ULTRASONIC TESTING IN NUCLEAR PLANTS
УЛЬТРАЗВУКОВИЙ КОНТРОЛЬ НА АТОМНИХ СТАНЦІЯХ

by Anmol Birring and James Williams

The approach for nondestructive testing inspections in nuclear power plants differs significantly from nonnuclear industries. In most industries, inspections are planned and specified by owners or insurance companies. Nuclear plant inspections are, however, regulated and must follow certain processes, regulatory requirements, and required codes. Codes classify components and specify inspection methods, coverage, and inspection intervals. With regard to ultrasonic testing (UT), there is a broad range of applications for nuclear plant components. Applications vary from simple corrosion/erosion scanning to more complex techniques that require UT of difficult-to-inspect materials. These include dissimilar metal welds of materials such as stainless steel, nickel alloy, and cast stainless steels. In addition, UT inspection can include tests of complex geometries such as nozzles and turbines. Furthermore, due to the nature of the nuclear plant operation, the inspection processes must be demonstrated to validate reliability and qualify equipment and personnel. Finally, this article highlights the importance of using notches as calibration reflectors when using refracted longitudinal waves and large aperture phased array probes to improve resolution.

Підхід до неруйнівного контролю на атомних електростанціях суттєво відрізняється від неатомних галузей. У більшості галузей промисловості обстеження плануються та призначаються власниками або страховими компаніями. Проте обстеження (контроль) на атомних станціях регулюються нормативними вимогами, положеннями обов’язкових стандартів та мають відповідати встановленим процедурам. Стандарти встановлюють класифікацію елементів, види, обсяги та періодичність контролю. Що стосується ультразвукового контролю (УЗ), існує широкий спектр застосувань для елементів атомних станцій. Застосування варіюються від простого виявлення корозії/ерозії до складніших методів, які потребують УЗ матеріалів, що важко піддаються контролю. До них належать композитні зварні шви з таких матеріалів, як нержавіюча сталь, нікелевий сплав і лита нержавіюча сталь. Крім того, УЗ контроль може застосовуватись до елементів зі складною геометрією, таких як сопла та турбіни. Крім того, через особливості експлуатації атомної станції необхідно продемонструвати процедури перевірки для підтвердження надійності та кваліфікації обладнання та персоналу. Нарешті, у цій статті підкреслюється важливість використання вирізів як калібрувальних відбивачів при використанні заломлених поздовжніх хвиль і фазованих решіток з великою апертурою для покращення роздільної здатності.

Introduction

As of March 2022, there were 56 nuclear plants in the US with 93 operating reactors. Of these, 62 are pressurized water reactors (PWRs) and 31 are boiling water reactors (BWRs) (NRC 2022). Safe operation of nuclear plants requires application of reliable nondestructive testing (NDT) inspections. Operation, inspection, and maintenance of nuclear plants is regulated by the US Nuclear Regulatory Commission (NRC) and therefore inspections must meet NRC requirements.

Application of NDT has reduced forced outages and extended plant lives. In addition, improved technologies have reduced planned outage periods, thereby increasing the availability factor. The intent of this article is a general introduction of the UT techniques applied in nuclear plants.

Components of Nuclear Power Plants

There are two basic types of nuclear power plants in the US: PWR and BWR. The major difference between them is the means of steam generation. In a PWR, steam is generated in a two-step process. In the first step, heat produced by fission of the nuclear fuel is transferred to reactor coolant (pressurized water) in the reactor pressure vessel and circulated to the steam generators to produce steam. The steam generator is a large heat exchanger that transfers heat from the reactor coolant to the water in the secondary circuit. The main difference between the PWR and BWR is that there is no separate steam generator and the steam from the boiling water reactor goes straight to the turbines.

Major components of the PWR are described in the Reactor Concepts Manual available on the US-NRC website (NRC 2020). They include the reactor vessel, primary piping, reactor coolant pump, pressurizer, steam generator, secondary piping, turbines, and generators. In the US, the three basic designs of the PWR are the Westinghouse, Combustion Engineering, and the Babcock and Wilcox. A typical PWR

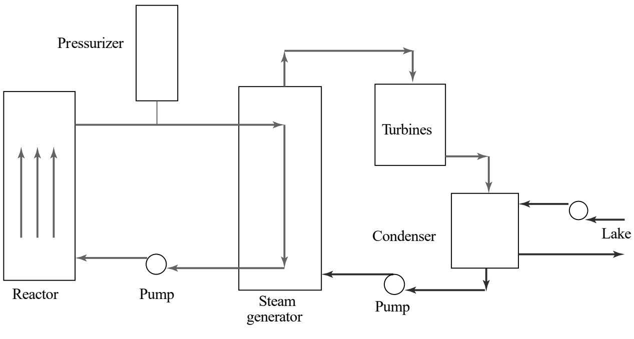

Figure 1. Pressurized water reactor circuit. Primary circuit is between the reactor and steam generator. Secondary circuit includes the steam generator, turbines, and condenser.

