GUIDE NO. AERB/NPP/SG/O-2 (Rev.1)

GOVERNMENT OF INDIA

AERB SAFETY GUIDE

IN-SERVICE INSPECTION OF NUCLEAR POWER PLANTS

ATOMIC ENERGY REGULATORY BOARD

AERB SAFETY GUIDE NO. AERB/NPP/SG/O-2 (Rev. 1)

IN-SERVICE INSPECTION
OF
NUCLEAR POWER PLANTS

Atomic Energy Regulatory Board
Mumbai-400 094
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FOREWORD

The Atomic Energy Regulatory Board (AERB) was constituted in 1983, to carry out certain regulatory and safety functions envisaged under Section16, 17 and 23 of the Atomic Energy Act, 1962. AERB has powers to lay down safety standards and frame rules and regulations with regard to the regulatory and safety requirements envisaged under the Act. The Atomic Energy (Radiation Protection) Rules, 2004, provides for issue of requirements by the Competent Authority for radiation installations, sealed sources, radiation generating equipment and equipment containing radioactive sources, and transport of radioactive materials.

With a view to ensuring the protection of occupational workers, members of the public and the environment from harmful effects of ionizing radiations, AERB regulatory safety documents establish the requirements and guidance's for all stages during the lifetime of nuclear and radiation facilities and transport of radioactive materials. These requirements and guidance's are developed such that the radiation exposure of the public and the release of radioactive materials to the environment are controlled; the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation is limited, and the consequences of such events if they were to occur are mitigated.

The Regulatory documents apply to nuclear and radiation facilities and activities giving rise to radiation risks, the use of radiation and radioactive sources, the transport of radioactive materials and the management of radioactive waste.

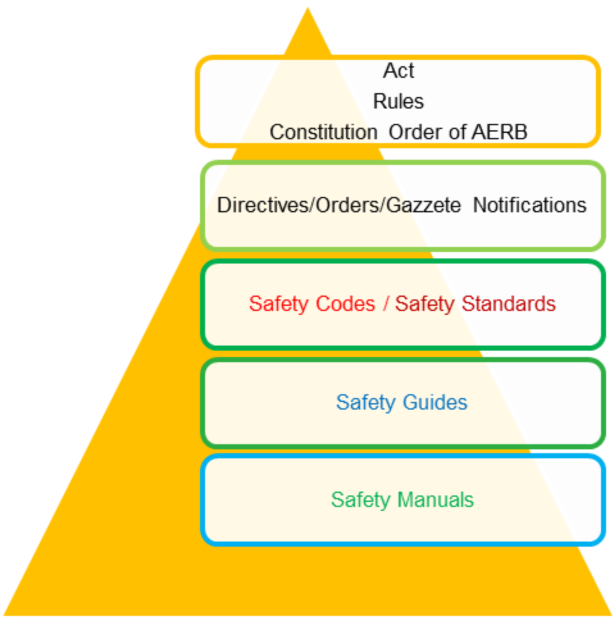

Fig. 1 Hierarchy of Regulatory Documents

Safety codes establish the objectives and set requirements that shall be fulfilled to provide adequate assurance for safety. Safety Standards provide models and methods, approaches to achieve those requirements specified in the safety codes. Safety guides elaborate various requirements specified in the safety codes and furnish approaches for their implementation. Safety manuals detail instructions/safety aspects relating to a particular application.

The recommendations of international expert bodies, notably the International Commission on Radiological Protection (ICRP) and the International Atomic Energy Agency (IAEA) are taken into account while developing the AERB Regulatory safety documents.

The principal users of AERB regulatory safety documents are the applicants, licensees, and other associated persons in nuclear and radiation facilities including members of the public. The AERB regulatory safety documents are applicable, as relevant, throughout the entire lifetime of the nuclear and radiation facilities and associated activities. The AERB regulatory safety documents also form the basis for AERB's core activities of regulation such as safety review and assessment, regulatory inspections and enforcement.

The AERB safety code on 'Safety in Nuclear Power Plant Operation' AERB/SC/O, stipulates the requirements to establish and implement the in-service inspection programme to examine plant structures, systems and components (SSCs) for detecting and identifying possible deterioration and take remedial action. The safety guide, AERB/NPP/SG/O-2 titled, 'In-Service Inspection of Nuclear Power Plants,' 2004, provides guidance to meet the requirements stipulated in the safety code for pressurised heavy water reactors (PHWRs) as well as BWRs. Considering the vast experience gained in the design, construction, safe operation and regulation of many PHWRs of different capacities and the BWRs, it was decided to review and revise the existing version of the safety guide. The contents of this safety guide in general are applicable to both PHWRs and BWRs. The reactor specific guidance's are provided in chapters 4, and 5; where sections 4A and 5A are meant for PHWRs and sections 4B and 5B are for BWRs. The guidance for inspection of major components of PHWRs like steam generators, coolant channels and feeders are provided in Chapter 14.

Safety related terms used in this safety guide are to be understood as defined in the AERB Safety Glossary (AERB/GLO, Rev.1). The special terms which are specific to this safety guide are included under section on 'Special Terms and Interpretation'. In addition, the terms already defined in AERB Safety Glossary AERB/GLO, Rev.1, and being used in this safety guide with a specific context and requires interpretation or explanation are also included in this section.

Appendix is an integral part of the safety guide, whereas references and bibliography are to provide information that might be helpful to the user. For aspects not covered in this safety guide, applicable and acceptable National and International codes and standards shall be followed. Industrial safety shall be assured through good engineering practices and by complying with the Factories Act, 1948 as amended in 1987, and the Atomic Energy (Factories) Rules, 1996.

This safety guide has been drafted by an in-house working group. The draft was further reviewed by a working group with specialists drawn from technical support organisations and institutions, and other consultants. The Comments obtained from all the major stake holders have been suitably incorporated. The safety guide has been vetted by the AERB Advisory Committee on Nuclear and Radiation Safety (ACNRS). AERB wishes to thank all individuals and organizations who have contributed to the preparation, review and finalization of the safety guide

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SPECIAL TERMS AND INTERPRETATION

SPECIAL TERMS

Analogous Components
Components of similar construction, carrying out similar function, located and exposed to similar environment.

Baseline Data
Data generated during PSI conducted on the plant systems and components.

Examination
An element of inspection which involves checking of materials, components, supplies or services to ensure its conformance with the specified requirements.

Flaw
An imperfection, discontinuity, irregularity or fault in the material of a component such as a crack, inclusion, or porosity, lack of penetration, lack of fusion, etc., or an indication that does not meet the acceptance criteria.

Potential
A possibility worthy of further consideration for safety.

Safety Report
A document provided by the applicant or licensee to regulatory body, containing information concerning the facility, its design, safety analysis and provisions to minimise the risk to the public and to the site personnel.

Specification
A written statement of requirements to be satisfied by a product, service, a material or process indicating the procedure by means of which it may be determined whether specified requirements are satisfied.

Testing
The determination or verification of the capability of an item to meet specified requirements by subjecting the item to a set of physical, chemical, environmental or operational conditions

INTERPRETATIONS

Recordable Indication
It is the level of indication for an individual NDT method at and above which the observed indications are required to be recorded and below which the indications are considered as absent and ignored. The level is specified in relevant NDT procedures. [In case of pressure tube, recordable indication has been defined as greater than or equal to 20% of the indication obtained from the reference standard for volumetric inspection using ultrasonic testing].

Reportable Indication
It is the level of indication which is more than recordable indication and should be reported to regulatory body, as a part of inspection report of SSCs. [In case of pressure tubes, reportable indication has been defined as greater than or equal to 50% of the indication obtained from the reference standard for volumetric inspection using ultrasonic testing]

Relevant Indications
These indications are caused by flaws requiring evaluation by a qualified inspector, typically with reference to an acceptance standard, by virtue of their size, shape, orientation or location of the observed discontinuity.

Significant Indications
Indications from an individual NDT method due to flaws which are greater than 100% of the reference size and not acceptable as per the required acceptance criteria of applicable codes/standards

CONTENTS

1 INTRODUCTION

1.1 General

The structure, system and components (SSCs) of Nuclear Power plant (NPP) are subjected to influences like stress, temperature, irradiation, hydrogen absorption, corrosion and fatigue during their operating life. These influences will result in degradation of materials used in the construction of SSCs which in turn will affect their fitness for service. The components of NPPs exposed to these influences need to withstand the effects of various normal and anticipated service conditions. The synergistic effects of the operating environment on materials of construction may not be amenable for accurate prediction. Nevertheless, trending of the degradation in material/ components with time during operation of the plant is very important for ensuring integrity and safety of the SSCs and that of the NPP. An inspection programme for periodic monitoring of the SSCs to detect, assess and trend their condition is an essential aspect of the safe operation of a NPP so that appropriate timely actions are initiated to restore them to a healthy state by way of repair or replacement. This safety guide elaborates the requirements of such inspections to be carried out both prior to start of operation, viz., Pre-Service Inspection (PSI) and during reactor operation, viz., In-Service Inspections (ISI) at prescribed regular intervals.

1.1.1 This Safety Guide supersedes AERB safety guide on 'In-Service Inspection of Nuclear Power Plants,' 2004, (AERB/NPP/SG/O-2).

1.2 Objectives

The objectives of this safety guide are to provide guidance to the Responsible Organisation (RO) with respect to following requirements:

(a) to establish PSI and ISI programmes and develop manuals for their satisfactory implementation at the NPPs from the design stage onwards;
(b) to focus on the importance of the PSI to provide baseline data for eventual comparison with the corresponding data of indications obtained in subsequent ISIs during operating life of the NPP to trend deterioration, if any, and take timely remedial actions;
(c) to stipulate the requirement of conducting PSI before commencement of operation of the plant and its extent including the PSI to be carried out on the SSCs in the event of their replacement or modifications;
(d) to lay down the periodicity and extent of ISI during the service life of the plant along with procedure for disposition of the inspection findings;
(e) to provide guidelines for the formulations and implementation of the inspection programmes using established and approved techniques, trained and qualified personnel and maintenance of inspection records; and
(f) to emphasise on the positive contribution of ISI in enhancing safety of the NPP by helping to predict and prevent equipment/component failures likely to be caused by degradation/deterioration during their operating life and to enhance the confidence in continued safe operation of the plant.

1.3 Scope

1.3.1 The safety guide is applicable to the SSCs of land based NPPs of PHWR and BWR types.

1.3.2 The safety guide outlines the provisions relevant to ISI of safety-related systems and pressure-retaining components including their supports. This guide also covers the classification of areas for inspection, extent of inspection, responsibilities of agencies involved; inspection techniques and procedures; personnel qualifications; scheduling of inspections; acceptance criteria; evaluation and disposition of inspection results; non-conformance control; independent verification; repairs, replacements, modifications; documentation; audit; first of a kind systems and inspection of some of the major components of PHWRs.

1.3.3 Though the safety guide is not intended for facilities other than NPPs and does not cover non-safety related SSCs of NPPs, the principles stated herein can be applied to such items in NPPs and to other nuclear facilities, including research reactors.

1.3.4 This safety guide does not cover the inspection requirements of equipment/components associated with electrical, control and instrumentation, and civil structures. The surveillance requirements of equipment/components associated with electrical, control and instrumentation are elaborated in AERB safety guide on, 'Surveillance of Items Important to Safety in Nuclear Power Plants,' (AERB/SG/O-8). The ISI of civil structures are separately covered in AERB safety manual, ‘In-Service Inspection of Civil Engineering Structures Important to Safety of Nuclear Power Plants,’ (AERB/NPP/SM/CSE-2).

1.3.5 This safety guide does not cover the inspection requirements for secondary cycle piping prone to flow assisted corrosion (FAC) consideration and non-nuclear pressure vessel examinations as required by Atomic Energy (Factories) Rules, 1996. However, a programme for in-service inspection of these components should be in place.

1.3.6 The aspects related to ageing management are separately covered in AERB safety guide on ‘Life Management of NPPs,’ (AERB/SG/NPP/O-14).

1.3.7 The safety guide is not deemed to be restrictive to the inclusion of additional requirements if found necessary based on the operating experience.

2 REQUIREMENTS FOR PSI AND ISI PROGRAMME

2.1 General

2.1.1 Pre-Service Inspection (PSI) data should be collected prior to commencement of initial operation (after erection and hydro-test) of NPPs to generate the base line data which can be used as reference for subsequent ISI carried out during operation of NPPs.

2.1.2 In-Service Inspection (ISI) of NPP should be carried out at regular intervals during its service life whose length should be chosen based on conservative assumptions, to ensure that any deterioration of the most exposed component is detected before it can lead to failure. Applicable code requirements, operating history, results of earlier examinations should also be considered while deciding the inspection intervals. The inspection schedule should provide for repetitions of the inspections spread over the operating lifetime of the nuclear power plant. The schedule of the ISI programme may be at regular intervals or, alternatively, at the varied intervals over the operating life of the plant to improve the correlation between inspection intervals and the inspection findings affecting the fitness for service of the SSCs involved. The intervals for evenly (regular) distributed inspections may be from a few years to ten years and for the variably distributed inspections, the intervals might be shorter in the early years of the plant's operation and can be lengthened or shortened as experience is gained.

2.1.3 The ISI programme should include periodic inspections of fluid-retaining and pressure boundary components, heat transport piping systems, reactor shutdown systems, decay heat removal systems and all other systems and components whose failure could jeopardise the functioning and integrity of safety systems and hence safety of the NPP. The extent of the ISI should ensure detection of unacceptable deterioration of the component(s) during operating life of the NPP.

2.1.4 Non-destructive examination techniques, which are extensively used for inspections are under continuous evolution. When new techniques are introduced for inspection, a comparison with the previous technique should be made. Such a comparison should provide a revised baseline data for future inspections.

2.1.5 Characterisation of flaws, their types and monitoring the growth of flaw sizes are an integral part of inspection. Alternate techniques may be employed to confirm some of the primary findings and the techniques should be validated to prove their efficacy for the intended application.

2.1.6 PSI and ISI should also be performed on such components or systems which are used beyond the conditions of proven experience and do not come under the purview of PSI/ISI, as per the normal governing selection criteria. For example, SSCs required for handling design extension conditions, some of the components given in Chapter 14, etc., come under this category.

2.1.7 Inspection programme involves several methods of testing (including leakage testing of systems) at regular intervals.

2.1.8 The extent of the inspection programme should commensurate with the safety significance of SSCs.

2.1.9 The inspection programme should be set on a firm and rational basis by considering not only the consequence of failure of the components but also the factors that determine the likelihood of such failures.

2.1.10 The inspection findings should be analysed and necessary corrective actions as per applicable codes and guides should be taken in a timely manner and reported to appropriate authorities.

2.2 Station Specific PSI and ISI Manuals

2.2.1 A comprehensive generic PSI/ISI manual should be prepared by Responsible Organisation (RO), wherever applicable. Based on this manual, station-specific PSI/ISI Manual should be prepared by the Plant Management (PM) and issued for implementation.

2.2.2 The PSI manual should be available at each station prior to start of PSI activities.

2.2.3 The ISI manual should be available at each station prior to first approach to criticality.

2.2.4 The PSI and ISI Programme manuals should include, but not limited to, the following:

(a) Scope of the PSI/ISI document.
(b) Applicable codes, standards and guides.
(c) Selection of Structures, Systems and Components (SSCs).
(d) Categorization of SSCs.
(e) Equipment, methods/techniques and procedures.
(f) Extent of Inspection.
(g) List of Inspection areas.
(h) Roles and responsibilities for implementation of the programme.
(i) Staffing, training and qualification/certification of personnel.
(j) Inspection intervals. (Applicable to ISI programme).
(k) Acceptance criteria for the observed indications.
(l) Reporting criteria for the observed indications.
(m) Data acquisition, interpretation and reporting of results.
(n) Evaluation, analysis and disposition of results.
(o) Additional examinations, repetitive examinations and successive inspections (Applicable to ISI programme).
(p) Non-conformance control.
(q) Independent verification of PSI/ISI results.
(r) Repair, replacement and modifications.
(s) Documentation.
(t) Audit.

2.2.5 The flow chart of activities to be carried out as part of PSI/ISI programme is given in Appendix-A**.**

2.2.6 The ISI programme manual should be reviewed and updated periodically at least once in five years to incorporate the state-of-the-art techniques, ~~and~~ experience gained in earlier ISI campaigns and due to the continuing advancements in technology.

2.3 Design Considerations for ISI

The design should take into consideration available state-of-the-art inspection techniques and ensure that the components are amenable to ISI so that the condition and the residual life of the components can be determined.

2.3.1 The considerations to be given at the design stage for effective implementation of ISI programmes:

(a) Ensuring accessibility to the components and feasibility of conducting inspections as per requirements.
(b) Optimisation of radiation shielding.
(c) Adequate provisions for removal, storage and installation of structural members,

shielding components, insulating materials, other equipment and components as necessary to perform the required examinations and tests.

(d) Adequate space in the plant layout for installation of supports, handling machinery, fixtures, platforms, etc., to facilitate removal, disassembly, reassembly, placing and mounting of inspection equipment and/or probes.

(e) Provision for conducting examinations by alternate methods.

(f) Provision of facilities for decontamination of systems, equipment and working areas.

(g) Built-in provisions to enable examinations remotely to reduce radiation exposure, especially in the case of reactor pressure vessels and core internals; coolant channels, feeders, etc., in the case of PHWRs; steam generators, etc.

(h) Weld configuration, surface finish of components, and minimisation of crud or corrosion product build-up by suitable design and selection of materials to enable ease of inspection.

(i) Provision for repair or replacement of systems or components due to observed defects or flaws in SSCs.

(j) Provision for adequate electrical power supply points and other services at convenient places for use of equipment and instruments required inside reactor building for inspection.

(k) Provision to facilitate entry of power and instrumentation cables from outside to inside the reactor building to enable the use of inspection equipment without bypassing the containment.

(l) In the case of some components of SSCs, where the required inspections cannot be carried out due to design and/or maintenance constraints, adequacy of measures taken at the design stage to ensure the healthiness of such components for the intended design life should be independently verified and documented.

(m) Provision for exposure of representative test coupons/specimens at appropriate locations for effective implementation of material surveillance programme to assess changes in the properties of material of construction of critical core components, brought about by prolonged exposure to service conditions like temperature, coolant chemistry, neutron irradiation, etc.

2.4 Considerations during Construction and Commissioning for ISI

The considerations should include the following:

(a) The geometry, configuration, profile and surface finish of weld joints.

(b) The need for change/deviation from original design and design provisions for ease of conducting inspections.

(c) All such changes/ deviations should be subjected to the same process of approval as applicable to new designs.

2.5 Maintenance of Documents, Calibration Standards and Archived Samples

2.5.1 As-built manufacturing and construction data should be compiled for all SSCs which are required to be subjected to inspections. The following documents, calibration standards and archive samples of materials should be available and maintained permanently at the plant:

(a) As-built drawings of the SSCs and their manufacturing details, such as materials used, locations of welds, technical justifications and approval for deviations from the original design.

(b) Records of inspections, examinations and tests including deviations, if any, from the recommended/standard procedures during and after manufacture of components with approvals for deviations.

(c) Calibration standards for tests using eddy current, ultrasonic or any other examination techniques, as applicable, as per the ISI requirements of the component.

(d) Archive samples of materials used in the construction of components which would be undergoing future examinations, and whose material properties are expected to change due to service conditions like thermal aging, irradiation embrittlement, corrosion, etc., as these properties might have a bearing on fitness for service evaluation. For example, off-cuts of pressure tubes; base metal and welds in case of reactor pressure vessels, etc.

2.6 Examination Procedure

2.6.1 Examinations should be carried out in accordance with documented and approved procedures for each inspection method to be employed.

2.6.2 The examination methods and techniques used should comply with the requirements of the applicable codes, guides and standards.

2.6.3 The examination procedure should be in accordance with the current practice for the examination of the component or system.

2.6.4 Trials using suitable mock-ups should be carried out to qualify the procedures, personnel and equipment for inspection of components with complex geometries.

2.6.5 Demonstration of reactor worthiness, either through mock-ups or site acceptance tests, (SAT) should be carried out for systems designed to inspect components like coolant channels, steam generator tubes, heat exchangers, etc.

2.6.6 Calibration checks for systems designed to inspect components should be carried out in the same configuration to be employed at the site.

2.7 Staffing, Training, Certification and Qualification of Personnel

2.7.1 An adequate number of trained and certified personnel should be deployed to carry out the inspections.

2.7.2 The inspection personnel should be trained and certified to operate special inspection systems, such as BARC Coolant channel Inspection System (BARCIS) to be used for inspection of coolant channels, before their deployment at site.

2.7.3 *Qualifications*

This section specifies the guidance for minimum qualification requirements for the three levels of inspection personnel for performing inspections, examinations and tests. The requirements for each level are limited to inspection activities and have no relation to organizational position or professional status. The capabilities for different levels of inspection personnel are summarized in Appendix-B.

3 EQUIPMENT, METHODS AND TECHNIQUES

3.1 Equipment

3.1.1 All equipment used for examinations and tests should be of acceptable quality, range, performance characteristics and accuracy in accordance with the applicable codes/standards.

3.1.2 All equipment together with the accessories should be calibrated as stipulated in the governing codes/standards. The equipment along with their calibration records should be properly identified. The validity of the calibrations should be verified regularly in accordance with the Quality Assurance (QA) programme.

