Assessing the Integrity of NPP Containment Pressure Boundaries

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ABSTRACT

Research is being conducted to address aging of the containment pressure boundary in light-water reactor plants. Objectives of this research are to (1) understand the significant factors relating corrosion occurrence, efficacy of inspection, and structural capacity reduction of steel containments and liners of concrete containments; (2) provide the U.S. Nuclear Regulatory Commission (USNRC) reviewers a means of establishing current structural capacity margins or estimating future residual structural capacity margins for steel containments, and concrete containments as limited by liner integrity; and (3) provide recommendations, as appropriate, on information to be requested of licensees for guidance that could be utilized by USNRC reviewers in assessing the seriousness of reported incidences of containment degradation. Activities include development of a degradation assessment methodology; reviews of techniques and methods for inspection and repair of containment metallic pressure boundaries; evaluation of candidate techniques for inspection of inaccessible regions of containment metallic pressure boundaries; establishment of a methodology for reliability-based condition assessments of steel containments and liners; and conduct of fragility assessments of steel containments with localized corrosion.

INTRODUTION

As of August 1999, 104 nuclear power reactors, producing about 20% of the electricity supply, were licensed for commercial operation in the U.S. The median age of these reactors is over 20 years, with 61 having been in commercial operation for 20 or more years. Expiration of initial operating licenses for several of these reactors will start to occur early in this century. Continuing the service of existing nuclear power plants (NPPs) through a renewal of their initial operating licenses provides a timely and cost-effective solution to the problem of meeting future electricity supply; however, a concern as plants approach the end of initial operating license period is that the capacity of the safety-related systems to mitigate extreme events has not deteriorated unacceptably due to either aging or environmental effects.

CONTAINMENT DESCRIPTION

Each boiling-water reactor (BWR) or pressurized-water reactor (PWR) unit in the U.S. is located within a much larger metal or concrete containment that also houses or supports the primary coolant system components. The basic laws that regulate the design (and construction) of NPPs in the U.S. are contained in Title 10, "Energy," of the Code of Federal Regulations (CFR). The reactor containment and associated systems are to provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. As such, the containment and associated systems are to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident (LOCA).

Concrete containments are metal-lined, reinforced concrete pressure-retaining structures that in some cases may be post-tensioned. The reinforced concrete containment shell, which generally consists of a cylindrical wall with a hemispherical or ellipsoidal dome and flat base slab, provides the necessary structural support and resistance to pressure-induced forces. Leak-tightness is provided by a thin steel liner (e.g., 6 mm) that is anchored to the concrete shell by studs, structural steel shapes, or other steel products. Depending on the functional design (e.g., large dry or ice condenser), NPP concrete containments can be on the order of 40 to 50 m diameter and 60 to 70 m high, with wall thicknesses from 0.9 to 1.4 m, dome thicknesses from 0.9 to 1.4 m, and base slab thicknesses from 2.7 to 4.1 m.

Currently operating metal containments in the U.S. are freestanding, welded steel structures that are enclosed in a reinforced concrete reactor or shield building. The reactor or shield buildings are not part of the pressure boundary and their primary function is to provide protection for the containment from external missiles and natural phenomena. Metal containments are typically fabricated using low-alloy or unalloyed steels. Typical metal containments range in diameter from 18 to 43 m and have shell thicknesses on the order of 25 to 51 mm.

TESTING, INSPECTION, AND MAINTENANCE AND LICENSE RENEWAL REQUIREMENTS

Testing

One of the conditions of all operating licenses for water-cooled power reactors is that the primary reactor containments shall meet the containment leakage test requirements set forth in Appendix J ("Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors") to 10 CFR Part 50. These test requirements provide for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components that penetrate containment of water-cooled power reactors, and establish the acceptance criteria for such tests. On September 26, 1995, the USNRC amended Appendix J to provide a performance-based option for leakage-rate testing. The amendment is aimed at eliminating prescriptive requirements that are marginal to safety and providing licensees greater flexibility for cost-effective implementation methods for regulatory safety objectives. Thus, either Option A—*Prescriptive Requirements* or Option B — *Performance-Based Requirements* can be chosen by a licensee to meet the requirements of Appendix J. Option B allows licensees with good integrated leakage-rate test performance histories to reduce the Type A (i.e., primary reactor containment overall integrated leakage rate) testing frequency from three tests in ten years to one test in 10 years. However, a general inspection of accessible interior and exterior surfaces of the containment, structures and components must be performed prior to each Type A test and during two other refueling outages before the next Type A test.

