Inspection of Containment Structures in the U.S.A.

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ABSTRACT

In the United States, until 1996, the containment structure visual inspections were performed as part of the containment integrated leak rate testing in accordance with Appendix J of 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors". In August 1996, the U.S. Nuclear Regulatory Commission amended its regulation, 10 CFR 50.55a, "Codes and Standards," to incorporate by reference Subsections IWE and IWL of Section XI of the ASME Boiler and Pressure Vessel Code for inspection of containment structures in light-water-cooled nuclear power plants. Subsection IWE provides the requirements for inservice inspection, repair, and replacement of Class MC pressure-retaining components (i.e., steel containments) and their integral attachments, and metallic shell and penetration liners of Class CC pressure-retaining components (i.e., concrete containments) and their integral attachments. Subsection IWL provides the requirements for preservice examination, inservice inspection, and repair of the reinforced concrete and post-tensioning systems of Class CC components.

The regulation endorsed the requirements of the 1992 Edition and 1992 Addenda of Subsections IWE and IWL with a few supplementary requirements for inspection of steel and concrete containments. A later amendment to the regulation also endorsed the use of the 1995 Edition and the 1996 Addenda of Subsections IWE and IWL with a consistent set of supplementary requirements. The regulation requires the owners of nuclear power plants to complete the first period inspections of their containments by September 9, 2001.

This paper discusses (1) instances of containment degradation found before 1996, as a result of visual examinations and ad hoc findings, (2) the need for periodic inservice inspections, and (3) issues raised during the implementation of the regulatory requirements.

1. INTRODUCTION

The containment structure (containment) is a vital engineered safety feature of a nuclear power plant. In normal operating conditions, the containment is subjected to operating and environmental stressors such as ambient pressure fluctuations, temperature variations, earthquakes, ice, and windstorms. In some containment designs, the principal leaktight barrier is surrounded by another structure, such as a shield wall or a shield building, which protects the containment from some of the external influences. The mechanical stresses and strains generated by transients under normal conditions and the effects of high-probability ($>10^{-2}$) external influences are a small fraction of the limiting conditions for which the containment is designed. However, the fatigue life of the containment can be affected by the significant number of cycles of such low-stress transients. The containment is also subjected to various types of internal degradation (aging degradation) depending on the inherent characteristics of the materials, fabrication processes, and construction methods. The rate and extent of the degradation are influenced by sustained environmental conditions such as temperature, humidity, water leakage, expulsion of chlorides, and acidic spills. Thus, the performance of a containment under design basis conditions as well as under higher loads due to severe accidents and earthquakes would be influenced by the containment's inherent capability and the various stresses and degradation mechanisms that act on it.

2.. SUMMARY OF MAJOR CONTAINMENT DEGRADATIONS

2.1 Concrete Containments

The following instances of significant degradation of reinforced concrete and prestressed concrete containments have been reported during the operation of the plants:

- In January 1985, anchor heads of vertical tendons at Joseph M. Farley Nuclear Power Plant were found to be cracked; three anchor heads were broken in pieces (Ref. 1). Metallographic and fracture examinations showed that the failures resulted from hydrogen stress cracking of the anchor head material. The contributing factors were the high hardness of the anchor head material [American Iron and Steel Institute material 4140 with a

Rockwell hardness (Rc) between 38 and 44], free water in the grease caps, and high stresses in the anchor heads. All of the cracked and broken anchor heads were replaced with newly designed ones, and the affected tendons were retensioned.

- The reactor vessel of Fort St. Vrain, which is not operating, is the only prestressed concrete reactor vessel in the United States. Approximately 6 percent of the prestressing tendons were equipped with load cells for continuous monitoring of prestressing forces. During an inspection in 1984, several wires in vertical, hoop, and bottom crossheads were found to be corroded, and some had failed. After an extensive investigation, the licensee determined that the most likely contributor to the wire failures was microbiological corrosion (Ref. 2).

- The base of the cylindrical wall of the containment at R. E. Ginna Nuclear Power Plant is unique in that it is supported on a series of neoprene pads, each consisting of two layers of neoprene separated and covered by three carbon steel shim plates. The pads are installed on the thickened basemat (ring beam) along the circumference of the cylinder. Thus the cylinder can move in and out under the loads (temperature, internal pressure, earthquakes) by deforming the neoprene pads. The vertical post-tensioning tendons are enclosed in stainless steel bellows when passing through the gap between the cylinder and the ring beam. The vertical tendons are coupled to the rock anchors at the bottom of the ring beam. An NRC staff member visiting the site noticed puddles of water near the base of the cylinder along the outside circumference of the cylinder. He noticed degradation of the asphalt coating on the wall and the insulation material between the pads. The licensee dewatered the area and constructed effective berms to alleviate water accumulation and subsequent degradation of the pads, tendons, and concrete. The licensee also reanalyzed the structure and confirmed through deformation measurements during leak rate testing that the containment behavior at the base (under the postulated loadings) will be within the established criteria.