3.1.3 *Calibration Standards*

(a) Reference blocks made according to acceptable standards should be used for calibration.

(b) In special cases, where such standards for calibration are not established, reference blocks made of identical material and surface finish subjected to the similar fabrication (construction) conditions as that of the component being examined, should be used.

(c) The reference blocks used during manufacturing stage, PSI and ISI should be the same, wherever practical.

3.1.4 Whenever an anticipated degradation mechanism is typical or considered difficult to be examined using standard reference blocks, reference specimens should be developed. The reference specimen should contain discontinuities and/or conditions that are comparable to existing or anticipated flaws or conditions. It should be used to demonstrate the ability of the inspection system to detect the anticipated degradation, for example, the reference specimens used for detecting nodular corrosion in coolant channels and Inter-Granular Stress Corrosion Cracking (IGSCC) in BWR piping.

3.2 Methods and Techniques

3.2.1 Methods and techniques used for the examinations should be in accordance with the requirements laid down in the applicable codes/standards used in design and construction of the components. The examinations are categorised as visual, dimensional, surface, volumetric and integrative. Components and the corresponding methods of examination normally considered for the ISI programme are given in Appendix-C.

3.2.2 *Visual Examination*

(a) The visual examination is used to check general condition of a part, component or surface including such conditions as scratches, wear, cracks, corrosion or erosion, distortion, position alignment and evidence of leakage. Optical aids such as magnifying glass, cameras, binoculars, borescope, fiberscope, telescope, CCTV and mirrors may be used for visual examination.

(b) Surface replica technique can be used as a visual examination method, provided the surface resolution of the replica is at least equivalent to that obtained by the visual observation.

(c) Surface preparation or decontamination should be carried out before visual examination in such a manner that evidences of degradation are not affected.

(d) Proper illumination and resolution as specified by the relevant standards should be ensured before carrying out visual inspection.

3.2.3 *Dimensional Examination*

The dimensional examination includes inspection for geometric sizes and configuration of a part or components to verify their compliance with the design specification. The variation in geometric sizes due to distortion, wear, alignment, creep, sag, corrosion, erosion, etc., can be measured by direct methods (e.g., scale, micrometer, Vernier callipers, gauges) and indirect methods (e.g., theodolite, ultrasonic and other suitable methods employing transducers).

3.2.4 *Surface Examination*

The surface examination is used to delineate or verify the presence of surface flaws or near-surface flaws or discontinuities using techniques like magnetic particle testing, liquid penetrant testing and eddy current testing.

3.2.5 *Volumetric Examination*

(a) The volumetric examination is used for qualifying a part or component or weld and for detection of a subsurface flaw or discontinuity, its orientation and size (length and depth). This examination is carried out using radiographic testing (RT), ultrasonic testing (UT), eddy current testing (ECT) and remote field eddy current testing (for tubing).

(b) Advanced techniques like Phased Array UT, Time of Flight Diffraction UT, etc., can be used as and when required, for detailed characterisation of the observed indications.

3.2.6 *Integrative Examination*

The integrative examination (e.g., leak detection, hydro-test, acoustic emission, strain measurement etc.) is used for monitoring component/system integrity.

3.2.7 *Alternative and Complementary Examinations*

(a) Alternative examination using a combination of methods or newly developed techniques may be substituted for an existing method, provided the results of alternative examination demonstrates equivalence or being superior to the existing method of examination.

(b) Examination using complementary techniques may be performed to authenticate /verify the results obtained using the existing technique.

4 CRITERIA FOR SELECTION AND EXTENT OF EXAMINATIONS

4.1 General

This chapter addresses the guidelines for the appropriate selection of components and the extent of examination for the PSI and ISI of PHWRs and BWRs. These two reactor types have been designed using different codes/standards and operate under different environments and conditions. Hence, there are differences in selection of components for PSI and ISI although the underlying philosophy for inspection is the same for both types of reactors. In view of this, selection criteria and extent of examination for ISI have been detailed separately for both reactors. This would be useful for preparation of station specific inspection manuals for each type of reactor.

4.2 Scope of ISI and its Optimisation

4.2.1 The selection of components, areas to be examined and the extent of examination for ISI in each type of reactor can be optimised based on the availability of relevant data collected during ISIs for life assessment of various SSCs from similar units. This can be done by considering the existence of similarity of components, service conditions and period of operation. In such cases, an appropriate justification should be prepared by the RO and submitted to the regulatory body for review and approval.

4.2.2 The criteria to be considered for evaluation of similarities are given below:

(a) *Design*

The components should have been manufactured as per the same set of design drawings. Design changes/DCRs if any should be evaluated for any possible effect on the component performance.

(b) *Material*

All materials used in the manufacture of various parts/components should be of the same specification.

(c) *Manufacturing Process*

The parts/components should be manufactured by the same process.

(d) *Storage and Preservation*

The storage and preservation should not have caused any significant degradation of the parts/components. In case some degradation has been observed during the storage and preservation, the relevance of the same should be duly evaluated with respect to the performance of the component and the subsequent inspection requirements.

(e) *Service Conditions*

(i) Similarity in operating parameters

The fluid flow, pressure, temperature and environment including irradiation should be the same for the components under consideration.

(ii) Coolant chemistry should be compatible with the material.

The pH, conductivity, dissolved oxygen, dissolved solids, chlorides and other relevant parameters affecting the corrosion or other degradation of the component should be in the same range.

(iii) Similarity in service period

The important parameters to be considered are Effective Full Power Years (EFPYs) and Hot Operating Years (HOYs).

4.2.3 The relevance of the above criteria should be suitably considered for each component/part based on the effect of degradation during optimization of the ISI.

4.2.4 Any differences in consideration of similarity of components should be evaluated with respect to service conditions. The selection of components and inspection intervals should ensure proper and timely detection of degradation of the component. Due credit for the earlier inspections of the similar components can be taken while considering the extent of inspection.

4A: CRITERIA FOR SELECTION AND EXTENT OF EXAMINATIONS FOR PHWRs

4A.1 Criteria for Selection of SSCs

4A.1.1 The SSCs or the portions thereof including the supports, satisfying the following criteria, should be selected for inspection:

(a) Pressure boundaries of reactor coolant or any other systems like piping, valves, pumps, etc., whose failure can result in a significant release of radioactive substances.

(b) Systems essential for safe reactor shut down.

(c) Systems essential for cooling of nuclear fuel in the event of process system failure due to operational occurrences.

(d) Systems and components whose dislodgement or failure might jeopardise the integrity of the system mentioned in the sub-section 4A.1.1.(a) or 4A.1.1.(b) above, or both, such as coolant channel garter springs, moderator inlet/outlet manifold, coolant pumps, flywheel of reactor coolant pump, etc.

(e) Systems whose failure could prevent fulfillment of adequate safety functions, e.g., emergency process water systems, etc.

(f) The fluid boundary of vessels including nozzles to vessel attachment welds.

(g) The supports for components (e.g., for piping, vessels, pumps and valves) and rotating machinery should be considered as part of the fluid boundary whose integrity is relied upon to withstand the design loads and seismic-induced displacements.

(h) Systems and components, including supports required for handling design extension conditions.

(i) Containment pressure retaining components and their integral attachments including liners, metallic expansion joints, penetration attachments, etc.

4A.1.2 The following systems with fluid/pressure boundaries should be subjected to inspection:

(a) The fluid/pressure boundaries of all components and piping including their supports.

(b) The fluid/pressure boundaries referred in the sub-section 4A.1.2.(a) should be considered as follows:

(i) Systems containing nuclear fuel or interconnected systems. These include all portion(s) which does not (do not) have,

(p) two additional barriers between fluid boundary and sheathing of nuclear fuel; or

(q) two additional barriers between fluid boundary and the outside atmosphere; or

(r) one barrier between fluid boundary and the sheathing of the nuclear fuel and the other barrier between it and outside atmosphere.

(ii) The number of barriers mentioned in the sub-section 4A.1.2.(b).(i) may be determined as follows:

(p) Metal boundary is equal to one barrier

(q) Valve, manual, automatic or remotely controlled, that remains closed during normal operating conditions equal to one barrier.

(r) Valve-self-closing, that may be open during Level-A operating conditions (see Annexure-I) -equal to half barrier.

(s) Manually operated valve that remains open during Level-A conditions should not be considered as a barrier, unless it is shown that the valve could be closed within a reasonable time period that is consistent with permissible radiation releases. In such cases the valve should be considered as equal to half barrier.

(t) Containment boundary-equal to one barrier

4A.2 Inspection Category of SSCs

Size of failure, stress intensity and fatigue usage factor should be considered for assigning the inspection category and for determining the extent of examination.

4A.2.1 Procedures for evaluation of size of failure, stress intensity, fatigue usage factor and determination of inspection category are given in Appendix–A.A.

4A.2.2 For mechanical couplings, the inspection category should be determined from Fig. 1(A) or Fig. 1(B) of Appendix-A.A based on the size of failure resulting from a complete failure of the connection. The inspection categories for the various items in a coupling might be different.

4A.2.3 For supports, the inspection category should be determined based on the failure classification of the supported component and are categorised as follows:

(a) All supports should be placed in Category-A where the supported component has a large failure size, and in Category -B where the supported component has a medium failure size.

(b) For a support having a component attachment weld, the inspection category should be determined from Fig. 1(A) or Fig. 1(B) of Appendix –A.A based on the size of failure resulting from a complete failure of the weld. Where a support has two or more component attachment welds, the inspection category for the welds might be different.

4A.3 Extent of Examination

4A.3.1 Pre-Service Inspection (PSI)

The PSI should be performed on all the SSCs, portions of systems and supports that have been selected to undergo ISI.

4A.3.2 In-Service Inspection (ISI)

(a) The extent of examination should be determined based on the inspection category, as explained in these sub-sections and the subsequent sub-section 4A.4.

(b) The ISI should be extended to include those SSCs, whose operating conditions or operating behavior differs significantly from that contemplated in the design. The inspection programme for such SSCs should be determined on a case-by-case basis.

4A.3.3 Inspection of an item designated as spares for replacement should be the same as that required for the original item. Spares for replacement means an item that satisfies all the following characteristics:

(a) The item would normally be replaced rather than repaired.

(b) A replacement for the item is available.

4A.4 Sampling Criteria

4A.4.1 Sampling:

(a) The selection of samples for examination should be from the regions:

  (i) containing the most significant acceptable flaws found during the PSI,

  (ii) that are most vulnerable to corrosion (e.g., two phase flow, outside surface attack) or erosion (e.g., turbulence, resonance and water hammer, etc.),

  (iii) exposed to the most severe conditions of service in terms of stress, particularly cyclic stress, and

  (iv) subjected to the highest amount of creep or irradiation or both.

(b) Successive examinations should be performed on the same selected regions (as given in the sub-section (a) above) during each successive inspection interval, except in such cases, where the examination results indicate a change or special operating conditions or degradation mechanisms not envisaged in design.

(c) Dissimilar metal weld joints (having different P numbers as per the Section IX, ASME Boiler and Pressure Vessel Code) should be treated separately from similar metal weld joints (having the same P numbers).

4A.4.2 General requirements for sampling:

(a) The selected area should be sufficient for inspection of a reasonable amount of base metal and if the area includes a weld joint then the full width of the weld zone.

(b) The selected area on each side of a weld joint should extend to a minimum distance of one-half the component thickness (base metal) from the point of maximum width of the final pass of the weld and should in no case be less than 13 mm, whichever is greater. For butt welds, the material thickness should be considered to be equal to the mean weld thickness. For fillet welds, the selected area for inspection should be considered from the edge of the weld.

(c) The selection of area for volumetric inspection where through-wall coverage is required should be:

  (i) the full wall thickness of component or

  (ii) it should not be less than one-third of the nominal wall thickness measured from the internal surface, provided that the external surface is inspected using surface examination methods.

(d) The selected area(s) for inspection where the weld joint contains weld intersections should cover all such intersections.

(e) The area showing the most significant indication in each pipe run[1] or component should be volumetrically examined. The examination should be extensive enough so that changes in the indication can be monitored. Engineering analysis considering stress state, flaw characterisation, and its location should be used to determine whether the indication is significant.

(f) A weld joint between piping and any other component should be considered as a piping weld for inspection purposes.

(g) Welds provided exclusively for additional sealing purposes may be excluded from inspection, but these areas should be covered during system leak checks/hydrostatic tests.

(h) Where the stress level is known to vary significantly along the length of a weld joint, the selection of area(s) for inspection should cover the region(s) having the highest stress.

4A.4.3 Selection of areas for inspection of equipment:

(a) Piping

The length of the selected area should be the full length of a circumferential weld or a branch intersection circumferential weld. For a longitudinal weld the selected area should be minimum 300 mm long or three times the mean weld thickness, whichever is greater, along the weld.

(b) Vessels

  (i) All the pressure retaining welds should be inspected. The length of the selected area should extend along the centre-line of the weld for a minimum of 300 mm or three times the mean weld thicknesses, whichever is greater.

  (ii) Where the total selected area for a weld joint is divided into two or more locations, the length of any selected area should not be less than three times the mean weld thickness.

  (iii) Nozzles, openings, and penetrations should be included in the selected area when their internal diameter is greater than $0.2\sqrt{Rt}$ where 'R' is the mean radius of shell or head, and 't' is the nominal thickness of shell or head.

[1] Pipe run: A length of piping that has common specification and extends to but not beyond a large component (e.g., vessel, pump and anchor) or a piping intersection/junction. The pipe run may extend beyond an intersection/junction where the pipe has an extruded outlet or a weld-on fitting for the branch connection.

(c) Mechanical couplings
- (i) *Bolting:* The selected area should include the complete fastener (e.g., bolt, stud, and nut) and the flange ligaments between threaded stud holes.
- (ii) *Other components*: The selected area should be determined on case-by-case basis.

(d) Pumps and Valves

All pressure retaining welds should be inspected. The length of the selected area should extend along the centre-line of the welds for a minimum of 300 mm or three times the average weld thickness, whichever is greater.

4A.4.4 Extent of Examinations and Inspection Categories

The extent of examination of the equipment in each inspection category should be as follows:

4A.4.4.1 Category-A

(a) Piping

(a).1 In the case of 220 MW(e) units the following welds should be subjected to inspection:
- (i) A minimum of one weld joint in each pipe run. The weld joint having the highest fatigue usage factor should be selected.
- (ii) Where fatigue usage factors are not calculated, the weld joint having the highest stress ratio should be selected.
- (iii) The representative weld joints between dissimilar materials, if any.
- (iv) Terminal weld joints in the pipe run.
- (v) Intersection welds.
- (vi) If the weld joints falling under the purview of ISI are not accessible then suitable alternate weld joints should be selected based on the criterion of stress intensity.
- (vii) In case any reportable indications are observed in a weld joint during PSI/earlier ISI/current ISI campaign then the same weld joint should be inspected for the next three successive ISI campaigns in addition to the scheduled quantum of inspection.
- (viii) If any generic degradation is observed in any weld joint then the observations should be reviewed in consultation with the regulatory body for enhancing the quantum of weld joints for ISI during the same inspection campaign.

(a).2 In the case of 540 MW(e) and 700 MW(e) Units:

The ISI quantum in each ISI interval should be completed based on the criteria as indicated below from (i) to (v) subject to a minimum quantum of 25% of the total welds:
- (i) A minimum of one weld joint in each pipe run. The weld joint having the highest fatigue usage factor should be selected.
- (ii) Where fatigue usage factors are not calculated, the weld joint having highest stress ratio should be selected.
- (iii) Terminal weld joints in the pipe run.
- (iv) Intersection welds
- (v) All dissimilar metal weld joints.
- (vi) If the weld joints which fall under the purview of ISI are not accessible, suitable alternate weld joint should be selected based on the criterion of stress intensity.
- (vii) Additionally, the welds where any reportable indications have been obtained during PSI/earlier ISI/current ISI campaign should be inspected

along with the scheduled quantum for the next three successive ISI campaigns.

(viii) If any generic degradation is observed in any weld joint, the observations should be reviewed for enhancing the quantum of weld joints for ISI during the same inspection campaign, in consultation with the regulatory body.

(b) Vessels

All pressure retaining welds should be inspected. For identical welds, the number of welds to be inspected may be reduced based on a review by the designers, subject to approval of the regulatory body.

(c) Heat Exchangers

- (i) For the shell, the sub-section 4A.4.4.1.(b) is applicable.
- (ii) The full length of 25% of the total number of tubes should be inspected. The selection should include vulnerable regions of the tubes prone to failures based on experience.
- (iii) For the other types of heat exchangers, like plate type/finned tube type, the inspections should be carried out as per the instructions of the manufacturer including gasket replacement and leak testing.

(d) Mechanical Couplings

- (i) All bolting.
- (ii) All ligaments between threaded stud holes.
- (iii) All other components.

(e) Pumps

- (i) All pressure-retaining welds in casing and casing to nozzle welds (where applicable).
- (ii) Flywheels of drive motors, fasteners and supports.
- (iii) Internal surface of pump (at least one pump of each type) as per the following:
  - *(p)* If any degradations are observed then the same should be evaluated and disposed of as per procedures given in the Section-7.
  - *(q)* If no degradation is observed, then it is recommended to select another pump of the same type for inspection in the subsequent inspection interval.
  - *(r)* In the case of primary coolant pumps (PCPs) the inspection intervals should be as given in the sub-section 5A.2.11.

(f) Valves

- (i) All pressure retaining welds should be inspected.
- (ii) Internal surface of valve (at least one valve of each type) as per the following
  - *(p)* If any degradations are observed then the same should be evaluated and disposed of as per procedures given in the Section-7.
  - *(q)* If no degradation is observed, it is recommended to select another valve of the same type for inspection in the subsequent inspection interval.

(g) Supports

- (i) All supports.
- (ii) All component attachment welds.
- (iii) Snubbers should be inspected as follows:
  - *(p)* Visual examination of all snubbers for degradation due to corrosion, erosion, mechanical damage/deformation, leakage etc.
  - *(q)* Free operability of all snubbers should be checked for a minimum stroke length based on the actual requirement or as per analysis with an additional 10% margin on stroke length.

(r) At-least one representative snubber of each capacity/type should be tested for its functionality which includes the drag force and sensitivity (locking acceleration/velocity).

(s) If any abnormality/unacceptable behaviour are observed during the functionality test then the root-cause-analysis for the unacceptable behaviour should be established and corrective actions should be incorporated for the affected snubber. If the root-cause of unacceptable behavior could not be established clearly, then all the snubbers of similar type/capacity should be tested for sensitivity and drag force.

(h) Rotating Machinery
All regions (e.g., flywheel and pump motor shaft) should be inspected for structural degradations.

4A.4.4.2 Category-B

(a) The SSCs falling in Category-B should be inspected as per the criteria in the sub-section 4A.4.4.2.(d) below, provided a leak detection system is in use,
   (i) that provides continuous monitoring for leakage; and
   (ii) has a sensitivity to detect and indicate incremental leakages in excess of 8g/s (480 g/minute).

(b) When it can be demonstrated to the regulatory body, that SSCs not complying with the condition mentioned in the sub-section 4A.4.4.2.(a) is acceptable, then they should be inspected as per the criteria given in the sub-section 4A.4.4.2.(d).

(c) Where the conditions of the sub-sections 4A.4.4.2.(a) and (b) cannot be satisfied, the SSCs should be treated as per the requirements of Category-A (see the sub-section 4A.4.4.1).

(d) The SSCs complying with the sub-sections 4A.4.4.2 (a) or (b) should be inspected in accordance with the following:

   (i) Piping
      (p) A pipe-run having one or more Category-B regions, the weld joint having the highest fatigue usage factor should be inspected. If a pipe-run has a Category-A region, then the inspection should be carried out as per Category-A requirements.
      (q) Where fatigue usage factors are not calculated, the weld joint having the highest stress ratio should be selected for inspection.
      (r) The quantum of ISI for each inspection interval should be completed based on the criteria as indicated above 4A.4.1
      (s) The minimum quantum of welds inspected should be 7.5% of total welds.
      (t) Additionally, in case any recordable indications are observed in a weld joint during PSI/earlier ISI/current ISI campaign, then the same weld joint should be inspected for the next three successive ISI campaigns along with the scheduled quantum.
      (u) Representative dissimilar metal weld joints.
      (*v*) Terminal weld joints.
      (*w*) Intersection welds.
      (*x*) If the weld joints falling under the purview of ISI are not accessible, then suitable alternate weld joints should be selected based on the criterion of stress intensity.

   (ii) Vessels
      (p) All pressure-retaining welds having the highest fatigue usage factor should be inspected.
      (*q*) The number of welds inspected should not be less than one-third of the welds in Category B.

(*r*) For identical welds, the number of welds to be inspected may be reduced based on a review by the designers, subject to approval of the regulatory body.

(s) Where fatigue usage factors are not calculated, the joints having the highest stress ratio should be selected for inspection.

(iii) Heat Exchangers

(p) For the shell the sub-section, 4A.4.4.1.(b) is applicable.

(q) The full length of 25% of the total number of tubes should be inspected. The selection should include the vulnerable regions of the tubes prone to failures based on experience.

(r) For the other types of heat exchangers, like plate type/finned tube type, the inspections should be carried out as per the instructions of the manufacturer including gasket replacement and leak testing.

(iv) Mechanical Couplings

(p) *Bolting:* 10% of the total number of fasteners in the joint, rounded to the next higher integer, should be inspected.

