The Nucleus ***60**, No. 1 (2023) 86-97*

www.thenucleuspak.org.pk

The Nucleus
ISSN 0029-5698 (Print)
ISSN 2306-6539 (Online)

Role of In-Service Inspections in Nuclear Power Plants (NPPs)

Mahmood Khan[1,2*], Muhammad Umer Farooq Awan[1,3], Khurram Lal[1], Muhammad Gulab[1], Shabbir Ahmed[1], Zaheer Ahmed[1]

[1]*National Centre for Non-Destructive Testing (NCNDT), Islamabad, Pakistan*

[2]*Institute of Space Technology (IST), Islamabad Pakistan*

[3]*Pakistan Institute of Engineering and Applied Sciences (PIEAS), Islamabad, Pakistan*

ABSTRACT

The National Centre for Non-destructive Testing (NCNDT) since its inception in 1995 has been providing Pre-Service Inspections (PSI) and In-Service Inspections (ISI) services to the nuclear power plants (NPPs) in the country. In Pakistan, a total of six NPPs are operational with a cumulative capacity of 3620 MWe. All the efforts are made to ensure the safe and reliable operation of NPPs and the provision of an uninterrupted supply of electricity to improve the life of the common man. While extreme care is taken during the design, construction and commissioning of the structures, systems and components, a constant health monitoring system has to be ensured throughout the life (i.e., 40-50 years) of NPPs in accordance with the design parameters. The plant-specific codes & standards explain the ISI methods and acceptance criteria for critical life-limiting components, pressure boundary components and welds etc. Recently remotely controlled, advanced mechanized inspection systems for the inspection of reactor pressure vessels (RPV) and steam generators (SGs) have been procured. These systems are being used during the re-fueling outages (RFOs) of NPPs. Different conventional and non-conventional NDT methods were used for inspection. The components where unacceptable indications are revealed by ISI are repaired, replaced, or isolated. The baseline data generated during PSI is used for the comparison and surveillance. The observations made during ISI are subjected to thorough review and analysis by qualified experts to obtain assurance that unacceptable degradation in component quality is not occurring and it remains fit for service.

Keywords: *PSI: Pre-Service Inspection, In-Service Inspection, NPP: Nuclear Power Plant, NCNDT: National Centre for Non-Destructive Testing, NDT: Non-Destructive Testing*

1. Overview of Nuclear Power Plants (NPPs) in the World

Our world's temperatures are rising due to human activities and now this climate change is threatening every aspect of human life. Fossil fuels are the major contributor (approximately 75 %) to the emissions of these greenhouse gases responsible for climate change [1]. Drastic reduction of the greenhouse gas emissions is unavoidable whereas the energy access and economic opportunity for billions of people is expanding simultaneously.

The world is looking for clean energy [2]. Climate change, energy storage and variable energy output are some of the challenges faced for the long-term use of renewable resources (solar, wind etc.) [3]. Nuclear power plants are promising solution as they provide carbon-free energy [4]. These plants have the lowest carbon footprint and are reliable and cost-effective [5]. Nuclear energy requires less fuel (0.5 kg of nuclear fuel produces the same energy as 1000 tons of coal), can provide an interrupted supply of power for a longer time and is a plausible solution to meet the increasing energy demand [6]. Around 10 % of the world's electric power is generated by a total of 442 nuclear power reactors that are operational worldwide whereas around 52 nuclear power plants are under construction [7, 8]. Consequently, the value of operating NPPs has increased and NPP life management, especially service life extension has become a worldwide issue.

The goal of this paper is to place non-destructive evaluation (NDE) in the context of NPP life management and life extension and to show its increasing role in ensuring plant long-term operation. The part of the NCNDT in the PSI & ISI of NPPs in Pakistan and recent advancements in the development of indigenous mechanized inspections are also discussed.

2. Nuclear Power Plants in Pakistan

There are seven NPPs in Pakistan with a total capacity of 3620 MWe. Of these seven NPPs, six are operational and are pressurized water reactors (PWRs) whereas the remaining is a pressurized heavy water reactor (PHWR) and is permanently shut down. The oldest nuclear power plant K-1, the only PHWR, started its operation in 1971. It is a CANDU type reactor which had a capacity of 137 MWe. After completing 50 years of successful and safe operation it was permanently shut down in 2021.

Four of the six operational plants are 300 MWe two-loop PWR reactors based on CNP-300 and are called C series reactors namely C-1, C-2, C-3 and C-4. These are situated in Mianwali and started operation in 2000, 2011, 2016 and 2017.

Fig. 1: Aerial view of Nuclear Power Plants in Mianwali

*Corresponding author: mahmoodkhan77@gmail.com

Table 1: Specifications of NPPs in Pakistan

Sr. No.	Characteristic	K-1	C-1	C-2	C-3	C-4	K-2	K-3
1.	Plant Type	CANDU (PHWR)	PWR	PWR	PWR	PWR	PWR	PWR
2.	Power Output	137s MWe	325 MWe	325 MWe	340 MWe	340 MWe	1100 MWe	1100 MWe
3.	Year of Operation	1971	2000	2011	2016	2017	2021	2022
4.	No. of Loops	2	2	2	2	2	3	3
5.	Codes and Standards for In-Service Inspections	ASME BPVC (Section XI)	ASME BPVC (Section XI)	ASME BPVC (Section XI)	ASME BPVC (Section XI)	ASME BPVC (Section XI)	RSE-M	RSE-M
6.	Fuel	Natural Uranium	LEU (Low Enriched Uranium)	LEU	LEU	LEU	LEU	LEU
7.	Status	Permanently Shutdown	Operational	Operational	Operational	Operational	Operational	Operational
8.	Location	Karachi	Mianwali	Mianwali	Mianwali	Mianwali	Karachi	Karachi

The other two reactors namely K-2 and K-3 are situated in Karachi and are Generation III 1000 MWe, three-loop PWR reactors which became operational in 2021 and 2022. These are based on the French codes RCC-M, RSE-M etc. Table 1 shows the major specifications of the NPPs of Pakistan for comparison.

3. Role of Non-Destructive Testing (NDT) in Nuclear Power Plants

3.1 Key Components of a Nuclear Power Plant (PWR)

NPPs involve the conversion of heat generated by nuclear fission into electrical energy. Nuclear fission occurs in the reactor pressure vessel (RPV) where the fuel is placed in fuel assemblies. The melting of fuel is to be avoided and its cooling is done through highly pressurized water (called primary coolant). Several radiations including neutrons, gamma radiations, alpha and beta particles are produced in the RPV. These radiations can lead to the embrittlement of the material which can cause the failure of the material. Boric acid is added to the water for fine control of the reactivity. Different gasses are also produced in the reactions and the chemistry of the coolant is to be controlled to avoid corrosion effects (oxygen, hydrogen etc.). For coarse control of the reactivity, control rods are used which are made of a material having a higher neutron absorption coefficient (silver-indium-cadmium control rods are used in Westinghouse design-based PWR NPPs). Different penetrations for instrumentation (neutron flux, temperature, pressure measurements etc.) thimbles are also present in the RPV. It is made up of carbon steel with an internal cladding of stainless steel.