Рис. 1. Схема реактора з водою під тиском. Головний циркуляційний контур між реактором і парогенератором. Другий контур включає парогенератор, турбіни і конденсатор circuit is shown in Figure 1. A BWR plant does not include a steam generator and secondary piping.

Components in a nuclear plant are classified as per the International Atomic Energy Agency (IAEA) safety classification (STUK 2000). They are Class 1, Class 2, and Class 3, with Class 1 being the most critical. Inspection requirements depend on the component class. Inspection frequency and NDT methods for each class are given in ASME XI as subsections IWB, IWC, and IWD, respectively (ASME 2019b).

Reactor

A reactor is a large heavy-wall vessel that contains the reactor core with nuclear fuel. The nuclear fuel by the process of controlled fission produces heat, which is transferred to the coolant. This cylindrical vessel has a hemispherical bottom and a removable top head. The top head is removed for refueling and makes access available for inspections, including NDT. The reactor vessel is constructed from manganese molybdenum steel, and all surfaces that come in contact with the coolant are clad with stainless steel. The typical PWR temperature range for outlet and inlet leg is 326 °C (619 °F) to 292 °C (558 °F), respectively, and at a pressure of 15.6 MPa (2263 psi).

A BWR operates at a steam temperature of about 287.8 °C (550 °F) and a pressure of 8.62 MPa (1250 psi).

A reactor is a Class 1 component. UT is applied extensively to the reactor pressure vessel shell welds and the nozzle welds connecting the reactor vessel to the primary piping.

Primary Piping

Primary piping in the PWR carries the coolant from the reactor to the steam generator and back to the reactor via the reactor coolant pump. In the case of BWR, the primary piping carries the steam directly from the reactor to the turbines. Primary piping materials are mostly cast stainless steel and wrought stainless steel. For PWRs, primary piping legs between the reactor are short and can be single-spool spun-cast stainless steel pipes. Primary piping is a Class 1 component.

BWR Recirculation and Residual Heat Removal Piping

BWR piping is mostly 304 and 316 stainless steel.

Steam Generator – PWR

A steam generator in a PWR plant is a heat exchanger that transfers heat from the reactor coolant and converts water to steam for the turbines. The primary piping nozzles and tubing of steam generators are Class 1 components.

Secondary Piping – PWR

Secondary piping in a PWR carries the steam from the steam generator to the turbines. Secondary piping material is normally carbon steel. Secondary piping is mostly Class 2.

Emergency Core Cooling Water Piping

Emergency cooling water piping circulates cooling water for emergency systems. This piping can be carbon steel or aluminum bronze. This piping is a Class 3 component. Some plants have experienced leaching in aluminum bronze piping.

Turbines

Turbines convert the energy in the steam to a rotary motion for power generation. Damage mechanisms can include stress corrosion cracking, corrosion fatigue, and stress fatigue. UT is performed on the steam turbine rotors and blade attachment areas. Common materials for turbines are steels with various amounts of nickel, chromium, molybdenum, and vanadium.

Condensers

Before the steam is sent back to the steam generator, it must be condensed. This operation is performed in a condenser, which is basically a large heat exchanger. Water from a lake, sea, river, or canal passes through the condenser tubes. Steam passes around the tubes and is condensed. Various tube materials are used in condenser tubes: these include copper nickel alloys, brass, and titanium. Tube inspection is done by eddy current testing. Almost no UT is performed in the condensers.

Regulations and Codes

Operation of nuclear plants is regulated by the NRC. The NRC publishes the code of federal regulations. Some of the most important regulations pertaining to nuclear plants are in 10 CFR, Part 50 (NRC 2006a). Furthermore, 10 CFR, § 50.55 is another publication that provides codes and standards applicable to nuclear plants (NRC 2006b). The code applicable to NDT is Section XI of the ASME Boiler and Pressure Vessel Code (BPVC). This code provides the information on in-service inspections for various classes of components. Required inspection and inspection intervals for the Class 1 components can be found in section IWB-2000 of ASME, Section XI. The inspections listed in ASME, Section XI are categorized as visual, surface, or volumetric. UT is a volumetric examination. The inspection interval depends on the type of component.

Specific nondestructive examination methods are described in ASME BPVC, Section V (ASME 2019a). Specific NDT procedures to meet the ASME codes are written by the inspecting organizations.

Application of Phased Array Technology in the NDT Industry

A major upgrade in UT beginning in 2005 has been the introduction of portable phased array ultrasonic testing (PAUT) systems. Previously, all weld inspections were done using single or multiple single-element probes. Phased array systems that existed (if any) were quite bulky and not commonly used.

