Aging Degradation of BWR Reactor Internals[a]

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ABSTRACT

Researchers at the Idaho National Engineering Laboratory recently performed an assessment of the aging of the reactor internals in Boiling Water Reactors (BWRs), and identified the unresolved technical issues related to the degradation of these components. Several failures in BWR reactor internals have been caused by a combination of susceptible materials, environment, preload stress, and flow-induced vibration. ASME Code Section XI inservice inspection requirements are insufficient for detecting aging-related degradation in reactor internals. Many of the susceptible locations are not accessible for inspection.

INTRODUCTION

This paper briefly summarizes the aging assessment of BWR reactor internals presented in the report edited by Shah and MacDonald (1988). More detailed descriptions and references are given in that report.

The BWR reactor internals, shown in Figure 1, comprise the core support structure, the feedwater spargers, the jet pump assemblies, and the steam separator and dryer assemblies. The reactor internals materials are stainless steel or Ni-Cr-Fe alloy (Inconel). The core plate and top guide studs, nut, wedges, and pins are made of Type 304 stainless steel, Type XM-19 stainless steel, and ASTM A193 (Grade 8A) high alloy steel bolting material. Some castings (CF-3 and CF-8 stainless steel) are also used; including the orificed fuel supports and portions of the jet pump (the inlet mixer adapter casting, the wedge casting, and the diffuser collar casing) and the steam separator and dryer assemblies.

DEGRADATION SITES AND MECHANISMS

This section describes the degradation sites and mechanisms for BWR reactor internals. Shah and MacDonald (1988), Brown and Gordon (1987, 1988), and Herrera and Stancavage (1988) provide more details.

Intergranular Stress Corrosion Cracking (IGSCC)

IGSCC can occur where a high tensile stress exists in material that is sensitive to attack (e.g., stainless steel and Inconel) in the corrosive environment of oxygenated (200 ppb), high temperature (288°C) water.

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Other factors are high residual stresses, high coolant conductivity, crevices (which concentrate corrosive ions), and cold working. Weld sensitization makes the microstructure of stainless steel more susceptible to IGSCC.

In February 1970, the hold-down beam assembly of a jet pump failed at the Dresden 3 plant. Similar cracks were discovered at the Quad Cities 2, Pilgrim, Millstone 1, and Vermont Yankee plants. IGSCC occurred across the ligament sections of the Inconel X-750 beams, originating at the highly stressed thread roots of a bolt hole. The final heat treatment procedure was subsequently modified to produce a less susceptible microstructure.

Cracks in stainless steel intermediate and source range monitor dry tubes, first discovered through visual inspections in 1984, are attributed to IGSCC and irradiation-assisted stress corrosion cracking (IASCC), accelerated by crevices. Thick oxide formations in the crevices are the stress sources. Sensitization does not occur in the heat-affected zone because the tube is thin and is cooled quickly and evenly after welding. Local fluence levels range from 5×10^{21} to 1×10^{22} n/cm^2 (> 1 Mev). The threshold for IASCC damage (Herrera and Stancavage, 1988) is 5×10^{20} n/cm^2 (> 1 Mev).

Cracking was discovered at creviced locations in preloaded Inconel 600 shroud head bolts at a number of BWR plants in 1986. The cracking occurred in the vicinity of a weld joining the 304 stainless steel collar to the Inconel 600 bolt. Initiation and growth of the IGSCC was apparently driven by the residual weld stress, but the cracking was sufficiently distant from the weld root that it was not affected by the heat-affected zone.

BWRs with jet pumps (all but 2 or 3 earlier BWRs) have 2 access holes in the shroud support (baffle) plate. After construction, the holes are closed with Inconel 600 cover plates welded over the holes. Intermittent short cracks were found in the weld heat-affected zone around the entire circumference of the covers at Peach Bottom Unit 3 in 1988. It is estimated that cracking extends as deep as 70% through the wall. The IGSCC can be attributed to high residual welding stresses, weld crevices, and less than ideal water quality.