(q) *Flange Ligaments:* 10% of the flange ligaments between threaded stud holes, rounded to the next higher integer, should be inspected.

(r) *Other Components:* The extent of the inspection should be considered on a case-by-case basis.

(v) Pumps

(p) Pressure-retaining welds of casings and casings to nozzle welds having the highest fatigue usage factors in this category should be inspected.

(q) Where fatigue usage factors are not calculated, the welds having the highest stress ratios should be selected for inspection.

(r) Flywheels of drive motors, fasteners and supports.

(s) The number of welds inspected should not be less than one-third of the welds in Category-B.

(s) The examination of internal surface of at least one pump of each type should be carried out as per the following:

(i) If any degradations are observed, then the same should be evaluated and disposed of as per procedures given in the Section-7.

(ii) If no degradation is observed, then it is recommended to select another pump from the same type for inspection in the subsequent inspection interval.

(vi) Valves

(p) Pressure-retaining welds having the highest fatigue usage factors in this category should be inspected.

(q) Where fatigue usage factors are not calculated, the welds having the highest stress ratios should be selected for inspection.

(r) The number of welds inspected should not be less than one-third of the welds in Category-B.

(s) Examination of internal surface of at least one valve of each type should be carried out as per the following:

(i) If any degradations are observed then the same should be evaluated and disposed of as per procedures given in the Section-7.

(ii) If no degradation is observed then it is recommended to select another valve from the same type for inspection in the subsequent inspection interval.

(vii) Supports

(p) All supports should be inspected.

(q) Where a support has more than one component attachment welds, at least one component attachment weld, having the highest fatigue usage factor, should be inspected.

(r) Where fatigue usage factors are not calculated, the component attachment welds having the highest stress ratio should be selected for inspection.

(s) Snubbers should be inspected as follows:

(i) Visual examination of all snubbers for degradation due to corrosion, erosion, mechanical damage/deformation, leakage, etc.

(ii) Free operability of all snubbers should be checked for a minimum stroke length based on the actual requirement or as per analysis with an additional 10% margin on stroke length.

(iii) At-least one representative snubber of each capacity/type should be tested for its functionality which includes the drag force and sensitivity (locking acceleration/velocity).

(iv) If any abnormality/unacceptable behaviour are observed during the functionality test, then the root-cause-analysis for the unacceptable behaviour should be established and corrective actions should be incorporated for the affected snubber. If the root-cause of unacceptable behavior could not be established clearly, then all the snubbers of similar type/capacity should be tested for sensitivity and drag force.

(viii) Rotating machinery (Flywheel and pump motor shaft)
The region falling in Category-B and has the highest stress should be inspected (e.g., PCP Flywheel and pump motor shaft). However, if the component falls in Category-A also, then only Category-A regions should be selected for inspection.

4A.4.4.3 Category-C1

(a) No ISI is required for SSCs falling in this category, provided that they do not have any dissimilar metal welds.

(b) When it can be demonstrated to the regulatory body that SSCs not complying with the conditions mentioned in the sub-section 4A.4.4.3.(a) above are acceptable, no ISI is required, except that called for in the sub-section 4A.4.4.5.

(c) Where the conditions of the sub-sections 4A.4.4.3.(a) and (b) cannot be satisfied, the SSCs should be treated as per requirements of Category-B (see the sub-section 4A.4.4.2).

4A.4.4.4 Category-C2

No ISI is required for SSCs falling in this category. However, depending upon their safety significance, appropriate inspection requirements should be followed.

4A.4.4.5 Additional Requirements for Corrosion and Erosion

(a) The following additional inspections should be performed to determine the deterioration due to corrosion or erosion, or both.

(b) Category-A Areas:

(i) Where the system or component, except for pumps and valves, is known to operate under conditions that can be classified as non-corrosive[2] and non-

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[2] Non-corrosive condition:
(a) where corrosion effects are known to be negligible; or
(b) where the corrosion effects are reduced by chemistry or temperature control so that the reduction in material thickness over the intended service life does not exceed 1.5 mm (0.06 inch) or 6% of the wall thickness, whichever is less.

erosive, or measured wall thickness data from inspection are available to prove that non-corrosive and non-erosive conditions exist, no further inspection is required.

(ii) *(p)* For pumps and valves, dimensional inspection for loss of material should be performed.

(q) For pumps and valves subjected to degradation due to corrosion related cracking, surface inspection of the internal fluid boundary should be performed. The selected area for inspection should include internal surface and all areas having the highest potential for the occurrence of corrosion-related cracking.

(iii) For SSCs, except pumps and valves, which do not satisfy the requirements of the sub-section 4A.4.4.5.(b).(i), the following dimensional inspections should be performed:

(p) If corrosive conditions exist in the system, the material thickness should be determined at the locations having the highest corrosion rate.

(q) If there is no area in the system that has a corrosion rate significantly greater than the average for the system, then the material thickness should be determined at the location having the highest stress intensity.

(r) If erosive conditions exist, the material thickness should be determined at the location in the system considered to have the highest erosion rate.

(c) Categories B and C1 Areas

The areas in these categories should be inspected according to the requirements for Category-A, given by the sub-section 4A.4.4.5.(b). However, no further inspection is required for systems that have an inspection area included in Category-A if the corrosion and/or erosion of this area are not greater than that of the inspected area of Category-A.

(d) Category- C2 areas

No additional inspection is required for systems and components that are in this category. However, depending upon their safety significance appropriate inspection requirements should be followed.

4A.5 Confirmatory Inspection[3]

4A.5.1 In the cases where inspections have been performed prior to the hydrostatic pressure test of the component, a confirmatory inspection should be carried out as per the conditions given in the following sub-sections after the component hydrostatic pressure test. If the results of confirmatory inspection indicate no significant change when compared with the results obtained during inspections carried out on the component prior to the hydrostatic pressure test, then the results of the inspections carried out before the hydrostatic pressure test should qualify as PSI.

4A.5.2 In the case of any degradation observed in the confirmatory inspection, the same should be considered as base line data (PSI). The extent of in-service inspection should be enhanced appropriately, subject to approval of the regulatory body.

4A.5.3 Provision for confirmatory examinations should be exercised only if permitted by the applicable codes/standards.

4A.5.4 The confirmatory inspection should employ the same methods as those used for the PSI.

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[3] *PSI requirements of BWR covered in the chapter 5B. Confirmatory test concept is not applicable for BWRs.*

4A.5.5 Areas/components identified for the confirmatory inspection should be a part of the ISI and should include the following:

(a) Piping
The number of inspection areas selected should be equal to at least 10% of the periodic inspection sample for the piping system. The area selected for inspection should include the areas with the most significant indications detected previously (prior to the pressure test).

(b) Vessels
(i) The full length of all major nozzle welds and,
(ii) the most significant indications detected previously in the longitudinal and circumferential joints such that at least 10 periodic inspection areas (or all if fewer than 10 exist) per component or 10% of the indications, whichever is greater, are inspected.

(c) Pumps
(i) Full inspection of at least 10% of all the pumps of a system subject to a minimum of one pump, including all Category-A & B pressure-retaining welded joints.
(ii) At least 10% of the locations with the most significant indications.

(d) Valves
(i) At least 10% of the valves (one valve minimum), including all Category A and B pressure-retaining welded joints in the selected valve; and `
(ii) at least 10% of the locations with the most significant indications.

(e) Supports
At least 10% of all the supports of equipment, vessels, pipe, etc.

(f) Steam Generator Tubes
Refer the sub-section 14AS.1.3 for quantum of tubes to be inspected and the sub-section 14AS.1.4 for selection of tubes.

(g) Heat Exchanger Tubes
10% of the tubes in the concerned heat exchanger should be inspected.

4A.6 Criteria for Exemption

4A.6.1 Piping, piping components and segments, which are of size 25NB (NPS 1) and smaller except for steam generator tubing.

4A.6.2 Components that are connected to reactor coolant pressure boundary, which are of such a size and shape, that upon their postulated rupture, the resulting loss of coolant from the reactor coolant system under normal plant operating conditions is within the capacity of the makeup systems that are operable using on-site emergency power. The emergency core cooling systems are excluded from the calculation of makeup capacity.

4A.6.3 Components in systems, which are completely filled with liquid at temperatures below a value corresponding to a vapour pressure of 340 kPa (50 psi) and whose supports are shown to be adequate to withstand the forces resulting from failure of the fluid boundary.

4A.6.4 Components/Welds or portions of welds that are inaccessible due to being encased in concrete, buried underground, located inside a penetration or encapsulated by a guard pipe, and maintenance constraints, as stated by the systems designer, with the approval of the regulatory body. However, their healthiness should be ensured during the system integrity checks.

4B: CRITERIA FOR SELECTION AND EXTENT OF EXAMINATIONS FOR BWRs

4B.1 Criteria for Selection of SSCs

4B.1.1 SSCs or the portions thereof including the supports, satisfying the following criteria, should be selected for inspection:

(a) Pressure boundaries of reactor coolant or any other systems whose failure may result in a significant release of radioactive substances.
(b) Systems essential for safe reactor shut down.
(c) Systems essential for safe cooling of nuclear fuel in the event of system failures or due to operational occurrences.
(d) Systems and components whose dislodgement or failure might may jeopardise the integrity of the system mentioned in 4B.1.1.(a) or (b) above or both such as reactor coolant pumps and their flywheels, stabilisers of reactor pressure vessel, core support structure, core shroud, etc.
(e) Systems whose failure could prevent fulfillment of adequate safety functions, e.g., emergency process water systems.
(f) The fluid boundary of vessels including nozzles to vessel attachment welds.
(g) The support for the components (e.g., for piping, vessels, pumps and valves) and rotating machinery should be considered as part of the fluid boundary whose integrity is relied upon to withstand the design loads and seismic-induced displacements.
(h) Systems and components including supports required for handling design extension conditions.
(i) Containment pressure retaining components and their integral attachments including liners, metallic expansion joints, penetration attachments, etc.

4B.1.2 The following systems with fluid/pressure boundaries should be subjected to inspection:

(a) The fluid boundaries of all components and piping including their supports.
(b) The fluid/pressure boundaries referred in the sub-section 4B.1.2. (a) should be considered as follows:

 (i) Systems containing nuclear fuel or interconnected systems. These include all portion(s) which does not (do not) have,
  (p) two additional barriers between fluid boundary and sheathing of nuclear fuel; or
  (q) two additional barriers between fluid boundary and the outside atmosphere; or
  (r) one barrier between fluid boundary and the sheathing of the nuclear fuel and the other barrier between it and outside atmosphere.

 (ii) The number of barriers mentioned in the sub-section 4B.1.2 (b) (i) may be determined as follows:
  (p) Metal boundary is equal to one barrier.
  (q) Valve, manual, automatic or remotely controlled, that remains closed during normal operating conditions equal to one barrier.
  (r) Valve-self-closing, that may be open during Level-A operating conditions (see Annexure-I) equal to half barrier.
  (s) Manually operated valve that remains open during Level-A conditions should not be considered as a barrier, unless it is shown that the valve could be closed within a reasonable time period that is consistent with permissible radiation releases. In such cases the valve should be considered as equal to half barrier.
  (t) Containment boundary-equal to one barrier.

4B.2 Safety Classification of SSCs

The SSCs are broadly classified based on the functional requirements and safety significance which are given below:

(a) Class 1
The SSCs required to prevent the release of a substantial fraction of core fission product inventory to the containment/environment are classified as Safety Class-1 and include those components that comprise the reactor coolant system pressure boundary. Excluded from Safety Class-1 are those fluid systems/components that are part of the coolant pressure boundary, but the failure of which would result in a loss of reactor coolant inventory well within the make-up capacity of normally operating coolant inventory control systems so as to maintain a coolant inventory sufficient for an orderly cool down to a safe shutdown state.

(b) Class 2
The SSCs that perform the safety function necessary to mitigate the consequences of an accident which would otherwise lead to release of substantial fraction of the core fission product inventory or activation product inventory into the environment are classified as Safety Class 2.

(c) Class 3
It incorporates those safety functions which transfer heat from other safety systems to the ultimate heat sink to ensure necessary services as a support function for Safety Class1 and 2 SSCs.

(d) Class Metallic Containment (MC) and Lined Concrete Containment (CC) Structures: The metallic containment structure (drywell) and the metallic lined pressure suppression chambers viz., the suppression chambers, the common chamber with their penetrations and support structures respectively, come under this category.

4B.3 Extent of Examination

4B.3.1 The SSCs are classified as per their safety functions as Classes 1, 2 and 3 as well as Metallic Containments and Liners of Class CC components and their supports as given in the sub-section 4B.2. Their different examination areas are classified under different categories as given in the Annexure-I.B based on the Design and Surveillance Code (ASME). The general examinations to be carried out on different components are given in the Annexure-II.B.

4B.3.2 The requirements for the extent of examination are to be considered as per the design, construction and surveillance codes followed. Relevant Articles and Tables for Classification of SSCs with examination category as per the Section XI of the ASME-Boiler and Pressure Vessel Code are given in the Annexure-III-B.

4B.3.3 Selection of piping welds for examination is based on the stress intensity factor (SIF) and fatigue usage factor. In the case of non-availability of stress analysis of the piping system for calculation of SIF, 100% coverage of the welds should be ensured for the Class-1 components.

4B.3.4 The number, frequency and extent of ISI of identical SSCs may be reduced by devising a sampling programme that will vary according to the design, the number of identical SSCs involved, operational requirements, or the existence and experience of identical units in a multiple unit plant. The sampling criteria should be consistent with the importance to safety of the component and the rate of degradation.

4B.3.5 For Class MC and liners of Class CC vessels the following are to be inspected:

(a) All areas subject to examination during any inspection interval are required to be examined before initial plant start-up.

(b) If protective coatings are to be applied to any of these items, the examinations must be performed after application of the coatings.

(c) When portion of a vessel or liner is repaired/replaced during the service lifetime of a plant, the PSI requirements for the repaired/replaced portion should be satisfied.

(d) When a system pressure test is required after completion of the repair/replacement activity, the PSI should be performed during, or upon completion of, the pressure test.

(e) Welds, made as part of repair/replacement activities, joining Class MC or Class CC components to items designed, constructed, and installed to meet the requirements of a particular section of the code and/or safety class should be examined as per the code requirements.

4B.4 Sampling Criteria

4B.4.1 Sampling criteria to be followed for selection of SSCs for ISI:

(a) The most significant acceptable flaws discovered during PSI.

(b) The areas most prone to corrosion or erosion or both.

(c) The areas having the most severe service conditions in terms of stress, particularly fatigue.

(d) The areas subjected to creep and irradiation.

4B.4.2 The SSCs like piping and pipe welds in large diameter piping i.e., 4 NB (100 mm) or larger, which are susceptible to outside surface attack due to proximity to nearby leak paths, proximity to chloride bearing material, existence of moisture or salt laden atmosphere etc., should be included in the ISI programme based on plant operating experience and its review. Such welds should be subjected to surface examination during each inspection interval.

4B.5 Criteria for Exemptions

(a) Annexure-IV.B refers to applicable articles for exemption from examination as per safety classification of components based on ASME standards, as a case study.

(b) Welds or portions of welds thereof that are inaccessible because of being covered or placed within concrete, located inside a penetration, buried underground or encased by a guard pipe, and maintenance constraints as stated by the systems designer, are exempted from volumetric examination, with the approval of the regulatory body. However, their healthiness should be ensured during system integrity checks.

5 SCHEDULING OF PSI AND ISI

The guidelines for scheduling of PSI and frequency of ISI are given in this chapter for PHWRs and BWRs. The operation of PHWRs is different from that of BWRs. PHWRs being amenable for on-line refueling have some flexibility in scheduling and frequency of ISI. The concept of biennial shutdowns (BSDs) followed in the case of PHWRs makes it feasible to evenly space ISI intervals. BWRs rely on refueling in batches and hence their ISI schedule and frequency generally follow the refueling outages to the extent possible.

Like the procedure followed in the previous chapter, the clauses specific to PHWRs begin with number 5A followed by numerals and clauses specific to BWRs begin with number 5B followed by numerals.

5A: SCHEDULING OF PSI AND ISI FOR PHWRs

5A.1 Pre-Service Inspection (PSI)

5A.1.1 A PSI should be performed on all identified SSCs specified in the sub-sections 4A.1.1 and 4A.1.2 after hydrostatic pressure test of the system, equipment or component, before the initial plant start-up to provide the baseline data.

5A.1.2 The PSI should be carried out using the same NDT methods, techniques and types of equipment as those intended for the ISI.

5A.1.3 In case a component or a portion thereof of the fluid boundary or a system is repaired, or replaced, inspection should be performed on the replaced component or the repaired portion of the fluid boundary, as per the applicable codes for construction, to gather the baseline data. This inspection will be considered as the PSI for the replaced/repaired component for the subsequent inspections.

5A.1.4 The inspections performed prior to hydrostatic or any other pressure test on component could qualify as PSI, provided a confirmatory inspection is carried out as per the conditions given in the sub-section 4A.5.

5A.1.5 The shop and field inspections performed during construction/manufacturing may form a part of the PSI, where inspection after final installation and testing is not practical, provided that:

(a) these inspections have been conducted under similar conditions and with equipment and techniques equivalent to those that are planned to be employed during the ISI,
(b) the inspections conducted before hydrostatic (or pneumatic) pressure tests have been followed by a confirmatory inspection after the hydrostatic test on a sample of inspection areas to demonstrate that no significant change has occurred,
(c) the inspections have been performed on components classified as pressure vessels after their hydrostatic (or pneumatic) test,
(d) the inspection records have been documented and identified in a form consistent with the requirements given in the Section-11 of this guide,
(e) the inspection results with proper technical justifications have been submitted to the regulatory body for their acceptance.

5A.1.6 The PSI should be extended to all safety related critical components that are subjected to ISI. Where this includes a sample of welds, the full length and width of the welded zone, pressure retaining bolting and the specified portion of the adjoining base material should be examined.

5A.2 In-Service Inspection (ISI)

5A.2.1 The inspection schedule should provide for repetitions of the ISI programme during the operating life of the unit. The inspection schedule should be evenly distributed into intervals during the operating life of the unit based on the observed degradation mechanism and the rate of degradation. The inspection interval should be conservatively designed to detect any deterioration before it could lead to failures and should be reviewed based on the inspection findings.

5A.2.2 The first inspection interval should be scheduled at 5 years and the subsequent inspection intervals at 10 years.

5A.2.3 Inspections on all the SSCs selected as per the criteria given in the Section 4A should be completed in an inspection interval.

5A.2.4 The inspection interval should be subdivided into inspection periods during which a required percentage of examinations should be completed, depending upon the component, the type of examination, or the accessibility allowed by the normal plant operations or by the scheduled outages, like BSDs.

5A.2.5 The quantum of inspection to be conducted in an inspection period should be judiciously spread over the entire inspection interval. All the inspections should be scheduled to be completed during the first inspection interval of 5 years, preferably in 2 or 3 BSDs. For the subsequent 10 year inspection intervals, the entire quantum should be distributed amongst the 5 BSDs.

5A.2.6 The sequence of examinations of components followed during the inspection intervals should be repeated so as to maintain a constant time interval between successive inspections of the component, to the extent practical.

5A.2.7 An extension of inspection interval is permitted to coincide with the upcoming scheduled outages (BSD) of the unit but should not extend beyond one year. The inspections carried out in the extended period, should not be accounted for the subsequent inspection interval and all the inspections of the subsequent inspection interval should be completed within the stipulated time.

5A.2.8 The examinations, which require disassembly of components such as pumps or valves (for the examination of their fasteners or their pressure boundary) might be deferred until the end of the inspection interval or could be scheduled to coincide with dismantling of the components for maintenance. Removal of covers or insulation for gaining access to inspection areas should not be construed as dismantling of equipment.

5A.2.9 The inspections necessitated based on results of examination(s) conducted on analogous components should not be deferred.

5A.2.10 If a system or component is subjected to conditions that differ significantly from those contemplated in the design specifications, or if the operating conditions results in a degradation mechanism not envisaged during the design, the ISI interval for such system or component should be determined on a special case basis and should be brought to the specific notice of the regulators for their approval.

5A.2.11 Primary Coolant Pumps Internal Surface Inspection Intervals

The inspection quantum and frequency given below applies only to those cases of inspection of primary coolant pumps (PCPs) involving dismantling for inspection of the internal surfaces of casings.

(a) In the case of 220 MWe twin reactor stations:

(i) For the first unit, the first inspection of the internal surface of one primary coolant pump (PCP) should be carried out within 10 years from the beginning of operation of the unit and subsequent inspection of one PCP at an interval of 15 years.

(ii) For the second unit, the first inspection of the internal surface of one PCP should be carried out after 15 years from the beginning of operation of the unit and subsequent inspection of one PCP at an interval of 15 years.

(b) In the case of 540MWe/700MWe twin reactor stations:

(i) A minimum of one primary coolant pump (PCP) should be inspected in one unit during the first inspection interval.

(ii) The subsequent inspections should be carried out at every 10 year intervals.

(c) However, if a PCP is opened for maintenance in the meanwhile, the inspection of the internal surface should be carried out. Such inspections will get credited for the period in which the inspection has been carried out.

5A.2.12 Dormant or Passive Systems:

The inspection interval for the dormant or passive systems, their components or portions thereof, that come under the purview of PSI and ISI should not exceed one-sixth of the operational life of the plant as specified in the design or 5 years, whichever is less.