The coolant from the RPV gets heated and is taken to the steam generator (SG) where it is passed through several U-tubes and returned to the RPV. Outside these U tubes, feedwater passes (which is at lower pressure) and gets converted to steam which is then taken to the turbine where it expands and rotates the turbine and electricity is generated. The U tubes are made up of Inconel (Nickel based alloy) and are thousands in numbers. This is an interface of the primary coolant with the feedwater. If the tubes get damaged due to corrosion or foreign material the primary coolant may be released into the feedwater and the radiation will be leaked. This can cause radiation exposure to the workers and the public. The number of steam generators equals the number of loops of PWR.

Pressurizer (PRZ) controls the pressure of the primary coolant during the power surges. It is a vessel having electric heaters at the bottom and a spray line at the top. It contains a water steam interface (water at the bottom and steam at the top) and their conversion, either way, controls the surges. It is also made up of carbon steel internally cladded with stainless steel. There is only one pressurizer in the primary coolant system. The primary coolant is circulated between the aforementioned equipment with the help of a reactor coolant pump (RCP). The RCP is equipped with a casing enclosed with an impeller and linked to the motor through a shaft and different seals are present to avoid leakages. The RCP is also provided with a flywheel to provide inertia to the pump in case of power loss to prevail cooling of the primary coolant. There is an RCP in each loop of the primary coolant system. This equipment is placed inside a containment to prevent the leakage of radiation to the environment in any unfortunate accident.

There are other supporting systems along with this main primary coolant system which are divided into different classes called Safety Class 1, 2 and 3 and non-safety class depending upon the consequences, in case they fail, on the safety of NPPs. The safety of NPPs is much more important than that of conventional power plants due to radiation safety concerns. The melting of the fuel is to be avoided to keep the fission reaction under control. With the advancement, several passive safety systems have been added to the reactor design to ensure the safe operation of NPPs.

3.2 Non-Destructive Testing

NPPs have different phases during their lifetime namely: design, manufacturing, installation, commissioning, operation and de-commissioning. During the design phase according to relevant codes and standards, NDT methods are specified for ensuring quality control during the manufacturing and as a damage assessment tool during the installation and operation phases. The basic goal is to find the discontinuities and defects in the material which can lead to the failure of the material. The sizes of these defects are to be measured and characterized. The specific sections of codes are referred to for the NDT methods employed for the PSI during installation and ISI during re-fueling outages (RFO). Inspections are made during the manufacturing and based on accept reject criteria repairs are made if necessary. Hence, during manufacturing using NDE methods quality is ensured. Afterwards, during installation inspections PSI is performed to provide baseline data for comparison during the inspections in ISI of NPPs.

RFOs are carried out once every 14-18 months depending upon the design to replace the depleted fuel from the peripheries with fresh fuel and the core is rearranged to attain uniform neutron flux during operation. During these outages along with the fuel replacement, several inspections (ISI) are carried out as per codes and standards for ageing assessment of components/equipment and different piping for life assessment and ageing management. In ISI, service defects are expected to appear and are targeted for detection and characterization. The service defects include fatigue cracks, corrosion, stress-corrosion cracking, flow accelerated corrosion, radiation embrittlement, erosion, wear etc. NPPs are an uninterrupted source of clean power and are often operated as base-load power plants. So, efforts are made to enhance their life and keep them operating based on the assessment made during these inspections. Hence these inspections are of utmost importance.

Different NDT methods are used for inspections based on the application including: visual testing (VT), radiographic testing (RT), ultrasonic testing (UT), eddy current testing (ET), magnetic particle testing (MT), dye-penetrant testing (PT), acoustic emission (AE). UT and RT are used for volumetric examinations whereas VT, PT, MT & ET are used for surface examinations. Also, ET is used mainly for the inspections of tubes in steam generators, condensers and heat exchangers. Highly skilled and qualified manpower equipped with specified reliable inspection equipment is required for these inspections. Procedures are developed and planning is done by the trained engineers to perform these inspections reliably and safely as health safety due to the radiation environment is of concern. Manual inspections are performed which rely heavily on the competency of the person performing them. These inspections do not require installations and hence are versatile and adaptable to sudden changes. Semi-mechanized and mechanized inspections are also performed to reduce operator-based errors and reliable data acquisition is performed but requires excessive installation and exceptionally trained and experienced manpower for understanding, successful execution and interpretation [9]. Before the inspections are performed, following points are needed to be known and are mentioned in the plant-specific codes and standards:

1. Type of expected flaw.
2. The frequency of inspections depends upon the nature of the component, the more critical the component the more frequent the inspections.
3. The method to be used for the detection of an expected flaw depends upon codes/standards, the PSI/ISI program and the nature of the defect (surface, internal).
4. Preparation of components including insulation to be removed, arrangement of scaffolding and planning of jobs in as low as reasonably achievable (ALARA) radiation level.
5. Qualification requirements of procedure, equipment or inspector, on open or blind samples as per codes and standards.

Based on the ISI assessment of the life of NPP is made and decision on the extension in the operation of the plant is made. Along with NDE different surveillance capsules are placed inside the RPV (of the same material as of RPV) for destructive testing. These include specimens of different testing including: toughness and fatigue to assess the effect of radiation embrittlement. Before the operation of NPP, testing is performed on unirradiated material for baseline data and subsequently, the samples are withdrawn during each RFO for the life assessment and condition monitoring. These all measures are taken to extend and ensure the safe operation of NPPs and are indicative of strict safety regulations imposed on NPPs.

4. National Centre of Non-Destructive Testing

NCNDT was established in 1995 with the mandate to provide quality NDT training and services to the industrial sector of the country. NCNDT is providing training and certification in different NDT techniques as discussed in previous section namely: UT, RT, ET, VT, MT and PT. NCNDT has trained 7517 personnel through 446 courses in conventional and advanced NDT techniques. Qualified and experienced personnel are also providing services in the NDT techniques to the national industry. The number of the qualified manpower in different techniques is shown in Figure 2.

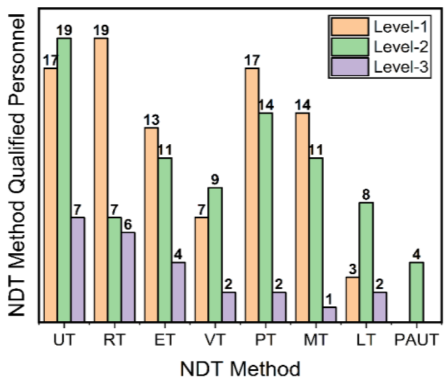

Fig. 2: Number of NDT Qualified Manpower

NCNDT is extensively involved in the PSI before operation and ISI of NPPs during their schedule re-fueling outages and is discussed in detail in next section. Equipment calibration services and third-party inspection and evaluation services are also being provided. Services related to mechanical testing (creep, fatigue, tensile testing, impact testing, hardness analysis), metallographic testing and material characterization are also available. These facilities are also being used to provide failure and life assessment analysis.

Recently, services of lifting equipment inspection and concrete testing have also been started. NCNDT is carrying out international collaboration with International Atomic Energy Agency (IAEA) for enhancement of its capabilities. NCNDT is accredited with ISO-17024 and ISO-17020 from Pakistan National Accreditation Council (PNAC). Cooperation with universities is also being carried out for research and development and several students are facilitated for their internship programs.