- The Mark I containment structures at Brunswick Steam Electric Plant (the drywells and tori of the two units) are constructed of reinforced concrete, with steel liner plates on the inside surfaces serving as leaktight membranes. During a routine inspection in January 1993, an NRC inspector found the liner plate of the Unit 2 drywell to be corroded at various spots at the junction of the base floor and the liner. The licensee later found additional corrosion areas in both the units. The sealing material along the circumference had degraded and allowed water to accumulate at the junction. Some corrosion of the liner plates in the tori had been found in earlier inspections. In the drywells, the licensee cleaned the gaps, repaired the corroded plate areas as necessary, and resealed the gaps with dense silicon elastomer. For the corroded areas in the tori, the licensee confirmed that the noncorroded thickness of the plates is adequate for leaktightness under the postulated loading conditions. The licensee periodically monitors the tori.

- During the 20-year surveillance of the prestressing system of the Calvert Cliffs Nuclear Power Plant, Unit 1 (June–July 1997), the licensee (Baltimore Gas & Electric Company) found a low liftoff value of the prestressing force for one of the three randomly selected vertical tendons. The low liftoff value was attributed to the uneven shim stack heights on the two opposite sides of the anchor-head. In accordance with the requirement in the plant's Technical Specification (TS), the licensee tested two vertical tendons adjacent to this tendon. During the liftoff testing of one of these tendons, noises were heard that indicated some of the tendon wires might have broken. A visual examination of the tendon showed that three wires had broken at 12.7–17.2 cm (5–7 in) below the bottom of the anchor-heads. Further examination of the wires at the top of other vertical tendons revealed additional breakage. The licensee expanded the liftoff testing and visual examination to 100 percent of the vertical tendons. Similar degradation of other vertical tendons was found. As a part of its corrective actions, the licensee replaced the degraded tendons and regreased all the vertical tendons.

Several incidents of higher-than-expected prestressing losses, grease leakage through concrete, and liner bulges have been observed in the prestressed concrete containments of operating reactors. The incidents are summarized in References 3 and 4.

2.2 Steel Containments

The U.S. Nuclear regulatory Commission (NRC) has received several reports of incidents of steel containment corrosion. They are briefly described in the following paragraphs.

- After observing and monitoring the water leakage around various containment penetrations and floors for more than 5 years at Oyster Creek Nuclear Plant, the owner of the plant took extensive ultrasonic thickness measurements of the drywell shell to find out if the shell was degraded (Ref. 5). The measurements were taken at various locations near the sand cushion and at a higher elevation. They showed that approximately 7.6 mm (0.3 in.) of the metal had been lost from the nominal 29.2 mm (1.15 in.) metal thickness of the drywell shell in the sandcushion areas. Measurements just above these areas in the shell indicated no reduction in the metal thickness. The licensee removed the sand from the areas affected by accumulated water and coated the drywell areas to minimize the possibility of future corrosion.

- The inside surface of the Boiling Water reactor (BWR) Mark I containment torus shell at Nine Mile Point Nuclear Station, Unit 1, which was designed and constructed as uncoated, was evenly corroded below the required nominal thickness in some areas of the torus (Ref. 6). The torus was locally pitted on the inside surface. The overall corrosion rate of the inside surface of the torus wall was estimated to be more than double the expected rate of 0.04 mm/year (1.57x10-3 in./year). The licensee periodically monitors the extent of corrosion to ensure that the remaining thickness of the torus meets the design requirements.

- The Mark I containment torus at FitzPatrick Nuclear Plant was corroded to varying degrees after the coating on the inside surface of the torus wall became severely degraded. The licensee periodically monitors the extent of corrosion to ensure that the remaining thickness of the torus shell in the corroded areas meets the design requirements.

- In 1989 the owner of the William B. McGuire and Catawba nuclear power plants reported significant coating damage and base metal corrosion on the outer surfaces of the steel shells of Pressurized Water Reactor (PWR) ice-condenser containments near the annulus floor levels (Ref.7). The steel shells of the containments have a nominal thicknesses of 25.4 mm (1 in.) near the annulus floors. The degraded areas varied in length from 5 to 10 m (16.5 to 33 ft) along their circumference. The average depth of corrosion was measured as 2.2 mm (8.7×10^{-2} in.), with pits up to 3.1 mm (0.12 in.) deep. The licensee took corrective actions to alleviate moisture accumulation and repaired the shell to ensure it was of adequate thickness.

- In 1990, the utilities reported additional instances of corrosion at higher elevations in the drywell of the Oyster Creek containment and on the inside surfaces of the McGuire containment (Ref. 8). The licensees have taken actions to alleviate the possibility of future corrosion, and periodically monitor the affected areas.

