Role of Non Destructive Techniques for Monitoring structural Integrity of Primary Circuit of Pressurized Water Reactor Nuclear Power Plant

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Abstract.

The safety of nuclear installations is ensured by assessing status of primary equipment for performing the intended function reliably and maintaining the integrity of pressure boundaries. The pressure boundary materials undergo material degradation during the plant operation. Pressure boundary materials are subjected to operating stresses and material degradation that results in material properties changes, discontinuities initiation and increase in size of existing discontinuities.

Pre-Service Inspection (PSI) is performed to generate reference base line data of initial condition of the pressure boundary. In-Service Inspections (ISI) are performed periodically to confirm integrity of pressure boundaries through comparison with respect to base line data. The non destructive techniques are deployed considering nature of the discontinuities expected to be generated due to operating conditions & degradation mechanisms. The paper is prepared considering Pressurized Water reactor (PWR) Nuclear Power Plant. The paper describes the degradation mechanisms observed in the PWR nuclear power plants & salient aspect of PSI & ISI and considerations in selecting non destructive testing. The paper also emphasizes on application of acoustic emission (AE) based condition monitoring systems that can supplement in-service inspections for detecting and locating discontinuities in pressure boundaries.Criticality of flaws can be quantitatively evaluated by determining their size through in-service inspection. Challenges anticipated in deployment of AE based monitoring system and solutions to cater those challenges are also discussed in the paper.

1. Nuclear Reactors for Power Generation:

Worldwide mainly Light Water Reactors (LWR) are mainly being used for Nuclear Power Generation. LWRs used for power generation are either Pressurized Water Reactors (PWRs) or Boiling Water Reactors (BWRs).

In Dec 2014, World Nuclear Association quoted International Atomic Energy Agency (IAEA) datathat worldwide 438 Nuclear Power Plants were under commercial operation generating 376 Giga Watt electricpower. There were 277 Pressurized Water Reactors (PWR) contributed 257 Giga Watt electric power, which amount to approximately 68% of total power generation from various types of reactors deployed for power generation [1].

Nuclear Reactor requires moderator to sustain the chain reaction and coolant to transfer the reactor heat. Moderator converts high kinetic energy neutrons (fast neutrons) into slow neutrons (called thermal neutrons) in order to facilitate interaction with the nuclear fuel causing chain reaction. The probability of fast neutron collision with small nuclei is much higher in comparison to larger nuclei [2]. Light water therefore,is an excellent moderator because of the hydrogen nuclei (protons) in water but its neutron absorption is also large.Coolant transports heat from the core to the steam generator. Light waterhas a high specific heat and is an inexpensive coolant.Light water is used as coolant and moderator in LWR.

2. Pressurized Water Reactors (PWRs):

Pressurized Water Reactors (PWR) designs are derived from designs originally developed for propelling submarines and large naval ships [3]. A PWR steam generating plant consists of two major systems to produce steam. First is Primary system that contains the reactor. The second is secondary side of the plant that contains the feed water/steam system, turbine and generator. PWR primary system usually consists of two to four loops connected with reactor pressure vessel (RPV) and Pressurizer. Each loop contains a Reactor Coolant Pump, Steam Generator and associated piping.Typical set up of Primary System Loop of PWR is shown in Sketch-1 along with main components. The reactor core is secured inside the Reactor Pressure Vessel. In the primary circuit radioactiveprimary water is kept sub-cooled at 155bars and pumped to the reactor core at temperatureof around 290^{0}C where it is heated by the fission energy to a temperature of around 325^{0}C. The heated water then flows to a steam generator where its thermal energy is transferred to secondary sidefor steam generation.

Light water as moderator is an important safety feature of PWRs due to its negative temperature coefficient [2]. The reactivity reduces with increase in temperature due to negative temp coefficient that makes PWR self regulating around the temperature set by control rods. PWRs are designed to be maintained in under moderated state, resulting into relatively small moderator volume and have compact cores [2].

3. Materials of equipment & piping:

Pressure boundary material of primary circuit in contact with the reactor coolant is mainly 300 series austenitic stainless steel or 300 series austenitic stainless steel clad on carbon steel or low alloy steel.Widely used material in PWR primary circuit pressure boundary equipment & piping are tabulated in Table-1.

