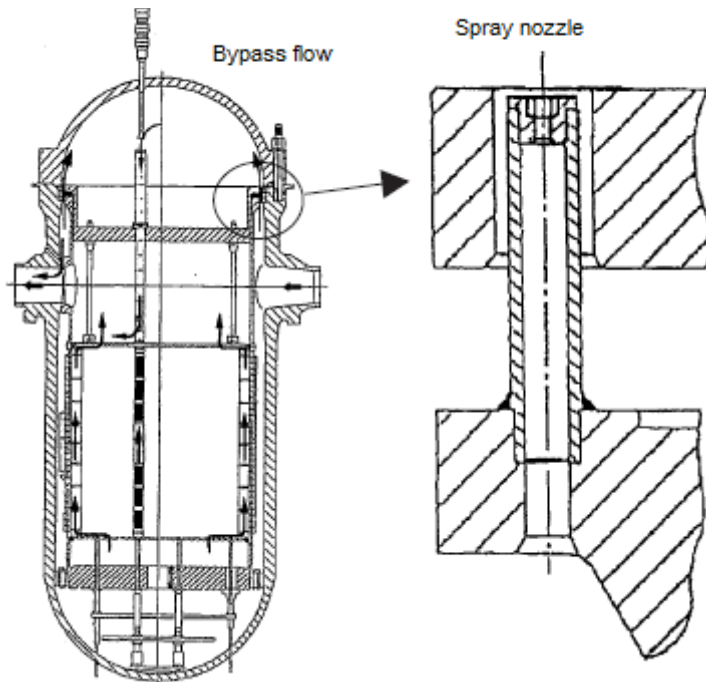


Fig.2 Temperature lowering by increasing bypass flow

In the United States, the Nuclear Regulatory Commission (NRC) intensified its supervision because, as shown in Figure 3, circumferential cracks were found in unit 3 of the Oconee nuclear power plant and other plants in the early 2000s.

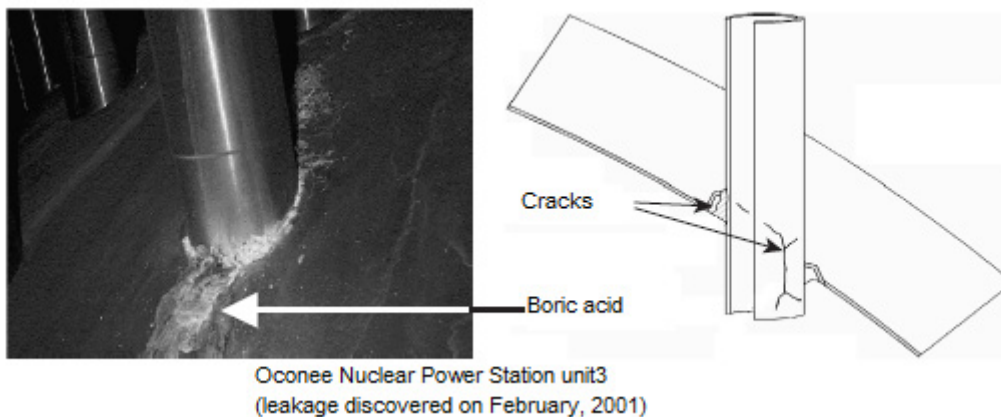


Fig.3 Leakage example in the USA

NRC, which uses the effective degradation year (EDY) expressed in reference to an upper head temperature of 315.6 deg C, has a guideline of 8 EDY for applying enhanced inspection [2].

Japanese electric power companies and manufacturers conducted studies in the same time period to develop a method to estimate the timing of PWSCC occurrence from temperature and stress. They obtained results nearly the same as those obtained by the NRC's method. Because it could not be denied that PWSCC would occur in units 3 and 4 of the Takahama nuclear power plant after about ten years, introduction of inspection equipment as well as maintenance and repair measures were contemplated to make assurance doubly sure. In these background, the Nuclear and Industrial Safety Agency (NISA) published an instructive document on inspection etc. [3] in December 2003 [4].