5B: SCHEDULING OF PSI AND ISI FOR BWRs

5B.1 Pre-Service Inspection (PSI)

5B.1.1 The general philosophy of PSI is to collect the base line data for SSCs for future reference. PSI provides information on the initial conditions and data for comparison for the subsequent examinations.

5B.1.2 The PSI data should be collected prior to commencement of initial operation but after the hydro-test.

5B.1.3 The PSI should use the same NDT methods, techniques and types of equipment as those planned to be used for the ISI.

5B.1.4 In case a component or a portion of the fluid boundary of a component or a system is repaired or replaced, the inspection and testing should be carried out as per the applicable construction code in line with the requirements of ASME B & PV Code, Section-XI for repair/replacement before clearing the component/system for operation. These inspections will be considered as the PSI for the replaced/repaired component for the subsequent inspections.

5B.1.5 The PSI should be extended to include the defined extent of examination of all of the pressure-retaining welds in all Classes 1, 2, 3, MC and CC components, regardless of the sample sizes applicable for the ISI.

5B.1.6 The PSI should be performed on all components which are listed in the ISI programme comprising essentially 100% of all pressure-retaining welds before the initial plant start-up. This should also include the PSI on both base metal and welds including the heat affected zone (HAZ) in the core-belt region of the reactor pressure vessel, using suitable volumetric examination techniques. This examination should include the regions of the base metal which have been repaired by welding to depths higher than 10% of vessel wall thickness.

5B.1.7 All welds in control-rod drive housings should be examined during PSI.

5B.1.8 The PSI of all supports of Classes-1, 2, 3 and MC components should be carried out after the first heating cycle. Examinations of supports for systems which operate at a temperature greater than 95°C during normal plant operation should be performed during or following the initial heat-up and cool-down of the system.

5B.1.9 For Class MC and metallic liners of concrete containment vessels (CC), the PSI should be performed following completion of the containment structural integrity test as per the requirements of applicable construction code. For un-bonded post-tensioned systems, all tests and examinations required in an inspection interval are to be performed before the initial start-up of the plant.

5B.1.10 Shop and field examinations may be considered as the PSI, if,

(a) the examination of vessel is performed after completion of the pressure test (hydro test or structural integrity test, as applicable) as per requirements of the construction code,
(b) the equipment and techniques used in shop and field/manufacturing inspections are equivalent to those planned to be employed during the ISI,
(c) records are documented, maintained and identified in a specific form consistent and in-line with the applicable requirements for the ISI,[4] and

[4] For format requirement of the Article IWA-6340 (Section XI) of the applicable code should be followed.

(d) the decision to consider shop and field inspections as the PSI is submitted to the regulatory body for acceptance, with proper technical justifications.

5B.1.11 The PSI should be extended to all safety related critical components that are subjected to ISI. Where this includes a sample of welds, the full length and width of the weld along with the HAZ, the pressure retaining bolting and the specified portion of the adjoining base material should be examined.

5B.2 In-Service Inspection (ISI)

5B.2.1 The ISI is carried out to ensure that the levels of reliability and availability of all plant SSCs that have a bearing on safety remain in accordance with the assumptions and intent of the design so that the safety of the plant is not adversely affected after the commencement of operation.

5B.2.2 The ISI programme should be established with the coverage and consideration for selection of SSCs as mentioned in sub sections 4B.3 and 4B.4. The inspection interval is further subdivided into inspection periods during which a required number of examinations must be completed, depending upon the component, the type of examination, or the accessibility allowed by the normal plant operations or by scheduled refueling outages. These examinations may be considered as a part of the total inspection required for the respective interval.

5B.2.3 The SSCs subjected to the ISI in accordance with sub sections 4B.3 and 4B.4 should be examined by visual, surface and volumetric methods as a rule. In addition, the integrity of the pressure-retaining components of Classes-1, 2 3 and MC should be checked by a system leakage test (or) the hydro-test, as applicable, in line with the applicable requirements of the code. (ASME B&PV code, Section XI, Division-1).

5B.2.3.1 The reactor coolant system (Class-1) pressure boundary components should be subjected to periodic system leakage test prior to plant start-up following a reactor refueling outage. In addition, the system leakage testes should be conducted for Classes-2 & 3 components in each inspection period. (Ref. Annexure-V.B).

5B.2.3.2 In addition to the above, the Class 1 pressure-retaining boundary which is not pressurised when the system valves are in the position required for normal reactor start-up, they should be pressurised and examined at or near the end of the inspection interval.

5B.2.4 All pressure retaining welds of pressure vessel and steam generators should be examined over the inspection interval. Examination of the parent material of the reactor core belt region by volumetric inspection should also be carried out to the maximum possible extent.

5B.2.5 A minimum percentage of quantum of weld joints as per the code requirements should be inspected for Classes-1, 2 and 3 piping during one ISI period. The weld examinations should extend over the heat-affected zone (HAZ) on either side of the welds.

5B.2.6 If any new inspection inspection/technique is to be introduced, then its effectiveness should be demonstrated prior to deployment at site.

5B.2.7 Some of the components may be exempted from surface and volumetric examinations either because of the size of their connections, or because of the number of barriers between the component and the fuel or the outside environment. In such cases, these components are not exempted from the examinations for evidence of leakage as part of the system leakage tests[5].

[5] For format requirement, Article IWA-6340 (Section XI) of applicable code should be followed.

5B.2.8 The ISI should be extended to include those SSCs whose operating conditions or operating behaviour results in degradation which was not envisaged in the design. The inspection programme for such SSCs should be determined on a special case basis.

5B.3 Inspection Schedule

5B.3.1 The scheduling for inspection of SSCs of Classes 1, 2 and 3 components, supports and Class MC structures should be done as follows:

(a) The inspections should be evenly distributed during the inspection intervals and the inspection schedule should be followed as given in the Table- I. There is a specified maximum percentage of examinations that is accepted as having been completed even if the percentage of examinations actually performed is in excess of this figure. These maxima are shown in the column 4 of the table under the heading 'Maximum percentage of examinations credited.' The inspection interval may be decreased or extended by as much as one year to enable an inspection to coincide with a plant outage during this period.

Table-I Distribution of Quantum of Inspections to be completed in an Inspection Interval
(Ref. 5B.3.1)
(Inspection to be evenly distributed for all classes and categories)

Inspection Interval	**Inspection Period (Calendar years)**	**Minimum Examinations to be Completed (%)**	**Maximum Examinations Credited (%)**
(1)	(2)	(3)	(4)
All	3	16	50
All	7	50 (Note-2)	75
All	10	100	100

Notes:

1. The above Table-I has been prepared considering the inspection interval as 10 years which is further sub-divided into three inspection periods (such as 3-4-3 years).
2. If the percentage for examination exceeds 34% in the first period i.e., 3-years then at least 16% of the required examinations should be performed in the second period i.e., in the next 4 years.
3. The sequence of components for inspection should be chosen in such a way that the successive inspections maintain the same sequence. The period between the two inspections of components should not exceed the inspection interval (i.e., 10 years).
4. All the examinations as per the inspection programme should be completed as per the inspection interval specified in the Table-I. The inspection interval may be decreased or extended by as much as one year to enable an inspection to coincide with a plant outage during this period.

(b) In the case of extension of operation of the reactor beyond the design period, the inspection interval in general may follow the same schedule; however, such condition should be specifically reviewed and approval obtained from the regulatory body.

(c) The RO should prepare plans and schedules for performing the examinations and tests as per requirement of the applicable code before the start of each inspection interval. The examination should be completed in accordance with the inspection plan and schedule indicated in the Table-I and should be approved by the regulatory body. The plans are required to have enough information to ensure that all of the applicable requirements are met. The major requirements include the following:

(i) The classification of the SSCs and their boundaries.
(ii) The identification of the components for examination and testing.

(iii) The code requirements by category and item number for each component and the examination or test to be performed.
(iv) The code requirements by category and item number for each component that is not being satisfied by the examination.
(v) Applicable code cases proposed for use and the extent of their application.

5B.3.2 The inspection plans and schedules should be prepared well in advance for the first and the subsequent ISI intervals which should also incorporate modifications to the code if any, the experience gained during the previous inspection intervals, generic lessons learnt from other reactors, etc. The first inspection interval is 10 years following the initial start of plant commercial service. The successive inspection intervals also will have 10-year duration as per the applicable code.

5B.3.3 The examinations, which require disassembly of components (such as disassembly of pumps or valves for the volumetric examination of fasteners like bolts and ligaments, body casing, etc.,) or the removal of fuel or core support structures in the reactor vessels to examine welds or nozzle radius sections, may be deferred until the end of each inspection interval or can be timed to coincide with dismantling for maintenance or other purposes except where, on the basis of results of examination conducted on analogous components, an earlier inspection becomes necessary. The dismantling of equipment should not be considered to include the removal of insulation or access covers.

5B.3.4 The results of the ISI should be compared with the baseline data obtained from the PSI to help in evaluating the effects of service-induced degradation.

5B.3.5 The SSCs subjected to conditions that differ significantly from those contemplated in the design specification, the periodic inspection interval should be determined on a special case basis.

5B.4 Dormant Systems

The components in systems or portions of systems that are dormant (i.e., systems required to function in a passive way) should also be subjected to ISI. The inspection interval for such systems should be less than or equal to half of the interval corresponding to that of the active systems. Examples of dormant systems are vent shafts (down comers), suppression pool, rupture diaphragms of common chamber, containment filtered venting system, etc.

6. ACCEPTANCE CRITERA, ADDITIONAL AND SUCCESSIVE INSPECTION

6.1 General

The goal of the inspection is to assure detection and sizing of the relevant flaws and their acceptance which do not jeopardize structural integrity of the components. The bases for the specific acceptance criteria based on the component, material of construction and the maximum allowable size of the flaw are stated in the relevant codes and standards. Establishing the acceptance criteria for the inspection results is an essential aspect of an inspection programme. This Section provides the guidance on establishing the acceptance criteria and requirements for the additional inspections, if necessitated by the observed unacceptable indications. The guidance on the disposition of unacceptable indications is given in the Section-7.

6.2 Acceptance Criteria

6.2.1 The acceptance criteria for visual, surface and volumetric examinations should be established for the disposition of flaws/defects as per the applicable codes/guides/standards before the start of PSI and ISI programme. For cases where the acceptance criteria are not available in the codes/guides/standard, the same should be established and submitted to the regulatory body for their review and approval prior to implementation of the inspection programme.

6.2.2 The reporting criteria should be established such that adequate margins exist between them and the acceptance criteria.

6.2.3 The indications that show no detectable change since the previous inspection should be considered acceptable.

6.2.4 For the disposition of material loss and loss rates due to erosion or corrosion, the acceptance criteria should be based on the limits specified by the designer for the components, piping and supports.

6.2.5 The acceptance criteria should be defined in the applicable NDE procedure.

6.3 Additional Examinations

6.3.1 Whenever a flaw/indication exceeding the acceptance criteria is found in a given sample size during the ISI, examinations should be performed on the specific problem area in an additional number of analogous components. The additional number should be approximately equal to the original sample size.

6.3.2 If additional examinations reveal flaws in the analogous components or areas exceeding the acceptance criteria, then all the remaining analogous components or areas should be examined. The extent of examinations should be the same as specified for the component or the item in the initial sample, except as modified by the requirements mentioned in the sub-sections 6.3.3 and 6.3.4.

6.3.3 Where the required piping examination in the sampling programme is limited to one loop or branch run of an essentially symmetric piping configuration and examinations indicate flaws exceeding the acceptance criteria, the additional examinations, mentioned in the sub-section 6.3.1 should include an examination of a second loop or branch run.

6.3.4 In the event that the examinations of the second loop or branch run indicate further flaws exceeding the acceptance criteria, the remaining loops or branch runs that perform similar functions should be examined.

6.3.5 In the case of any unacceptable indication found in a component of a reactor unit the corresponding area in the identically designed components of other reactor unit/units at the site should be inspected during the upcoming planned reactor cold shut down, BSD or refueling outage, as the case may be, whichever is earlier.

6.3.6 In case any unacceptable indication(s) found in a component(s) is generic in nature, analogous components in units of similar design should be inspected. The time period permitted to complete this additional inspection will depend on the significance of unacceptable indication and will be subject to approval by the regulatory body.

6.3.7 Complementary and/or supplementary examination/techniques can be employed to assist in the determination of the nature of the indication (location, size, shape, orientation, etc.). If these techniques are to be employed for successive inspections, the original requirements with respect to the type, extent and frequency of inspection as given in this safety guide should be followed (Sections 5A and 5B). Consideration should be given to applicability of these techniques to the inspection of other units following the requirements of the sub-sections 6.3.1 to 6.3.6.

6.4 Repetitive/Successive Inspection

6.4.1 Where examination of a component results in the evaluation of flaw/indications in accordance with the provisions of the Section-7 and qualifies the component as acceptable for continued operation, that portion of the component containing such flaws should be examined during each of the next three successive inspection periods, as an additional requirement over and above the original examination schedule.

6.4.2 In the event that the re-examinations required by the sub-section 6.4.1 indicate that the flaws remain essentially unchanged for three successive inspection periods, the component examination schedule may revert to the original schedule for successive inspections.

6.4.3 If growth of the flaw(s) is observed during the re-examinations as required by the sub-section 6.4.1, fitness of the component for continued service should be assessed taking into account the projected size of the flaw and the anticipated changes in the material properties till the next scheduled inspection, as per the requirements of the applicable codes and standards. The projected size of the flaw should be based on the conservative estimate of the observed flaw growth rate. The assessment as per the requirements given in the sub-section 7.5 should be submitted to the regulatory body for review and acceptance.

7 EVALUATION AND DISPOSITION

7.1 This section provides guidance on the evaluation and disposition of the inspection findings with respect to the acceptance criteria and submission of the disposition report to the regulatory body.

7.2 The results of the inspection should be analysed to determine their compliance with respect to the acceptance criteria given in the applicable codes and standards.

7.3 The inspection results should be interpreted by the qualified and certified Non-Destructive Examination (NDE) personnel in the respective technique and specific activity, to ensure that:

(a) All the indications have been categorised as relevant or non-relevant.
(b) All the relevant indications have been further investigated for the nature, size, shape, location and orientation. Additional investigations if necessary have been carried out using alternate inspection techniques to assist the characterisation of the indications.
(c) The trend analysis has been carried out.

7.4 If results of the examination do not satisfy the acceptance criteria then the concerned component should be analysed for fitness for service till the next ISI. If the analyses show that continued operation of the component is not possible then the component should be repaired or replaced as per the sub-section 10.2.

7.5 Fracture mechanics based analysis may be employed to assess the fitness for service of the component as per the applicable codes and standards. The procedure for analysis is given in the Appendix-D.

7.6 The proposal for disposition of the inspection findings for retaining the components in service as per the sub-sections 7.4 and 7.5 should include the following:

(a) A report on the evaluation of the component for fitness for service including the trending analysis of the relevant parameters of inspection findings and anticipated service induced changes in the material properties to determine acceptability of components for continued operation till the next scheduled inspection.
(b) The plans for additional and successive inspections as per requirements of the sub-sections 6.3 and 6.4.
(c) The action plan for repair or replacement of the component, if required.
(d) Interim actions, if any, like operation with restrictions, updating acceptance criteria, other condition assessment programmes, etc.
(e) Seeking the approval of the regulatory body for continued operation based on the evaluation report.

8 NON-CONFORMANCE CONTROL

8.1 There could be instances of non-conformance in different phases of activities with respect to materials, components, systems or processes. This Section gives the guidelines for handling such cases of non-conformances.

8.2 Procedures should be established for controlling, recording and reporting of the non-conformance to specified requirements of materials, parts, components, systems or processes.

8.3 These procedures should also provide for clear identification and segregation of the non-conforming items (physical segregation, tagging, etc.,) for preventing their inadvertent use and modification in processes etc.

8.4 The non-conformance control should address the following:

(a) The details of non-conformance and its impact on the system/process.
(b) The reasons for the non-conformance.
(c) The disposition of the observed non-conformance along with technical analysis and justification.
(d) Corrective actions to avoid recurrence.
(e) Documentation.

8.5 A Non-Conformance Report (NCR) for each case of non-conformance should be prepared by the Operating Organisation (OO) and submitted to the RO for review and acceptance. It should include the following:

(a) The statement of non-conformity with relevant drawings and sketches.
(b) The reasons for the non-conformity.
(c) Corrective actions to avoid recurrence.
(d) A proposal for disposition along with justification and evaluation report.

8.6 The RO should review and approve the non-conformance and the disposition. The RO approval for the disposition of non-conforming item(s) should be documented.

8.7 All the non-conformances which are of safety significance along with their disposition should be submitted to the regulatory body for review and acceptance.

9 INDEPENDENT VERIFICATION

9.1 An independent verification of PSI/ISI procedures and results should be carried out either by the RO or by the qualified personnel designated by the RO, belonging to other OOs, prior to the initial start-up of the unit for the PSI and before the start-up of the unit for the ISI.

9.2 The results of PSI and ISI should be confirmed by the concerned OO and reviewed by the RO and then submitted to the regulatory body for acceptance and approval.

10 REPAIRS, REPLACEMENTS AND MODIFICATIONS

10.1 General

If the flaws/indications recorded during the PSI or ISI render a component unacceptable as per the provisions of the Section 6, such a component should be repaired or replaced. Also, the SSCs might undergo modifications depending on changes in certain operational requirements. Whenever SSCs require repair, modification, alteration or addition, the following guidelines should be followed:

10.2 Repairs and Replacements

10.2.1 The repair of the components and their subsequent inspections should be in accordance with the applicable design codes and standards.

10.2.2 The replacement of a component or its part should satisfy the provisions and requirements of the codes, standards and other special instructions followed during the construction of the component or the part to be replaced. Alternatively, the replaced components or the part should satisfy the requirements of later editions of codes and standards or new codes and standards, or portions thereof, provided that the following conditions are satisfied:

(a) The requirements affecting the design, fabrication and examination are reviewed to ensure that the original safety requirements are not diluted by the application of the later version(s) of the code or the new codes.

(b) Fit and tolerances of the mechanical interfaces as per the revised or newer editions of codes or standards or new codes and standards would not adversely affect the performance requirements.

(c) The materials of construction of the replaced components or repaired parts are compatible with the rest of the system, operating environment and process fluids.

10.2.3 Components that are repaired or replaced should be re-examined in accordance with the provisions of the sub-sections-5A.1.3 or 5B.1.4, using the equipment, methods and techniques listed in the Section-3 and the Appendix-C, of this safety guide. If a new technique was adopted for detection of the unacceptable flaws/indications/degradation of the original component, then the re-examination should be carried out using the same examination methods/techniques that were used for detection and characterization of the flaws/indications/degradation. The results of the re-examination should be considered as the baseline data for the subsequent ISI of the replaced component.

10.3 Modifications

10.3.1 The SSCs that are modified or added after the initial plant start-up should be reviewed to check for any change in the inspection category, the extent of examination, potential failure consequences, the hazard to health and safety, etc.

10.3.2 If there is a change in any of the above aspects, the PSI and ISI programme of the modified system/component should be reviewed and amended to conform to the requirements of this safety guide.

11 DOCUMENTATION

11.1 The documentation is an important aspect of the inspection programme. The documents necessary for implementation of the PSI and ISI programmes should be developed and maintained by the OO. The documents should be readily retrievable and should comply with the following:

11.2 The documents should be clearly identifiable by the date, name of the reactor unit/OO and the competent authority issuing the document.

11.3 The documents should be developed and maintained from the design stage onwards. The list of such documents is given in the Appendix –E.

11.4 The relevant information as per the Appendix- E.1(d) to (h), surface condition, etc., should be documented to enable repetition and generation of authentic results during the subsequent examinations.

11.5 All recordable indications and pertinent information concerning the flaws/indications (e.g., nature, location, size, shape, orientation, etc.,) should be documented.

11.6 The comparisons of results of current and previous examinations should be documented to enable trending of results.

11.7 Records, which are directly applicable to an individual component, should be maintained for the life of the component. Other records should be maintained for the life of the reactor unit.

11.8 The detailed procedures for inspection should be clearly identified in the PSI and ISI programmes and the procedures should include the requirements given in the Appendix-E.2.

11.9 The PSI/ISI procedures and results will require independent verification as given in the sub-section 9.1.

11.10 The examination procedures and the PSI and ISI programme documents should be revised and updated incorporating the important lessons learnt, once in 5 years.

12 AUDIT

12.1 General

All activities related to the PSI and ISI need to be audited to ensure adherence to the codes, standards, specifications, established procedures, instructions and other applicable documents, for the review of their adequacy and effectiveness of the programme.

12.2 Audit Personnel (Auditors)

(a) The PSI and ISI programmes should be audited by appropriately trained personnel having the required authority, organizational freedom and are not directly responsible for the area being audited.

(b) The audit personnel for the PSI and ISI programme should have adequate knowledge and experience in the
   - (i) field of PSI and ISI programme,
   - (ii) NDT methods and tests (specific to PSI and ISI),
   - (iii) applicable inspection codes/standards, and
   - (iv) audit procedures.

12.3 Certification of Audit Personnel

(a) The auditors for external and internal audits should be appointed by the RO and the OO respectively. The appointment of auditors should be based on their qualification, knowledge and experience of the PSI and ISI programmes.