Mechanized scanning with a single-element probe was done by raster scans and the data reconstructed into B- and C-scans. Single-element probes would include refracted angles such as 45°, 60°, and 70°. Instead of three separate probe angles, phased arrays use a single probe that sweeps the beam over a range of angles and displays the image as a sectorial scan. In addition, phased array data is taken in a line scan instead of a raster scan, thus simplifying the scanning process. The phased array sectorial presentation provides a cross-sectional view that is much easier to evaluate and interpret as compared to reconstructed B- and C-scans from conventional single-element probes. Phased array technology is now routinely used for inspection of reactor vessel welds, steam line welds, steam turbines, and other components such as bolts.

In-Service Inspections

This section covers the in-service inspection of the various components of nuclear systems.

Pressurized Water Reactors

UT is performed on nuclear plant components for volumetric inspections. A PWR shell is generally 100 to 150 mm thick carbon steel with stainless steel clad. Phased array ultrasonic inspection is done from inside the PWR. The carbon steel shell welds are inspected using shear waves. Required inspection intervals can be found in section IWB-2000 of ASME BPVC, Section XI.

All shell, head, shell-to-flange, head-to-flange, and repair welds are subjected to a 100% volumetric examination during the first interval. Ensuing test intervals require fewer beltline region, head, and repair weld tests. All nozzle-to-safe and butt welds with dissimilar metals (such as ferritic steel nozzle to stainless steel or to heat-resistant nickel chromium alloy) are subjected to volumetric and surface tests at each interval. Inspection volume coverage of each weld is given in Section XI. For similar and dissimilar weld metal welds, NPS 4 (DN100) or larger nozzles, and piping, the examination volume is limited to one-third of thickness from the inner diameter (ID) surface. A typical nozzle-to-safe end dissimilar metal carbon steel alloy 182 stainless steel weld configuration is shown in Figure 2. Alloy 182 is a nickel alloy weld material. This material is anisotropic, and shear waves are ineffective for such welds. Dissimilar metal nickel alloy welds are inspected by refracted longitudinal wave (L-wave) mode. Inspection is done from the inside surface or the outside surface depending on access (Figure 3). Refracted L-wave can penetrate such materials, but the inspection is limited to one-half the vee path of the beam due to mode conversion upon reflection. Another important aspect of PAUT is probe frequency and active aperture, which define resolution and play an important role in inspection quality. A good inspection requires high-resolution images with good flaw definition. Higher frequencies and a large active aperture reduce beam spread, focus the beam, improve resolution, and result in high-definition images (Birring 2021).

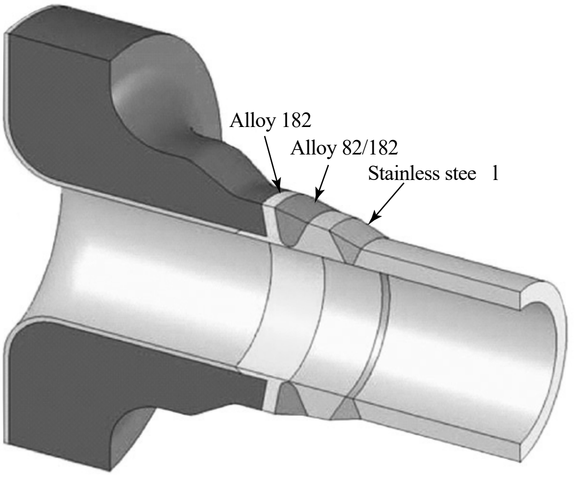

Figure 2. Nozzle-to-safe end dissimilar metal weld. Buttering (shown in yellow) is nickel alloy 182. Weld material (shown in red) is nickel alloy 82/182 weld electrode. Inspection of these welds is done from the inside of the nozzle using phased arrays in the refracted L-wave mode.

Рис. 2. Композитний зварний шов від насадки до безпечного кінця. Стик — це нікелевий сплав 182. Зварювальний матеріал — зварювальний електрод із нікелевого сплаву 82/182. Контроль цих зварних швів виконується з внутрішньої сторони патрубків за допомогою фазованих решіток у режимі заломленої L-хвилі

One of the issues related to PWR is primary water stress corrosion cracking (PWSCC). This is discussed in detail in NUREG/CR-7187 (NRC 2014). Alloy 82/182/600 exposed to reactor coolant water (or steam) is susceptible to PWSCC. Cracking initiates from the inside surface of such welds. Inspection of such internal cracks initiating from the inner surface is typically done from the outside surface using phased arrays. However, due to limitations in some configurations, the inspection is done from the ID surface.