2.2. Status of standardization

During this time period, the private sector conducted studies on the maintenance of reactor vessel head penetrations according to the inspection and evaluation guidelines on reactor internals prepared by the Thermal and Nuclear Power Engineering Society. Regarding boiling water reactors (BWRs), a report was published in 2004, and work is in progress to reflect the results in the next version of the maintenance standard following the 2002 version. Studies on PWRs are also being conducted, with a report expected to be published in the near future. Generally, standards shall be reflected in regulations according to the procedure outlined in Figure 4.

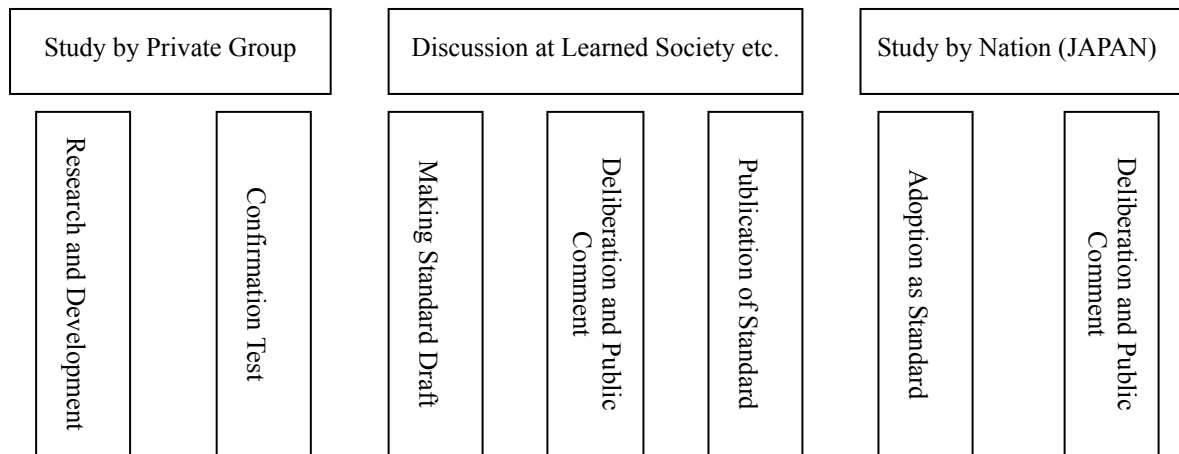


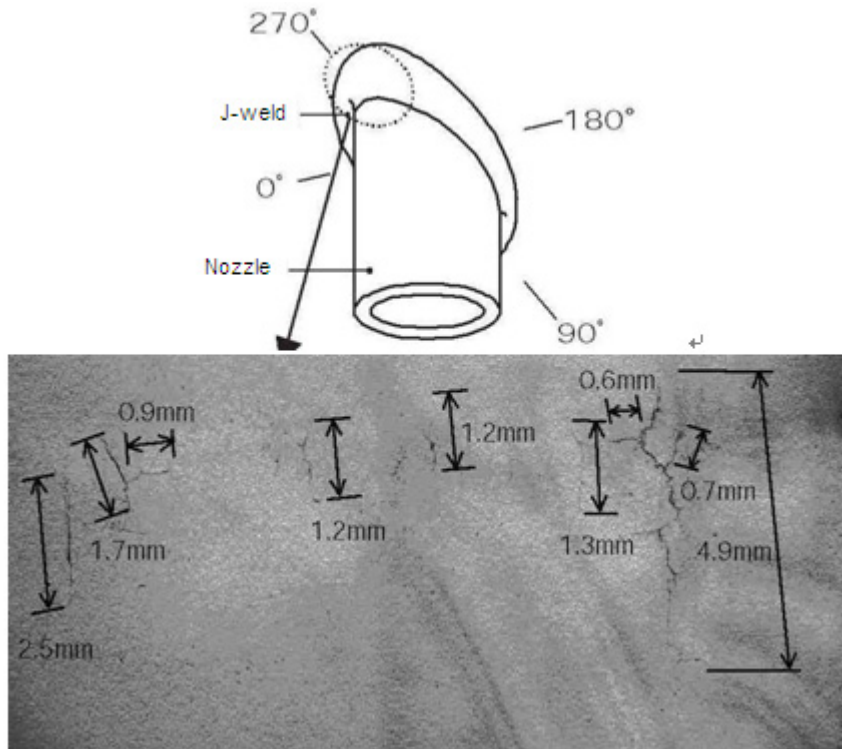
Fig. 4 Procedure of technological development and standardization

3. Vessel head leakage at Ohi unit 3 and its repair

3.1. Outline of leakage incident

Leakage from reactor head penetrations occurred at Ohi unit 3 on May 4, 2004 (see Fig. 5). The consideration so far indicates that the J-welded surface was under compressive residual stress due to buffing conducted for liquid penetrant testing (PT) at the time of reactor manufacturing (the buffing itself was not an obligation). The repairing method was applicable to both the nozzle and the J-weld, what non-penetrating cracks were detected by inspection.