Inspection

Appendix J requires a general inspection of the accessible interior and exterior surfaces of the containment structures and components to uncover any evidence of structural deterioration that may affect either the containment structural integrity or leak-tightness. On August 8, 1996, the USNRC published an amendment to 10 CFR Part 50.55a ("Codes and Standards") to require that licensees use portions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for containment in-service inspection. The regulations were amended to assure that critical areas of containments are routinely inspected to detect and to take corrective action for defects that could compromise structural integrity. The amended rule became effective September 9, 1996 with a five-year implementation period. Specifically, the rule requires that licensees incorporate the 1992 Edition with the 1992 Addenda of Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," and Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants,“ of Section XI, of the ASME Code into their in-service inspection plans. In addition, several supplemental requirements with respect to the concrete and metal containments were included in the rule (e.g., expansion of evaluation of inaccessible areas of concrete containments to include metal containments and liners of reinforced concrete containments, and prevention of duplicate examinations required by both the periodic routine and expedited examination program).

Maintenance and License Renewal

Proper maintenance is essential to the safety of NPP containments, and a clear link exists between effective maintenance and safety. To reduce the likelihood of failures due to degradation, the "Maintenance Rule" was issued by the USNRC as 10 CFR 50.65 ("Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants") on July 10, 1991. As discussed in the rule summary, in order to maintain safety, it is necessary to monitor the effectiveness of maintenance, and to take timely and appropriate corrective action, when necessary, to ensure that the maintenance process continues to be effective for the lifetime of NPPs, particularly as plants age. The rule requires that plant owners monitor the performance or condition of structures, systems, and components (SSCs) against owner-established goals, in a manner sufficient to give reasonable assurance that such SSCs are capable of fulfilling their intended functions. It is further required that the licensee take appropriate corrective action when the performance or condition of a SSC does not conform to established goals. Subsequently, on May 8, 1995, the USNRC published a final rule amending 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," that contained requirements an applicant must meet to renew an operating license. The final rule is intended to ensure that important SSCs will continue to perform their intended function in the period of extended operation. Only passive, long-lived structures and components are subject to an aging management review for license renewal, and the USNRC license renewal review focusses on the adverse effects of aging.

DEGRADATION AND OPERATING EXPERIENCE

NPP containments provide resistance to external challenges and confinement of radioactive materials in the event of an accident. They are designed to withstand loads from a number of low-probability external and internal events, such as earthquakes, tornadoes, and LOCAs. Loads incurred during normal plant operation therefore generally are not significant enough to cause appreciable degradation. However, these structures are susceptible to aging by various processes, depending on the operating environment and service conditions, that can cause their strength and stiffness to deteriorate.

Degradation is considered to be any phenomenon that decreases the load-carrying capacity of the containment, limits its ability to contain a fluid medium, or reduces its service life. Service-related degradation can affect the ability of a NPP containment to perform satisfactorily in the unlikely event of a severe accident. The root cause for component degradation can generally be linked to a design or construction problem, inappropriate material application, a base- or weld-metal flaw, maintenance or inspection activities, or a severe service condition. Degradation of the containment metallic pressure boundary (i.e., steel containments and liners of reinforced concrete containments) can be classified as either material or physical damage. Material damage occurs when the microstructure of the metal is modified causing mechanical property changes (e.g. intergranular corrosion and low-temperature exposure). Material damage to the containment metallic pressure boundary is not considered likely, however. Physical damage occurs when the geometry of a component is altered by formation of cracks, fissures, or voids, or its dimensions change due to overload, buckling, corrosion, erosion; or formation of other types of surface flaws. Degradation due to either general or pitting corrosion represents the greatest potential threat to the containment pressure boundary. General corrosion can affect structural capacity by reducing the net section available to resist loads, and pitting can completely penetrate the component to compromise its leak-tight integrity.