IGSCC was formerly thought to occur only in sensitized areas of stainless steel welds; however, cracking has been found in non-sensitized weld areas of jet pump inlet riser safe ends (at the juncture of the recirculation piping and the reactor vessel). General Electric (GE) discovered IGSCC on both the creviced and noncreviced sides of the weld. Inspections of creviced low carbon stainless steel safe ends at a BWR plant with a lower lifetime average coolant conductivity, revealed cracking (Brown and Gordon, 1987). Thus the role of high conductivity as a contributor to IGSCC is strongly suspected.

Other areas of potential IGSCC are the weld between the lower shroud and the shroud support (baffle) plate, the upper shroud near the core spray inlet tee, and the beam-to-plate welds in the core plate. Their creviced geometries make these areas susceptible to IGSCC. IGSCC cracks in recirculation nozzle safe end welds subsequently propagated into the base metal; thus there is a concern that IGSCC in the welds of reactor internals to the reactor vessel might also propagate into the reactor vessel wall base metal.

Accumulated BWR data indicates that IGSCC has been the degradation mechanism that has caused most of the significant repair/replacement efforts. A summary of IGSCC in BWR reactor internals is listed in Table 1. The materials are stainless steel unless otherwise noted.

Based on relatively short-term tests, Brown and Gordon (1987) state that the use of hydrogen water chemistry (HWC) is very beneficial in mitigating IGSCC. However, the degree of success appears to be highly plant dependent and the long-term benefit still needs to be established.

TABLE 1. HISTORY OF IGSCC IN BWR REACTOR INTERNALS

Location	Detected	Contributing Factors
Jet pump holddown beam	1970	High applied stress (in screw threads), Inconel X-750
Core spray sparger	1978	Cold work, sensitization, installation stresses
Junction box between vessel wall and shroud	1982	Sensitized heat-affected zone of weld
Core spray sparger arm weld	1982	Sensitized heat-affected zone of weld
Jet pump instru-mentation penetrations	1984	Sensitized heat-affected zone of safe-end weld
Neutron monitor dry tubes	1984	High conductivity and fluence, crevices, oxide wedging
Jet pump safe ends	1985	High conductivity, crevices, cold work
Shroud head bolts	1986	High conductivity, crevices, residual weld stress, Inconel and stainless steel
Access hole cover in baffle plate	1988	High weld residual stress, crevices, Inconel 600

IASCC

IASCC, like IGSCC, requires a combination of susceptible material (e.g., austenitic stainless steel), a conductive environment, and tensile stress (much lower stress for initiation than for low radiation). Fast neutron fluence changes the microstructure of originally annealed austenitic stainless steel to make it prone to cracking in the presence of a corrosive environment and stress. Laboratory tests have shown that high tensile stresses, crevices, and high fluence levels accelerate cracking (Kass, 1974, and Gerber, 1986).

Several failures of highly irradiated reactor internals have been attributed to IASCC. These include failures of early fuel rod cladding, neutron source holders, and control rod absorber tubes. Some more recent failures have suggested that cracking can occur in highly irradiated areas with very little applied stress. Failures of a control rod blade handle and neutron monitor dry tubes suggest that cracking will begin in stainless steel irradiated to about 5×10^{21} n/cm^2 (> 1 Mev). Jacobs (1987) reports a threshold for IASCC of 8×10^{20} n/cm^2, while Brown and Gordon (1987) report 5×10^{20} n/cm^2 (> 1 Mev) based on neutron monitor dry tube experience.

The locations with the highest neutron flux are the fuel supports, shroud, and top guide. Gerber (1986) concluded that based on an estimated 40-year fluence of 1×10^{22} n/cm^2 (> 1 Mev) at the bottom center of the top guide, rather large IASCC cracks (> 76 mm long) would have to develop before failure would occur from a crack initiated at a slot in the bottom beam. However, this study did not include several factors, e.g., radiation-induced swelling, multiple beam failures, refined material properties, and refined fluence estimates. The significance of combining these factors is not known.