5. Role of NCNDT in NPPs

NCNDT has been involved in the inspections carried out in the NPPs in Pakistan. Starting from the In-Service Inspection of conventional island components of C-1 in 2001. NCNDT started inspecting welds of piping and Class 1, 2 and 3 components in 2010. Meanwhile, NCNDT has participated in and performed inspection of welds of piping in the Pre-Service Inspections of C-3, C-4, K-2 and K-3. Radiographic inspection of the welds of the steam generator using an indigenously developed source positioning tool has been performed for C series NPPs. In 2017 mechanized inspection of the steam generator was performed and in 2023 mechanized inspection of the reactor pressure vessel is going to be performed using specialized mechanized equipment.

Table 2: Inspections Performed by NCNDT

Sr. No.	Inspection	Started In	Details
1.	Inspection of piping of non-nuclear safety class	2004	This includes PT, MT, UT and RT of the welds.
2.	Inspection of piping welds of safety class 3	2006	This includes PT, MT, UT and RT of the welds.
3.	Inspection of piping welds of safety class 2	2006/2008	This includes PT, MT, UT and RT of the welds of various auxiliary and safety systems.
4.	Inspection of piping welds of safety class 1	2010	This includes VT, PT, UT and RT of the: Primary coolant piping Valves in the primary system Auxiliary piping in the main primary system Secondary side of steam generator Main steam system (MSS) valve components MSS piping
5.	Inspection of tubes of condenser, heat exchangers, generator, air coolers	2002	This includes the eddy current (ET) inspections of the tubes.
6.	Inspection of reactor coolant pump welds	2008	This includes the UT of nozzles to safe end and safe end to main coolant piping welds.
		To be performed in 2023	The RT of the nozzles to safe end and safe end to piping welds using specialized source positioning tools for K-2 and K-3.
7.	Inspection of pressurizer welds		This includes the UT of pressurizer welds.
		To be performed in 2023	This includes RT of upper head to nozzles and manhole welds, surge nozzle to bottom head and upper head to shell circumferential welds to be performed using specialized source positioning tools for K-2/K-3. This includes the RT of nozzles to safe end dissimilar welds
8.	Inspection of steam generator welds	2016	This includes the RT of steam generator primary nozzles to safe end and safe end to piping welds using specialized source positioning tool for C-series.
		To be performed in 2023	This includes the RT of steam generator primary nozzles to safe end and safe end to piping welds using specialized source positioning tools of K-2/K-3.
9.	Inspection of the RPV closure head	2018	This includes the VT of the closure head cladding.
10.	Mechanized inspection of steam generator	2017	This includes the mechanized eddy current inspection of the SG tubes.
11.	Mechanized inspection of reactor pressure vessel	To be performed in 2023/2024	This includes the UT, RT of the RPV welds, CCTV inspection of the RPV cladding in the most irradiated area, UT and ET of RPV closure head studs and ligaments and control rod drive mechanism omega welds.

Overall NCNDT has participated in:

1. 15 RFOs of C-1
2. 09 RFOs of C-2
3. 04 RFOs of C-3
4. 04 RFOs of C-4
5. 01 RFO of K-2
6. 01 RFO of K-3

NCNDT has also provided NDE inspection services in NPPs as a subcontractor of China Nuclear Operations Ltd. (CNPO), China during PSI and first ISI of C-3 and C-4 and during the PSI of K-2 & K-3. Fig. 3 shows the major milestones of inspections performed by NCNDT in chronological order.

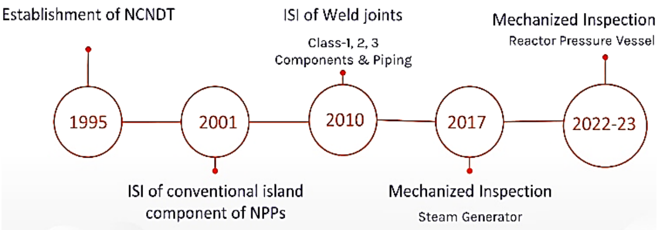

Fig. 3: Inspections Performed by NCNDT in NPPs

Table 2 shows the details of inspections performed by NCNDT.

6. Inspections in Nuclear Island

In this section, inspections of the critical components performed on the nuclear island are discussed.

6.1 Reactor Pressure Vessel

The reactor pressure vessel is made from fine-grained ferritic steel (ASME SA 508, 16MnD5 RCC-M). The material has excellent weldability and high toughness [10]. It has a height of approximately 40 ft. It is internally lined with austenitic steel cladding typically around 3-10mm. The RPV has inlet and outlet nozzles depending upon a number of loops/steam generators. The RPV has a closure head through which control rods are inserted and vents are present. It is made up of shells (3 in Westinghouse-based design) welded together and a hemisphere at the bottom. A flange is welded at the top of the upper shell for the closure head to be bolted to the vessel [11]. Several welds are present and are discussed in section 6.1.3. The RPV contains highly pressurized water at a high temperature. The core is present in the RPV and is situated at the centre of the containment. The water is highly radioactive as it is heated through nuclear fission. A typical PWR Westinghouse-based RPV is shown in Fig. 4.

The inspections of RPV are performed underwater during the RFO and hence the reactor internals must be removed prior to the inspections. Specialized mechanized/robotic equipment has been developed by different NDT companies to perform the inspections. For these inspections several techniques are required including [12]:

1. Visual and Eddy current testing as surface examination.
2. Ultrasonic testing and radiographic testing as volumetric examination.

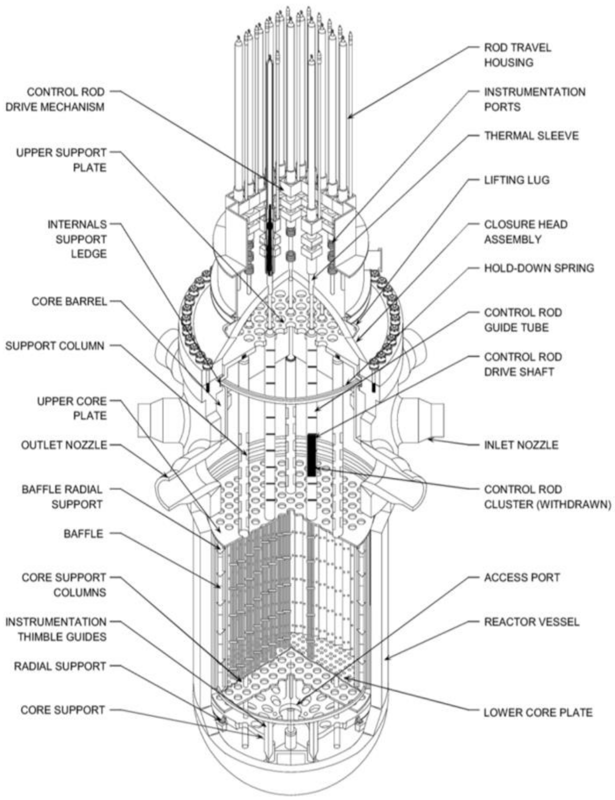

Fig. 4: A typical PWR RPV [13]

In the past such examinations were performed manually usually from the outside of RPV whereas now advanced automated technology is available to perform the examinations from the inside of RPV which includes:

1. Automated ultrasonic examination with conventional UT probes.
2. Automated ultrasonic examination with phased arrays UT (PAUT) probes.
3. Automated ultrasonic examination with conventional and phased arrays UT probes.