As NPPs age, degradation incidences are occurring at an increasing rate, primarily due to environmental-related factors. Since 1986, there have been over 36 reported occurrences of corrosion of metallic pressure boundary components. In two cases, thickness measurements of the walls of steel containments revealed areas that were below the minimum design thickness, and two cases have been reported where corrosion has completely penetrated the liner of reinforced concrete containments. There have been four additional cases where extensive corrosion of the liner has reduced the thickness locally by nearly one-half [1]. Also, there have been incidences of transgranular stress corrosion cracking in bellows.

ASSESSMENT OF AGED/DEGRADED CONTAINMENT PRESSURE BOUNDARIES

As a part of the overall USNRC research program to benchmark existing design criteria and evaluate containment performance under severe accident conditions, research is being conducted related to condition assessment and repair practices, inspection of containment metallic pressure boundaries, and performance of degraded containments (i.e., corroded) during events at or beyond the design basis.

Condition Assessment, In-Service Inspection, and Repair Practices

Condition assessments are an essential element of continued service evaluations performed by qualified engineers and authorized personnel who determine the adequacy of components for intended use [2]. The decision-making process begins with an understanding of the in-service condition of each component or system. Condition assessments provide essential information for continued service evaluations and involve detecting damage, classifying types of damage present, determining root cause of the problem, and quantifying the extent of degradation. Knowledge gained from a condition assessment serves as a baseline for evaluating the safety significance of any damage present and defining in-service inspection programs and maintenance strategies. A degradation assessment methodology is available for characterizing the in-service condition of containment pressure boundary components [3].

If undetected, degradation effects may reduce the margin a containment has to accommodate accidents. An essential element in the assessment of the integrity (or in the determination of available safety margin) of a containment structure is knowledge of the damage state of its materials of construction. In-service inspections and testing are performed to measure the current state of damage. Application of nondestructive evaluation techniques while the component remains in-service (i.e., in-service monitoring) can provide valuable information for assessing the current condition of a degraded component, estimating its remaining useful service life, and making informed aging-management decisions. The most common nondestructive examination (NDE) techniques in civil structures are visual inspection, liquid penetrant, magnetic particle, ultrasonic, eddy current, and radiography [4].

Whenever damage is detected, corrective actions are taken to identify and eliminate the source of the problem and thereby halt the degradation process. When significant wall thinning, cracking, surface defects, or leakage is detected and containment structural or leak-tight integrity is potentially jeopardized, defective areas are either evaluated, repaired, or replaced before the plant is returned to service. The primary mechanism of concern to the containment pressure boundary is corrosion. Methods to prevent the occurrence of corrosion primarily include the application (or maintenance) of coatings to exposed steel that is at risk, and use of cathodic protection systems (i.e., impressed current or sacrificial anode). Repair methods generally include: (1) defect removal by mechanical means in which the unacceptable flaw is reduced and the resultant section thickness created by the removal process remains equal to at least the minimum design thickness; (2) repair welding in which the design section thickness is reestablished (e.g., cladding); and (3) component replacement with items that meet acceptance standards. Repair options for restoring damaged bellows include replacement of penetration assembly, bellows replacement, installation of new enveloping bellows, in-place welding repairs, removal of severe dents, and blending the surface. Detailed information on repair of the containment pressure boundary components is available [5].

