UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

January 12, 2006

NRC INFORMATION NOTICE 2006-01: TORUS CRACKING IN A BWR MARK I CONTAINMENT

ADDRESSEES

All holders of operating licenses for nuclear power reactors having boiling water reactor (BWR) Mark I containments, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to inform the owners of BWR Mark I containments about the occurrence and potential causes of the through-wall cracking of a torus in a BWR Mark I containment. Recipients are expected to review the information for applicability to their facilities and consider appropriate actions to avoid similar problems. However, the measures suggested in this IN are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

On June 27, 2005, with the plant operating at 100-percent power during a licensee inspection of reactor core isolation cooling system torus suction piping, James A. FitzPatrick Nuclear Power Plant (FitzPatrick) personnel discovered a torus leak near a torus support. The plant's torus is a large doughnut-shaped steel structure that is partially filled with water and designed to act as a pressure suppression chamber (see Figure 1). The torus geometry and supports at the location of the torus crack are shown in Figure 2. The leak was located about 5 feet below the waterline and just below the high-pressure coolant injection (HPCI) turbine exhaust pipe.

The leak was characterized as a slight seepage with streaking and a small puddle below the leak. Subsequent nondestructive examination determined that the leakage was from a small through-wall torus crack which was x-shaped with an approximate 4.6 inch maximum length. The licensee determined that operability of the primary containment was not assured and declared an Unusual Event and subsequently shut down the reactor (see Event Notification 41815, Reference 1).

To correct this condition, the licensee installed an approximately 13 inch outer diameter torus repair plate with a full-penetration weld joint. Pressure testing and inspection of the torus and drywell were completed after the repairs were completed. An NRC special inspection team

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reviewed the licensee's repair methods, root cause and extent-of-condition determinations, and corrective actions before the reactor was restarted (see NRC Inspection Report 05000333/2005009; Accession No. ML053610132).

BACKGROUND

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a "Codes and Standards," incorporates by reference Subsections IWE and IWL of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for inspection of steel and concrete containments with certain modifications and limitations. These subsections require licensees to inspect the pressure-retaining components of containments at periodic intervals. Subsection IWE of the ASME Code is applicable to the inspection of the FitzPatrick containment, consisting of a steel drywell, a steel torus, and connecting vents.

The NRC also requires licensees to perform leak rate testing of the containment pressure-retaining components and isolation valves according to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Option B of Appendix J is a performance-based regulation permitting licensees to set test frequencies based on the performance of the components. The pertinent testing requirement is the containment integrated leakage rate test (ILRT) requirement (Type A test). Based on the results of the earlier Type A tests and using the risk-informed methodology described in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," the licensee had previously been granted a license amendment to use a 15-year interval for the ILRT, with the next test to be performed by March 2010.

10 CFR Part 50, Appendix J, Option B, III.A, requires that a general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration which may affect the containment leak-tight integrity must be conducted prior to each test, and at a periodic interval between tests based on the performance of the containment systems. These test requirements provide for periodic verification of the structural integrity of the primary reactor containment.

DISCUSSION

The FitzPatrick licensee performed a root cause investigation of the event, and after eliminating a number of possible causes (thermal fatigue, clearing load phenomena, metallurgical discontinuity, weld defects, corrosion, flow-induced phenomena, flow-accelerated corrosion, cavitation, and direct jet impingement), the licensee concluded that the most likely cause for the initiation and propagation of the crack was the hydrodynamic loads of the turbine exhaust pipe during HPCI operation coupled with the highly restrained condition of the torus shell at the torus column support (see Figure 2). The cracking occurred in the heat-affected zone of the lower gusset plate of the ring girder at the torus column support (Figure 3 shows the HPCI turbine exhaust pipe entering the torus and the approximate location of the crack). The licensee concluded that the crack was initiated by cyclic loading due to condensation oscillation during HPCI operation.