All studs and threaded stud holes in the closure head studs undergo surface and volumetric tests at specified inspection intervals. Any integrally welded attachments are required to have surface (or volumetric) tests of welds at specified test intervals. UT of the pressurized water reactor is performed after removing the head and internals and using a remotely operated vehicle (ROV) or by placing a tripod-type device with extended arms on top of the reactor. The reactor is filled with water during the examination. The front end of these systems includes the cameras, phased array probe, and other sensors. This system is used for inspection of vessel seam welds, flange-to-shell welds, nozzle-to-shell welds, nozzle-to-piping welds, meridional (lower head) welds, shell-to-lower head welds, and nozzle inner radius. ROV systems are neutrally buoyant and are controlled remotely. One such system

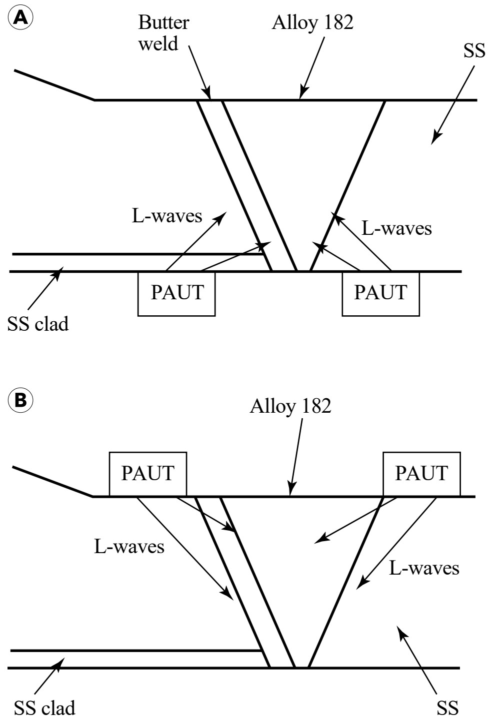

Figure 3. PAUT of dissimilar metal welds using refracted longitudinal waves: (a) from ID surface; (b) from OD surface.
Рис. 3. УЗФР (ультразвуковий контроль за допомогою фазованих решіток) зварних швів із різнорідних металів з використанням заломлених поздовжніх хвиль: (а) зсередини; (b) ззовні

is shown Figure 4. Prior to inspections, the systems are qualified on actual mock-ups of pressurized water reactors to ensure detection of the discontinuities.

Boiling Water Reactors

BWR vessels consist of a cylindrical shell and a hemispherical bottom head containing both the control rod penetrations and the in-core instrumentation penetrations. The top head is hemispherical and includes nozzles with bolting flanges attached. BWRs by design have limited access for internal in-service inspection.

A main issue with BWRs (and especially the older reactors) is the limited access for internal in-service inspection. Typically, a large percentage of the vessel's weld lengths were exempted as inaccessible.

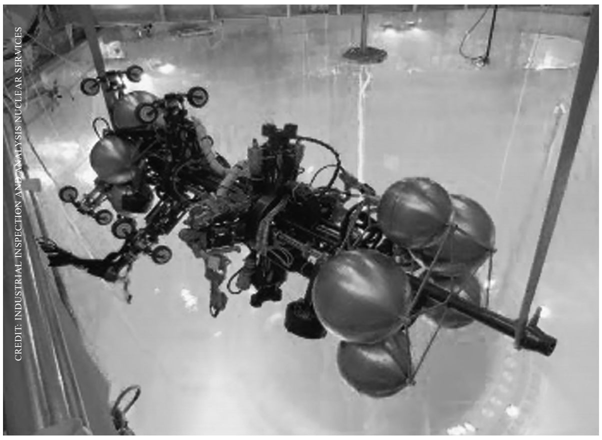

Figure 4. Nuclear reactor nozzle scanner.
Рис. 4. Сканер патрубків ядерного реактора

The main difficulty in the inspection of boiling water reactors is access to the welds below the top of the core elevation in the boiling water reactor. The inaccessibility results from the narrow clearance between the vessel sidewall and shroud surrounding the core barrel. Increased coverage of the reactor interior requires the use of maneuverable robotic devices (Birring et al. 1990).

The internal clearance is typically in the 150 to 300 mm (6 to 12 in.) range at the top and can be less at the bottom of the vessel. Clearance varies from the top (inlet) to the bottom (outlet) of the jet pumps. For this reason, BWR inspections have always been limited to elevations above the top core level. The limited examination is, however, not enough for the evaluation of aging plants and for license renewal. For this reason, there is always a need to increase the test coverage of welds in BWRs. To increase ultrasonic inspection coverage, several companies in the nuclear industry have designed robotic tooling to access these welds from the inside surface. Several inspection agencies funded programs to conduct additional inspections to increase coverage. These systems combine mechanical scanners with advanced UT data acquisition and analysis systems. Software tools are integrated into these systems for acquisition, real-time data processing, and test tracking. The tracking tool provides a real-time display of the surface location of the scan module by using a drawing of the reactor or examination area.

During a test, the scan module's surface location is recorded on the selected test area drawing. When the inspection is complete, the user can check coverage, display, evaluate, and create reports.