Inspection of Containment Pressure Boundaries

Inspection of NPP structures can be difficult because there are a number of functionally different components in a variety of environments. In the previous section several techniques commonly used to assess civil structures were identified. Application of these techniques, however, generally requires that at least one surface of the component inspected be accessible and the techniques are most effective when an approach is utilized in which the structures have been prioritized with respect to such things as aging significance, environmental factors, and risk. Guidance on component selection is available [6,7]. Once the components have been selected for inspection, however, there are several locations in NPPs where performing the inspections may not be straight forward. Inspection of inaccessible portions of metal pressure boundary components of containments (e.g., fully embedded or inaccessible containment shell or liner portions, the sand pocket region in Mark I and II drywells, and portions of the shell obscured by obstacles such as platforms or floors) represents one of these conditions.

Embedded metallic portions of the containment pressure boundary may be subjected to corrosion resulting from groundwater permeation through the concrete; a breakdown of the sealant at the concrete-containment shell interface that permits entry of corrosive fluids from spills, leakage, or condensation; or in areas adjacent to floors where the gap contains a filler material that can retain fluids. Although no completely suitable technique for inspection of inaccessible portions of containment metallic pressure boundaries has been demonstrated to date, preliminary investigations have been conducted using high frequency acoustic imaging technology [8,9], magnetostrictive sensor technology [10], and ultrasonic multimode guided wave techniques [11].

Exploratory analytical and experimental simulations have been conducted to investigate the feasibility of high frequency acoustic imaging techniques for detecting and locating thickness reductions in the metallic pressure boundaries of NPP containments. The analytical study used an elastic layered media code (OASES) to perform a series of numerical simulations to determine the fundamental two-dimensional propagation physics. The analytical simulation suggests that for the case of steel-lined concrete containments, the thin steel liner and additional concrete backing contribute to give unacceptable loss of signal to the concrete. For embedded steel containments, analytical simulation suggests that significant degradations (i.e., $\geq$2 mm) of containment thickness below the concrete/air interface provide reasonable backscatter signal levels of approximately –15 dB. The experimental study utilized a commercial ultrasonic testing system to carry out several full-scale tests of steel plates 25 by 203 by 914 mm, some partially embedded in concrete. Scattered signals from simulated degradations of different size and shape (i.e., rectangular, semi-circle, and "V" shaped) were investigated (Fig. 1). By altering the test setup to bistatic measurements, the "self noise" (e.g., inner wedge reverberant field and through thickness standing echoes) was eliminated. Overall, experimental results showed that the measurement system displayed a dynamic range of 125 dB with measurement variability less than 1-2 dB. Based on these results, a 4-mm-deep round-faced degradation embedded in 30 cm of concrete has expected returns of –73 dB relative to input and should be detectable.

Magnetostrictive sensors are devices that launch guided waves and detect elastic waves in ferromagnetic materials electromagnetically to determine the location and severity of a defect based on timing and signal amplitude. The feasibility of applying magnetostrictive sensor technology to inspection of plate type materials and evaluating its potential for detecting and locating thickness reductions in the containment metallic pressure boundary has been investigated. Pulse-echo sensor data for notches ranging from 10- to 30-cm long are presented in Fig. 2. The notches were placed into a 6.11–m-long by 1.23–m-wide by 6.35-mm-thick plate at a distance equal to 4.06 m from the probe end of the plate. Limited analytical studies suggest that a low-frequency A_0 mode wave (below approximately 0.5 MHz-mm, that corresponds to approximately 40 kHz in a 12.7-mm-thick plate or 20 kHz in a 25.4-mm-thick plate) would be best suited for inspection of containment pressure boundaries that are either backed on one or both sides by concrete. Results indicate that guided waves provide an effective means of inspection of the metallic pressure boundary in a NPP and are capable of performing global, long-range inspection of plates, including areas that are difficult to access because of the presence of other equipment or attachments, or the presence of concrete on one or both sides.