In the early nineties, companies implemented the NDT technique for automated remote testing of reactor pressure vessel welds from inside with conventional UT technique and recently PAUT technique has been implemented and qualified.

6.1.1 Objectives of RPV-ISI

The following are the main objectives of the In-service inspections of RPV:

1. To perform RPV examination according to the relevant standards (e.g., ASME Code Sec. XI, RSE-M).
2. To examine RPV welds and other critical areas (core region, distribution ring, core barrel support lugs, baffle to former bolts) based on current knowledge, experience from

similar examination programs, in-depth design information and good engineering practice.

3. Evaluate the integrity of the reactor pressure vessel.

6.1.2 RPV Degradation Mechanism

The inspection of RPV is critical due to the action of the following degrading mechanisms [14-18]:

1. Radiation embrittlement causes a significant decrease in the ductility of a material, which decreases the fracture toughness. It occurs mostly in the active height of the fuel assemblies i.e., in the intermediate and lower shell of RPV and slightly above and below it. It affects the region where the end-of-life fluence (time integrated neutron flux) is around 10^{21} n/m^2.
2. Thermal ageing also causes a decrease in ductility by the embrittlement due to copper precipitation etc.
3. Temper embrittlement due to the impurity of phosphorus concentration which causes phosphorous segregation-induced weakening of grain boundaries.
4. Fatigue which is caused by the transients. These transients fluctuate the temperature and pressure to another set of loadings (cyclic loading) which initiates and propagates the cracks due to vibrations etc.
5. Corrosion of different types including:
   a. Intergranular attack (IGA), stress corrosion cracking (SCC), flow assisted corrosion crack.
   b. Corrosion of welds.
   c. General corrosion and pitting.
   d. Corrosion due to boric acid.
6. Wear (maintenance operations, opening and closing of the RPV)

6.1.3 RPV Weldments

As explained previously RPV comprises of shells and a hemispherical bottom head. The upper shell has a flange for the closure head placement. The performance of the RPV weldments (both base metal and HAZ) is critical to the safe and efficient operation of the nuclear reactor. Weldments have a different microstructure than the base metal, they are more susceptible to failure and their inspections are more important. Certain RPV weld regions are vulnerable to stress corrosion cracking and other environmental degradation.

1. Cracking and erosion in connection welds.
2. Nozzle cracking in dissimilar metal welds.
3. RPV structural integrity is highly dependent on welding residual stress profiles (stress distributions in the region of the weld).

Major welds of RPV include:

1. Longitudinal shell welds
2. Circumferential welds (shell to shell, shell to bottom head)
3. Nozzles to shell welds
4. Dissimilar nozzle to safe end weld

Along with the inspections of these welds (which are carried out by UT, except nozzle to safe end weld carried out by both UT and RT), an inspection of the cladding in the most irradiated area (as explained before) is carried out by VT, ET and UT for detection of surface and subsurface cracks. Inspection using UT, ET and VT is performed for RPV studs, nuts, threads, ligaments and washers in the flange. VT of the closure head cladding is also performed as required by the codes and standards.

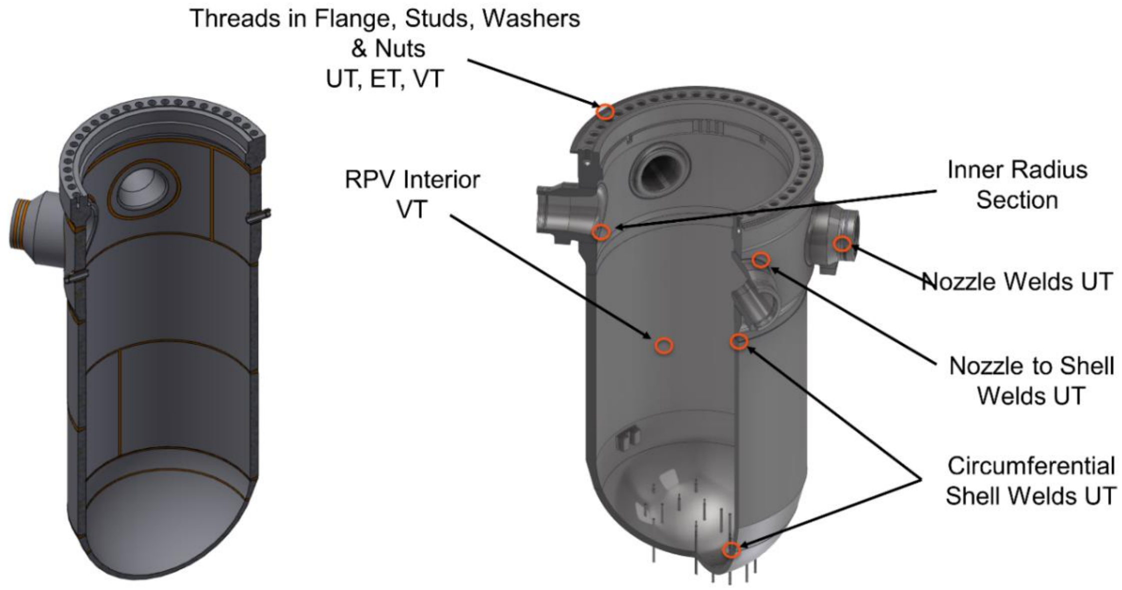

Fig. 5: A typical 325 MWe RPV

6.1.4 325 MWe NPP – ASME

A typical 325 MWe Westinghouse PWR RPV with its weldment and inspection to be performed is shown in Fig. 5. The inspection of Westinghouse-based RPV is executed on ASME B&PV Code Section XI.

6.1.5 1000 MWe NPP – RSE-M

A typical 1000 MWe French code RCC-M PWR RPV with its weldment and inspection to be performed is shown in Fig. 6. The inspection of such RPV is based on the RSE-M code.