Primary System Piping

The design of coolant piping differs according to reactor design. This can include wrought stainless steel, ferritic steel with stainless weld overlay, and ferritic steel with metallurgically bonded and cast stainless steel. Manual UT is the method applied for inspection of welds in stainless steel pipes. Welds are inspected with both shear and longitudinal waves. Shear waves work well for base metal defects such as inner surface diameter toe cracks. Refracted longitudinal waves are applied to detect discontinuities inside the weld volume that are difficult to penetrate with shear waves. Cast stainless steel is by far the most difficult piping material to inspect. Preferential grain structure in cast stainless steel makes shear waves ineffective. Longitudinal waves are affected less than shear waves. Angle-beam longitudinal waves are therefore used for the testing of cast stainless pipes instead of shear waves. Typically, dual-element low-frequency refracted L-wave transducers or phased arrays in the refracted L-wave mode are applicable.

A significant issue of BWR piping has been the occurrence of intergranular stress corrosion cracking (IG-

SCC). The piping is mostly 304 and 316 stainless steel. A combination of material, environment, and stress can lead to IGSCC. The first noted incidence of IGSCC was in December 1965, and since then cracking has been found in several plants (NUREG 2000). IGSCC is found in the heat-affected zone of the welds. Cracking has been found in small-diameter piping, large-diameter piping, and in Inconel-safe ends welded to austenitic stainless steel. Detection of IGSCC is difficult because of its undefined shape, which follows the grains. Substantial efforts in research and development have led to the development of test procedures and inspector training for detection and sizing of such cracking. These programs are conducted on samples with actual IGSCC. Inspectors are first trained in UT techniques and then tested on samples with actual discontinuities.

Carbon steel pipes with stainless steel weld overlay clad can also be difficult to inspect. Unless the weld overlay is machined, the clad results in random reflections from the weld overlay at the ID surface, making inspection in the second leg practically impossible.

Steam Generators

Steam generators are heat exchangers that are part of the PWR loop. High-temperature, high-pressure water from the reactor enters the steam generator. Hot water flows through the steam-generator tubes to produce steam for the turbines. Steam flows from the secondary loop to the turbines.

Inspections conducted in steam generators focus on the inlet/outlet nozzles, which are located at the bottom of the steam generator. Inspection is done by inserting an inspection device through the nozzle. These devices have special tooling that first passes through the nozzles and once inside, expand to inspect the nozzle. The front end of this tooling includes cameras, phased array probes, and eddy current probes. As always, the tooling must be demonstrated and qualified on mock-ups.

Secondary Piping

A degradation mechanism found in PWR and BWR carbon and low-alloy steel piping is flow-accelerated corrosion (FAC). This corrosion mechanism occurs when the protective oxide layer dissolves by the fast-flowing water. FAC occurs when three conditions are met: temperature, pH level, and steel composition.

Two common methods applied for this inspection are ultrasonic thickness testing and radiographic testing (RT). The main advantage of RT is that the inspection can be done without the removal of insulation.

Emergency Core Cooling Water Piping

An example of material degradation in piping is shown in Figure 5. Material degradation can result in loss of strength and toughness. In some cases, material degradation can result in a reduction in ultrasonic velocity and an increase in ultrasonic backscatter. In addition, material degradation may or may not include corrosion. Detection of such damage when associated with corrosion can be challenging. A form of material degradation is selective leaching in aluminum bronze. Leaching occurs in cast aluminum bronze components in water service, whereby aluminum leaches out and forms aluminum hydride, leaving a copper-rich volume. Advanced UT techniques have been used with some success for the detection of selective leaching. These include measuring velocity drop using time of flight diffraction software and full matrix capture.

Turbines – Solid Rotor

The common rotor for steam turbines is the solid (or boreless) rotor. Bored rotors were previously used as they ensured removal of defective material from the center, which was left during the steel making and forging process. With improvements in steel making, the bored rotors have been phased out and replaced by solid rotors (Viswanathan 1989). The main advantage of a solid rotor is its lower level of stress compared to the bored rotor. The lower stress level makes the boreless rotor more tolerant of larger discontinuities.

A solid rotor is tested by using a combination of L-wave and transverse wave transducers. Inspection of solid rotors requires that the selected angles cover the maximum possible material volume. Specific scan plans are developed to ensure coverage. As an alternative, phased arrays that sweep a range of angles can be used. Transverse cracking in low-pressure rotors initiates from corrosion pits and can grow during service by fatigue corrosion. Transverse cracking is detected by magnetic particle testing on the outside rotor surface.