(b) The auditors certified by either a national or an international certification agency should be engaged for the audit. If such certified auditors are not available then the auditors can be appointed by conducting an internal oral/written/practical test(s) to assess their competency and should be approved by the RO.

(c) Alternatively, services of any recognised certification agency to conduct such audits can be engaged by the RO. Only the certified/RO approved audit personnel should be appointed for carrying out the audit work.

12.4 The activities of the PSI should be audited every year by an internal audit and twice by an external audit during the PSI stage.

12.5 Activities of the ISI should be audited once in a year by an internal audit and once in 3 years by an external audit.

12.6 The audit should be conducted covering all the aspects as given in the Appendix–F.

12.7 The report of the audit should be reviewed by the RO and corrective actions, if any, should be implemented to maintain the adequacy of the inspection programme.

13 FIRST OF A KIND (FOAK) SYSTEMS

13.1 The First of a Kind Systems (FOAKs), as the name suggests, are those systems being designed and introduced for the first time into the nuclear reactor units of a Power Station. Some of these systems are meant to enhance the safety features and are expected to be available to handle postulated off-normal situations like, extended station black outs, design basis accidents, beyond design basis accidents, etc. As such, most of these systems are passive but remain poised for operation in case of any need. The Station might not have sufficient proven operating experience of such systems. However, it is important and necessary to ensure that such systems are available in healthy condition throughout the operating life of the plant. Information regarding healthiness of such systems and the trending of degradation due to operating environment is mainly derived from the findings of the R&D activities carried out at the design stage. These systems need to be monitored further for ensuring the adequacy of their performance during operation.

13.2 Since the FOAKs present in each station might not be the same as those present elsewhere, their PSI and ISI requirements including the extent, method, frequency of inspections, inspection procedure and acceptance criteria should be documented in the station specific inspection manuals and approved by the regulatory body. The ISI requirements of these systems will change if there is a change of environment or change in the degradation mechanism assessed during their use. In the absence of such changes the prevailing requirements should be applied.

13.3 For systems built with materials that are being used for the first time in reactor environment, the extent of inspection, methods, frequency and acceptance criteria must be enumerated by the designer based on the possible degradation mechanism and put up to the regulatory body for approval.

13.4 For systems designed and built with well-known materials according to the construction codes already in use, with well-known trend of degradation in the operating environment, the inspection will be governed by the requirements of the code, as enumerated in this safety guide.

14 INSPECTION OF STEAM GENERATORS, COOLANT CHANNELS AND PHT FEEDER PIPES

Some of the components might not come under the purview of ISI based on the rules of selection for examination given in the Section-4A. But from the considerations of consequences of failures, material of construction and operating experience, these components should undergo in-service inspections to ensure their continued fitness for service under all conditions of reactor operation. The PSI and ISI requirements of such components are given in this Section.

14AS Steam Generators

14AS.0 General Requirements

(a) This Section provides the requirements for the PSI and ISI of mushroom type steam generators (SGs).

(b) In case of any variation from the requirements given in this Section or the special condition not addressed for other types of steam generators, then the RO should prepare an inspection programme (both PSI and ISI) for such cases considering the surveillance requirements of construction codes and the intent of this guide and get it approved by the regulatory body.

14AS.1 Steam Generator Tubes

(a) The standard examination technique/methods should be used for the detection of anticipated degradations and their trending.

(b) In case it is not possible to employ the standard examination technique/methods, alternative examination methods can be employed provided it can be shown or demonstrated that the alternative inspection technique/method is equivalent to the standard technique for the anticipated degradation mechanism for the concerned component and the same should be approved by the regulatory body.

14AS.1.1 Calibration Standards

(a) The calibration standards and specimens should conform to the material and geometry of the tubes/component being inspected and the anticipated degradation mechanisms.

(b) The calibration standards should contain engineered discontinuities anticipated in the tubes, viz., volumetric and planar flaws, and should enable calibration of the probes for sizing and characterisation of the observed indications.

(c) The calibration standards/calibration mockups should replicate all the features of the internal structure of the SGs that the probe might encounter during inspection, like support plates, grid supports, etc.

14AS.1.2 Pre-Service Inspection

14AS.1.2.1 One of the SGs of the reactor unit, and in case different makes of SGs are being used in the reactor unit, one each of a particular make of SGs should be inspected by carrying out volumetric examination of the full length of all the tubes from the inlet tube sheet rolled joint to the outlet tube sheet rolled joint for detecting flaws and reduction in wall thickness along the length of the tubes.

14AS.1.2.2 For the remaining SGs of the reactor unit, volumetric examination of the full length of the tubes, including tube sheet rolled joints, for a minimum of 25% of the total number of tubes in each of the SGs should be carried out. Tubes should be selected for inspection as per the criteria given in the sub-section 14AS.1.4.

14AS.1.2.3 The PSI of the SGs should be performed preferably after hydrostatic pressure test of the PHT system (including the SGs) prior to start-up of the reactor.

14AS.1.2.4 The inspections of the SGs carried out prior to hydrostatic pressure test of PHT system can be considered as the PSI provided a confirmatory inspection of a minimum of 10% of tubes of each SG is carried out after the PHT system hydrostatic pressure test (including the SGs) is performed and the results indicate no significant changes.

14AS.1.2.5 The confirmatory inspection, should employ the method(s) proposed to be used for the ISI and should specifically include all the tubes/regions with significant indications detected during inspections at the manufacturing stage and inspections carried out at site, if any.

14AS.1.3 In-Service Inspection

14AS1.3.1 Extent of Examination

(a) The volumetric examination of the full length of the tubes from the inlet tube sheet rolled joint to the outlet tube sheet rolled joint should be carried out for detection of flaws and reduction in of wall thickness.
(b) A minimum of 25% of the total number of tubes in each SG should be subjected to the ISI. The sample should include the tubes inspected during the PSI.
(c) The tubes for inspection should be selected in accordance with the sub-section 14AS.1.4
(d) In addition to the above, the tubes with recordable indications greater than 20% of nominal wall thickness (NWT) observed during the PSI/previous ISI should be inspected.
(e) In case any generic degradation is observed in any of the SGs during inspection, the same should be notified to the regulatory body. The quantum of inspection and interval in such cases should be reviewed by the designers and approved by the regulatory body.

14AS.1.4 Selection of Tubes for the PSI and ISI

14AS.1.4.1 The total number of tubes to be selected for the PSI and ISI should be from two categories, viz., specific sample and random sample, as defined in the sub-section 14AS.1.4.2

14AS.1.4.2 All tubes with the allowable defects as per manufacturing inspection records and those having higher risk of in-service degradation based on the design conditions and the operating experience of similar SGs should be categorized under specific sample. The rest of the tubes will form the random sample.

14AS.1.4.3 The specific sample should be inspected during the PSI and subsequently in each ISI.

14AS.1.4.4 The number of tubes in the random sample should be at least 50% of the required total, chosen to cover all the areas of SG in a progressive manner over the inspection intervals. The selection can be spread over different SGs such that the representative tubes from all regions of the SGs get inspected in their design life.

14AS.1.4.5 In the case of a plant with multiple SGs of different makes, tubes for inspection from each make of SG should be selected in such a manner that, when taken together, the representative tubes from all regions of the SGs of a particular make get inspected at least once during their design life.

14AS.1.5 Metallurgical Evaluation

14AS.1.5. Case A: Where lead reactors are available[6]

(i) One of the tubes should be removed from the vulnerable region of the SG and/or the tube-to-tube-sheet joint for metallurgical examination to assess the effect of deposits on the tubes or/and the tube-to-tube sheet joint[7]. The removal of the tube should be done from one of the SGs belonging to the lead reactor units during each inspection interval commencing from the third ISI interval onwards.

(ii) If any unanticipated degradation is detected during the inspection of SG tubes, a portion of the tube containing the defect should be removed for metallurgical evaluation to assess the condition of the tube. The regulatory body should be notified regarding such an evaluation.

(iii) Non removal of the tubes as per the above requirements should be justified with reasons and measures taken to prevent such degradation. The justification should be submitted to the regulatory body for review and acceptance.

14AS.1.5.2 Case B: Where lead reactors are not available[8]

(i) One of the tubes from the vulnerable region and/or area of the tube sheet from one of the SGs of the reactor unit should be removed for metallurgical examination to check for the degradation due to deposits on the tube/tube sheet during each inspection interval commencing from the third ISI interval onwards (Refer foot note 7).

(ii) If any unanticipated degradation is detected during the inspection of SG tubes, a portion of the tube containing the defect should be removed for metallurgical evaluation to assess the condition of the tube. The regulatory body should be notified regarding such an evaluation.

14AS.2 Steam Generator Internals Inspection

14AS.2.1 The tubes and tube support structure of the accessible regions in the secondary side of the SG should be inspected using validated visual inspection aids once in every ISI interval. This inspection should cover such regions of the tube support whose degradation/failure/dislodgement might jeopardize the integrity of the tube bundle.

14AS.2.2 If the inspection as per 14AS.2.1 is not possible, then all tubes in one of the SGs of a particular design or make should be examined by suitable methods to check the integrity of the baffles/supports to the extent possible. Alternatively, a suitable justification should be provided supporting healthiness of the tube supports.

14AS.2.3 In addition to inspection as per the sub-section 14AS.2.1, visual examination of secondary side of the tube-sheet of all SGs should be carried out once in an ISI interval to check for the presence of any deposits on the tube sheet.

14AS.2.4 Samples of the deposit from the secondary side of tube-sheet should be analysed for their chemical composition during every ISI interval for all SGs.

14AS.3 Steam Generator Secondary Side Inspection

14AS.3.1 Pre-Service Inspection

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[6] e.g., All 220MWe units except the cases of RAPS-2, MAPS 1 &2

[7] (i) Controlled conditions should be maintained so as to ensure that metallurgical degradation of the tubes does not take place by the accumulated deposits on the tube sheet surface.
(ii) Frequent removal of deposits on the tube sheet and tube support plate regions by using methods like sludge lancing should be carried out as part of the ISI.

[8] e.g., 540MWe and 700MWe units.

14AS.3.1.1 All weld joints on the primary and secondary sides including the nozzle to vessel weld joints of all the SGs should be inspected for volumetric and surface discontinuities.

14AS.3.1.2 If the PSI is carried out before the hydro test, then a confirmatory inspection should be carried out as per requirements given in the sub-section 4A.5.

14AS.3.2 In-Service Inspection

14AS.3.2.1 All the weld joints on the primary and secondary sides including the nozzle to vessel weld joints of one SG from each loop should be inspected for volumetric and surface discontinuities in every ISI cycle, thus covering all the 4 SGs in a period of two consecutive ISI cycles for each reactor unit. If any indication is observed in any of the SGs during inspection then all the 4 SGs of the unit should be examined.

14AS.3.2.2 The extent of examination during the inspection of welds should be as per the sub-sections 4A.4 in general and 4A.4.3 (b) in particular.

14AS.3.2.3 In addition to the requirements in the sub-sections 14AS.3.2.1 and 14AS.3.2.2, indications recorded during the PSI/earlier ISI/current ISI, should be revisited for trending in the subsequent inspections.

14AS.3.2.4 If any indication exceeding the acceptance criteria is observed during the inspection in any of the SGs, the relevant areas of all the 4 SGs belonging to the unit should be examined.

14AS.3.2.5 The additional inspections according to the sub-section 6.3 should be carried out if a flaw/indication of generic nature is found during the ISI.

14AS.3.2.6 For the flaws or indications that have been disposed of as per provisions of the Section-7 for continued operation, the portion of the shell containing such flaws should be inspected according to the sub-section 6.4.

14AS.4 Mechanical Couplings (Fasteners/ Boltings)

All fasteners, ligaments between threaded stud holes and all the other pressure retaining components should be inspected during the PSI and ISI.

14AS.5 Inspection Intervals

14AS.5.1 The first ISI interval should be of 5 years and the subsequent ISI intervals should be of 10 years as per the sub-section-5A.2.2 of this safety guide.

14AS.5.2 Altered Intervals:

If the SGs are subjected to conditions that differ significantly from those contemplated in the design specifications, the sample size for the tubes for the ISI and inspection intervals for the shell and tubes should be reviewed and, if found necessary, the ISI programme should be revised with the approval of the regulatory body.

14AS.6 Acceptance Criteria

14AS.6.1 The acceptance criteria for the tubes and the criteria for plugging of the tubes having unacceptable indications should be as per the requirements of the surveillance criteria specified in the design specifications.

14AS.6.2 The acceptance criteria for indications observed in the primary and secondary side weld joints should be as per the requirements of the construction code followed.

14AS.6.3 The acceptance criteria for the indications observed in the SG tubes should be included in the station specific inspection PSI/ISI programme document.

14AS.6.4 If the proposed acceptance criteria are not in conformity with the design specifications, they should be justified, reviewed and put up to the regulatory body for approval.

14AS.7 Evaluation and Disposition of Findings

14AS.7.1 The observed indications should be evaluated and disposed of as per the guidance given in the Section-7.

14AS.7.2 If the tubes with unacceptable indications are required to be continued in service for some reason, then a proposal for their disposition, in line with the guidance given in the Section-7, should be submitted by the RO to the regulatory body for approval.

14AC Coolant Channels

14AC.1 Pre-Service Inspection (PSI)

14AC.1.1 The pre-service inspection data is intended to form the base line for assessing the generic degradation of pressure tubes. Pressure tubes should be subjected to PSI after hydro-test/hot conditioning and prior to criticality in the case of new reactors and before hydro-test/hot conditioning in the case of reactors with en-masse coolant channel replacement (EMCCR). The information obtained during the PSI should be compiled along with the fabrication and installation details.[9]

14AC.1.2 The manufacturing history dockets should be prepared for the individual coolant channels comprising of manufacturer's inspection records for each pressure tube on volumetric and dimensional inspection, chemical analysis, corrosion tests, mechanical properties and design concession reports/non-conformance reports (DCRs/NCRs), if any. Further, offcuts of sufficient lengths (~ 300 mm) of representative pressure tubes manufactured from each ingot (3 samples from each ingot, representing tubes manufactured from the top, middle and bottom of the ingot) should be made available to generate initial material properties data. The licensee should verify that the above records are complete and ensure the preservation of catalogued off-cuts till the end-of-life of the coolant channels.

14AC.1.3 A minimum of 18 coolant channels should be subjected to PSI. A minimum of one representative channel from low flux zone, 2 channels from medium flux zone and 12 channels from high flux zone should be selected. The channel selection for the PSI and subsequent ISI can be based on the criteria given in the Annexure I.A. The following parameters should be measured and recorded for all the 18 coolant channels during PSI.

(a) *Internal Diameter:*
The internal diameter should be measured along the full length of the pressure tubes (at 6-12 o' clock & 3-9 o' clock position) with axial spacing separated by not more than 250 mm or one-half of the fuel bundle length, whichever is less.

(b) *Wall Thickness:*
The wall thickness should be measured along the full length of the pressure tubes at not less than 3 equally spaced circumferential positions with axial spacing separated by not more than 250mm or one-half of the fuel bundle length, whichever is less.

[9] In the case of new reactors, PHT hot conditioning & hydro test are carried out with light water, followed by fuel loading. In the case of reactors with EMCCR, PHT hydro test and hot conditioning are carried out with heavy water, with the fuel loaded in the core.

(c) *Volumetric Examination*:
The volumetric examination should be carried out for the full volume of the pressure tubes including the areas adjacent to the rolled joints (*Inspection region:* burnish mark of North rolled joint to burnish mark of South rolled joint).

(d) *Deflection Profile/Sag Measurement:*
The deflection profile of the pressure tubes should be measured along their full length at axial spacing separated by not more than 250 mm or one-half of the fuel bundle length, whichever is less.

(e) *Garter Spring (GS) Location:*
  (i) All GSs should be detected and their locations should be recorded for all coolant channels during installation.
  (ii) For the new reactors, all GSs should be detected and their locations should be recorded for all coolant channels after hot conditioning of the PHT system.

(f) *Channel Length:*
The position of the E-faces of all coolant channels, end fittings with reference to a datum line after fuel loading should be measured and recorded at both ends. Based on this measurement, the channel position on bearings needs to be worked out and recorded.

14AC.1.4 In addition to the above, following PSI should be carried out for the rolled joint region[10]. The extent and quantum of examination are given below:

(a) The burnish mark positions from the 'E-face' should be measured & recorded for all pressure tubes during installation.
(b) The rolled joint regions should be visually inspected from ID after rolling and recorded for all pressure tubes after installation.
(c) Twenty four rolled joint regions should be inspected from ID by UT for volumetric flaws. In the case of a reactor with EMCCR, clearance type rolled joints may be selected for these inspections. Representative rolled joints should be selected for the inspection in case rolled joints are made using more than one rolling tool.

14AC.1.5 The data obtained from post-installation inspections/dimensional checks including garter spring (GS) locations and the minimum front gap measurements (kept for thermal expansion and axial creep) should be recorded for all channels. With respect to recording, reporting, evaluation of inspection results, the requirements of the sub-section 14AC.2 should be met.

14AC.2 In-Service Inspection (ISI)

A periodic ISI should be carried out to assess health of the pressure tubes and trend their degradation during their expected service life.

14AC.2.1 Sample Size and Criteria for Selection of Coolant Channels
(a) A minimum of 15 channels should be inspected in the first inspection interval. At least one representative channel from low flux zone, 2 channels from medium flux zone and 12 channels from high flux zone should be selected for inspection. Selected channels should be a subset of channels which had undergone PSI.
(b) At least 10 out of these 15 channels should be selected for subsequent ISI in such a way that all the 15 channels inspected during the first ISI interval get inspected at

[10] *Rolled joint region*:
(a) The region from the end of a pressure tube to 300mm inboard of the burnish mark of the pressure tube
(b) In the case of reactors with EMCCR, the rolled joints of pressure tubes at one end are of zero clearance type and at the other end are of clearance type
(c) In the case of new reactors, the rolled joints at both ends of pressure tubes are of zero clearance type.

least thrice during the entire coolant channel life with some overlap, enabling trending of the parameters essential for assessing the fitness for service/life management of the coolant channels.

(c) In addition, the channels having station specific problems, which may affect their service life, should be included for the ISI.

(d) If a degradation mechanism is identified in any other unit and can be correlated to a particular channel, that channel should also be included for the ISI.

14AC.2.2 Inspection Intervals

The periodic inspections should be carried out as per the schedule given in the table below:-

Table-II Periodic Inspection Schedule

Inspection Interval	Interval Duration (FPYs)	Allowable Inspection Window (FPYs)
First	0-4	2-4
Second	5-12	10-12
Third	13-18	16-18
Fourth	19-24	22-24
Fifth	25-30	28-30

[The first ISI interval runs from the date of criticality of the reactor to the last day of the 4th FPY. Similarly, the second inspection interval runs from the 1st day of the 5th FPY to last day of the 12th FPY.]

14AC.2.3 Extent of Examination

The following inspections on the representative pressure tubes should be carried out during the periodic inspections:

(a) *Volumetric Examination*

The volumetric examination should be carried out for the full volume of the pressure tubes including the areas adjacent to the rolled joints (*Inspection region*: burnish mark of North rolled joint to burnish mark of South rolled joint).

(b) *Determination of the pressure tube to calandria tube (PT-CT) gap* either by:

(i) Determination of garter spring location and analysis to determine the remaining gap, or

(ii) The pressure tube to calandria tube gap measurement.

The locations of gap measurements should be axially separated by not more than 250mm or one-half of the fuel bundle length, whichever is less.

(c) *Internal Diameter*

The internal diameter should be measured along the full length of the pressure tubes (at 6-12 o' clock & 3-9 o' clock position) with axial spacing separated by not more than 250mm or one-half of the fuel bundle length, whichever is less.

(d) *Wall Thickness*

The wall thickness should be measured along the full length of the pressure tubes at not less than 3 equally spaced circumferential positions with axial spacing separated by not more than 250mm or one-half of the fuel bundle length, whichever is less.

(e) *Sag/Deflection Measurement*

The sag/deflection of the pressure tube should be measured with axial spacing separated by not more than 250mm or one-half of the fuel bundle length, whichever is less.

*[**Note:** In the case of 540/700MWe PHWRs, special attention should be given for the sag measurement in the channels near reactivity devices such as poison injection units (PIUs) and/or horizontal flux units (HFUs)].*

(f) *Axial Creep Measurement*
The elongation of each pressure tube due to axial creep should be measured during each BSD to ensure that till the next BSD:
(i) coolant channels will remain positioned on their bearings,
(ii) sufficient gap is available between the hardware to avoid their interference due to differential creep between adjacent channel pairs,
(iii) sufficient creep gap is available for free expansion of channels, and
(iv) channels are accessible to the fueling machine.

14AC.2.4 Calibration Specimens for Inspection

(a) *Volumetric Inspection by UT*
(i) A reference specimen should be made from a pressure tube of the same material, fabrication process, nominal diameter and wall thickness as that of the pressure tube to be inspected and should have following notches as given in the table below:

Table-III: Location and Direction of Notches in the Reference Pressure Tube

Notch: Location and Direction	Length (mm)	Width (mm)	Depth (% of PT Wall Thickness)
Internal surface, longitudinal	6.0 ± 0.5	0.15 ± 0.05	2
Internal surface, circumferential	6.0 ± 0.5	0.15 ± 0.05	2
External surface, longitudinal	6.0 ± 0.5	0.15 ± 0.05	2
External surface, circumferential	6.0 ± 0.5	0.15 ± 0.05	2

(ii) Nodular Corrosion Inspection by UT
The reference specimens containing nodules of depth in the range of 50±10µ, 100±10µ and 200±10µ should be used for calibration of the ultrasonic transducers used for detection of nodules.