A limited investigation to demonstrate the feasibility of using the multimode guided wave technique for identification and location of thickness reductions in the metallic pressure boundary of NPP containments has been conducted. Experiments utilized a bare plate with two defects, a plate with concrete but no defects, and a plate embedded in concrete with one defect. Each plate was 25 by 203 by 914 mm. The specimens provided a benchmark for studying several aspects of guided wave inspection, including sensitivity, transmission ability across defects, inspection reliability, and penetration ability. The plates were interrogated using both horizontal shear [electromagnetic acoustic transducer (EMAT)] and Lamb (piezoelectric transducer) guided waves. Horizontal shear (SH) guided waves have particle displacements in the shear horizontal direction, which is perpendicular to the propagation direction. The grid distance of the EMATs used in the experiment was 12.7 mm, which determines that the corresponding frequency for generating the non-dispersive SH wave mode is around 200-250 kHz. The obtained waveforms are shown in Fig. 3 for the plate containing two defects and the plate containing a defect embedded in concrete. The reflected echoes for the bare plate with two defects indicates that both defects can be detected by using the SH waves. Unlike SH waves, Lamb waves have particle displacements in both parallel and perpendicular directions to the propagation direction. The frequency and wedge angle determines the generated Lamb wave mode. In order to obtain a fairly uniform energy distribution across the plate thickness for Lamb waves, a 38° wedge angle was utilized. The tone burst frequency was 565 kHz. Figure 4 presents the pulse echo signals for the plate with two defects and the plate containing one defect embedded in concrete. For the plate embedded in concrete but without defects, multiple echoes are received from the plate-concrete interface before the backwall echo (BWE). This indicates one of the disadvantages of this Lamb wave mode in that it is sensitive to the plate-concrete interface.

Fragility Assessment of Steel Containment Subjected to Internal Pressurization

Fragility analysis is an essential ingredient of a fully coupled risk analysis. A probabilistic safety assessment (PSA) is a structured framework for evaluating uncertainty, performance, and reliability of an engineered facility. The move toward quantitative risk assessment has accelerated in recent years as the benefits have become increasingly apparent in many fields [12]. The recently issued Regulatory Guide 1.174 [13] defines the NRC's position on risk-informed decision-making regarding proposed changes to the licensing bases of operating nuclear plants.

The PSA process is initiated with the identification of limit states (LS) or conditions in which the system ceases to perform its intended function(s) in some way. For structural components and systems in NPPs, such limit states may be either strength or deformation-related, as large (inelastic) deformations affect the integrity or operability of mechanical or electrical systems that are attached to or otherwise interface with the structure. With the limit states identified, the limit state probability is expressed as,

$$P[LS] = \sum P[LS|D = x]\, P[D = x] \tag{1}$$

in which D describes the intensity of demand on the system (hazard), and $P[LS|D = x]$ is the conditional limit state probability, or the fragility, of the system.

The fragility displays, in probabilistic terms, the capability of an engineered system to withstand a specified event with intensity x (sometimes referred to as a review-level event), one that often is well in excess of the design-basis event. Thus, it defines safety margins probabilistically against specific identified events for decision and regulatory purposes in a manner that effectively uncouples the system analysis from the hazard analysis. The fragility modeling process leads to a median-centered estimate of system performance, coupled with an estimate of the uncertainty in performance. The fragility of a structural component or system often is modeled by a lognormal cumulative distribution function (CDF), described by,

$$FR(x) = \Phi\, [\ln (x/m_C)/\beta_C] \tag{2}$$

in which $\Phi(\ )$ = standard normal probability integral, m_C = median capacity (expressed in units that are consistent with the demand, x, in Eqn. (1), and β_C = logarithmic standard deviation, which is approximately equal to the coefficient of variation (COV) in capacity, V_C, when $V_C < 0.3$ and provides a measure of uncertainty in capacity.

The strengths of steel and concrete structural materials and components are random variables, and their median (or mean) strengths are well in excess of the nominal values specified for NPP design [14]. If these median strengths are used in structural analysis in lieu of specified nominal strengths, one often can obtain a reasonable estimate of the median capacity, m_C, in Eqn. (2) [15]. The uncertainty in capacity displayed by Eqn. (2) arises from numerous sources. Some of these uncertainties (denoted by COV β_R) are inherent (aleatory) in nature, and are essentially irreducible under current engineering analysis procedures. Other uncertainties (denoted by COV β_U) arise from assumptions made in the analysis of the system and from limitations in the supporting databases. Such knowledge-based (epistemic) uncertainties depend on the quality of the analysis and data, and generally can be reduced, at the expense of more comprehensive (and costly) analyses. The role of epistemic uncertainty on fragility can be displayed in one of two ways. In the first, a family of fragilities is generated, one for each modeling assumption. In the second, the aleatory and epistemic uncertainties are combined in the form $\beta_C^2 = \beta_R^2 + \beta_U^2$, and only one (mean) fragility curve is generated The second approach is taken herein.