Turbines – Blade Attachment

The mechanism of crack initiation and growth in the attachment of turbine disk to blade (steeples) depends on three variables: operating temperature, stresses, and environment. Stress corrosion cracking, combined with fatigue, is the primary mechanism. Initially, cracking

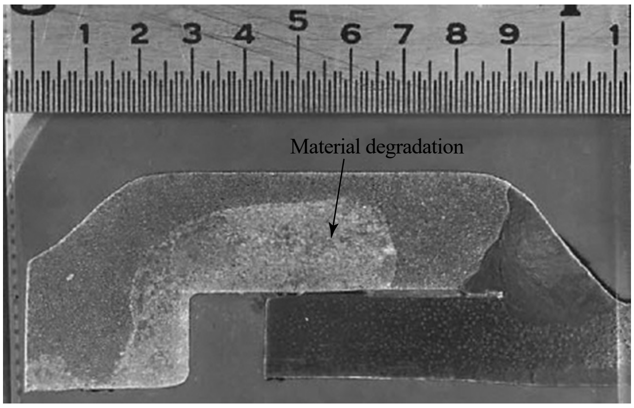

Figure 5. Example of material degradation. Material degradation can result in a decrease of ultrasonic velocity and an increase in backscattering.

Рис. 5. Приклад деградації матеріалу. Деградація матеріалу може призвести до зменшення швидкості ультразвуку та збільшення зворотного розсіювання

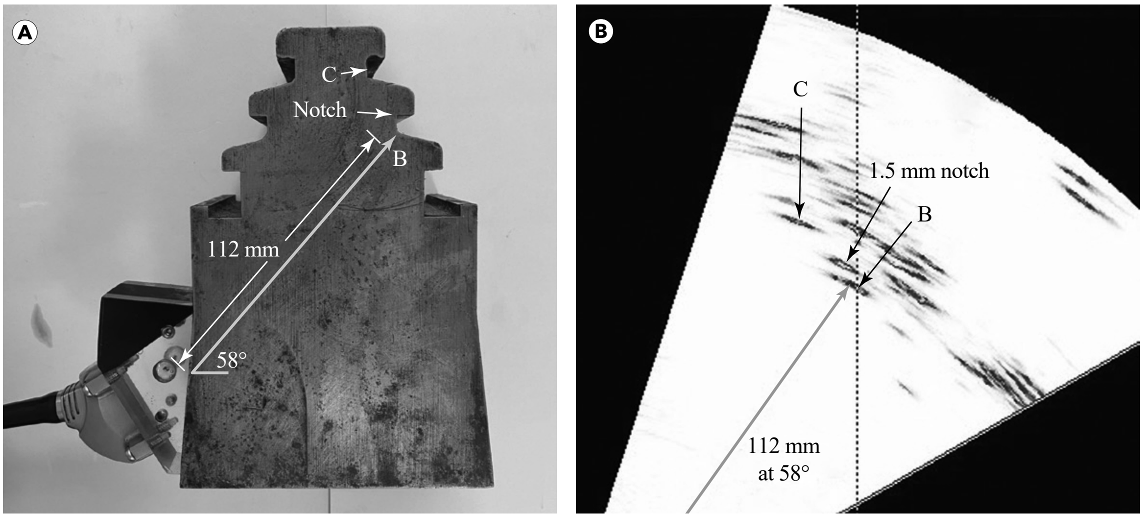

Figure 6. Phased array inspection of turbine disk blade attachment area: (a) PAUT probe setup on the calibration block with notches; (b) phased array image showing detection of 1.5 mm notch. Inspection requires high-resolution PAUT probes to resolve between the notch and geometries B and C. The probe is 5 MHz, 32×1.0 mm.

Рис. 6. Перевірка зони кріплення лопатки диска турбіни фазованою решіткою: (a) установка зонда УЗФР на калібрувальний блок з насічками; (b) зображення з фазованою решіткою, що демонструє виявлення виїмки 1,5 мм. Для контролю потрібні зонди УЗФР з високою роздільною здатністю, щоб визначити межі між виїмкою та геометріями B і C. Зонд має частоту 5 МГц, 32×1.0 мм

in a low-pressure rotor grows slowly by stress corrosion. When the stress intensity KI exceeds the threshold stress intensity Kh, crack growth is predominantly due to fatigue. Crack growth rates in this mode are significantly high because of vibratory loads. Generally, failure can be imminent when the threshold for fatigue crack growth Kth is reached. Therefore, it is important that nondestructive tests detect cracks before their stress intensity reaches Kth. The test method applied to detect steeple cracking depends on the geometry. Dovetail-shaped turbine blades are inspected by UT only. Phased array or multiple single-element transducers are used for inspection. Side-entry steeples allow access for surface inspection.

Figure 6 shows UT inspection of blade attachment areas using phased array. The PAUT probe is placed at a specific location so that the area of interest is illuminated. The presence of a crack will show as a discrete signal. For the inspection, the PAUT probe is placed on a stationary arm. Data is acquired while the turbine rotates on turning rolls.