3. FRAMEWORK FOR SYSTEMATIC INSPECTIONS OF CONTAINMENTS

The findings summarized in Section 2 above were identified by containment tendon inspections according to Regulatory Guide 1.35 (Ref. 9), by ad hoc inspections by the plant personnel, or NRC inspectors, or, to a lesser degree, by inspections required before the leak rate testing of the containments pursuant to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Even when found, there was no formal procedure for documenting the degradations and implementing corrective actions.

The ASME Subgroup on metal containment inspection and the Working Group on concrete containment inspection of Section XI of the Boiler and Pressure Vessel Code have been developing the requirements for inservice inspection (ISI) of steel and concrete containments since 1979. The first edition of Subsection IWE on inservice inspection of metal containments was published in 1981, and the first edition of Subsection IWL for inservice inspection of concrete containments was published in 1989. The NRC staff's review of the 1989 edition of Subsection IWE found that it was more focused on the weld inspections in metal containments than on base metal inspections. As described in Section 2.2 above, all significant degradation of metal containments involved the base metal corrosion. Consequently, the NRC staff found the use of the 1989 Edition of Subsection IWE unacceptable for ISI of metal containments. In the 1992 Edition of the Code, the Subgroup revised the requirements to reflect experience.

In August 1996, the NRC amended its regulation, 10 CFR 50.55a, "Codes and Standards," (the rule) to incorporate by reference the 1992 Edition and the 1992 Addenda of Subsections IWE (Ref. 10) and IWL (Ref. 11) of Section XI of the ASME Boiler and Pressure Vessel Code for inspection of containment structures in light-water cooled nuclear power plants. Subsection IWE provides the requirements for inservice inspection, repair, and replacement of Class MC pressure retaining components (i.e., steel containments) and their integral attachments, and metallic shell and penetration liners of Class CC pressure retaining components (i.e., concrete containments) and their integral attachments.

Subsection IWL provides the requirements for preservice examination, inservice inspection, and repair of the reinforced concrete and post-tensioning systems of Class CC components.

4. DISCUSSION OF THE INSPECTION REQUIREMENTS

The following discussion illustrates how various provisions of Subsections IWE and IWL (1992 Edition 1992 Addenda), as augmented by the requirements of the rule minimize the potential of significant containment degradation going undetected.

4.1 Subsection IWE

Article IWE-1000, "Scope and Responsibility," lists the components of containment covered by the requirements of the subsection. It also specifies the containment areas that must be available for inspection and defines the containment areas that need augmented examination. Several incidents summarized in Section 2.2 above forced the plant owners to assess the potential for degradation in inaccessible areas, where the environment could cause degradations. IWE-1240 requires the plant owners to make such an assessment. This requirement is further augmented in 10 CFR 50.55a(b)(2)(ix)(A).

Article IWE-2000, "Examination and Inspection," provides the requirements for preservice examinations, inspection schedules, and examination requirements of in-scope containment components. In the 1992 Edition of the Subsection, the details of component examination requirements are specified in six Tables. The rule made the implementation of the requirements for surface examination of pressure retaining and dissimilar welds (Examination Categories E-B and E-F) optional, considering the operating experience related to these welds. However, when the visual examinations or containment leak rate test indicate degradations, the owners are required to follow up the findings with applicable surface and volumetric examinations. During implementation, the industry groups argued that the IWE examination requirements for seals and gaskets and pressure-retaining bolting duplicate the requirements of 10 CFR Part 50, Appendix J. The NRC staff received a number of requests for relief from implementing these requirements and approved the requests on their merits.

Article IWE-3000, "Acceptance Standards," describes the acceptance standards for the examinations performed in accordance with IWE-2000. The acceptance standards for nondestructive, and visual examinations, as well as the acceptance standards for each Examinations Category are provided. Most acceptance standards are given in qualitative terms; for example, when VT-3 examinations of non-coated steel surfaces indicate evidence of cracking, discoloration, wear, pitting, excessive corrosion, arc strikes, gouges, surface discontinuities, dents and other signs of surface irregularities, the suspect areas shall be accepted based on an engineering evaluation or corrected by repair or replacement. Quantitative acceptance criteria are provided for two types of degradations: (1) when the thickness of the base metal is found to be reduced by more than 10% of the nominal shell (or liner) thickness, and (2) when the dissimilar welds indicate flaw sizes in excess of those specified in IWB-3514. Thus these types of acceptance standards could be characterized as condition-based rather than prescriptive. However, from the standpoint of detecting degradations, these acceptance standards are adequate.