(b) *Inspection by ECT*
The same calibration standard as used for volumetric inspection by UT should be used for calibration of the ECT system.

(c) *Dimensional Inspection*
(i) An appropriate calibration specimen for the dimensional measurement should be available. Whenever the measurement is sensitive to the material, calibration specimen should be made from the same material as that of the coolant channels installed in the reactor.
(ii) For measurement of sag/deflection profile, internal diameter and wall thickness, the calibration specimens should be such that they cover the entire range of the expected values of these parameters.
(iii) Measurement method/technique used for axial creep measurement should be

calibrated for the full range of the expected values.

14AC.2.5 Mockup Facility for Volumetric/Dimensional Inspection

(a) The RO should develop appropriate mockup facilities to verify the performance and accuracy of the equipment and techniques used to perform the required inspections/measurements on the coolant channels, viz., internal diameter, wall thickness, sag/deflection profile, PT-CT gap, volumetric inspection for detection of flaw/nodular corrosion, axial creep, sliver sampling, etc. The mockup facility should be used for rehearsal of operations before actual deployment of the equipment in the reactor. Actual procedures and equipment like fuelling machines and control system, coolant channel inspection system drives, etc., should be used as a part of the mockup facility for rehearsal and also to check the overall accuracy.

(b) Essentially the mockup facility should have a full length coolant channel assembly of the same material as that installed in the reactor with the simulated flow and pressure condition. The mockup facility should have flexibility to cover the entire range of values expected during various inspections/ measurements.

(c) Alternative techniques like ECT, replica method, etc., should be available to confirm the nature of indications. The mockup facility should also cater to the calibration and qualification of these alternative techniques before their deployment in the reactor.

14AC.2.6 Recording of Inspection Data

All raw data/signals should be recorded during the volumetric and dimensional inspections of the coolant channels and should be made available (preferably in the digital form) for evaluation and inter-comparison of inspection results at a later stage, if required.

14AC.2.7 Reporting of Inspection Results

(a) *Volumetric Inspection*
The following criteria should be followed while reporting the volumetric inspection results:
   (i) Any indication having a signal response greater than or equal to 50% of the corresponding indication from the calibration notch in the reference specimen as defined in the sub-section 14AC.2.4.(a) on volumetric inspection reference specimens, should be reported.
   (ii) Any cluster of indications observed on the pressure tube, should be reported irrespective of the signal amplitude.
   (iii) Any loss of back wall echo should be reported.
   (iv) For the ultrasonic indications meeting any one or more of the above criteria, the following information should be reported:
      (p) Signal characteristics.
      (q) Axial and circumferential locations of the indication.
      (r) Signal amplitude with respect to the calibration notch in reference specimen.
   (v) When alternative inspection methods (i.e., other than ultrasonic inspection) are to be used, the method and recording criteria should be submitted to the regulatory body for approval prior to their use for inspection

(b) *Dimensional Inspection*
A review report on the findings of the dimensional inspection should be submitted by the RO to the regulatory body containing the following:
   (i) The maximum/minimum values of internal diameter, the axial location of maximum/minimum internal diameter, the maximum change in the internal diameter since the previous inspection and the rate of change in diameter.

(ii) The minimum wall thickness, the axial location of minimum wall thickness, the maximum change in the wall thickness since the previous inspection and the rate of change in wall thickness.
(iii) The axial locations of the garter springs and their displacement, if any, with respect to the corresponding PSI locations.
(iv) The minimum PT-CT gap, the location of the minimum gap, the maximum change in the gap since the previous inspection and the rate of change in gap.
(v) The most extreme axial bearing position and the predicted rate of change.
(vi) The maximum sag/deflection of the pressure tube, the location of the maximum sag/deflection, the maximum change in sag/deflection since the previous inspection, the rate of change of sag/deflection and curvature profile.
(vii) The measured or computed gap between the sagged coolant channel and the nearest calandria internal component, the rate of closure of the gap and the estimated time for the gap closure.

14AC.2.8 Evaluation of Inspection Results

The RO should be responsible for evaluation of the results at each stage of the ISI programme. Wherever necessary, validated computer codes and programmes designed for the specific inspection activity should be used. The results from each inspection should be evaluated for acceptance. For the evaluation of volumetric examination results, the following should apply:
(a) The reportable indications whose amplitude exceed the reference level should be evaluated for their dimensions and nature i.e., volumetric/planar.
(b) Alternative NDE technique should be employed for characterisation of the indication.

14AC.2.9 Acceptance Criteria

(a) *General*
(i) Relevant indications satisfying the conditions mentioned in the sub-section 14AC.2.9.(b) and meeting the dimensional parameters given in the sub-section 14AC.2.9.(c) should be considered acceptable.
(ii) When a previously accepted indication in a pressure tube has not shown any growth during the subsequent inspection, the fitness for service assessment of the pressure tube should be revised considering the anticipated changes in the parameters, such as material properties, H_{eq} and dimensions of the pressure tube, used in the previous assessment.
(iii) Relevant indications or the dimensional conditions that do not comply with the criteria given this sub-section should be considered acceptable, provided that the fitness for service of the pressure tubes is demonstrated to the regulatory body as per the requirement given in the sub-section 14AC.2.9.(d).

(b) *Volumetric Inspection*
(i) Any indication ≤ reference level is acceptable.
(ii) Any indications that are not nodule-like are acceptable.

(c) *Dimensional Inspection*
The pressure tubes should be acceptable provided that the following conditions are predicted to exist up to the next inspection:
(i) The internal diameter is within the maximum specified in design.
(ii) The wall thickness is not less than the minimum specified in design.
(iii) No contact between PT and CT.
(iv) No contact between the sagged coolant channel and the calandria internals such as PIUs/HFUs.

(v) The coolant channel remains on its bearings and sufficient gap is available in the shock absorber stud assembly to accommodate the axial creep elongation till the next BSD.

(vi) Sufficient feeder-to-feeder and/or feeder-to-hardware gap are available between the adjacent coolant channels to permit their free expansion till the next BSD/PSD.

(vii) The differential axial creep elongation between the adjacent coolant channels remains within the limits to allow their refuelling.

(d) *Fitness for Service*

The relevant indications or the dimensional conditions that do not comply with the acceptance criteria should be acceptable; provided it is demonstrated to the regulatory body that,

(i) the integrity of the pressure tubes is still adequate; and

(ii) the predicted deterioration will not seriously reduce the integrity of the pressure tubes before the next scheduled inspection.

14AC.2.10 Additional inspection

(a) Where the findings of the ISI do not meet the acceptance criteria given in the sub-section 14AC.2.9, additional coolant channels should be inspected in the same reactor. The extent of additional inspections should be determined as part of the disposition as referred to in the sub-section 14AC.2.11.

(b) The need to extend inspection to the coolant channels in other similar reactors units in light of the unacceptable findings in one of the reactor units, should be taken into consideration by the RO.

(c) When there is a reason to suspect a coolant channel during the course of inspection, all the coolant channels which are likely to come under the same suspicion should also be included for such inspection after due evaluation of the coolant channel which was found suspect in the first instance.

14AC.2.11 Disposition of the Inspection Results

When the result of an inspection does not comply with the acceptance criteria mentioned in the sub-section 14AC.2.9, evaluation and further actions should include:

(a) Notifying the regulatory body.

(b) Further inspection by alternative non-destructive method (s), wherever necessary, to assist in the determination of the characteristics (nature, size, shape, location and orientation) of the indication or dimensional condition.

(c) The appraisal of the inspection results, as specified by the sub-section 14AC.2.9 to determine their disposition, i.e., acceptance, repair or replacement.

(d) The submission of the proposed disposition to the regulatory body for acceptance. The disposition should also include consideration of changes to the extent and frequency of inspection. Consideration should be given to the inspection of other units in the same reactor group.

(e) Obtaining acceptance of the disposition from the regulatory body prior to returning the reactor to operation.

14AC.2.12 Successive Inspection

(a) The region of the pressure tube with an indication exceeding the reference level should be inspected during the next three BSDs to assess the growth of the indication. If no growth of the indication is observed, the inspection schedule can be reverted to the normal ISI schedule.

(b) The reportable indications should be revisited in the next inspection campaign.

14AC.3 Material Surveillance

The licensee should develop a material surveillance programme to study specific characteristics of coolant channels over their operating life.

The material surveillance programme should essentially include:

(a) In-situ determination of H_{eq} in irradiated pressure tubes by sliver sampling technique or by any other established non-destructive technique.
(b) The determination of material properties in irradiated condition of the representative pressure tubes and garter spring(s) by removing them from the reactor core.
(c) The guidelines for selection of coolant channels for material surveillance is given in the Annexure-II.A.

14AC.3.1 In-Situ Sliver Sampling

(a) *Sample Size*
In each reactor unit, sliver sampling should be carried out in 6 pressure tubes in each campaign.

(b) *Locations of Sliver Samples*
From each pressure tube, 6 sliver samples should be taken from the main body. The selection of the locations for measurement of H_{eq} along the length of the pressure tube should be based on the guidelines given in the Annexure-III.A.

(c) *Sampling Frequency*
The sliver sampling should be carried out as per the following frequency:

Table-IV Slivering Sample Frequency

Slivering Interval	Interval Duration (FPYs)	Allowable Slivering Window (FPYs)	
		220 MWe PHWRs	540/700 Mwe PHWRs
First	0-4	NA	2-4
Second	5-12	10-12	10-12
Third	13-18	16-18	16-18
Fourth	19-24	22-24	22-24
Fifth	25-30	28-30	28-30

(d) *Selection of Pressure Tubes for Sliver Sampling*
The pressure tubes for the sliver sampling should be selected based on the guidelines given in the Annexure-I.A.

(e) *Fitness for Service of the Slivered Pressure Tubes*
The fitness of the slivered pressure tubes for continued service should be established by carrying out their volumetric examination at the slivered locations.

14AC.3.2 Removal of Pressure Tubes and Garter Springs for Assessment of Material Properties:

14AC.3.2.1 General Requirements

(a) In order to assess the generic degradation of coolant channel materials and for the assessment of material properties in the irradiated condition, after about 12 - 14 effective full power years (EFPYs) of reactor operation, one pressure tube along with its 2 rolled joint regions and 4 garter springs, representing the materials installed in the core and operating conditions defined by neutron flux spectrum, irradiation temperature history and neutron fluence should be removed from each reactor unit and sent for post irradiation examination (PIE).

(b) The subsequent removal of pressure tubes and garter springs for material surveillance, should be after every 5 or 6 years of reactor operation. The cognizance of the available PIE data (subject to the regulatory body approval), the latest ISI findings and sliver sampling results should be taken into account before their removal.

(c) The material properties data generated from the irradiated pressure tube and garter springs removed from one reactor unit can be used for safety assessment of the other reactor units, on a case by case basis, with approval from the regulatory body subject to the following:
   (i) The available irradiated material property data on the representative materials is sufficient for the purpose.
   (ii) The removed pressure tube and garter springs represent in all respects the material and operating conditions of the coolant channels, as given in the sub-section 14AC.3.2.1.(a), of the reactor unit(s) under consideration.
   (iii) The design and construction of the reactor units are identical.
   (iv) The H_{eq} and H/D pickup rates are similar as derived from the sliver sampling results of the units under consideration.

(d) The pressure tubes with any abnormal ISI findings (like blister/nodules, flaws, etc.) and abnormal sliver sampling results, should be considered for removal after a review.

(e) The extent of post-irradiation examination should be as per the sub-sections 14AC.3.2.2 and 14AC.3.2.3.

14AC.3.2.2 Initial Material Surveillance on Un-Irradiated Pressure Tube, Garter Springs and Girdle Wire

To generate the base line data on un-irradiated material properties as a part of material surveillance, the following requirements should be satisfied:

(a) The offcuts of sufficient lengths (~300mm) of the representative pressure tubes manufactured from each ingot (3 samples should be made available from each ingot, representing tubes manufactured from the top, the middle and the bottom of the ingot) and an appropriate number of garter springs should be catalogued and preserved such that the same could be used for generation of the base-line material property data for comparison as and when required.

(b) The RO should establish the following properties/parameters, required for the evaluation of fitness for service of pressure tubes, on the un-irradiated pressure tube material:
   (i) Hydrogen concentration (H_{eq}).
   (ii) Mechanical properties like Ultimate Tensile Strength (UTS), Yield Strength (YS), % elongation (% EL).
   (iii) Fracture toughness (K_{IH}, K_{IC}, etc.,).
   (iv) Delayed Hydride-Cracking Velocity (DHCV).
   (v) Critical Crack Length (CCL).
   (vi) Hydride orientation and distribution.
   (vii) Autoclaved oxide layer thickness on inside and outside surface.
   (viii) Microstructure and texture.

(c) The properties, listed against the sub-section 14AC.3.2.2 (b) (ii) to (v), should be measured at different temperatures ranging from the ambient to the reactor operating temperature. It is permitted to hydride the material to assess DHCV at the relevant temperatures. The CCL can be estimated from the results of the fracture toughness tests, however, the conservatism of such an estimation may have to be established by suitable methods.

(d) A sufficient number of measurements should be carried out to establish the variation in material properties along the full length of the pressure tube.

(e) The RO should establish the following properties/parameters on the un-irradiated garter spring/girdle wire material:
- (i) Average length of spring and girdle wire.
- (ii) Mechanical properties, YS, UTS and % elongation for the girdle wire.
- (iii) Compliance of the spring.
- (iv) Residual elongation of the spring at various incremental loads.
- (v) Crush strength per coil.
- (vi) Hydrogen concentration (H_{eq}).
- (vii) Hydride orientation in the hook region and in the coils.
- (viii) Estimation of stress relaxation, if any.

14AC.3.2.3 Periodic Material Surveillance for Irradiated Pressure Tube, Garter Springs and Girdle Wires

PIE should be carried out to the extent given below to assess the trend of degradation of the material properties and health of coolant channels as a part of material surveillance programme:

(a) The measurement of properties listed in the sub-section 14AC.3.2.2 (b) (except microstructure and texture) on the irradiated pressure tube including the rolled joint stubs[11].

(b) The measurement of properties listed in the sub-section 14AC.3.2.2 (e) on the irradiated garter springs and girdle wire.

(c) A sufficient number of measurements should be carried out to establish the variation in the material properties and H_{eq} along the full length of the irradiated pressure tube.

(d) The properties should be measured at different temperatures ranging from the ambient to pressure tube operating temperature.

(e) The following additional examinations should be carried out along the length of the irradiated pressure tubes and on the rolled joint stubs, as applicable:-
- (i) Visual examination for corrosion patches, evidence for nodular corrosion, crud deposits, garter spring marks, etc.
- (ii) Dimensional measurements for internal diameter and wall thickness.
- (iii) Volumetric examination using UT for flaws.
- (iv) Measurements of sag/bow.
- (v) Gamma scanning to establish the neutron flux profile.
- (vi) Metallography to establish the orientation of hydrides and oxide layer thickness.
- (vii) Extent of H/D pick up.

(f) The following additional examinations should be carried out on the irradiated garter springs and girdle wires:
- (i) Visual examination for identification of corrosion patches, evidence for

[11] The extent of parameters to be generated from the rolled joint stubs may be limited to generation of H_{eq} profile along the stub length, fracture toughness over a range of temperatures and hydride orientation, considering the limited amount of material available.

nodular corrosion, oxidation, hydride blisters, etc.

(ii) Metallographic examination for measurement of oxide layer thickness at 12-o'clock, 3-o'clock, 6-o'clock and 9-o'clock positions of the spring with respect to its mounting on the pressure tube,

(iii) Measurement of equivalent hydrogen isotope concentration (H_{eq}) at these locations.

(iv) Measurement of gamma activity to ascertain the axial location of the springs on the pressure tube.

14AC.3.2.4 Procedures for Evaluation

Material testing procedures as per the applicable standards should be available and followed for evaluation of the following properties/parameters:

(a) Equivalent hydrogen isotope concentration (H_{eq}).
(b) Mechanical properties, viz., YS, UTS and % elongation.
(c) Fracture toughness, K_{IC}.
(d) Delayed Hydride-Cracking Velocity (DHCV).
(e) Threshold stress intensity factor for initiation of delayed hydride crack growth, K_{IH}, in both axial and radial directions.
(f) Oxide layer thickness.
(g) Hydride orientation.
(h) Stress relaxation, residual elongation, compliance, crush strength, residual tension of garter springs and ductility, UTS and YS of girdle wires.
(i) Texture and microstructure.

14AC.3.2.5 Recording of Material Surveillance Results

All the data collected/generated during material surveillance should be recorded to permit comparison with previous and future inspections.

14AC.3.2.6 Reporting of Material Surveillance Results

(a) The RO should submit to the regulatory body a report describing the results of the material surveillance tests performed and the subsequent evaluation(s) to demonstrate compliance with the acceptance criteria.
(b) The results of H_{eq} measurements should be reported to the regulatory body within 120 days after completion of that stage of surveillance (sliver sampling).
(c) The results of material properties testing should be reported to the regulatory body within one year of the commencement of surveillance programme.

14AC.3.2.7 Evaluation of Material Surveillance Test Results and their Disposition

The RO should use the material surveillance test results to demonstrate fitness for continued service of the pressure tube and submit the same to the regulatory body for acceptance. It should be demonstrated that the integrity of the coolant channels and the LBB are assured under all service conditions until the next inspection considering the following points:

(a) The predicted H_{eq} in the rolled joint region is within the acceptable limits.
(b) The predicted H_{eq} in the pressure tube body should be such that hydrides will not be present in the tube during reactor operation and the rate of change per operating year, is within the allowable limits, based on the reactor operating temperature.
(c) The fracture toughness values, K_{IC} and K_{IH} should be greater than the reference lower-bound fracture toughness values at all temperatures and conditions of reactor operation.
(d) The DHCV should be lower than the reference upper- bound value.

(e) The garter springs will maintain their integrity and possess sufficient residual tension to ensure their location and to prevent PT-CT contact till the next surveillance measurements.

14AF PHT FEEDER PIPES

14AF.0 General

The PHT feeder pipes fall under Category C2 based on their size and may not warrant inspections. However, the following inspections should be performed based on their safety significance and operating experience.

14AF.1 Pre-Service Inspection (PSI)

(a) *Visual Examination*

The visual examination of all feeder pipes and their supports should be performed and a record should be made of any observations arising from the inspection relating to the pressure retaining integrity of these components. The visual examination of the supports of the feeder pipes should also be carried out during or following the initial system heat up and cool down.

(b) *Wall Thickness Measurement*

(i) The wall thickness measurements of all the inlet and the outlet feeder elbows/bends including the extrados, the intrados and their weld joints near the high pressure feeder coupling (HPFC) end should be carried out in all reactor units. The selected areas for inspection should be chosen from the susceptible regions likely to experience the maximum reduction in the wall thickness.

(ii) The wall thickness measurements for a minimum of 20 representative first bends from the header ends and the feeder-stub-bend/feeder-bend depending on the layout, which are prone to the maximum reduction in the wall thickness, should be carried out.

(iii) The wall thickness measurements for the representative feeder pipe lengths of at least 400mm downstream of orifices/venturis and the first bends of minimum 10 feeder pipes from the headers which are prone to the maximum reduction in wall thickness (e.g., bends) based on the operating experience should be carried out.

(c) *Inspection for Flaw Detection*

(i) The visual, surface and volumetric examinations of a minimum of 10 representative elbows/bends including the weld joints near the HPFC end should be carried out. These examinations should be carried out taking special efforts to cover the intrados portion of feeders near the HPFC end utilising smaller diameter probes.

(ii) The visual, surface and volumetric examinations of a minimum of 20 representative header-to-feeder stub weld joints of SA 333 Gr.6 material should be carried out.

(iii) The visual, surface and volumetric examinations of all dissimilar metal weld joints should be carried out.

14AF.2 In-Service Inspection (ISI)

The ISI programme should consider the material of construction of the feeder pipes and their vulnerability to the degradation mechanism. The following examinations on the different components of the feeders should be carried out:

(a) *Visual Examination*

(i) The accessible feeder piping and their supports, including the near-by regions of supports of adjacent feeders.

(ii) Any component in close proximity that might cause fretting damage to the adjacent feeder pipes or the feeder elbows, e.g., HPFC clamp hardware.

(b) *Wall Thickness Measurement*

(i) The wall thickness measurements of the feeder pipes, especially at the elbows/bends near the HPFC end. The sample selection for inspection, applicable to each reactor unit, should be based on the sampling criteria given in the sub-section 14AF.2.2.

(ii) The wall thickness measurements of the first bend region (feeder-stub-bend/feeder-bend) from the header end on a minimum of 4 sample feeders.

(iii) The wall thickness measurements of a minimum of 5 representative feeder pipe portions of at least 400 mm length downstream of the orifices/venturis which are prone to the maximum reduction in the wall thickness.

(c) *Inspection for Flaw Detection*

(i) The visual, surface and volumetric examinations of a minimum of 5 representative feeder elbows/bends, including the weld joints, near the HPFC end should be carried out to the extent practicable. Special efforts should be taken to cover the intrados portion of the feeders near the HPFC end utilising smaller diameter probes.

