UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
OFFICE OF NEW REACTORS
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS
WASHINGTON, DC 20555-0001

August 9, 2012

NRC INFORMATION NOTICE 2012-15: USE OF SEAL CAP ENCLOSURES TO MITIGATE LEAKAGE FROM JOINTS THAT USE A-286 BOLTS

## ADDRESSEES

All holders of an operating license or construction permit for a nuclear power plant issued under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

All holders of combined licenses issued under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

All holders of and applicants for an independent spent fuel storage installation license under 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

## PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to inform addressees of potential issues associated with the installation of seal cap enclosures (enclosures) to mitigate leakage from A-286 bolted connections in nuclear power plant piping. A-286 is a precipitation-hardened, iron-based super alloy specified as American Society for Testing and Materials (ASTM) A453, Grade 660 material. The NRC expects recipients to review the information in this IN for applicability to their facilities and consider taking action, as appropriate. Suggestions contained in this IN are not NRC requirements; therefore, no specific action or written response is required.

## DESCRIPTION OF CIRCUMSTANCES

Callaway Plant, Unit 1

Between 1985 and 1987, Callaway Plant, Unit 1, installed enclosures on four swing check valves located on the chemical and volume control system charging header to mitigate gasket leakage from the bolted body-to-bonnet flange joint. The enclosures were not part of the pressure boundary. Each valve was constructed from austenitic stainless steel bodies and bonnets, which were connected using bolting composed of A-286. In the unified alloy numbering system, it is designated as UNS S66286.

In 1992, all four enclosures were removed for valve and bolting examination and maintenance. A redesigned enclosure was installed on each valve at that time. During the 2002 refueling outage, one of the valves fitted with the redesigned enclosure exhibited evidence of boric acid leakage from the upper enclosure weld. The enclosure was permanently removed and the valve bonnet and bolting were replaced, but no bolting issues were identified. During the 2004 refueling outage, a second valve fitted with the redesigned enclosure exhibited boric acid leakage. The licensee removed the enclosure to perform maintenance on the valve. During disassembly, 3 of the 12 valve bonnet closure studs failed with the application of negligible force. A metallurgical analysis concluded that the studs most likely failed because of intergranular stress corrosion cracking (SCC). This valve was repaired and the enclosure was not reinstalled. Although inspections found no evidence of leakage or bolting degradation on the two remaining swing check valves, the licensee removed the enclosures and replaced the bolting on these valves.

<u>South Texas Project, Unit 2</u>

In February 1997, South Texas Project Nuclear Operating Company (STPNOC) discovered steam wisping from the body-to-bonnet gasket of a swing check valve located in the 8-inch nominal diameter safety injection piping at South Texas Project, Unit 2 (STP-2). This valve is constructed from an austenitic stainless steel body (SA182, Type 316) and bonnet (SA240, Type 316) joined using type A-286 bolts. As a corrective action, the licensee welded a metal enclosure over the bonnet and the studs of the check valve to mitigate the leakage (Figure 1). This enclosure is designed to contain leakage only; it is welded to, but not part of, the reactor coolant system pressure boundary.

During a January 1999 outage, the licensee identified boric acid deposits on the insulation for the check valve. The licensee took no action at this time because it did not observe active leakage. In March 2001, the licensee cleaned the check valve and performed a liquid penetrant examination on the welds, joining the enclosure and valve body based on the findings in January 1999. No liquid penetrant indications were noted. In April 2010, the licensee observed a 6-inch steam plume emitting from the check valve during plant startup. The licensee inspected the seal cap welds and did not observe weld defects or boric acid deposits. The licensee concluded that condensation in the enclosure bowl on top of the bonnet caused the steam plume, so it took no action at that time.

During the October 2011 refueling outage, the licensee identified water and boric acid crystals on the outside surface of the enclosure and on the valve bonnet (Figure 2). The licensee performed liquid penetrant testing and identified flaws on the weld joining the enclosure to the bonnet. Subsequently, the licensee repaired the fillet weld.

In April 2012, the licensee again found boric acid crystals in the weld joining the enclosure to the bonnet. Subsequently, the licensee removed the enclosure on the check valve and performed ultrasonic and visual examinations of the bolts. It did not find any degradation on the bolts. The enclosure was subsequently reinstalled. Permanent repair of the valve and removal of the enclosure are currently scheduled for the next refueling outage planned in 2013.

**BACKGROUND**

Alloy A-286 is procured to meet the requirements of ASTM A453 (American Society of Mechanical Engineers (ASME) SA453), Grade 660, or ASTM A638 (ASME SA638), Grade 660. This alloy has been used in a variety of nuclear applications including reactor vessel internals bolting, control rod drive mechanism parts, reactor coolant pump shafts and bolting and other applications. In many applications alloy A-286 has performed satisfactorily. The resistance of this alloy to general corrosion is similar to that of 300 series stainless steels. It is not susceptible to boric acid corrosion. There have been several instances, however, in which this alloy has failed in service because of SCC.

NRC Information Notice (IN) 90-68, "Stress Corrosion Cracking of Reactor Coolant Pump Bolts," dated October 30, 1990, discusses service failures of A-286 bolting that attached reactor coolant pump-turning vanes to the pump shaft in an international nuclear plant and SCC of A-286 bolting in reactor vessel internal components at four different Babcock and Wilcox designed reactors. IN 90-68, Supplement 1, dated April 14, 1994, describes SCC failures of A-286 reactor coolant pump bolting at a Westinghouse designed reactor.

Additional service failures of A-286 bolting caused by SCC include cracking of top guide bolts discovered at an international nuclear plant in 1982 and, later, at ABB-Atom boiling-water reactors (BWRs). The plants used the bolts to attach guide bars to the top guide or core grid, which aligns the top end of the fuel assemblies. The bolts that failed were highly loaded; however, lower stressed components (less than 30 percent of yield strength) made of alloy A-286 did not experience cracking.