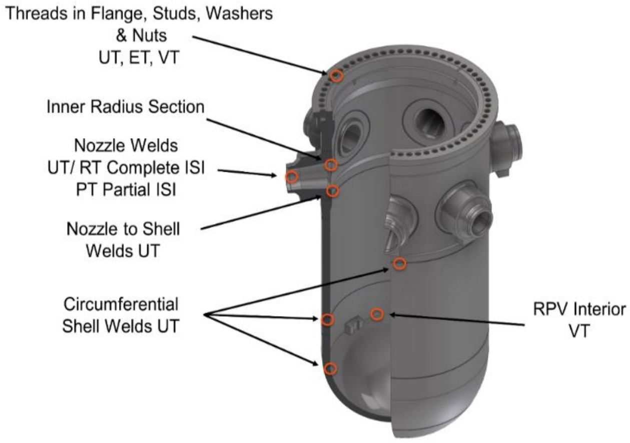

Fig. 6: A typical 1000 MWe PWR RPV

6.1.6 RPV Inspections

Specialized automated inspection systems for RPV have been developed by different NDT companies which is adjustable to different RPV designs (for different diameter and depth). The system constitutes a manipulator which provides 6 or 7 degrees of freedom (DOF) and is connected to the mainframe of the equipment. The system is placed on the RPV flange using the polar crane and RPV guiding studs. It is aligned to the RPV at 0° keyway. It is submersible up to 30 meters. The components are designed to be radiation resistant. The system is remotely operated from the distance. The system has different end effectors to carry out the inspections and are connected to the manipulator depending upon the required inspection. The end effectors include:

1. Nozzle inner radius end effector for ultrasonic testing of nozzle to safe end welds, nozzle inside surface and nozzle inner radius area.
2. Shell end effector for ultrasonic testing of shell welds and lower head weld.
3. Tangential end effector for ultrasonic testing of nozzle to shell welds.
4. RT end effector to place the gamma radiation source at the welds center and to displace the water from surroundings for the radiographic testing of nozzle to safe end welds and safe end to primary coolant piping welds for 1000 MWe NPPs.

These end effectors are remotely interchanged from the refueling bridge via remote docking system (RDS) to Manipulator placed at the RPV flange during the inspections and are maneuvered during the inspections. The system is linked with data acquisition and data analysis systems, which can be performed remotely. These systems complete the inspections in a shorter period to reduce the refueling outage time.

6.2 Steam Generator

Steam Generator (SG) is a U-tube heat exchanger in which pressurized primary coolant from RPV passes through the tubes and boils the feedwater on the outside of the tubes. It has approximately 60 ft in height and is the biggest component of NPP. The tubes are thin and are made up of inconel (nickel-based alloy) 600 which has been replaced by inconel 690 and 800 due to its better corrosion resistance. Thousands of such tubes are present in the steam generator which pass through a tube sheet at the bottom and U bend is at the top. At the bottom of the steam generator manholes are provided along with the primary coolant inlet and outlet nozzles. The details of such tubes for a typical 325 Mwe and 1000 MWe PWR reactor are shown in Table 3.

Table 3: Tube Parameters of Typical 325 and 1000 MWe Steam Generator

Steam generator tubes parameters	K-series PWR Reactor	C-series PWR Reactors
Material	Inconel 690	Inconel 800
Tubes quantity	5835	2977
OD (outer diameter)	17.48mm	22.00mm
Wall thickness	Row1-2, 1.04mm	1.2
	Row3-112, 1.02mm	1.2
Straight tube length	9600mm-9772.49mm	7035 mm (avg)
Shortest tube length	19459.34mm	19233 mm
U-bend tube radius of row 1	82.55mm	62 mm
U-bend tube radius of row 112	1520mm	1364 mm

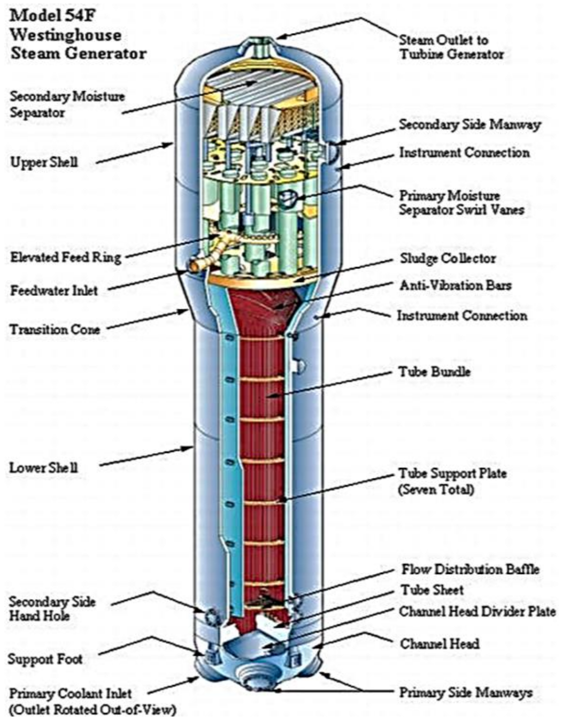

Fig. 7: A Typical Westinghouse Steam Generator [19]

The secondary side of a typical SG includes the "J" rings through which feed water enters the SG, passes through the annulus and then at the shell side around the tubes where steam is produced. It then passes from the top region where chevron-type and swirling vane-type moisture separators are present and reduces the moisture of the steam (minimum 99.75%) to avoid degradation of turbine blades. The steam then leaves the SG through different safety valves to the turbine. A typical SG is shown in Fig. 7.

6.2.1 Objectives of SG ISI

SG tubes not only transfer the heat between primary and secondary systems but also act as a physical boundary between radioactive and non-radioactive coolants. If one or multiple tubes puncture/burst due to a routine ageing process (highly unlikely) or any sudden unforeseen event (due to different degrading mechanisms as discussed in 5.2.2) then both closed loops primary and secondary loops will be mixed and radiation will start to spread and contaminate the secondary side equipment. This spread of radiation will not only disrupt the normal operation of NPPs by damaging the equipment but also be hazardous to the environment and plant personnel. The sudden burst of the steam line will lead to the depressurization of the secondary side. This will produce excessive pressure difference and can cause the rupture of a large number of tubes. The rupture of tubes will depressurize the primary system which can cause loss of coolant accident (LOCA) and subsequently melting of the core [20].

To avoid this gradual or sudden failure of this boundary regular inspections of SG tubes are performed using Eddy Current Testing (ET) technique. Whereas the Nozzles welds with the safe end are inspected through Radiographic Testing (RT) technique.

6.2.2 Steam generator degradation mechanisms

Corrosion is one of the major degrading mechanisms of SG. The chemistry of feedwater is very important and is needed to be controlled to avoid ageing. When steam is formed if there are any impurities present, these will be left behind and get accumulated. These accumulated impurities initiate the corrosion phenomenon. The corrosive products are also produced from the flow accelerated corrosion in the wet steam areas and from the general corrosion of the carbon steel [21]. Different degradation mechanisms occur in the SG as discussed below [22-25].

1. Localized corrosion causes the pitting of tubes both on tube sheets and tubes and is very dangerous.
2. Primary water stress corrosion cracking (PWSCC) under constant (in Heat Affected Zone) and increasing stress (in case of rupture), is observed more frequently in the upper region of the U-bend and is initiated from the inside radius or in crevice regions.
3. Outer diameter stress corrosion cracking and intergranular attack cause circumferential or axial cracks. These occur at the tube-to-tube sheet crevices, tube support plate and free span.
4. Thinning due to accumulation of sludge on tube sheet. The sludge prevents the contact of coolant with the surface increasing its temperature. Wetting and drying alternating regions are formed at which the corrosion is maximum.
5. Denting due to compressive force applied by magnetite accumulated in the gap between intermediate support plate and tubes from accelerated corrosion of carbon steel.
6. Fretting wear occurs due to the vibrations produced as the fluid bends and exerts stresses on the tubes. Cracks and thinning of tubes occur due to these vibrations. These occur at the contact points of anti-vibration bars and tubes etc.
7. High cycle fatigue due to high-stress levels and flow-induced vibrations. It occurs at the upper support plate if tubes are clamped.
8. Foreign material intrusions FMI (any material that is not part of the system or component as designed like bolts, broken parts, debris etc.) if not carefully traced and removed during the refueling outages may get stuck at the tube bundle [26]. Relative motion between these loose parts and the tube bundle results in wear, thinning or rupture of the tubes.
9. Wastage occurs due to the phosphate chemistry. It occurs at the tube sheet crevices, sludge pile, support plates and anti-vibration bars and general thinning is produced.