The requirements for repair, replacement, and system pressure tests are provided in Articles IWE-4000, IWE-5000, and IWE-7000. IWE-5240 requires the plant owners to perform visual examination VT-2, after major repair or replacement (R/R). VT-2 examination requires the plant owners to perform a surface examination to detect leakage during a relevant leak rate test. The plant owners argued that the implementation of this requirement is an unnecessary burden as the examinations and tests performed after such major R/R as part of the respective construction code confirms the adequacy of the R/R. Recognizing the validity of the complaint, the NRC authorized plant-specific relief, and suggested the code committee to provide an alternative, that would not duplicate the surface examinations, at the same time to confirm the adequacy of the R/R activities.

4.2 Subsection IWL

Article IWL-1000, "Scope and Responsibility," describes the scope of the Subsection and lists the items exempt from inspection. The scope for inspection of pressure-retaining components of concrete containments, i.e. metallic liner and penetrations is excluded from the scope of this subsection. These pressure-retaining components are within the scope of Subsection IWE. Thus, the two subsections cover the inspection requirements for concrete and steel containments.

Article IWL-2000, "Examination and Inspection," provides the requirements for pre-service examination, personnel qualification, examination schedule and examination requirements for the containment concrete and the post-tensioning system. The concrete examination methods are as described for VT-3C and VT-1C requirements. These requirements are equivalent to the VT-3 and VT-1 examination requirements of IWA-2200 and IWA-2300 applicable to steel surfaces of Class 1, 2, and 3 and Class MC pressure-retaining components. For concrete surface examinations, the

article recommends that the examiners look for degradations described in ACI 201.1R (Ref. 14). The requirements for examining the post-tensioning system are based on Regulatory Guide 1.35 (Ref. 9).

Article IWL-3000,"Acceptance Standards," describes the acceptance standards for preservice and inservice examinations. For determining the acceptability of concrete surface degradations, the article relies on the judgement of the owner designated Responsible Engineer. Without making it a regulatory requirement, the NRC staff has advocated the use of ACI 349.3R (Ref. 15) for establishing the acceptance criteria for concrete surface degradations. For the unbonded post-tensioning system, the acceptance criteria are based on those in Reference 9.

The requirements for repairs, system pressure tests, and replacements are given in Articles IWL-4000, IWL- 5000, and IWL-7000.

Thus, Subsections IWE and IWL of Section XI of the ASME Boiler and Pressure Vessel Code, as supplemented by the rule, provide a reliable method of ensuring the structural integrity of the containments in the United States.

5. IMPLEMENTATION OF CONTAINMENT INSPECTION REQUIREMENTS

The ASME committees that developed the code requirements for inservice inspection of containments are comprised of engineers involved in design, construction, and inspection of nuclear facilities, and representatives of nuclear utilities, research laboratories and regulatory agencies. The impact of these requirements did not become apparent until the issuance of the "for comment" version of the rule in January 1994. The difficulties and hardships related to the various requirements of the 1992 Edition and the 1992 Addenda of Subsection IWE were not recognized until the time the utilities started developing plans for implementing the requirements of the rule.

The Nuclear Energy Institute (NEI), as a representative of the owners of nuclear power plants, with the assistance of the Electric Power Research Institute (EPRI), spearheaded the efforts to initiate a dialogue with the NRC. As a first step, NEI sought clarifications of issues related to (1) Repair & Replacement, (2) General Topics, (3) Implementation Schedule, (4) Acceptance Criteria, (5) Personnel Qualifications and Certification Topics, and (6) Augmented Examinations. These issues and their resolutions are documented in Reference 12. EPRI, later, identified a number of Code requirements that were either considered hardships without commensurate increases in the level of quality and safety, or where cost-effective alternatives might be available. These requirements and NRC's responses to the issues are documented in Reference 13.

In January 2000, the NRC amended the rule, incorporating by reference the use of the 1995 Edition with the 1996 Addenda of the Code. The 1998 Edition of Subsections IWE and IWL substantially changed the requirements for examination by delegating a number of responsibilities to the Owner of the plant. On a plant-specific basis, the NRC has permitted the use of the 1998 Edition of the Code with certain supplementary requirements. The NRC is in the process of amending the rule to incorporate by reference the 1998 Edition through the 2000 Addenda of the Code.

6. CONCLUDING REMARKS

Containment structures in the United States are designed and constructed to conservative design criteria and highly demanding quality assurance requirements. However, plant experience indicates that the containments are not immune to age-related degradations when subjected to adverse environmental conditions. This realization dictated a need for containment inspection in a systematic manner.

Implementing the containment inspection rule did encounter a number of challenges. However, with the constructive comments and suggestions from the industry, it was possible for the NRC staff to develop approaches which could enhance safety without imposing unnecessary hardships on the industry.

Disclaimer: Opinions expressed in this paper are those of the author and do not necessarily reflect the views of the Nuclear Regulatory Commission.

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