These condensation oscillations induced on the torus shell may have been excessive due to a lack of an HPCI turbine exhaust pipe sparger that many licensees have installed. The licensee could not pinpoint exactly when of the crack started. Subsequent HPCI system operation helped propagation of the crack. The licensee indicated that no detrimental torus condition was noted during the general visual examination performed (per the 1998 edition of Subsection IWE of Section XI of the ASME Code) during the refueling outage in 2002, and no leakage was observed from this torus area during a walkdown by plant personnel on April 19, 2005. As part of its assessment of the IWE inspection program, the licensee noted that the IWE inspection program is only capable of identifying conditions that are visually detectable. Based on its assessment, the licensee established a corrective action to address the need for augmented inspections in areas where high operating stresses may exist or high fatigue cycling is likely.

The NRC staff is aware of several instances where the torus and drywells of BWR Mark I containments have been subjected to pitting and general corrosion (see References 2 and 3, INs 86-99 and 88-82 and the related supplements). However, this is the first occurrence of a through-wall crack known to the NRC staff. The following measures could reduce the possibility of such an event in the future or enable early detection of similar degradation:

- Many licensees have installed HPCI turbine exhaust pipe condensing spargers as one possible way to adequately distribute the operational type hydrodynamic loads and ensure that the stresses developed by these loads are within the acceptable limits.

- 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that measures be established to assure that applicable regulatory requirements and design basis are correctly translated into specifications, drawings, procedures and instructions and any changes are subject to commensurate design control measures. Design changes (any past or planned modifications) that affect the operational and accident loads imposed on the containment torus need to be subjected to commensurate design control reviews to ensure that the critical areas meet the acceptance criteria of the design specifications.

- The combined operation of the HPCI system and the safety relief valve (SRV) discharges during the northeast grid blackout disturbance of August 2003 may have initiated the crack (although this could not be conclusively determined). The HPCI system operated for approximately 14.5 hours and SRVs lifted five times over a period of 28 hours following the grid blackout disturbance. The FitzPatrick licensee established a corrective action to address the need for augmented inspections in areas where high operating stresses may exist or high fatigue cycling is anticipated. Such actions may be warranted after such stress-inducing events or after a strong seismic event (i.e., operating basis earthquake).

- 10 CFR Part 50, Appendix J, provides for periodic verification of the leak-tight integrity of the primary reactor containment as specified in the technical specifications. Subsection IWE of the ASME code requires periodic inspection of the containment surfaces. These inspection programs are focused toward detecting structural deterioration that could affect either structural integrity or leak-tightness. The torus through-wall cracking in this event was revealed by water leakage. Water leakage is readily indicated for through-wall cracks below the torus water line. BWR Mark I containments also have areas above the water line, where only air or gas would leak, and cracking at these locations would not be as easily detected. Because cracks can affect structural integrity or leak tightness, licensee containment inspection programs are required to consider the potential for such cracking, in addition to detecting general and pitting corrosion-induced degradation to ensure that containment integrity is maintained as specified in technical specifications.

REFERENCES

1. Event Notification Report No. 41815, posted July 1, 2005, available on the NRC Web site at http://www.nrc.gov/reading-rm/doc-collections/event-status/event/2005/20050701en.html#en41815.

2. IN 86-99, "Degradation of Steel Containments," dated December 8, 1986 (Agencywide Documents Access and Management System (ADAMS), Accession No. ML0312502480), and Supplement 1, dated February 14, 1991 (Accession No. ML031250234).

3. IN 88-82, "Torus Shells with Corrosion and Degraded Coatings in BWR Containments," dated October 14, 1988 (Accession No. ML031150069), and Supplement 1, dated May 2, 1989, http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/1988/in88082s1.html.