NDT Reliability and Performance Demonstration Initiative

Two industries that mandate NDT reliability are aerospace and nuclear, but both apply it differently. The aerospace industry uses tests on a large number of flawed and unflawed samples to validate a 90/95 reliability (that is, 90% probability of detection with a 95% confidence level). The nuclear industry uses performance demonstration. NRC 10 CFR, § 50.55a provides quality assurance and reliability requirements when NDT is performed in nuclear power plants and requirements for performance demonstration initiative (PDI). Details of the performance demonstration are given in Mandatory Appendix VIII of ASME XI. Appendix VIII, «Performance Demonstration for Ultrasonic Examination», was incorporated into ASME XI in the 1989 edition. Specifically, PDI requires procedures, equipment, and personnel to demonstrate reliability and be qualified by performance demonstration on pressure-retaining components. To simulate real-life inspections, performance demonstration is done on mock-up samples. Mock-ups can include piping, nozzles, reactor vessel welds, and so forth with embedded flaws. Personnel for detection and sizing are qualified for detection if the results of the performance demonstration satisfy the detection requirements of ASME XI, Appendix VIII. As an example, in one case, personnel are qualified if no surface-connected flaw greater than 0.25 in. (6.35 mm) in height is missed and no embedded flaw greater than 0.50 in. (12.7 mm) is missed. Through-wall depths must be sized to 0.15 in. (3.81 mm) root mean square (RMS) and length sized to 0.75 in. (19.05 mm) RMS. Details for performance demonstration are also provided for dissimilar metal welds and welds where examination from both sides is not possible. PDI requires that NDT personnel must be recertified on a three-year interval. Performance demonstration and qualification is conducted at the Electric Power Research Institute in Charlotte, North Carolina.

Nuclear Plant Aging

Nuclear plants were built with a design life of 40 years. However, with high costs of plant retirement and replacement, a need arises for extending plant life past 40 years (NRC 2021). This requires quantitative assessment of aging as related to component integrity. Major categories of aging mechanisms are radiation embrittlement, fatigue, intergranular and irradiation-assisted stress corrosion cracking, corrosion, and corrosion erosion (IAEA 2003, 2005; Shah and Liu 2002). These mechanisms result in degradation of material, cracking, and loss of material.

Material degradation is caused by embrittlement and resulting loss of toughness. Toughness can be assessed by a surveillance program, where samples placed in the reactor are removed at specified time intervals for conducting tensile and charpy tests. Ultrasonic methodology such as velocity changes and scattering can be applicable on a case-by-case basis. The primary volumetric NDT method for cracking is UT. This can include conventional manual UT, tip diffraction, and PAUT. The main issue with inspection is not just testing, but reliability. Manual UT is highly operator dependent and should be used carefully after assessing technician skill. Techniques based on crack tip signals are not applicable for austenitic steels and cast stainless steel. Creeping wave method is another approach used sometimes for surface inspections. Outer diameter (OD) creep wave probes are high-angle refracted L-wave probes. Inner surface ID creep is shear wave mode converted to high-angle L-wave. There are no creeping waves in elastic media, and the term «creeping wave» is nothing but a misinterpretation of high-angle refracted longitudinal signals (Achenbach 1999; Blanshan and Ginzel 2008). Obsolete and incorrect methods result in loss of inspection reliability and confidence and should be discarded.

PAUT is the primary methodology for ultrasonic inspections. Even with PAUT, there can be quality issues if proper probes are not selected. Small-aperture PAUT probes, such as 10 mm to inspect 25 mm thickness, will result in large beam spread and poor resolution. For high-resolution and high-quality PAUT, a probe active aperture of at least 0.8X the thickness for shear waves and 1X the thickness for refracted longitudinal waves is recommended (Birring 2021).

To achieve the highest resolution, high frequencies such as 5 MHz should be used for both carbon steel and nickel alloy steels. Therefore, when inspecting a 30 mm thick alloy 182 weld, inspection should be done using a 5 MHz, 32×1 mm probe with a 32:128 phased array machine. A 64:128 phased array instrument should be used when using a 64-element probe on heavy-wall vessel thickness. Lower frequencies should be used only when sound penetration can be a problem in materials, such as cast stainless steel.

Another issue with refracted L-wave is the choice of calibration reflectors. Codes allow both side-drilled holes or notches as calibration reflectors for piping. They work well for shear waves as they produce almost equivalent sensitivity. This is not the case with refracted longitudinal waves, where L-wave mode converts on inner surface notches and results in a significantly reduced signal as compared to side-drilled holes. The sensitivity difference between notches and side-drilled holes can be as high as 16 dB at a refracted angle of 45° (Birring 2016). Therefore, when using refracted L-wave, notches should be used for surface sensitivity and side-drilled holes for volumetric sensitivity.

Advanced and reliable NDT technologies are therefore required for the assessment of aging and plant life extension.