As a result of the investigation, it was found that the leakage at Ohi unit 3 was caused by cracks that penetrate J-welds. Because no abnormal PT records were found in the manufacturing process, it was estimated that the leakage was caused by blowholes etc. that existed very close to the surface or PWSCC induced by high tensile residual stress associated with local areas where buffing was not performed. It is difficult to make further estimation at present because the cracked surface portion was removed in the course of investigation. After replacing the vessel head with new one several years later, the cause will be investigated by collecting samples from the head.



**Fig. 5 Cracks on reactor vessel head of Ohi unit3
(after removing 3mm surface)**

3.2. Establishment of repairing method

In response to the incident in USA, a method to repair the nozzle of the vessel head was developed with regard to repairing non-penetrating cracks at the base material part of the nozzle and J-welds. When the Ohi unit 3 incident occurred, studies on structural strength evaluation and techniques of welding on cracks had already been completed. Based on these, we could decide a repairing method for weld-penetrating cracks and pursue technological establishment, standardization, and authorization.

There are major restrictions in repairing cracks that penetrate J-welds. One is that we need to establish a temper bead welding method that does not require post weld heat treatment (PWHT) for restoring the cracked area after removing all cracks penetrated. This is because the vessel head is made of low carbon alloy steel. Another restriction is that it is difficult to manufacture a remote-control device that allows removal and restoration of J-welds in narrow areas.

From the above-mentioned reason, we adopted a method of direct welding on cracks (see Fig. 6), which is a permanent repair method adopted in the United States. This method isolates the environment causing PWSCC propagation and prevents leakage.

Moreover, a compact weld structure is allowed because the weld strength required can be determined by considering only the local area around the cracks of the J-weld. This is different from the situation of joint welds, for which contribution to the structural strength of the vessel head is required.

This means that the vessel head with cracks must have enough strength to prevent its collapse regardless of whether the welding is carried out. Whereas a joint weld can be likened to a bridge that connects both sides of a river, the adopted welding method is likened to filling a hole in the bridge. As a precondition, the strength of the bridge itself must be assured by a separate means.

Identifying the requirements of structural strength and weld functionality is essential not only for clarifying the issues to be addressed in technological establishment but also for proving conformity with standards and obtaining authorization.

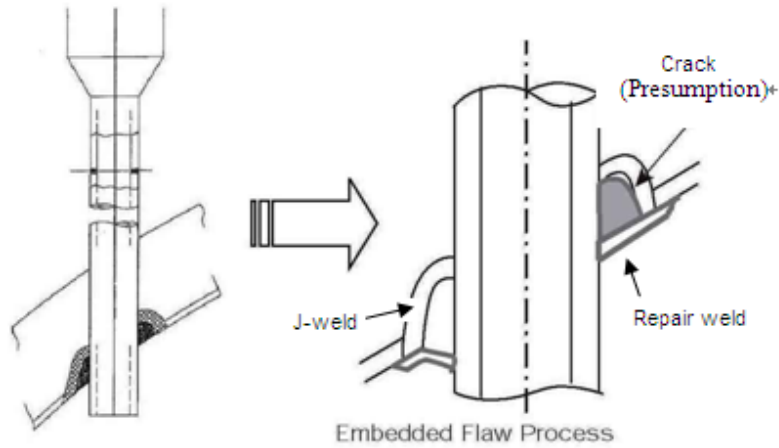


Fig.6 Repairing method applied at Ohi unit 3

The welding procedure condition was determined experimentally by using a test plate having simulated cracks created by electric discharge machining (EDM). A "proper weld" means that it correspond to the Ministry of International Trade and Industry Ordinance No. 123, which is a technical standard for welds (hereinafter referred to as "the Ministerial Ordinance on Welds"). We will describe it more in detail in the following.