Fatigue

The reactor internals component most susceptible to high-cycle fatigue is the jet pump. A jet pump may experience over 10^{10} cycles during 40 years (EPRI, 1986). Although fatigue has not yet caused any jet pumps to fail, the riser support braces have been identified as a potentially life-limiting area. Feedwater sparger fatigue failures due to high cycle flow-induced vibration (FIV) have occurred in numerous BWRs, including the Millstone 1, Humboldt Bay, Dresden 2 and 3, Quad Cities 2, and Peach Bottom 2 and 3 plants.

Thermal Embrittlement

CF-3 and CF-8 stainless steels and their welds may be subject to thermal embrittlement with prolonged exposure at temperatures of 288 to 316°C. The microstructures of these alloys contain a significant (up to 30%) percentage of the ferrite phase. When the ferrite content exceeds 15%, a significant loss of impact properties has been observed in specimens tested at 400 to 500°C for selected periods of time, and these data have been extrapolated to LWR temperatures at longer times. Thermal embrittlement has not been fully investigated, especially at LWR temperatures and long times, and is currently being examined by a Westinghouse owners group and by an NRC-sponsored program at the Argonne National Laboratory (Chopra and Chung, 1987).

INSERVICE INSPECTION AND SURVEILLANCE

ASME Section XI (Table IWB-2500-1) calls for visual examinations of accessible surfaces of core support structures and welds of interior attachments to the reactor pressure vessel at refueling outages. Visual examinations are useful for determining the locations and the nature of deteriorated areas. These locations can then be subjected to more detailed examinations. However, visual examinations can only examine the surface, and thus in many cases cannot predict locations of impending failures. Tight cracks would probably be overlooked in these inspections. Limited space and high radiation fields have resulted in the development of remote video cameras for underwater visual examinations of reactor internals. The present ASME Code requirements are not sufficient to adequately detect all aging-related degradation of reactor internals. Many of the susceptible locations are not accessible for inspection, e.g., the interior of the jet pumps.

GE has a program to place small specimens of representative reactor internals materials with induced cracks into autoclaves connected to the recirculation systems of operating BWRs. These specimens are then periodically removed to assess crack initiation and growth in the actual BWR environment.

SUMMARY AND RECOMMENDATIONS

Reactor internals subcomponents susceptible to degradation are listed in order of concern in Table 2. IGSCC is thought to be the overall life-limiting mechanism. Prompt initiation of HWC could be beneficial in reducing IGSCC, but the long-term effect has not yet been demonstrated. The other primary mechanisms of concern are IASCC, fatigue, and thermal embrittlement. Currently required ASME Code inservice inspections exempt many potential areas of degradation from inspection because of inaccessibility. Recommendations for understanding and detecting the aging of BWR reactor internals are:

1. Research is need to assess the effects of crevices, cold work, stress, and cumulative fluence on IGSCC and IASCC damage.

2. Data are needed to assess the long-term benefits of HWC in the core region, including the effect on fatigue crack initiation and growth.

TABLE 2. DEGRADATION PROCESSES FOR BWR REACTOR INTERNALS

	Degradation Sites	Stressors	Degradation Mechanisms
1	Top Guide	Radiation, thermal stress, corrosive water	IASCC, IGSCC
2	Core shroud and baffle plate	Corrosive water, radiation	IGSCC, IASCC
3	Core plate	Corrosive water	IGSCC
4	Jet pumps	Corrosive water, FIV, temperature	IGSCC, fatigue, thermal embrittlement
5	Feedwater spargers	FIV, corrosive water	Fatigue, IGSCC
6	Fuel assembly supports	Corrosive water, FIV, radiation, temperature	IGSCC, fatigue, IASCC, thermal embrittlement
7	Steam separator and dryer	Corrosive steam, FIV, temperature	IGSCC, fatigue, thermal embrittlement

3. Inservice inspections should be performed at critical locations (e.g., the top guide bottom beam) throughout plant life to provide early detection of IGSCC and IASCC. Methods to detect cracks in inaccessible locations such as bolts and in attachment welds to vessel walls are also needed.