Conclusions

Unlike the nonnuclear industry, where selection and application of inspection methods are made by owners and insurance companies, nuclear plant component inspections are regulated. The regulations apply to inspection method, coverage, and inspection intervals. The regulations and code require that methods, equipment, and personnel prove reliability of the test on mock-ups and samples with flaws. Further, inspectors must requalify periodically. The main challenge with nuclear plant inspection is aging and plant life extension past 40 years of design life. NDT methods play an integral and important role in this license-renewal exercise. Specifically, application of PAUT is gaining wider usage and replacing conventional ultrasonic single-element probes.

With regard to effective application of phased arrays, two important points are highlighted. First is the selection of phased array probes with a large aperture and higher frequency to achieve high resolution. The second pertains to selection of calibration reflectors when using refracted L-wave.

Notches and side-drilled holes should be used separately to establish sensitivity — notches for ID surface sensitivity and side-drilled holes for volumetric sensitivity.

ACKNOWLEDGMENTS

The authors wish to thank Jesse Delgado, Nuclear Services at Industrial Inspection and Analysis, San Antonio, Texas, for providing information on the ROV system for nozzle inspection.

AUTHORS

Anmol Birring: NDE Associates Inc., Houston, TX; nde@nde.com

James Williams: South Texas Electric Generating Station, Wadsworth, TX; jwilliams@stpegs.com

CITATION

Materials Evaluation 80 (6): 24-30
https://doi.org/10.32548/2022.me-04269

REFERENCES

1. Achenbach, J.D., 1999, *Wave Propagation in Elastic Solids*, eighth impression, Elsevier, Amsterdam, Netherlands
2. ASME, 2019 a, *Boiler and Pressure Vessel Code, Section V*, Nondestructive Examination, American Society of Mechanical Engineers, New York, NY
3. ASME, 2019b, *Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, Division 1, Rules for Inspection and Testing*

of Components of Light-Water Cooled Plants, American Society of Mechanical Engineers, New York, NY

4. Birring, A., 2016, «Selection of Calibration Reflectors for Corrosion Resistant Alloy Piping», *Materials Evaluation*, Vol. 74, No. 3, pp. 360-366
5. Birring, A., 2021, «Optimizing Probe Active Aperture for Phased Array Weld Inspections», *Materials Evaluation*, Vol. 79, No. 8, pp. 797-804, https://doi.org/10.32548/2021.me-04220
6. Birring, A.S., E. Peterson, E.A. Franke, and A.E. Crouch, 1990, «Robotic Devices for NDE of Component Interiors», *Proceedings from the Tenth Annual EPRI NDE Information Meeting*, Palo Alto, CA
7. Blanshan, B., and E. Ginzel, 2008, «The Truth behind Creeping Waves», *Materials Evaluation*, Vol. 66, No. 5, pp. 465-470
8. IAEA, 2003, IAEA-TECDOC-1361: *Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety, Primary Piping in PWRs*, International Atomic Energy Agency, Vienna, Austria
9. IAEA, 2005, IAEA-TECDOC-1470: *Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: BWR Pressure Vessels*, International Atomic Energy Agency, Vienna, Austria
10. NRC, 2000, NUREG-1719: *Pipe Cracking in U.S. BWRs: A Regulatory History*, US Nuclear Regulatory Commission, Washington, DC
11. NRC, 2006a, *10 CFR, Part 50 – Domestic Licensing of Production and Utilization Facilities*, US Nuclear Regulatory Commission, Washington, DC
12. NRC, 2006b, *10 CFR, § 50.55a, Codes and Standards*, US Nuclear Regulatory Commission, Washington, DC
13. NRC, 2014, NUREG/CR-7187: *Managing PWSCC in Butt Welds by Mitigation and Inspection*, US Nuclear Regulatory Commission, Washington, DC
14. NRC, 2020, «Pressurized Water Reactor (PWR) Systems», Reactor Concepts Manual, USNRC Technical Training Center for Educators, US Nuclear Regulatory Commission, Washington, DC, last updated 25 November 2020
15. NRC, 2021, *10 CFR, Part 54 – Requirements for Renewal of Operating License for Nuclear Plants*, US Nuclear Regulatory Commission, Washington, DC
16. NRC, 2022, «Power Reactors», US NRC website (nrc.gov/reactors/power.html), US Nuclear Regulatory Commission, Washington, DC, last updated 31 March 2022, accessed April 2022
17. Shah, YN., and YY. Liu, 2002, «Generic Aging Management Programs for License Renewal of BWR Reactor Coolant System Components», 10th International Conference on Nuclear Engineering (ICONE 10), Arlington, VA
18. STUK, 2000, YVL 2.1: *Nuclear Power Plant Systems, Structures and Components and their Safety Classification*, Radiation and Nuclear Safety Authority, Helsinki, Finland
19. Viswanathan, R., 1989, *Damage Mechanisms and Life Assessment of High Temperature Components*, ASM International, Materials Park, OH

Premission to Reprint 25.07.2022:
The American Society for Nondestructive Testing, Inc.
CITATION: *Materials Evaluation* 80 (6): 24-30
https://doi.org/10.32548/2022.me-04269