(ii) The visual, surface and volumetric examinations of the representative weld joints between the header-to-feeder stubs, whose selection is based on the criteria as given in the sub-section 14AF.2.3.

(iii) The minimum number of weld joints to be chosen for inspection:

(*p*) 120 for PHWRs having SA 106 Gr. B feeder stub material.

(*q*) 20 for PHWRs having SA 333 Gr. 6 feeder stub material.

(iv) The visual, surface and volumetric examinations of the pipe-welds between dissimilar metals should be performed. A minimum 25% of the total number of dissimilar metal weld joints should be subjected to inspection. In case of detection of any flaw in any of the weld joints, the quantum of the weld joints for inspection should be enhanced in consultation with the designers.

(v) Based on the operating experience, the above quantum for the inspection for flaw detection should be reviewed by the designers and subject to approval by the regulatory body.

14AF.2.1 Indications recorded during the PSI/the earlier ISI/the current ISI, should be revisited for trending in the subsequent ISI.

14AF.2.2 Sampling Criteria

14AF.2.2.1 For all the 220 MWe PHWRs (Except MAPS-2) and 540 MWe PHWRs with SA 333 Gr.6, as the Material of Construction of the Feeders:

(a) A minimum of 20 feeder elbows/pipes, especially at the bends, selected based on the following criteria should be subjected to inspection:

(i) 5 outlet feeder elbows/bends from high velocity considerations.

(ii) 5 outlet feeder elbows/bends from high stress intensity factor considerations.

(iii) 2 feeder elbows/bends based on high survey factor (product of maximum velocity and maximum stress intensity range).

(iv) 4 inlet feeder elbows/bends based on the criteria mentioned in (i) and (ii) above.

(v) 4 feeder elbows/bends from the group that might be susceptible to higher loads during seismic activity based on the analysis.

14AF.2.2.2 For MAPS-2 with SA 106, Gr. B, as the Material of Construction of the Feeders:

(a) A minimum of 120 feeder elbows/pipes, especially at the bends, selected based on the following criteria should be subjected to inspection:

(i) 50 outlet feeder elbows/bends from high velocity considerations.

(ii) 20 outlet feeder elbows/bends from high stress intensity factor considerations.
(iii) 20 feeder elbows/bends based on high survey factor (product of maximum. velocity and maximum stress intensity range).
(iv) 20 outlet feeder elbows based on the feeder wall thinning experience.
(v) 6 inlet feeder elbows/bends based on the criteria mentioned in (i) and (ii) above.
(vi) 4 feeders elbows/bends susceptible to higher loads during seismic activity.

14AF.2.2.3 For 700 MWe PHWRs with SA 333 Gr.6, as the Material of construction of the Feeders:

The feeders in a 700 MWe reactor units operate with 2-3% boiling at coolant channel outlets, by design. The routing of the feeders is different from that of the 540 Mwe reactor units, because of which the length and nature of the path of coolant flow are different. Though the material of construction of the feeders is the same as that used in other reactor units, because of the afore said reasons, the performance of these feeders could be different from that of the units already in operation. Keeping this in view the following quantum of feeders are required to be inspected in the initial stages, which can be reviewed, based on the operating experience, as per the sub-section 14AF.2.2.4.

(a) A minimum of 120 feeder pipes, especially at the elbows/bends, selected based on the following criteria, should be subjected to ISI:
  (i) 50 outlet feeder elbows/bends from high velocity considerations.
  (ii) 20 outlet feeder elbows/bends from high stress intensity factor considerations.
  (iii) 20 feeder elbows/bends based on high survey factor (product of maximum velocity and maximum stress intensity range).
  (iv) 10 outlet feeder elbows/bends susceptible to higher loads during seismic activity.
  (v) 20 inlet feeder elbows/bends (based on the criteria given in (i) to (iv) above). The breakup of the number of inlet feeders is as given below:
    (*p*) 8 inlet feeders from high velocity considerations.
    (*q*) 4 inlet feeders from high stress intensity factor considerations.
    (*r*) 4 inlet feeders based on high survey factor (product of maximum velocity and maximum stress intensity range).
    (*s*) 4 inlet feeders having higher seismic load contributions.

14AF.2.2.4 As a minimum, ISI is to be carried out on the feeders listed in the sub-section 14AF.2.2.3 for the first 2 ISI intervals. Based on the results, the quantum of feeders for inspection, should be reviewed for each 700 MWe PHWR unit by the RO, subject to approval by the regulatory body.

14AF.2.2.5 The feeder pipes/elbows whose remaining wall thicknesses are below the acceptance limits and are retained in-service based on the evaluation of residual life considering the erosion rate estimated as per an earlier inspection, should be inspected in addition to the normal quantum specified for inspection in the above sub-sections. The period and frequency of inspection should be decided based on the outcome of the residual life analysis.

14AF.2.2.6 In case, there is change in the size or material of construction of the feeder pipes or the operational condition of the reactor (like, flow reduction in one channel leading to increased flow in the adjacent channels etc.,), then the quantum of inspection, should be appropriately modified based on design considerations on a case to case basis with the concurrence of the regulatory body.

14AF.2.2.7 The visual, surface and volumetric examinations of header to feeder stub weld joints should be carried out. The selection of feeders and quantum of examination should be based on the following:

(a) Criteria for selection of header to feeder stub weld joints:
- *(p)* Sizes of feeder stubs.
- *(q)* Angular orientations on the headers.
- *(r)* Total header lengths (vertical as well as horizontal feeders).
- *(s)* Accessibility of the welds.

(b) The quantum of examination should be as follows:
- *(p)* A minimum of 100 feeder to header stub weld joints in one ISI interval (applicable to RAPS-2, MAPS and NAPS).
- *(q)* A minimum of 20 feeder to header stub weld joints in one ISI interval (applicable to KAPS onwards).

14AF.3 Inspection Interval

14AF.3.1 The scheduling of PSI and ISI should comply with the requirements given in the Section-5A.

14AF.3.2 The weld joints having any indications and cleared for continued operation, should be inspected for the next 3 consecutive inspection campaigns. If the indication shows no change (values within the error band for the techniques used), then the inspection frequency can be reverted to the original.

14AF.4 Calibration Standard

(a) The calibration standard should be of the same material, grade, nominal diameter and thickness as that of the feeder pipe to be inspected.
(b) The calibration standard should cover the range of thicknesses to be measured due to wall thinning.
(c) The wall thickness measurement method(s) used should have adequate sensitivity to reliably detect and measure all the calibration artifacts in the standard.

14AF.5 Reporting Criteria

14AF.5.1 The following should be reported for each selected area of inspection:
(a) The minimum wall thickness.
(b) The location of the minimum wall thickness.
(c) The maximum rate of change of wall thickness since the last inspection and from the PSI[12].
(d) The region of maximum rate of change of wall thickness.

14AF.5.2 All the indications of rate of reduction of pipe wall thickness higher than that specified in the design manual should be evaluated by the designer and reported to the regulatory body.

14AF.6 Acceptance Criteria

(a) For the visual inspection, the requirements given in the sub-section-6.2.1 should apply.
(b) The indications obtained in volumetric and surface examinations should be evaluated from the point of view of fitness for continued service of the concerned group of feeder pipes.

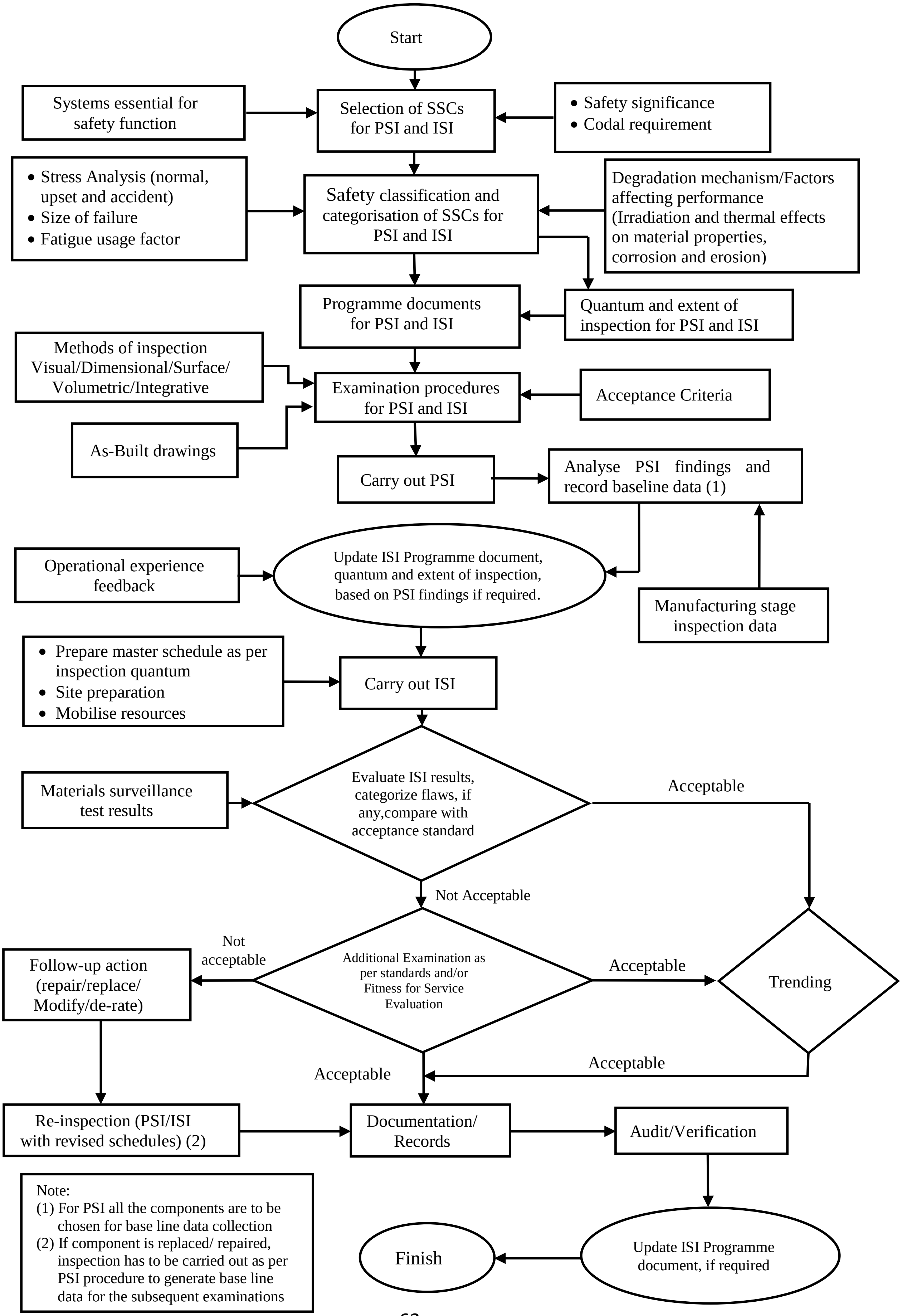
[12] In case the measured value of initial wall thickness of feeder pipes is not available, nominal wall thickness as per the pipe schedule can be used for evaluating the wall thickness reduction and reduction rate.

(c) If the projected wall thickness of the feeder pipes is not less than the minimum allowed design wall thickness at the end of the next inspection interval, then the same feeder pipes are acceptable for use.
(d) If the projected wall thickness by the end of the next inspection interval does not satisfy the design requirements, then fitness for service of the feeder pipes should be assessed till the end of the next inspection interval and submitted to the regulatory body for approval.
(e) The evaluation and disposition of inspection results that do not satisfy the acceptance criteria should be as per the guidelines given in the sub-section 7.6.

APPENDICES AND ANNEXURES

APPENDIX-A

FLOW CHART FOR PSI AND ISI OF NUCLEAR POWER PLANTS



Note:
(1) For PSI all the components are to be chosen for base line data collection
(2) If component is replaced/ repaired, inspection has to be carried out as per PSI procedure to generate base line data for the subsequent examinations

APPENDIX-B

REQUIRED CAPABILITIES OF INSPECTION PERSONNEL

[Refer sub-section 2.7.3]

B.1 Capabilities of Different Levels of Inspection Personnel

(a) *Level I Qualified Person*:

A Level I qualified person should have the capability to perform examinations and tests in accordance with documented procedures. Capabilities with regard to associated inspection procedures, tools, measuring/test equipment, etc., should include the following:

(i) Familiarity with the tools and equipment to be employed.
(ii) To demonstrate proficiency in the use of equipment and tools.
(iii) To assure correct calibration status of measuring/test equipment.
(iv) To assure proper condition for use of measuring/test equipment.
(v) To assure approved status of the procedure for inspection, examination and tests.

(b) *Level II Qualified Person*:

A Level II qualified person should have all the capabilities of a Level I qualified person. He/She should also have the capability to identify test category or class in question. In addition, a Level II qualified person should have capability to:

(i) Plan for inspections, examinations and tests/mock-ups.
(ii) Setup tests including preparation and setup of related equipment as appropriate.
(iii) Supervise or maintain surveillance over inspections, examinations and tests.
(iv) Supervise and certify lower level personnel.
(v) Report findings of inspection, examination and tests.

(c) *Level III Qualified Person*:

A Level III qualified person should have all the capabilities of Level II qualified person. He/She should also have the capability to identify test category or class in question and be well versed with the theoretical and practical aspects of the examination technique/method. In addition, a Level III qualified person should have capability to:

(i) Devise an inspection procedure and suggest alternate, complementary procedures.
(ii) Evaluate adequacy of specific programme for training and test the personnel for inspection, examination and the test method.
(iii) Prepare test procedures and suggest complementary techniques.

B.2 Education and Experience

The minimum requirements of education and experience of personnel for each level of qualification should be as given below. Other factors demonstrating capability to do the given job are prior experience and performance or satisfactory completion of testing.

(a) *Level I*

Personnel holding Level-I certificate in NDT certification in accordance with practices of IS-13805-2004 or other practices acceptable to the regulatory body.

(b) *Level II*

Personnel holding Level-II certificate in NDT certification in accordance with IS-13805-2004 or other practices acceptable to the regulatory body.

(c) *Level-III*
Personnel holding Level-III certificate in NDT certification in accordance with IS-13805-2004 or other practices acceptable to the regulatory body.

B.3 **Capability for Certification**

The capabilities of a personnel for certification should be determined by an appropriate evaluation of his/her education, experience, training, test results or capability demonstration. In order to evaluate the capabilities, the personnel should have passed both written and practical examinations conducted by a national body or its authorised body and received written certificate from the national body issued to this effect.

B.4 **Written Certificate of Qualification**

The qualification of personnel should be certified in writing in an appropriate form including the following information**:**

(a) Identification of person being certified.
(b) Date of certification and date of certificate expiry.
(c) Level of capability.
(d) Activities certified to perform.
(e) Basic criteria for certification.
    (i) records of education, experience and training.
    (ii) test results.
    (iii) results of capability demonstration.
(f) Results of periodic evaluation.
(g) Results of physical examination, when required.
(h) Signature and seal of the authorised body issuing the certificate.

B.5 **Re-evaluation of Performance**

(a) On the job performance of personnel for inspection, examination and testing should be re-evaluated at periodic intervals not exceeding 3 years. Re-evaluation should be by evidence of continued satisfactory performance or redetermination of the capability.
(b) During this evaluation or, at any time, if it were determined by the RO that the capabilities of an individual are not in accordance with the qualifications specified for the job, that person should be removed from the inspection activity until such a time as the required capability has been demonstrated.
(c) The RO should identify any special physical characteristics needed by the qualified personnel for the performance of each inspection activity. Such qualified inspection personnel having these special physical characteristics should have themselves examined by suitable authorised agencies at regular intervals which do not exceed one year.
(d) Any personnel who have not performed inspection, examination or testing activities in their qualified method/technique for a period of one year should be re-evaluated by re-determination of required capability.

B.6 **Records**

Documented evidence should be provided to ensure that qualified personnel have performed the inspections. Systematic records of the qualified personnel should be maintained by the RO.

B.7 **Responsibilities of Inspection Personnel of RO should be as follows:**

(a) Witness or otherwise verify all the examinations and carry out additional examinations, if necessary, to ensure that all applicable requirements have been met.
(b) Assure that the NDE methods are used to adopt appropriate examination techniques as applicable.

(c) Assure that the examinations are performed in accordance with the approved procedures by the qualified personnel.
(d) Certify the examination records only after verifying that all the requirements have been met and that the records are correct.

B.8 **The functions of different levels of inspection personnel in a project and operating plants are summarised in the Table-1 given below.**

Table-I: Functions of Different Levels of Inspection Personnel

Sl. No.	Project Function	Levels		
		Level-I	Level-II	Level-III
1.	Recording inspection, examination and testing data.	✓	✓	✓
2.	Implementing inspection, examination and testing procedures.	✓	✓	✓
3.	Planning inspections, evaluation and tests, setting up tests including related equipment.	✓	✓	✓
4	Evaluating the validity and acceptability of inspection, examination and testing results.		✓	✓
5	Reporting inspection, examination and testing results.		✓	✓
6.	Supervising equivalent or lower level personnel.		✓	✓
7.	Qualifying performance of lower level personnel.		✓	✓
8.	Evaluating the adequacy of specific performance used to train and test inspection, examination and testing personnel.			✓
9.	Qualifying same level personnel.			✓

<u>Note</u>: ✓ Means applicable

APPENDIX-C
METHODS OF EXAMINATION

Table-I: COMPONENTS AND CORRESPONDING METHODS OF EXAMINATION

Sl. No.	Components to be Examined.	Methods of Examination.	
1	Piping weld joints Vessel weld joints	Visual, Surface and Volumetric Visual, Surface and Volumetric	
2	Pumps and Valves - Pressure retaining welds - Interior surfaces	 Visual, Surface and Volumetric Visual and Surface#	
3	Piping Equipment Supports Snubbers Component attachment welds Others	Visual and Dimensional@ (wherever applicable). Functional Tests - Free operability over entire stroke length, sensitivity, and drag force measurement in tension as well as in compression. Visual and Surface# Visual	
4	Flywheel.	Visual, Surface and Volumetric	
5	Mechanical Couplings - Bolts - Studs - Nuts - Bushing	Size, ≤ 25.4 mm Visual Visual Visual Visual	Size, >25.4 mm Visual, Surface and Volumetric Visual, Surface and Volumetric Visual, Surface and Volumetric Visual, Surface and Volumetric
6	Other components	Visual	Visual, Surface and Volumetric
7	All components - Corrosion - Erosion	Dimensional Dimensional Dimensional	
8	Heat Exchangers -Tube side -Channel side weld joints -Shell side weld joints including nozzle to vessel welds	 Volumetric (ECT or relevant technique). Visual, Surface and Volumetric Visual, Surface and Volumetric	
9	Steam Generators -Tube side -Channel & shell side weld joints including nozzle to vessel welds.	 Volumetric Visual, Surface and Volumetric	
10	Coolant Channel	Visual, surface, volumetric (for flaws and nodules), chemical (for hydrogen isotope equivalent concentration, H_{eq}), PIE. Dimensional (wall thickness and ID), Refer the Section 14AC of this Guide.	
11	Feeder Pipes/elbows	Visual, UT thickness gauging, volumetric, surface, feeder gap.	
12	Fueling Machine components -Fueling machine head -Sealing plug	 Visual and Integrative Visual and Hydrostatic test	

Notes:
Inspection is required only for a material susceptible to stress corrosion cracking and pitting.
@ Verification of hot set and cold set points in case of hanger supports.

APPENDIX- D

FRACTURE MECHANICS BASED ANAYLSIS

[Refer sub-section 7.5]

The guidelines for the assessment of fitness for service of a flawed component based on fracture mechanics analysis are as follows:

D.1 The stresses in the region containing the flaw should be analysed for all conditions of operation and postulated accident conditions. The different levels of service conditions are defined in Annexure-I.

D.2 The conditions leading to the maximum stress at the flaw location should be selected along with the shape (circumscribing elliptical or circular), size and orientation such that the projected plane of the flaw is perpendicular to the direction of the maximum stress.

D.3 The values of the material properties used in the analysis should be those measured, but where such data do not exist, conservative material property should be assumed for the material (type and grade) containing the flaw.

D.4 In the case of degradation in the material properties due to operating environment such as irradiation, hydrogen embrittlement or hydrogen pick up, etc., the material surveillance samples should be used for obtaining the properties. Where such surveillance samples are not available, the material properties should be obtained from the data available in literature or generated in-house (if that is acceptable to the regulatory body) for the type and grade of material of the component containing the flaw.

D.5 The service induced flaw growth and changes in material properties should be factored in for the assessment of the fitness for service and residual life of the component till the next inspection.

D.6 In cases where the evaluation methods used in analysis are not as per the acceptable standards, the same should be put up for review and acceptance of the regulatory body.

APPENDIX-E
DOCUMENTATION
[Refer Sub- Section 11.3 & 11.4]

E.1 The following documents should be developed and maintained from the design stage onwards:

(a) Approved records of the:
  (i) list of components,
  (ii) material specifications,
  (iii) component specifications,
  (iv) as-built drawings,
  (v) manufacturing/fabrication processes,
  (vi) heat treatment records,
  (vii) installation and commissioning specifications and
  (viii) acceptance of deviations from specifications (NCRs).