The fragility assessment is illustrated using the Sequoyah Unit 1 containment, which was selected because it was one of the reference plants in the NUREG-1150 risk study [16] and has been thoroughly analyzed in several independent studies. Sequoyah Unit 1 is a PWR ice-condenser containment designed for an internal pressure of 74 kPa. Its internal diameter is 35 m, the springline height is 35 m, and the elevation of the apex of the spherical dome is 53 m. The steel in the shell is A516/Grade 60 plate, varying in thickness from 35 mm at the basemat to 12 mm at the springline. Vertical and circumferential stringers are welded to the exterior of the shell at approximately 1.2 m vertical and 3 m horizontal intervals.

The containment must confine radioactive material in the event of an accident, so the performance limit is loss of shell integrity or ability to perform this essential function [17]. This performance limit must be related to structural limit states that can be identified from nonlinear finite element analysis, along with local or general structure or material failure criteria.

Tests of internally pressurized scaled model containments have indicated that the governing failure mode invariably is one of tensile instability. In a recently published study [15], the tensile instability limit state for containment fragility analysis was defined by,

$$\varepsilon_p = \varepsilon_f f_1 f_2 f_4 \tag{3}$$

In which ε_p = effective plastic strain, ε_f = uniaxial limit strain, f_1 = factor to correct uniaxial limit strain for triaxiality effects, f_2 = factor that accounts for bias and uncertainty in the finite element analysis, and f_4 = factor to account for the reduction in steel ductility as a result of corrosion. The commercially available nonlinear finite element program ABAQUS was used to perform the numerical experiments of the pressurized containment leading to the fragilities. This analysis is described in more detail elsewhere [15]. Four cases are illustrated: the uncorroded (as-built) case, which is used as a benchmark, cases where there is postulated 10 % and 25% loss of containment shell thickness behind the ice basket, and the case where there is 50% loss adjacent to an upper floor.

The fragility of the containment in the as-built condition is shown in Fig. 5. The mean (and median) capacity is 455 kPa, and the logarithmic standard deviation β_C is 0.04. The median is consistent with values obtained elsewhere by other investigators. The estimated 5-percent and 2-percent exclusion limits are 427 kPa and 421 kPa, respectively. For comparison, median fragilities based on simplified criteria such as first yielding or 2% strain in the circumferential direction, which can be modeled by simple yield analysis of the shell, are 290 kPa and 365 kPa, respectively. These simplified analyses lead to conservative estimates of the margin of safety. Fragilities for the other cases are also presented in the figure. In comparison, for a postulated 25% loss of shell thickness behind the ice basket the mean (median) capacity is 386 kPa, and the logarithmic standard deviation has increased to 0.06. The 5-percent and 2-percent exclusion limits for this postulated degraded condition have decreased to 352 kPa and 345 kPa, respectively, or by approximately 18% from the as-built condition.