CONTACT

Please direct any questions about this matter to the technical contacts below or to the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

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Attachments: Figure 1: Typical BWR with Mark I Containment and
Figure 2: Cutaway Side-view of Torus Shell and Support
Figure 3: Photograph of HPCI Turbine Exhaust Pipe Entering the Torus

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

Figure 1 Typical BWR with Mark I Containment

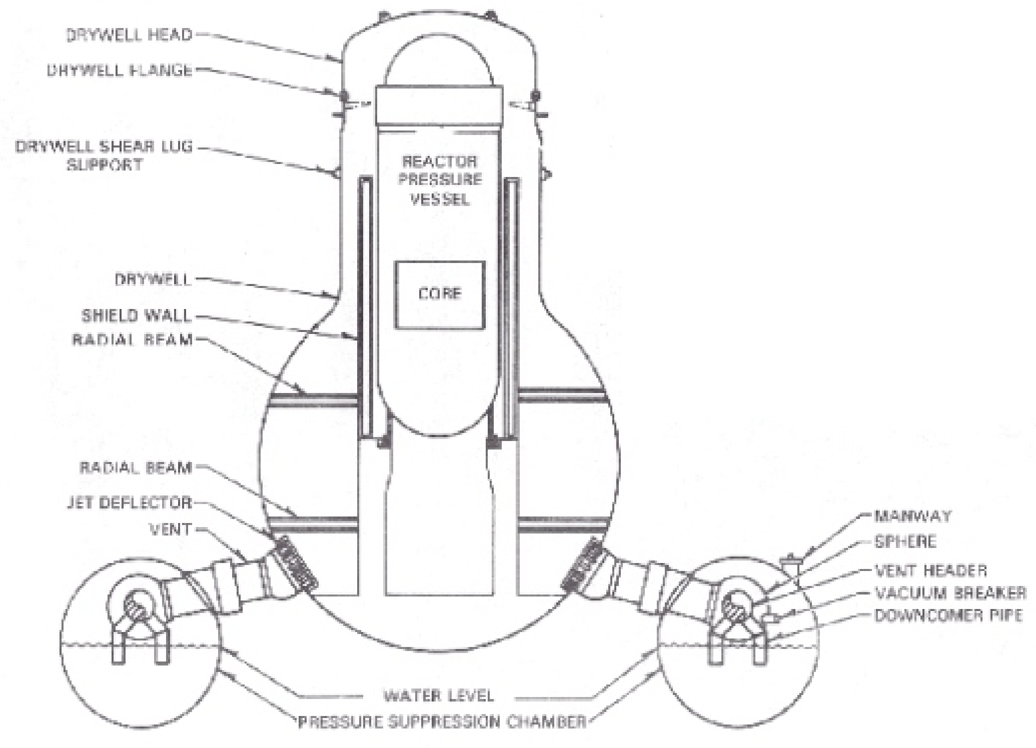

Figure 1 the cross-section of a pressure suppression chamber (or torus) is 29.6 feet wide. The pressure suppression chamber (torus) holds approximately 790,000 gallons of water.

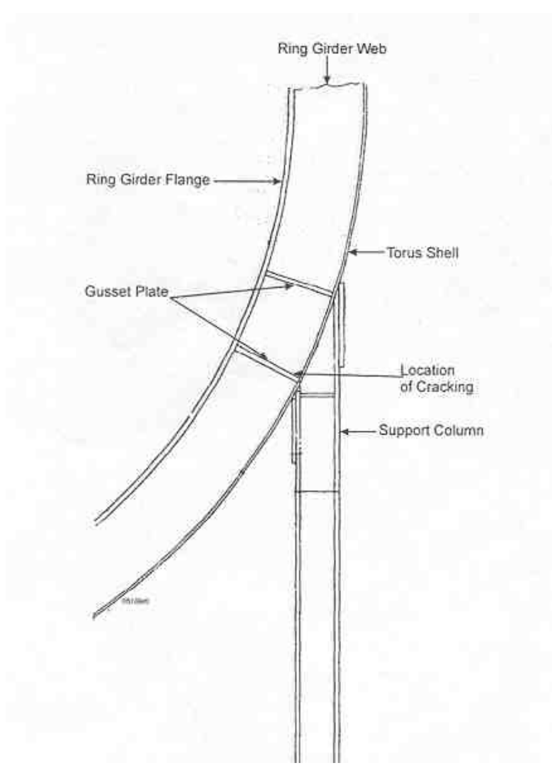

Figure 2
A cutaway side view of the torus shell and support showing the location of the cracking.