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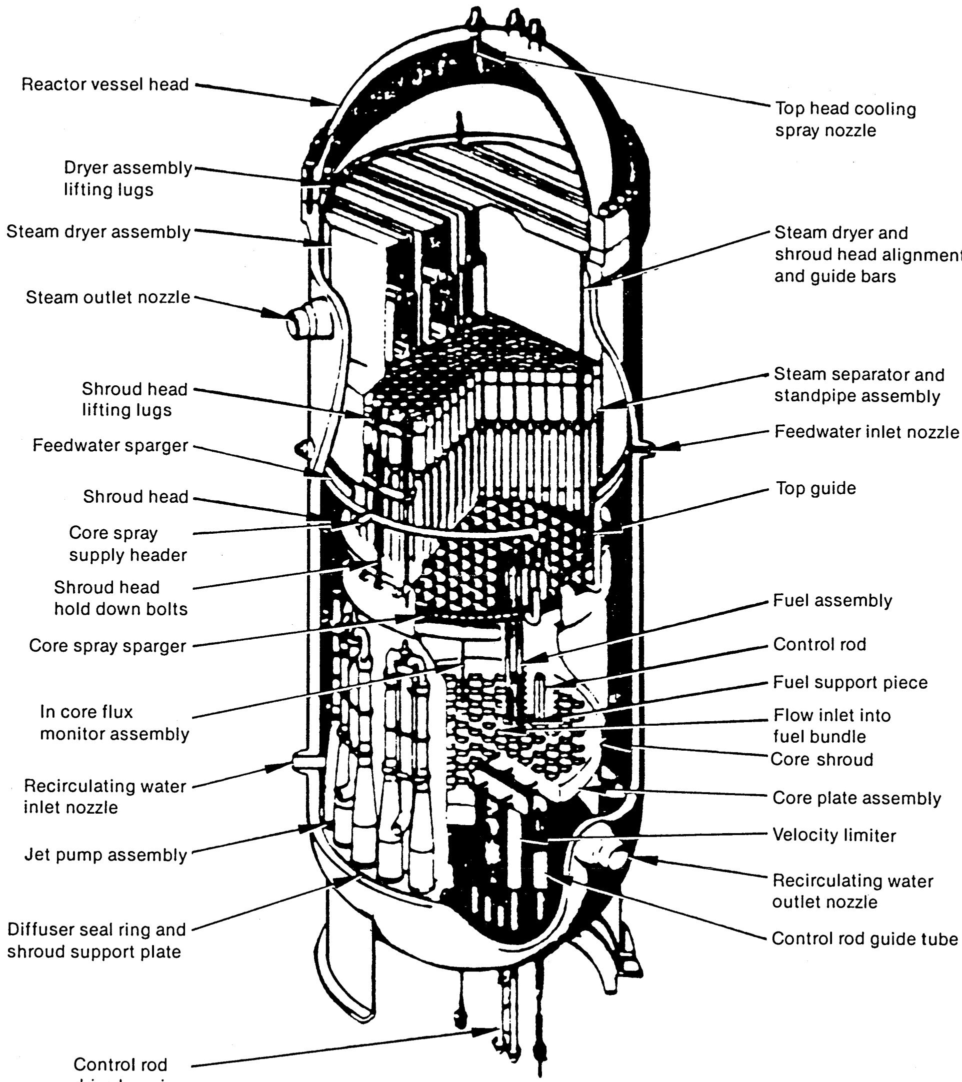

Figure 1. BWR reactor internals.

NOTICE