Laboratory studies have shown the susceptibility of A-286 to SCC in reactor coolant environments. In general, susceptibility increases with applied loading and with dissolved oxygen content in the environment. For high-purity, low-oxygen environments similar to pressurized-water reactors (PWRs), A-286 may not be susceptible to SCC unless loaded above the yield strength. For high-purity reactor-coolant environments that have higher oxygen content typical of BWR coolant chemistries, susceptibility has been established at loading levels of 60 percent of the yield strength.

NRC report, NUREG-6923, "Expert Panel Report on Proactive Materials Degradation Assessment," published in March 2007, notes that "…the role of impurities, including oxygen introduced during plant shutdown and possibly consumed only slowly in confined crevices, in helping crack initiation is clear from all the evidence available. Once initiated, cracks grow relatively easily even in well-controlled pressurized-water reactor (PWR) primary water…."

Service experience with A-286 bolting that is not wetted (i.e., external to the reactor coolant system) has been good, with no reported failures caused by SCC.

**DISCUSSION**

The environment inside of an enclosure that is installed on a leaking flange is not necessarily similar to the high-purity, low-oxygen environment inside a PWR reactor coolant system. When the enclosure is installed, it is full of air, but if the joint is leaking, the enclosure can slowly fill with leaking reactor coolant. The leaking reactor coolant initially will be a reducing environment, but the oxygen in the trapped air will dissolve and saturate the borated water. The environment inside the enclosure will consist of hot, oxygen-saturated water, which will be much more oxidizing than PWR normal coolant chemistry. Since there is no mechanism for exchanging the water in the enclosure, the enclosure is similar to a dead leg connected to the reactor coolant system through a tortuous leak path. The water in the enclosure will remain oxygen saturated until all of the oxygen is consumed by electrochemical reactions with the metal surfaces in the enclosure. Electrochemical reactions that cause SCC are likely to occur with the A-286 bolting and enclosure attachment welds.

As stated previously, laboratory studies have investigated the susceptibility of A-286 to SCC in reactor-coolant environments. The data from these studies indicate that cracking increases with increased oxygen content. Since the enclosure environments are likely to be higher in oxygen than typical reactor-coolant environments, it is likely that cracking of A-286 materials in an enclosure environment will be more severe than that identified in these studies. Because of the increased oxygen content in the enclosures, it is unclear that mitigating factors identified in laboratory testing, such as reducing tensile stresses, will preclude cracking in enclosure environments. The best and potentially only information currently available concerning cracking of A-286 material in enclosure environments is from Callaway, in which three studs failed, and from STP-2, in which no cracking was observed.

In addition to the A-286 bolting, the enclosure attachment welds exhibited multiple cracks at both Callaway and STP-2. The enclosures at Callaway and STP-2 exhibited leakage of reactor coolant through the 300-series austenitic stainless steel enclosure attachment welds. STP-2 performed a penetrant test and identified indications in the enclosure attachment weld and concluded that the leakage was caused by fabrication defects such as a slag inclusion or porosity. However, penetrant testing is not a sufficient technique, by itself, to identify the metallurgical nature of indications. The enclosure attachment welds at STP-2 were penetrant tested at least twice before the recent findings in 2011, once during original installation and once in March 2001. Previous examinations identified no weld defects or indications. STPNOC did not perform a metallurgical evaluation of the defects in the attachment welds.

Austenitic stainless steels, in general, are not susceptible to SCC in a PWR coolant environment, but are susceptible to SCC in hot oxygenated water. The enclosures at Callaway and STP-2 contained hot reactor coolant that was in contact with the oxygen-containing atmosphere trapped within the enclosure. It is possible that the leakage through the enclosure attachment welds at Callaway and STP-2 resulted from SCC that was caused by the pressure stresses and exposure of the attachment welds to the aggressive, hot, oxygenated environment inside the enclosure.

Detection of failures of the welds joining the valve to enclosure may be possible by detection of leaks. However, in both instances above, failures were detected by insulation removal and identification of boric acid deposits. No inspection techniques are currently identified to permit detection of bolt failure without first removing the enclosure. Failure to identify bolting failures could result in a loss-of-coolant accident.

In summary, this IN alerts licensees that failures of A-286 bolting and the enclosure-to-valve welds may occur because of the unique environment which exists within valve enclosures. Bolting failures may challenge the structural integrity of the primary system pressure boundary and may result in a loss-of-coolant accident. Because of the differences between the environment within enclosures and the environments in which laboratory testing was conducted, it is unclear if mitigating techniques, such as reducing bolt tensile stresses by reducing the torque on bolts, will prevent crack formation. Additionally, at the present time, inspection of bolting requires removal of the enclosure.

**CONTACT**

This IN requires no specific action or written response. Please direct any questions about this matter to the technical contact listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) and Office of New Reactors (NRO) project managers.

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Enclosure: Figures

Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under NRC Library, Document Collections.

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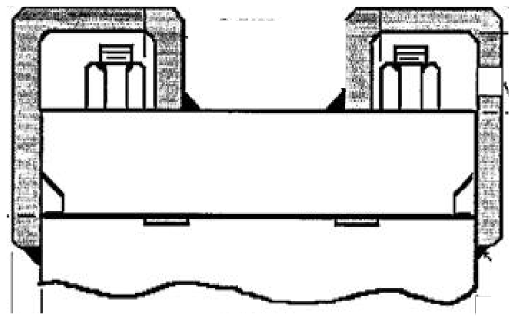

**Figure 1: Diagram of the enclosure**


**Figure 2: Photo of the enclosure**