3.3. Repair standards and approval

In July 2000, the Ministerial Ordinance on Welds was revised to specify weld performance with four provisions: (1) weld geometry, (2) weld crack prevention, (3) defect-free welds, and (4) weld strength. In applying these provisions to the repair of reactor vessel heads, they can be interpreted as follows. The purpose of provision (1), weld geometry, is to prevent weld failure due to unexpected stress resulting from discontinuous weld geometries etc. With regard to the repair of vessel heads, the purpose of welding is to achieve a weld geometry that does not compromise environmental isolation and leak prevention. More specifically, the purpose is to completely cover the J-weld and its surrounding area to enable environmental isolation and leak prevention, regardless of the surface condition of the J-weld. Provision (2) requires that welding does not cause existing cracks in the

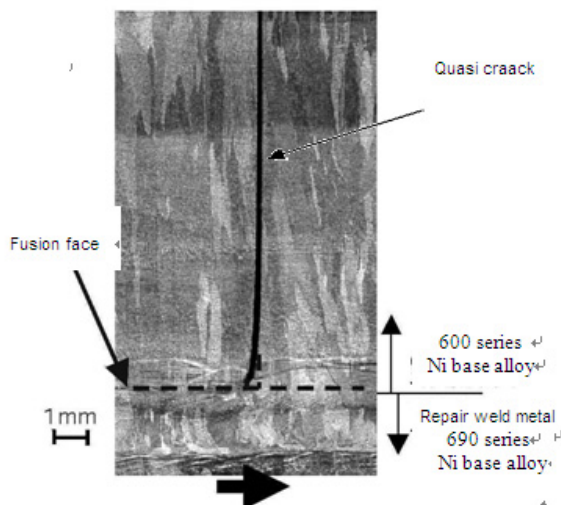


Fig.7 Macro structure of repair weld of the test specimen with quasi-crack

fusion face to grow or new cracks to occur (see Fig. 7). Provision (3) requires that there be no defects other than cracks to be repaired by welding. Provision (4) requires that the tensile strength of welding metal be evaluated and the thickness required to prevent leaks be ensured. Compliance with the ordinance requires meeting these provisions (see Fig. 8).

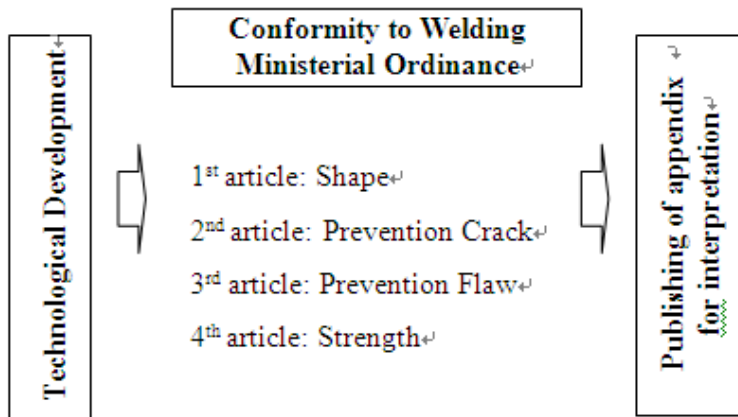


Fig.8 Establishment and Standardization of Welding Technology

These provisions were issued as an interpretation supplement to the technical standards for welding by the Nuclear and Industrial Safety Agency on October 22, 2004.

Maintenance standards for in-service equipment with cracks are available as standards for structural strength. However, repair work is considered as construction work, and therefore the design and construction standards will be used (there may be different interpretations). However, if this interpretation is applied, there are no standards that can be directly applied to the studied event.

To obtain approval, structural design must meet the Ministry of International Trade and Industry Ordinance No. 62 (hereinafter, Ordinance 62), technical standards for structural design, and the Ministry of International Trade and Industry Notification No. 501 (hereinafter, Notification 501). However, no provisions for crack growth are included in Notification 501. Therefore, we obtained approval for specially designed facilities in Article 3 of Ordinance 62 to carry out design that is not pursuant to Notification 501. This approval for specially designed facilities, which corresponds to the standard for the evaluation of cracked areas, involves evaluation of structural strength and crack growth.