4. Significance of pressure boundary integrity:

Sequential barriers to prevent the escape of fission products in nuclear reactors are cladding, primary circuit pressure boundary and the containment building. The nuclear fuel is seal welded inside the cladding, which forms part of fuel assemblies contained in reactor core. The reactor core is secured in Reactor pressure vessel, which is critical part of pressure boundary. In case of leak in pressure boundary there is release of active coolant. Break in pressure boundary may cause loss of coolant accident (LOCA) resulting in overheating of reactor core and in worst case melting of the reactor core. The probability of occurrence of such pressure boundary break is minimized throughout the working life of reactor pressure boundary, since such events could give rise to large exposure of radioactivity to the public. In view of these consequences, integrity of pressure boundary is essential for safe operation of nuclear power plant.

5. Material Degradation Mechanisms:

The pressure boundary is subjected to operating conditions in service,exposing materials to irradiation, stresses, high temperature and corrosionresulting inmaterial properties degradation. Degradation mechanisms cause inception and growth of discontinuities, erosion, corrosion that result into cracking and thinning of the material undergoing degradation. Situation further worsens in case of degradation mechanism causing embrittlement. The integrity of the pressure boundary also depends on aspects like quality of manufacturing of equipment, piping and their aggregation into steam generation plant. The quality of manufacturing is referred with respect to properties (mechanical, corrosion etc) and soundness of base metal and welds. Superior quality of manufacturing ensures larger margins between propertiesachievedand properties specified. Larger margins facilitate pressure boundaries to withstand material degradation for longer service life.

Degradation mechanisms mentioned in Table -2are the degradation observed in the PWR primary Circuit. Some of the locations are identified in the table-3 but degradation is not limited to the locations identified in table-2, it can occur at any location where favorable conditions for degradation are formed.

6. Pre-Service inspection (PSI) and In-Service Inspections (ISI):

Pre-service inspections are performed after installation & commissioning of primary equipment and piping into steam generation plant. Non-destructive examinations (NDE) are performed during PSI for recording the condition of pressure boundary and generate the Base Line Data. ISI is performed at periodic intervals during the service life of nuclear power plant to monitor & record condition of pressure boundary of primary circuit. ISI is concentrated to the areas which are prone to failure due to high stress, fatigue, corrosion or identified from design considerations.The primary circuit pressure boundary contain active coolant, the activity level of which further increases in case of fuel cladding failure as fission products are carried overthrough the coolant into the primary circuit. Fission products are accumulated at stagnant locations and increase the activity level around those areas called as hot spots. In view of radioactivity issue faced in ISI, remotely operated tools are deployed for ISI to keep radiation dose to the inspection personnel as low as reasonably possible (ALARA). The base line data generated in PSI serves as reference to evaluate gradual deterioration (if any) in condition of the pressure boundary. In order to have one to one comparison of examination results, the NDE used for PSI should be same as planned for subsequent ISI.

7. Non Destructive examination (NDE) for PSI & ISI:

NDE is performed during PSI and ISI to detect, monitor the cracking or thinning before it reaches critical dimension. It facilitates further analysis and corrective action to ensure safe operation of the plant.

Visual examination (VE), liquid penetrant examination (LPE), ultrasonic examination (UE), eddy current examination (ECE), magnetic particle examination (MPE) and thickness gauging are performed during PSI & ISI. VE is the basic examination to assess the surface condition, leakage and general mechanical & structural condition of components and supports. Visual inspectionincludesremote operated televisedcameras, bore scopes, periscopes etc. LPE is performed to detect surface discontinuity as well as to assess surface discontinuity depth qualitatively after visual examination. MPEis performed to detect surface and shallow subsurface discontinuities in ferromagneticmaterials.

Radiation exposure control and access to examination area are the two issues at the time of ISI. The remotely operated tools are required to be qualified to demonstrate that whole area of interest is examinable and specified sensitivity of examination is achievable. Examinations are performed underwater and from distant place to restrict the dose to inspection personnel. In view of these aspects NDE like UE, VE and ECE are preferable techniques for ISI. LPE is performed at SS weld surfaces, which are dry and accessible. MPE find limited use on accessible CS and LAS welds in equipment and piping.

UE is performed for PSI and ISI of base metal, welds in the equipment and piping. This is due to the reason that ultrasonic examination is most suitable examination for detecting, locating and sizing the planer defect. Remote operated televised camera designed to perform under water examination is used for visual examination of the internal surfaces of reactor pressure vessel, accessible primary pressure boundary and vessel head. ECE is deployed for examination of steam generator tubes.