6.2.3 Steam Generator Weldments

Typical welds in a steam generator are shown in Fig. 8.

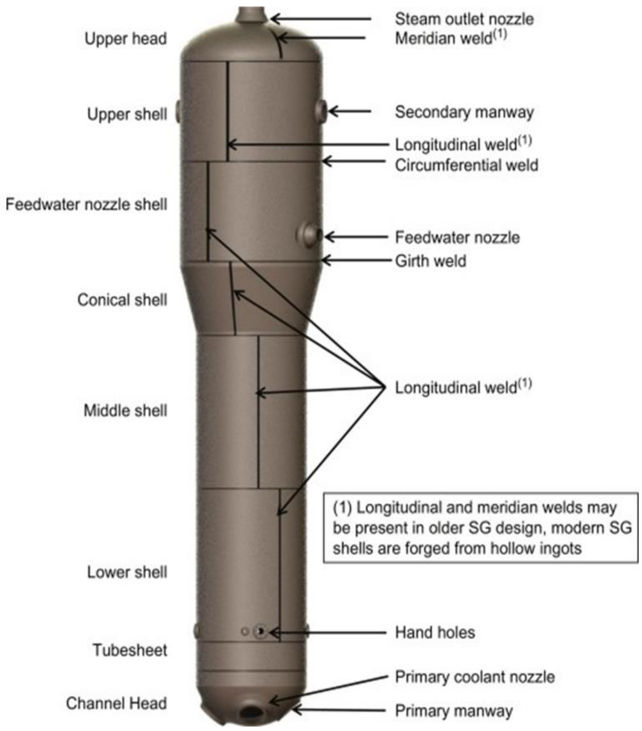

Fig. 8: SG Weldments [27]

6.2.4 Steam Generator Inspections

Different examinations are performed during the ISI as per applicable codes and standards. Some of these include Eddy current examination of steam generator tubes and radiographic examination of primary inlet and outlet nozzles to safe end and safe end to piping welds.

6.2.4.1 Eddy Current Testing of Steam Generator Tubes

Specialized equipment is used for Eddy current examination of tubes in which a push-pull mechanism is used for the insertion and retrieval of the probe in the tubes. Through the signal analysis defects are recognized and sized by comparison from calibration samples with known defects and a decision on the plugging of the tubes is made based on the applicable codes and standards. The system of ET examination of steam generates tubes includes [28]:

1. Manipulator/robot for probe positioning
2. Eddy current instrument
3. Eddy current probe
4. Calibration standard
5. Pushers for transporting the probe
6. Remote control and communication transmission equipment
7. Computers for data acquisition and analysis
8. Video surveillance system

A typical ET examination system of tubes is shown in Fig. 9.

Fig. 9: Typical ET SG Tubes Inspection System

6.2.4.2 Radiographic Inspection of Primary Nozzles Welds

The radiographic examination of the primary nozzles' welds is performed through specialized designed tools. The mounting plate of the tool is installed on the manhole whereas it has a pipe along with a centering mechanism to place the gamma-ray source at the centre of the weld for panoramic exposure.

6.3 Pressurizer

It is a cylindrical vessel of approximately 50 ft in height and has hemispherical heads at the top and bottom. The material is carbon steel (18MND5/16MND5) internally lined with stainless steel. A large number of electric heaters are placed at the bottom of the pressurizer (PRZ). It has saturated water (heated through heaters) at the bottom and saturated steam at the top. There is a surge nozzle at the bottom of PRZ which links it with the primary coolant system. Normally the water comprises 50/60% volume of PRZ. It controls the pressure of the primary coolant in case of transients through the action of heaters and sprays line. Different safety nozzles are present at the top head along with the spray nozzle and manhole. Different instrumentation penetrations are also provided to measure the temperatures and water level etc. [27]. A typical PRZ is shown in Fig. 10.

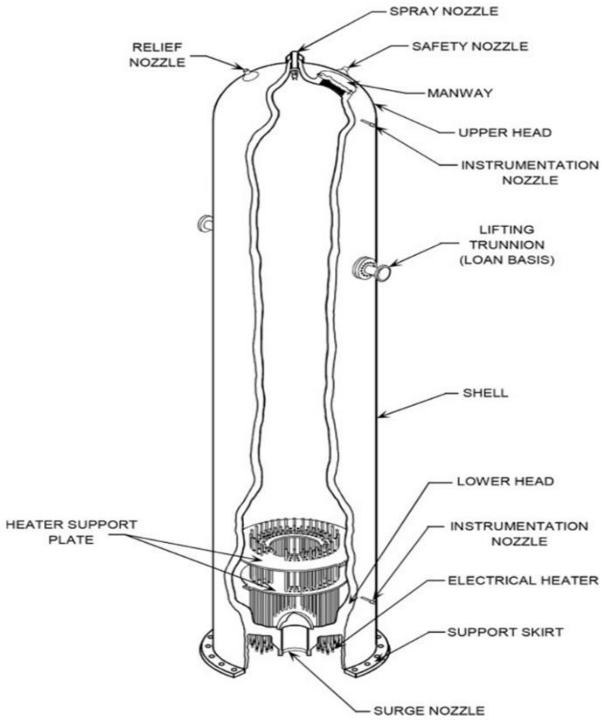

Fig. 10: A typical Pressurizer [29]

In the case where the primary coolant temperature decreases the water somehow contracts and flows from PRZ into the primary coolant system. This is called an out-surge. This decreases the water level which results in expanding the steam hence pressure in the pressurizer decreases. Resultantly, the electric heaters are actuated to boil more water into the steam to increase the pressure and restore the conditions. Contrarily, when the temperature of the primary coolant increases the water flows into the pressurizer and compresses the steam in PRZ hence increasing its pressure. This is called an in-surge. The spray line is actuated when pressure decreases below a set point and relatively cold water from RCP is sprayed in the PRZ. This results in the condensation of steam at the top and allows the coolant to expand and restore normal conditions. Automatic pressure relief valves are present to depressurize the system in case of the pressure gets to such a value that cannot be controlled through heaters or spray. If the pressure gets beyond the control of pressure relief systems (i.e. heaters, spray or safety valves) the reactor gets tripped automatically [30].

6.3.1 Objectives of Pressurizer In-Service Inspection

As explained before the pressurizer maintains the pressure of the primary coolant. Different safety systems are provided with the pressurizer to safely operate the nuclear power plant. Defects in heaters may lead to their ineffective working and if large number of heaters get damaged then this can become a serious issue. Therefore, tracking, maintenance and replacement (if required) of the malfunctioning heaters is to be performed to maintain the pressure of the system. Different components of the pressurizer are subjected to inspections including the heaters, nozzles welds, circumferential welds, cladding etc.