More specifically, the approval requires modeling of crack geometry after welding, evaluation of crack growth in the vessel head, nozzle and welds, and evaluation of strength after crack growth (evaluation of brittle fracture and primary local stress for the vessel head; evaluation of plastic collapse for the nozzle; and evaluation of leak prevention for the welds). These requirements are nearly the same as those of the maintenance standards (see Fig. 9).

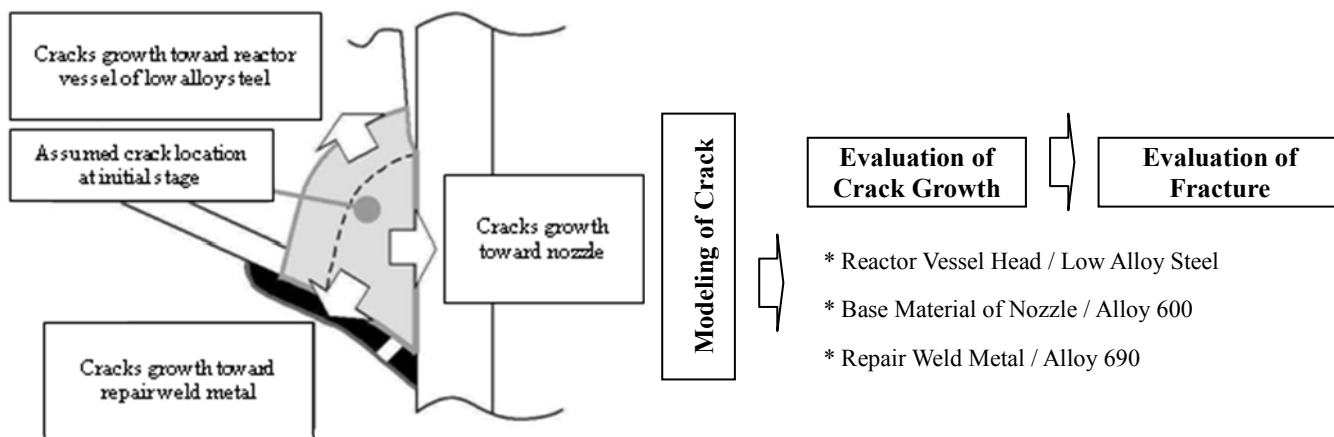


Fig.9 Evaluation of Fatigue Crack Growth after Repair

4. Prospects for technology development and standardization

This document focuses on reactor vessel heads for repairing and preventive maintenance of alloy 600 against PWSCC. It is necessary to establish maintenance measures to ensure the safety of other areas and the efficiency of repair by combining applicable inspections, drastic measures such as replacement, and damage repair measures according to the structural and functional characteristics of the applicable area. It is also necessary to carry out technological development in a planned manner, taking into account the lead time between the stages of developing and establishing technology.

Standards should deal with technological development in general and therefore need to be discussed and studied from various perspectives. With regard to reactor vessel heads, interpretation supplements to the technical standards for welds were issued, and they can be generally applied to PWR vessel heads. With regard to evaluation of structural strength, the J-weld joining the vessel to the pipe is not included in the areas subject to evaluation according to the maintenance standards. Therefore, generalizing and standardizing the methods of stress analysis and evaluation of the stress intensity factor K will facilitate the application and approval processes.

In June 2004, the revision of Ordinance 62 and the Ministerial Ordinance on Welds was undertaken to specify weld performance [5]. In the near future, private standards will be used more commonly. Therefore, mid- to long-term measures are required, such as including the described welding method in the welding standards of the Japan Society of Mechanical Engineers.

5. Summary

While commercial standards are being introduced to protect Alloy 600 against PWSCC, concrete steps to apply measures to actual plants, such as leak prevention and environmental isolation, were shown with regard to water leakage at unit 3 of the Ohi nuclear power station. In addition, in the approval process, active discussion was held in a rigorous and efficient manner. This is an example that provided significant incentives for technological development and standardization in the private sector. Needless to say, proper maintenance that has been practiced over the years made a significant contribution. This includes the welding repair of cracks in the neutron detector housing at the Tokai Daini power plant in 1999 [6].

We hope that activities for maintenance in a virtuous circle of technology development, standards, regulations, and implementation will be promoted by actively pursuing technological development with priority given to ensuring safety and by developing standards and applying them to actual plants. Such standards should define the areas requiring technological development and conditions of applying the developed technology.

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