8. Acoustic Emission Monitoring:

Acoustic emission (AE) monitoring is considered passive non destructive technique as waves are generated inside the material being monitored under load or pressure. Whereas, in other non destructive techniques the source emitting thewaves are generally applied to the material. Release of energy in form of pressure waves, leakage noise and collision with boundary is picked up by acoustic sensors installed. AE monitoring is being used for leak detection, loose part monitoring in the nuclear plants. Another promising usage of AE is monitoring discontinuity propagation at the time of hydro testof equipment and piping. ASTM E 569 provides guidelines for conducting AE monitoring during hydro test or load test.It provides AE source location,source activity and intensity [6]. The size of the critically active intense source can be determined by subsequent ultrasonic examination. This AE monitoring should be deployed during hydro test of nuclear equipment to identify critical areas (if any) and generate the portrait of initial condition of the equipment welds and base metal which can be monitored further during PSI and ISI.ASTM E 1139 – 02 may be extended for monitoring of pressure boundary of PWR. The standard recommends its application to nuclear reactor also.

9. Challenges and Solutions:

The challenges foreseen in implementation of AE monitoring system on PWR pressure boundary of primary circuit are accessibility, radiation related issues, plant noise, optimization of number of sensors, quality of AE signals generated from materials involved etc.A large portion ofequipment and piping pressure boundaryremains inaccessible for mounting of the sensors. The most critical equipment is reactor pressure vessel, which is installed in the reactor pit and thermal insulation is placed over the pressure boundary.High temperature, no direct accessibility and radiation compels sensor placement on wave guides and dry contact. ASTM E 650recommends minimum pressure of 0.7Mpa for sensor mounting. Experimentation is required to determine the effectiveness of sensor placement using wave guide and dry contact to establish optimum contact pressure for maximum AE signal transmission [8]. Radiation resistant sensors are available which can withstand a radiation dose of up to 1000MRad and temperature of up to 540 ^{0}C [9]. Still it is recommended that sensors should be placed at locationswhich are accessible for sensor replacement to facilitate monitoring throughout plant life.

Pressure boundary areas expected to experience high stresses, fatigue or corrosion and susceptible for maximum degradation are to be identified. Sensors installation restricted to highly susceptible areas facilitate optimization of the number of sensors and associated cabling.Initially few susceptible areas like belt line region of RPV and surge line (piping connecting Pressurizer to hot leg) may be considered for implementation of AE monitoring system. The reactor pressure vessel belt line region and welds near this region are expected to experience fatigue and variousembrittlements, which compound the complexity. The surge line is exposed to thermal fatigue which may lead to failure. Measures need to be initiated at plant design stage to facilitate mounting of wave guides in these regions and associated cabling to control room. Plant noise during reactor operation is anticipated to interfere with the AE signals emitted. Experimentation at Watts Bar Unit 1 established that the coolant flow noise could be filtered out and AE signals are detectable under reactor operating condition [10].

Acoustic Emission signal characteristics are significantly influenced by material and stress conditions. It is observed that ductile material of reactor pressure vessels results in weak acoustic emissions [11]. In case of stainless steel piping it is anticipated that acoustic signals may be weak and sensor arrangement shall take care to pick up the signals.

10. **Conclusion:**

Implementation of AE monitoring during shop hydro test will evaluate equipment and piping globallyand furnish information about critical areas. The subsequent NDE can be concentrated to critical areas to determine the size of discontinuity and its location along the thickness, which is critical for further analysis. It will boost user confidence in equipment & piping pressure boundary integrity prior to deployment in steam generating plant. The data generated in AE monitoring during shop hydro test will serve as reference for further PSI & ISI. RPV for VVER reactors manufactured at SKODA plant are subjected to AE monitoring during the hydro test. AE monitoring is being used by RPV manufactured at plants in Czech Republic, Hungary & Slovakia [10].

Continuous AE monitoring during reactor operation needs extensive mock ups, experimentation and design efforts before implementation. Literature survey provides positive finding from experiments conducted by researchers and their findings. Few critical locations as mentioned above may be identified in the pressure boundary and development work can be initiated for implementation.In case filtering reactor operation noise become distant dream for meaningful implementation of continuous monitoring, monitoring can be performed during hydro test after refueling outage to collect information about selected critical areas. Once reliability of AE monitoring is established, ISI may be localized or its quantum may be reduced to critical areas only. This can definitely help in reducing the radiation exposure to in-service inspection personnel.