SUMMARY AND DISCUSSION

Activities that address aging of the containment pressure boundary in light-water reactor plants are summarized. Testing and inspection requirements are summarized. Current and emerging nondestructive examination techniques and a degradation assessment methodology for characterizing and quantifying the amount of damage present are noted. The use of time-dependent structural reliability analysis methods to provide a framework for addressing the uncertainties attendant to aging in the decision process are discussed (i.e., methods help provide assurances that degraded metallic pressure boundaries will be able to withstand future extreme loads during the desired service period with a level of reliability that is sufficient for public safety). The impact of aging (i.e., loss of shell thickness due to corrosion) on steel containment fragility for a pressurized water reactor ice-condenser plant is presented. Results indicate that values at the 2- and 5-percentiles exclusion limits for this postulated degraded condition are well in excess of the design basis of 74 kPa (i.e., by factors of 5.7 in the as-built condition and 4.7 under the postulated degradation). These large margins of safety are due to a number of factors. Design material strengths are substantially less than the likely values in service; design is based on the assumption of elastic behavior, which does not account for additional capacity beyond yielding that is provided by the large ductility of carbon steels; conservative assumptions are made regarding structural response; and factors of safety in the range 1.5 to 2.0 are used, depending on the safety check. Thus, even in a deteriorated condition, the containment still may retain sufficient capacity integrity to withstand challenges from events at or beyond the original prescriptive design basis with a high level of confidence. Other studies have shown that a decrease in the median fragility of 15% is likely to lead to an increase in the limit state probability of a factor of two or less [18]. Such increases would not mandate immediate corrective action, but would require that a periodic inspection program be initiated to track cumulative impacts of such degradation over time [13].

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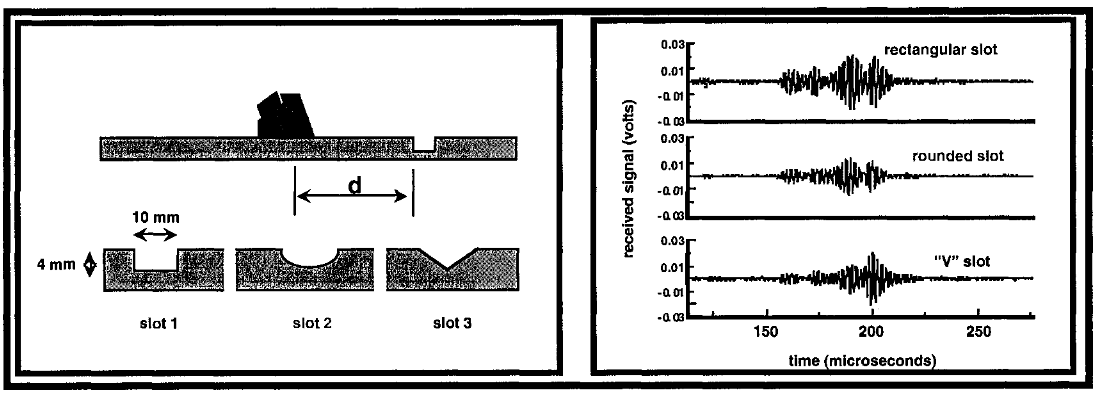

Figure 1 Experimental results evaluating defect geometry.

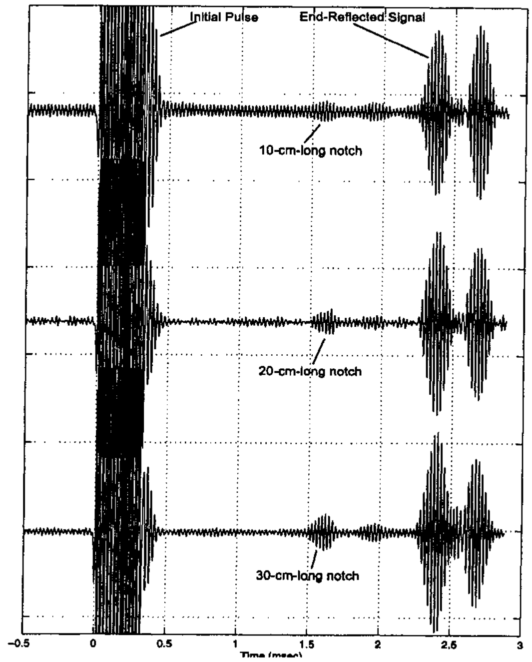

Figure 2 Pulse-echo magnetostrictive sensor data.