6.3.2 Pressurizer Degradation Mechanisms

Different degradation mechanisms of pressurizer components include [31-34]:

1. Low cycle thermal fatigue in vessels, nozzles and heaters.
2. Flow-induced vibrations.
3. Boric acid corrosion through coolant leakage from electric heater sheath and instrumentation penetrations.
4. Primary water stress corrosion cracking (PWSCC) in the vessel.
5. Thermal embrittlement in spray heads.
6. Electrical ageing (burnout) for heaters.
7. Thermal stratification in the surge nozzle. This occurs when two fluids of different temperatures meet. This occurs for fluid flowing with low velocity. This results in the application of different loads on the nozzle and piping due to temperature differences.
8. Thermal stripping of the surge nozzle due to temperature variation at the interface causes high cycle fatigue cracks.
9. Thermal cycling in the surge nozzle occurs due to transients and the interaction of turbulence in the fluid with the stratified fluid. This also causes fatigue cracks. The damage is maximum when the velocity is small with a large temperature difference. The potential for the maximum damage is during the heat-up and cooldown of the reactor.

6.3.3 Pressurizer Weldments

Pressurizer welds include circumferential shell welds, shell to upper and lower head welds, spray line and safety valve nozzles to upper head welds, surge nozzle to bottom head welds, dissimilar nozzles to safe end welds and manhole to upper head welds etc.

6.3.4 Pressurizer Inspections

According to RSE-M (French code for in-service inspection of mechanical components), radiographic examinations of nozzles to head, upper shell to head weld, manhole to upper head and nozzles to safe end dissimilar welds are to be performed. Visual examination of the internal surface of the pressurizer and acoustic examination of heaters is to be performed. Ultrasonic examination of the lower head to shell weld is to be performed.

The radiographic examinations of upper head to shell, upper head to nozzles and manhole, surge nozzle to bottom head, the source is to be placed inside and complete examination can be performed in a single exposure. Specialized source positioning tools have been developed by inspection companies for this purpose. Tools are installed at the manhole and provide a pathway for the source guide tube to pass through it. For the upper head to shell and upper head to nozzles welds tool of hook shape is used and the source is placed at the centre of the weld for the former and the centre of the upper head for the latter. Whereas for the lower head to surge nozzle a sophisticated tool, in the shape of a hanging umbrella, is used which is lowered in the PRZ using steel rope to the lower support plate of heaters. The centering is ensured by acquiring support from the shell. Panoramic exposure for all the welds is taken. The positioning tools are shown in Fig. 11. The dissimilar welds are exposed by using a double wall single image both on and off-axis for weld and buttering examination [35].

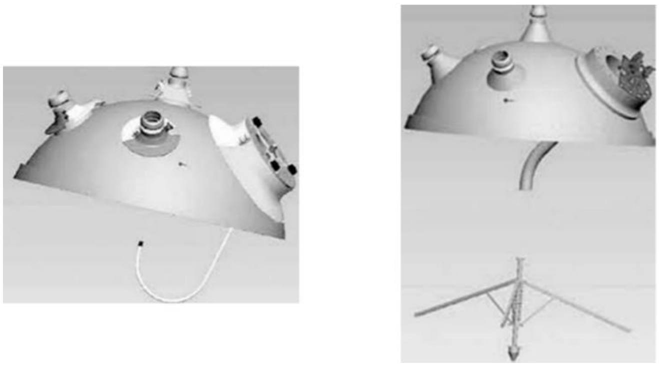

Fig. 11: Radiographic Examination Tools for RT [35]

6.4 Reactor Coolant Pump

In each loop of PWR, a Reactor Coolant Pump (RCP) is present to circulate the primary coolant. It is a single-stage centrifugal pump. The height of a typical RCP is around 30 ft. The suction is vertical with a horizontal discharge. It has a carbon steel casing internally lined with stainless steel provided with inlet and outlet nozzles. The casing encloses the impeller which is connected to the motor via a shaft. A flywheel is provided to keep the water flowing in case of power loss and to avoid any accident such as a core meltdown. As the primary coolant is radioactive hence leakages from the pump are to be avoided. Three seals are provided to the shaft to avoid leakages. Water at a higher pressure is inserted in the RCP to cool the seals and bearings and this also stops the primary coolant from leaking out.

6.4.1 Reactor Coolant Pump Degradation Mechanisms

The degradation mechanisms of RCP include [36]:

1. Fatigue due to transients and shaft vibrations. Thermal stresses and alternating bending stresses can also initiate fatigue cracks.
2. Stress corrosion cracking in the pump casing.
3. High cycle fatigue for pump shaft.

4. Corrosion for pump studs.
5. Foreign material intrusion induced damages.

6.4.2 Reactor Coolant Pump Weldments

RCP weldments include the inlet and outlet nozzles to safe end welds and safe end-to-piping welds.

6.4.3 Reactor Coolant Pump Inspections

Radiographic examination for casing welds is performed using specialized source positioning tools. The tools are fixed at the casing flange and they provide a guiding path to the source through a pipe. The source is placed at the centre of the weld and off-axis position (for buttering examination as per the standard requirement if any) for panoramic exposures. Visual examination of the casing's internal surface is also to be performed.

6.5 Safety Class Piping Welds

The primary coolant piping is made up of austenitic stainless steel. Ultrasonic testing and dye penetrant testing are performed for piping welds as per codes and standard requirements. The main degradation mechanism for the piping is fatigue (due to pressure shocks, thermal fatigue and excessive vibrations), thermal ageing, primary water stress corrosion cracking, boric acid corrosion, thermal stratifications and atmospheric corrosion [37, 38].

6.6 Non-nuclear Safety Class Components and Piping

NCNDT has been performing inspection activities of equipment and piping in the conventional island since RFO-1 of C-1. The key components include: the main steam turbine (HP/LP turbines), steam condensers, moisture separator reheaters, lube oil coolers, HP/LP heaters, exciter/generator air coolers, cross-over piping, governing valves and generators etc.

7. Conclusion

This paper discusses the non-destructive examinations performed during the In-Service Inspections of nuclear power plants with a brief introduction to NPPs in Pakistan, NCNDT and the history of NCNDT in NDT examinations. The examinations have been thoughtfully decided and recommended in the plant codes and standards with the intervals. The NDT examinations have been thoughtfully decided and recommended in the relevant plant codes and standards with their respective intervals in which they must be performed repeatedly for condition assessment. The results of these examinations are compared with baseline (PSI) data and analysis is performed for the life assessment based on the defects' sizes and their growth.

References

[1] D.J. Wuebbles and A.K. Jain, "Concerns about climate change and the role of fossil fuel use," *Fuel Processing Technology,* vol. 71, no. 1, pp. 99-119, 2001.

[2] P.A. Owusu and S. Asumadu-Sarkodie, "A review of renewable energy sources, sustainability issues and climate change mitigation," *Cogent Engineering,* vol. 3, no. 1, p. 1167990, 2016.

[3] T. Trainer, "Some problems in storing renewable energy," *Energy Policy,* vol. 110, pp. 386-393, 2017.

[4] I.C. Change, "Mitigation of climate change," *Contribution of working group III to the fifth assessment report of the intergovernmental panel on climate change,* vol. 1454, p. 147, 2014.