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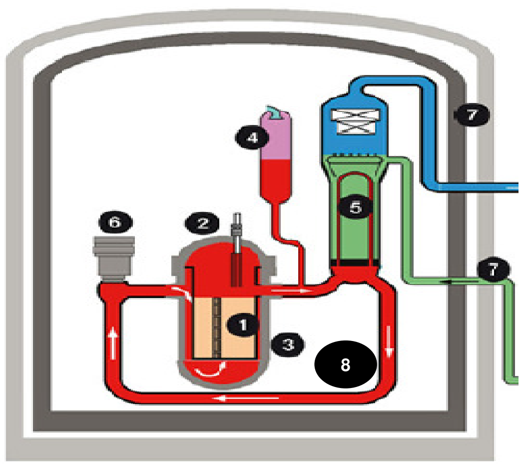

1. Reactor Core
2. Reactor Control Mechanism
3. Reactor Pressure Vessel
4. Pressurizer
5. Steam Generator
6. Reactor Coolant Pump
7. Containment
8. Piping

Sketch -1: Typical Primary system of PWR

Table – 1PWR Primary Circuit Pressure Boundary Materials

Sr. No.	Equipment/piping	Material
1.	**Reactor Pressure Vessel (RPV)**	Low Alloy Steel (SA 533 or SA 508) clad with 300 Series Austenitic Stainless steel. Inconel cladding near penetrations in RPV head.
2.	**Steam Generator (SG)**	
	Channel Head	Carbon or Low Alloy Steel Cladded with 300 Series Austenitic Stainless Steel.
	Tube Sheet/Header	Low alloy steel clad with 300 series austenitic stainless steel or Inconel on primary side.
	Tubes	Inconel / Stainless steel.
3.	**Pressurizer**	Low Alloy Steel (LAS) Cladded with 300 Series Austenitic Stainless Steel.
4.	**Reactor Coolant Pump**	300 Series Austenitic Stainless Steel Castings/Forgings.
5.	**Valves**	300 Series Austenitic Stainless Steel, Corrosion Resistant Hard Surfacing and Packing.
6.	**Piping**	Stainless steel or Carbon Steel (CS) Cladded with 300 Series Austenitic Stainless Steel.

Table-2 Degradation Mechanisms ofPWR Primary Circuit Pressure Boundary

Sr No.	Pressure Boundary	Degradation Mechanism	Locations	Influencing Parameters
1.	**Reactor Pressure Vessel**			
1.1.	Belt line Region	Irradiation Embrittlement	Ferritic base metal, weld and heat affected zone (HAZ).	Chemistry of material, manufacturing process, neutron flux, energy of neutrons, irradiation time.
1.2.	Belt line Region	Thermal Aging	Ferritic material.	Diffusion of alloying elements/impurities resulting in increase in DBTT. Cu content in steel.
1.3.	Weld & HAZ Region	Temper Embrittlement	Quench &tempered ferritic material.	P content (well above 0.02%) due to segregating to grain boundaries. (Tempering at 450-500^0 C)
1.4.	RPV Head	Primary Water Stress Corrosion cracking (PWSCC)	CRDM penetrations.	Residual stress, primary water chemistry and Alloy 600.
1.5.	RPV and RPV head	Boric Acid Corrosion	Leakage from sealing surfaces.	Primary Water Leakage, Hot CS or Las surface.
2.	**Steam Generator pressure boundary**			
2.1.	SG tubes [4].	PWSCC	From primary Side (i.e. tube ID) at roll Transition zone (RTZ), Bend regions & dented locations.	Susceptible tube microstructure, high applied or residual tensile stress and a corrosive environment.
2.2.	SG tubes [4].	Outer diameter Stress Corrosion Cracking (ODSCC)	From secondary side (i.e. tube outer diameter),crevices at tube sheet and tube support plates, sludge pile and even free span.	tensile stress, material susceptibility and corrosive environment (high temperature water containing aggressive chemicals)
		Fretting, Wear	Tube support location due to wear and corrosion.	Tube vibrations, rubbing between contact surfaces, thinning fatigue crack.
		High Cycle Fatigue	Tube with dents at the top tube support plate.	High vibration amplitude and low fatigue strength.
		Denting	Mechanical deformation of tubes at CS support plates due to voluminous corrosion products in U bend region at top support plate.	Degree of superheat, water chemistry (bulk chloride and oxygen concentration).
		Pitting	Top or within the cold leg sludge pile region.	Secondary water chemistry (chlorides & sulphates from leakage in condenser & ion exchangers) and sludge pile Leaks from
3.	**Primary Piping [1]**			
3.1.	Piping	Fatigue	Welds & base metal of surge line.	Thermal transients, thermal stratification, thermal striping, Vibrations
3.2.		Thermal Aging	Casting and welds of austenitic stainless steel.	Delta ferrite decomposition resulting loss of fracture toughness and impact value.
3.3.		Stress Corrosion Cracking (Inter-granular & trans-granular)	Stainless steel outside surface	Chloride contamination in combination with water condensation. Sensitized austenitic stainless Steel in oxidizing environment.
3.4.		Flow Accelerated corrosion (FAC)	CS & LAS Piping	Material, flow velocity, fluid temp, fluid oxygen content and component geometry.