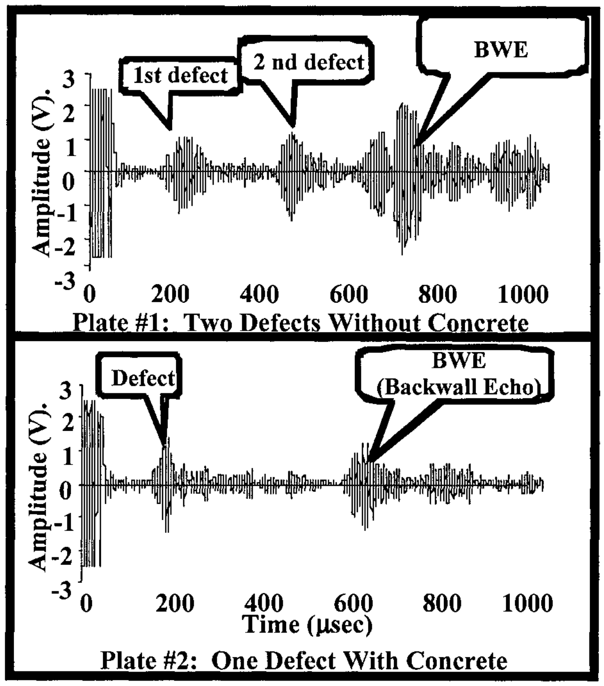

Figure 3 Experimental results using horizontal shear waves.

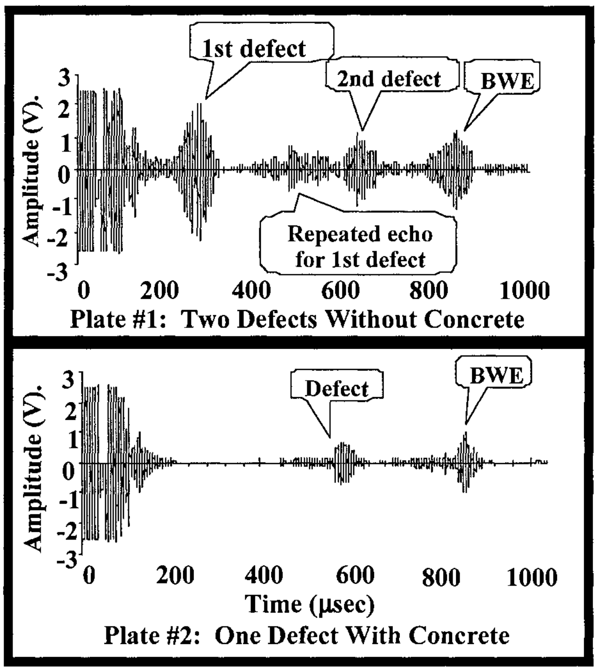

Figure 4 Experimental results using Lamb guided waves.

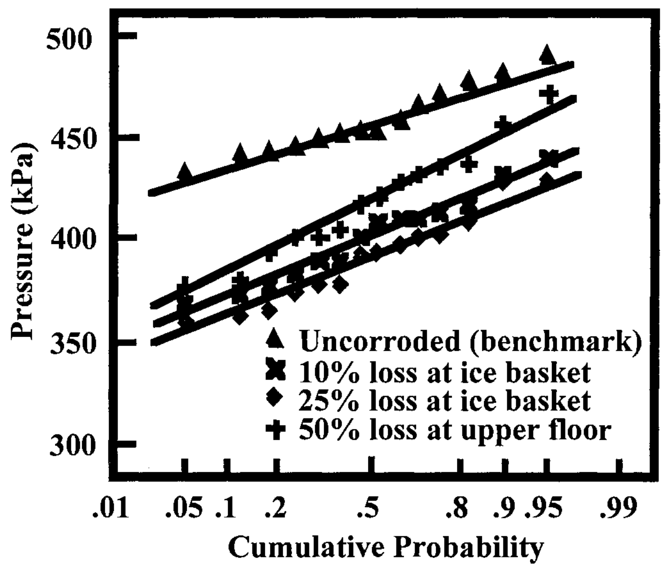

Figure 5 Containment fragilities for postulated corrosion.