[5] E. AKYÜZ, "Advantages and disadvantages of nuclear energy in Turkey: Public perception," *Eurasian Journal of Environmental Research,* vol. 1, no. 1, pp. 1-11, 2017.

[6] D. Basu and V. Miroshnik, "Advantages of Nuclear Power,", pp. 7-21, 2019.

[7] *Nuclear Power Reactors in the World*. Vienna: International Atomic Energy Agency, 2021.

[8] D.S. Siqueira, J. de Almeida Meystre, M.Q. Hilário, D.H.D. Rocha, G.J. Menon, and R.J. da Silva, "Current perspectives on nuclear energy as a global climate change mitigation option," *Mitigation and Adaptation Strategies for Global Change,* vol. 24, no. 5, pp. 749-777, 2019.

[9] J. Enkvist, A. Edland, and O. Svenson, "Human factors aspects of non-destructive testing in the nuclear power context A review of research in the field," Sweden 1104-1374, 1999, Available: http://inis.iaea.org/search/search.aspx?orig_q=RN:30026316.

[10] J.H. Kim, "Reactor nozzle inspection based on the laser-guided mobile robot," *International Journal of Advanced Robotic Systems,* vol. 15, no. 1, p. 1729881417748443, 2018.

[11] *Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: PWR Vessel Internals*. Vienna: International Atomic Energy Agency, 2007.

[12] J. Freixa, "SBLOCA with Boron Dilution in Pressuirzed Water Reactors, Impact to the Operation and Safety," 2007.

[13] Y. Bouveret, M. Algarotti, L. Hernandez, J.P. Landez, and M. Lutsen, "PWR reactor vessel in-service inspection according to RSEM," in *Proc. ECNDT*, 2006.

[14] *Integrity of Reactor Pressure Vessels in Nuclear Power Plants: Assessment of Irradiation Embrittlement Effects in Reactor Pressure Vessel Steels*. Vienna: INTERNATIONAL ATOMIC ENERGY AGENCY, 2009.

[15] W. Corwin, R. Nanstad, D. Alexander, R. Stoller, J. Wang, and G. Odette, "Thermal embrittlement of reactor vessel steels," Oak Ridge National Lab.1995.

[16] A. Sarkar, B.K. Kumawat, and J.K. Chakravartty, "Low cycle fatigue behavior of a ferritic reactor pressure vessel steel," *Journal of Nuclear Materials,* vol. 462, pp. 273-279, 2015.

[17] J. Congleton, T. Shoji, and R.N. Parkins, "The stress corrosion cracking of reactor pressure vessel steel in high temperature water," *Corrosion Science,* vol. 25, no. 8, pp. 633-650, 1985.

[18] J. Park, O. Chopra, K. Natesan, W. Shack, and W. Cullen Jr, "Boric acid corrosion of light water reactor pressure vessel materials," in *Proceedings of the 12th International Conference on Environmental Degradation of Materials in Nuclear Power System-Water Reactors*, pp. 459-468, 2005.

[19] S. Šadek and D. Grgić, "Operation and Performance Analysis of Steam Generators in Nuclear Power Plants," in *Heat Exchangers-Advanced Features and Applications*: IntechOpen, 2017.

[20] P. MacDonald, V. Shah, L. Ward, and P. Ellison, "Steam generator tube failures," Nuclear Regulatory Commission, 1996.

[21] K. Kang and L. Kupca, "Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Steam Generators," Tech. Rep. TECDOC-1668, IAEA1999.

[22] S.J. Green and J.P.N. Paine, "Materials Performance in Nuclear Pressurized Water Reactor Steam Generators," *Nuclear Technology,* vol. 55, no. 1, pp. 10-29, 1981.

[23] P. Berge and J.R. Donati, "Materials Requirements for Pressurized Water Reactor Steam Generator Tubing," *Nuclear Technology,* vol. 55, no. 1, pp. 88-104, 1981.

[24] L. Obrutsky, J. Renaud, and R. Lakhan, "Overview of steam generator tube-inspection technology," *CINDE Journal,* vol. 35, no. 2, pp. 5-8,10-13, 2014.

[25] S. Mahey, "Flow-Induced Vibrations of foreign objects inside steam generator's tube bundle," University of Waterloo, 2017.

[26] D. Urjan, "Foreign Material Exclusion Program at CNE Cernavoda Nuclear Generating Station," 2008.

[27] T. Sollier, "15-Nuclear steam generator inspection and testing," in *Steam Generators for Nuclear Power Plants*, J. Riznic, Ed.: Woodhead Publishing, pp. 471-493, 2017.

[28] S.Y. Ismail and S. Lee, "A Review of APR 1400 Steam Generators Tube Degradation and State-of-The-Art Inspection Techniques."

[29] T.W. Kerlin and B.R. Upadhyaya, "Chapter 10 - Reactor thermal-hydraulics," in *Dynamics and Control of Nuclear Reactors*, T. W. Kerlin and B.R. Upadhyaya, Eds.: Academic Press, pp. 111-127, 2019.

[30] N.R. Commission, "Pressurized Water Reactor (PWR) Systems," *Reactor Concepts Manual, US NRC Technical Training Center,* vol. 603.

[31] T.J. Katona, "19-Plant life management (PLiM) practices for water-cooled water-moderated nuclear reactors (WWERs)," in *Understanding and Mitigating Ageing in Nuclear Power Plants*, P. G. Tipping, Ed.: Woodhead Publishing, pp. 633-705, 2010.

[32] Y. Jeong and S. Hwang, "Materials management strategies for pressurized water reactors (PWRs)," in Materials Ageing and Degradation in Light Water Reactors: Elsevier, 2013, pp. 315-334.

[33] M. Hrazsky and M. Mikus, "Nuclear Power Plant Mechanical Components Ageing Management Programs in Slovakia," 2007.

[34] H. Sohn, J.Y. Yang, H.S. Lee, and B.J. Park, "20 - Sensing solutions for assessing and monitoring of nuclear power plants (NPPs)," in *Sensor Technologies for Civil Infrastructures*, vol. 56, M. L. Wang, J. P. Lynch, and H. Sohn, Eds.: Woodhead Publishing, pp. 605-637, 2014.

[35] J. Hao, D. Deng, J. Wang, S. Wang, and X. Hua, "Radiography Inspection Technology of CPR1000 Nuclear Power Unit Pressurizer," in *Energy Materials 2014*, Cham, Springer International Publishing, pp. 539-544, 2016.

[36] V. N. Shah, A.G. Ware, D. A. Conley, P. E. Macdonald, and J. J. Burns, "Aging assessment of PWR surge and spray lines and LWR coolant pumps," *Nuclear Engineering and Design,* vol. 118, no. 3, pp. 329-342, 1990.

[37] B. Liang, G. Li, and G. Wang, "Periodic remaining life evaluation program of PWR pressurizer surge line accounting for thermal stratification effect," 2007.

[38] A. Ballesteros, R. Sanda, M. Peinador, B. Zerger, P. Negri, and R. Wenke, "Analysis of events related to cracks and leaks in the reactor coolant pressure boundary," *Nuclear Engineering and Design,* vol. 275, pp. 163-167, 2014.