Figure 3: Photograph from inside the torus of HPCI turbine exhaust pipe, with the approximate location of the crack at the gusset. This view is from inside the torus. The HPCI turbine exhaust pipe (in the center of the photo) is approximately 24 inches in diameter.

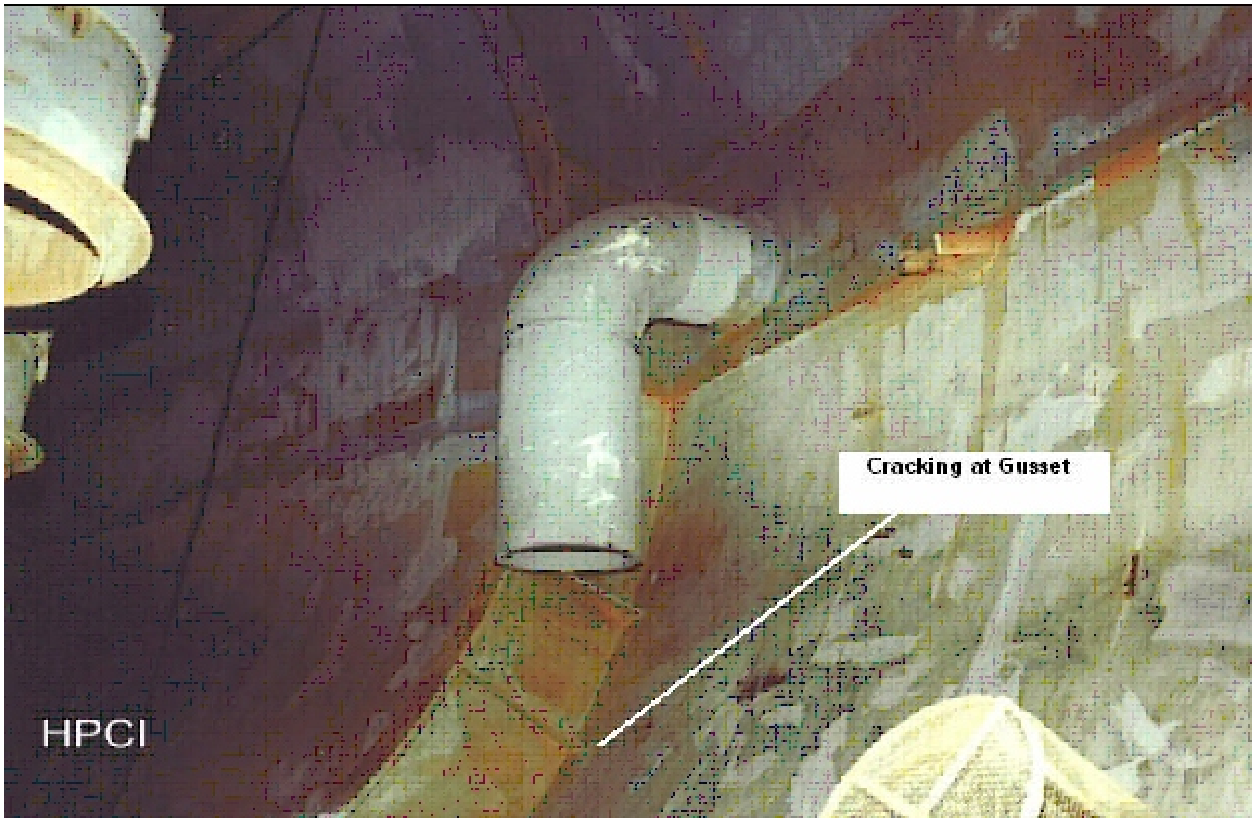

Figure 3