(b) The catalogue of samples of materials stored as archive samples, e.g., PT off-cuts, garter springs, feeder pipes and elbows, CT off-cuts, etc.
(c) PSI data and reports.
(d) PSI and ISI programme documents containing criteria for selection of components, location and extent of areas to be examined and frequency of ISI.
(e) Approved and verified PSI and ISI procedures.
(f) Sensor type, calibration records and charts.
(g) Sensitivity standards.
(h) Acceptance criteria.
(i) All amendments and revisions of the ISI document.
(j) Verified ISI records and reports.
(k) Calibration records.
(l) Record of all unacceptable flaws/indications found during PSI and ISI, details of their disposition, corrective actions taken, details of any repairs or replacements and subsequent examinations.
(m) Details of collective radiation dose received to aid in planning of inspections.

E.2 The detailed examination procedure should be clearly identified in the PSI and ISI programme documents. The procedures should be verified as given in the sub-section 9.1 and should include the following:

(a) Scope of the examinations.
(b) Applicable codes and standards.
(c) Acceptance criteria.
(d) Supporting documents.
(e) Qualifications of the inspection personnel.
(f) Methods and equipment to be used for the inspections.
(g) List of components to be examined.
(h) Requirements for calibration and re-calibration.
(i) Technique sheet(s).
(j) Format for data recording.
(k) Minimum recording level indication, if applicable.
(l) The data to be recorded.

APPENDIX-F
AUDIT
[Refer Section 12]

F.1 **Audit Plan**

The audit should be planned in advance. The plan should include scope of the audit, composition of the audit team, activities to be audited, sections/agencies to be notified and the applicable documents.

F.2 **Audit Notification**

The organisation to be audited should be notified well in advance before scheduling the audit. The notification should be in writing and should include information such as the scope and schedule of the audit, names of the auditor and the documents to be reviewed.

F.3 **Pre-Audit Meeting**

A pre-audit meeting should be conducted with the concerned sections to apprise them of the scope and schedule of the audit and to establish the channels of communications.

F.4 **Performance of Audit**

The following aspects should be audited:

(a) PSI and ISI programmes for completeness and adequacy.
(b) Completeness and adequacy of procedures and instructions.
(c) Implementation of the procedures and instructions in the areas being inspected.
(d) Records of training of the inspection personnel and their level of qualifications.
(e) Availability and updating of records like inspection, reports, and history cards.
(f) Compliance with the applicable codes and standards.

F.5 **Post-Audit Meeting**

A post-audit meeting should be held by the audit team with the audited organisation to present the audit findings, to finalise the action points and to conclude the audit.

F.6 **Audit Report**

A report should be prepared immediately after the audit. The report should include the following information:

(a) The purpose of audit.
(b) The list of standards, procedures, or other documents used as bases for auditing.
(c) The list of audit team members.
(d) Pre-audit meeting details.
(e) Summary of audit findings.
(f) Suggestions for correcting the non-conformances or deficiencies in the PSI/ISI programme.

F.7 **Follow-up Activity**

(a) By the Audited Section:
The audited section should review and report to the auditing organisation the progress achieved in completing the suggested corrective actions.

(b) By the Audit Organisation:
The audit organisation should evaluate the response and confirm that the suggested corrective actions are accomplished as scheduled.

ANNEXURE-I

DESIGN, SERVICE AND TEST LIMITS

(Excerpts from NCA-2142.4 ASME BPVC, Section III, Division 1, 2015)

I.1 **Design limits:**

The limits for design loadings should meet the requirements of the design specification/code.

I.2 **Service Limits**:

The design specification may designate Service Limits as defined in I.2.1 through I.2.4 below.

I.2.1 Level A

Those sets of limits which must be satisfied for all loadings considered as Level A service as identified in the design specifications to which the component or support may be subjected to during the performance of its specified service function.

I.2.2 Level B

Those sets of limits which must be satisfied for all loading considered as Level B service as identified in the design specifications for which these service limits are designated. The component or support must withstand these loadings without damage requiring repair.

I.2.3 Level C

Those sets of limits which must be satisfied for all loading considered as Level C service as identified in the design specifications for which these service limits are designated. These sets of limits permit large deformations in areas of structural discontinuity which may necessitate the removal of the component or support from service for inspection or repair of damage to the component or support. Therefore, the selection of this limit should be reviewed by the RO for compatibility with established system safety criteria (NCA-2141).

I.2.4 Level D

Those sets of limits which must be satisfied for all loading considered as Level D service as identified in the design specifications for which these service limits are designated. These sets of limits permit gross general deformations with some consequent loss of dimensional stability and damage, which may require repair or removal of the component or support from service. Therefore, the selection of this limit should be reviewed by the RO for compatibility with established system safety criteria (NCA-2141).

I.3 **Alternate Service Limits**

Components or supports may be designed using more restrictive service limits than specified in the design specification. For example, Level B service limits may be used where Level C service limits have been specified.

I.4 **Test Limits**

The loadings which include pressure tests as per design specification/code should meet the following:

(i) The limits for test loadings should meet the requirements of the design specification/code.
(ii) The selection of limits for other tests defined by the owner [NCA2142.3 (b)] should be included in the design specifications.

APPENDICES AND ANNEXURES FOR PHWRs

APPENDIX-A.A

DETERMINATION OF INSPECTION CATEGORIES, SIZE OF FAILURE, FATIGUE USAGE FACTOR AND STRESS INTENSITY

[Refer sub-section 4A.2]

A.A.1 Inspection Categories

(a) Determine the failure size classification, large, medium and small as given in the sub-section A.A.2.
(b) Determine the fatigue usage factor (FUF) as per the procedure given in the sub-section A.A.3.
(c) Determine the stress intensity classification as given in the sub-section A.A.4.
(d) Locate the intersection of the stress intensity ratio and fatigue usage factor in Fig. 1 (A) or Fig. 1 (B), as appropriate, to determine the inspection category (A, B, C1 or C2).

FIGURE-1 (A): DETERMINATION OF INSPECTION CATEGORIES (A, B, C1, and C2)

FOR *MEDIUM* FAILURE SIZE

Stress Intensity Ratio, R_{SA} or R_{SM} (see Section A.A.4)

Stress Intensity Ratio	Low Fatigue Usage Factor	Medium Fatigue Usage Factor	High Fatigue Usage Factor	Type of Stress Intensity
> 2/3	**Cl**	**B**	**A**	High
≤2/3	**C l**	**B**	**B**	Medium
≤1/3	**C2**	**C1**	**Cl**	Low

FUF → 0 ≤0.01 ≤0.1 >0.1

Note: [See A.A.3 for Fatigue Usage Factor (FUF)]

FIGURE-1 (B): DETERMINATION OF INSPECTION CATEGORIES (A, B, C1, and C2)

FOR *LARGE* FAILURE SIZE.

Stress Intensity Ratio, R_{SA} or R_{SM} (see Section A.A.4)

Stress Intensity Ratio	Low Fatigue Usage Factor	Medium Fatigue Usage Factor	High Fatigue Usage Factor	Type of Stress Intensity
> 2/3	**C1**	**A**	**A**	High
≤2/3	**C l**	**B**	**A**	Medium
≤1/3	**C2**	**C1**	**C1**	Low

FUF→ 0 ≤0.01 ≤0.1 >0.1

Note: [See A.A.3 for Fatigue Usage Factor (FUF)]

A.A.2 Size of failure

A.A.2.1 For components of systems located inside the containment boundary, considering Level A and B conditions (Annexure-I), the size of failure can be determined as follows:

(a) Piping, Pumps and Valves:
The size of failure is expressed as the ratio of the maximum energy release rate from the failure being considered to the maximum energy release rate from the most severe failure considered during the design of the systems that directly transport heat from the nuclear fuel.

Where the maximum energy release ratio (R_E) is not determined by detailed analysis, it should be determined as follows:

$$R_E = \frac{A_F}{A_D} \times \frac{h_F}{h_D} \times \frac{\sqrt{P_F}}{\sqrt{P_D}} \times \frac{\sqrt{r_F}}{\sqrt{r_D}}$$

Where,

A = Flow cross-sectional area (guillotine failure).
h = Enthalpy at operating temperature and pressure minus the enthalpy of saturated liquid at atmospheric pressure.
P = Operating pressure (gauge).
r = Density of fluid.
Subscript F = Conditions for Failure location being considered.
Subscript D = Conditions for design maximum energy release rate.

The failure size classification should be as follows:

Small failure: $R_E \leq 0.1$
Medium failure: $0.1 < R_E < 0.3$
Large failure: $R_E \geq 0.3$

(b) Vessels:
A size of failure based on energy release rate should be determined as mentioned above, using flow cross-sectional areas of the lines connected to the vessel that would continue to supply fluid to the failure. A failure size based on containment pressure rise should also be determined. It should be expressed as the ratio of the immediate containment pressure rise caused by the instantaneous release of fluid in the vessel, assuming a sealed containment, to the lowest pressure rise required to cause closure of the containment ventilation system.

Where the containment pressure rise ratio (R_v) is not determined by a detailed analysis, it should be determined as follows, assuming a uniform vapour distribution and no fluid heat loss:

$$R_v = {}^{P_c}/_{P_s}$$

Where,

R_v = Containment pressure rise ratio
P_c = Containment Pressure rise = $P \times V_T / V_C$
P_s= Lowest containment pressure rise to cause the closure of containment ventilation system.
V_c = Containment volume.
V_T = Vapour volume at pressure, $P = W_g \times V_g$.
V_g= Specific volume of saturated vapour at pressure P.

P = Operating pressure (gauge)
W_g = Weight of vapour produced, $W_l\left(h - h_f\right)/h_{fg}$
W_l= Maximum weight of liquid in vessel (considering the operating conditions).
h = Enthalpy of liquid at operating condition.
h_f = Enthalpy of saturated liquid at normal containment pressure, P
h_f = Enthalpy of evaporation at pressure, P.

The failure size classification based on containment pressure rise should be as follows:

Small failures: R_v <1.
Medium failure: 1≤R_v<3.
Large failure: R_v>3

If the two failure sizes determined from R_E and R_v differ, the larger of the two should be taken as the size of failure.

(c) Associated pipelines connecting to a failed pipe or component:
Flow from a pipeline connected to a failed pipe or component need not be included in computing the size of failure, if the line incorporates two remotely operated or self-closing valves in series, provided that such valves and their power supplies are,
   (i) not connected to the same power source; and
   (ii) sufficiently remote from one another that both would not be subjected to damage or malfunction from the same failure.

A.A.2.2 For components in systems located outside or forming a part of containment boundary, the size of failure can be determined as follows:

For the components in systems classified under the sub-section 4A.1.1 (a), located outside or forming a part of containment boundary, all failures capable of causing a radiation hazard equal to or exceeding the prescribed dose limits for a serious process failure (Category 2 event or above, as defined in AERB safety guide on, 'Design Basis Events for PHWR', AERB/SG/D-5) should be classified as large.

A.A.2.3 For components of the systems classified in the sub-section 4A.1.1 (b), the size of failure can be determined as follows:

(a) Failure that may prevent functioning of the systems as mentioned in the safety report should be considered as large.
(b) Failures that do not prevent functioning of the system as mentioned in the safety report should be considered as small.

A.A.2.4 For components of the systems classified in the sub-section 4A.1.1 (c), the size of failure can be determined as follows:

(a) The size of failure should be classified according to the total effect of the initiating and resulting failures.
(b) For dislodgement, that may lead to failure of a system classified in the sub-section 4A.1.1 (a), the relevant rules in the sub-section A.A.2.1 or A.A.2.2 above should be used to determine the size of failure.
(c) For dislodgement that may lead to failure of a system classified in the sub-section 4A.1.1 (b), the rules in the sub-section A.A.2.3 above should be used to determine the size of failure.

A.A.3 Fatigue Usage Factor (FUF)

A.A.3.1 The fatigue usage factor used in establishing inspection categories should be determined by the rules given in the Section III, ASME Boiler and Pressure Vessel Code.

A.A.3.2 FUF classification should be as follows:

(a) Low fatigue – FUF $\leq$ 0.010
(b) Medium fatigue – FUF > 0.010 $\leq$ 0.1; and,
(c) High fatigue – FUF > 0.10.

A.A.3.3 Where the FUF is not calculated, fatigue classification should be medium except in the following cases:

(a) Where operational experience under similar conditions indicates susceptibility to fatigue cracking, the fatigue classification should be high; or
(b) When (a) is not applicable and construction code of record does not require calculation of conservative value of fatigue usage, fatigue classification should be low.

A.A.4 Stress Intensity

A.A.4.1 Stresses or stress intensities should be used to calculate stress ratios as follows:

For Level A and B conditions (Annexure-I), the calculated stress or stress intensity should be compared to the allowable stress or stress intensity such that,

$$\textbf{Stress ratio } \mathbf{R_s} = \frac{\textbf{Calculated stress or stress intensity}}{\textbf{Allowable stress or stress intensity}}$$

(a) $\mathbf{R_s}$ should be determined for all stresses and stress combinations that are calculated to meet design requirements for normal, upset and emergency conditions. These stress ratios are determined for each potential inspection area, and the highest ratio then determines the stress classification.
(b) Primary and secondary stresses, including thermal expansion stresses, should be included in the evaluation.

A.A.4.2 The use of more than one method of stress calculation may produce different calculated stresses or stress intensities. The calculated stress intensity to be used in the sub-section A.A.4.1 should be taken from the most precise method employed. The following methods are listed in order of precision:
(a) Methods of analysis more refined than those given in the Section III, ASME Boiler and Pressure Vessel Code, (e.g., finite element analysis) with inputs and modeling assumptions justified and documented; or
(b) Methods using the rules given in the Section III, ASME Boiler and Pressure Vessel Code.

A.A.4.3 Where the stress analysis considers more than one condition, several values of $\mathbf{R_s}$ should be obtained. The maximum value of $\mathbf{R_s}$ should be expressed as R_{SM}.

For material where the value of S_a corresponding to 10^6 cycles is > S_m, the ratio R_{SM} can be calculated as follows:

$$R_{SA} = R_{SM}\frac{S_m}{S_a}$$

Otherwise,

$$R_{SA} = R_{SM}$$

R_{SA}= adjusted stress ratio.
S_m is defined in article NB-3200 Section III of ASME B&PV code.
S_a is defined in article NB-3222 Section III of ASME B&PV code.

Stress classifications of inspection areas should be as follows:

(i) low stress intensity – areas where $R_{SA} \leq 1/3$.
(ii) medium stress intensity – areas where $R_{SA} > 1/3$ and $\leq 2/3$; and
(iii) high stress intensity – areas where $R_{SA} > 2/3$.

ANNEXURE-I.A

GUIDELINES FOR SELECTION OF PRESSURE TUBES FOR VOLUMETRIC INSPECTION, DIMENSIONAL INSPECTION AND HYDROGEN EQUIVALENT (H_{EQ}) MEASUREMENTS

[Refer sub-section 14AC.1.3 and 14AC.3.1 (d)]

I.A.1 The pressure tubes should represent the material installed in the core with respect to:

(a) Chemical composition, manufacturing process and metallurgical characteristics.
(b) The neutron flux spectrum, neutron fluence.
(c) Irradiation temperature history.

I.A.2 The channels for the inspections should be selected considering the following:

(a) Channel power
(b) Channel having high power to flow ratio
(c) Even distribution within the core
(d) Even distribution among coolant headers
(e) Even distribution among AGMS headers
(f) High initial hydrogen
(g) High axial creep
(h) Low axial creep
(i) History of fuel failure
(j) Proximity to adjuster
(k) High diametric creep
(l) Difficulty faced during on-power refueling
(m) Tubes with flaw indications observed during PSI/ISI
(n) Mechanical and metallurgical parameters
(o) Rolled joints having DCR on helium leak rates
(p) Channels having DCRs
(q) Channels having any station specific problem that may affect the service life of the coolant channel.

ANNEXURE-II.A

GUIDELINES FOR SELECTION OF COOLANT CHANNELS FOR MATERIAL SURVEILLANCE

[Refer sub-section 14AC.3]

II.A.1 The pressure tubes and the garter springs chosen for material surveillance should represent the material installed in the core, with respect to chemical composition, material properties, metallurgical parameters like grain size, grain aspect ratio, cold work, initial hydrogen concentration and any other parameter that governs the in-reactor material performance.

II.A.2 If there is a modal distribution of the parameters that govern the selection of the installed coolant channel material, then the selection should represent each modal type.

II.A.3 If more than one design of garter spring is used, surveillance programme should address all designs, even though they may be made of the same material.

II.A.4 The selected coolant channel for pressure tube and garter spring surveillance should have:

(a) Neutron fluence within 10% of the highest fluence channel in the core.
(b) The highest H_{eq}.
(c) The maximum creep deformation,
(d) Power within 5% of the highest rated channel.

II.A.5 The garter springs selected for testing should have the highest predicted load and from the coolant channel having the lowest measured or predicted PT-CT gap.

II.A.6 Sufficient amount of archive material should be available for the pressure tube and the garter springs chosen for material surveillance for establishing the baseline information.

II.A.7 A justification for the selection of the coolant channel for material surveillance based on the statistical analysis of the as-manufactured data, operating parameters and data from previous ISI campaigns should be provided to the regulatory body by the RO.

ANNEXURE-III.A

GUIDELINES FOR SELECTION OF LOCATIONS FOR MEASUREMENT OF H_{EQ} ALONG THE LENGTH OF A PRESSURE TUBE

[Refer sub-section 14AC.3.1 (b)]

The recommended bases for selection of locations for measurement of H_{eq} along the length of the pressure tube are as follows:

III.A.1 A sufficient number of measurements should be carried out so as to establish the H_{eq} profile along the length of the pressure tube.

III.A.2 The measurements should be carried out at the following 6 locations along the length of pressure tube body:

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Table-VII: Locations Along the Pressure Tube (in mm) for Slivering for Different Types of PHWRs

220 MWe	540 MWe	700 MWe
830	800	800
1250	1800	1800
2000	2800	2800
3000	3800	3800
3800	4800	4800
4800	5700	5700

Note: a) Distances are given for the center of the sliver cut from the PT inlet end.
b) The expected accuracy in location of the slivering tool is ±50mm.

III.A.3 For trending of H_{eq} in the main body of the pressure tube, slivering should be carried out thrice in the same channel. For such repetition, slivering locations should be varied axially by about 150 mm from the earlier scraped locations.

III.A.4 In case it is not possible to sliver the channels, the H_{eq} for the pressure tubes should be evaluated by using proven models to predict the upper-bound values with respect to the number of reactor hot operating years.

ANNEXURES FOR BWRs

ANNEXURE-I.B

EXAMINATION AREAS OF COMPONENTS AND SUPPORTS

Examination Area and its Category based on Safety Classification Classes 1, 2 and 3 as well as Metallic Liners of Class CC and Supports of Classes 1, 2 and 3 including Metallic and Concrete Component Supports
[Refer sub-section 4B.3.1]

Sl. No.	Examination Category	Examination Area
1		**Class1 Components**
1(b)	B-A	Pressure-Retaining Welds in Reactor Vessels
1(c)	B-B	Pressure-Retaining Welds in Vessels Other Than Reactor Vessels.
1(d)	B-D	Full Penetration Welded Nozzles in Vessels
1(e)	B-F	Pressure-Retaining Dissimilar Metal Welds in Vessel Nozzles
1(f)	B-G-1	Pressure-Retaining Bolting, Greater Than 2 in. (50 mm) in Diameter
1(g)	B-G-2	Pressure-Retaining Bolting, 2 in. (50 mm) and Less in Diameter
1(h)	B-J	Pressure-Retaining Welds in Piping
1(i)	B-K	Welded Attachments for Vessels, Piping, Pumps, and Valves
1(j)	B-L-2	Pump Casings
	B-M-2	Valve bodies
1(k)	B-N-1	Interior of Reactor Vessel
	B-N-2	Welded Core Support Structures and Interior Attachments to Reactor Vessels
	B-N-3	Removable Core Support Structures
1(l)	B-O	Pressure-Retaining Welds in Control Rod Drive and Instrument Nozzle Housings
1(m)	B-P	All Pressure retaining components
1(n)	B-Q	Steam generator tubing
2		**Class 2 Components**
2(b)	C-A	Pressure-Retaining Welds in Reactor Vessels
2(c)	C-B	Pressure-Retaining Nozzle Welds in Pressure Vessels
2(d)	C-C	Welded Attachments for Pressure Vessels, Piping, Pumps, and Valves
2(e)	C-D	Pressure-Retaining Bolting, Greater Than 2 in. (50 mm) in Diameter
2(f)	C-F-1	Pressure-Retaining Welds in Austenitic Stainless Steel or High Alloy Piping
2(g)	C-F-2	Pressure-Retaining Welds in Carbon or Low Alloy Steel Piping
2(h)	C-H	All Pressure-Retaining Components
3		**Class 3 Components**
3(b)	D-A	Welded Attachments for Pressure Vessels, Piping, Pumps, and Valves
3(c)	D-B	All Pressure-Retaining Components
4		**Class MC and Metallic Liners of Class CC Components**
4(b)	(E-A)	Containment Surfaces
4(c)	(E-C)	Containment Surfaces Requiring Augmented Examination
4(d)	(E-G)	Pressure-Retaining Bolting
5.		**Class 1, 2, 3, and MC Components Supports**
5(b)	F-A	Supports
6		**Class CC Concrete Components**
6(b)	L-A	Concrete
6(c)	L-B	Unbound Post-Tensioning System

ANNEXURE-II.B

METHODS OF EXAMINATION TO BE CARRIED OUT ON DIFFERENT COMPONENTS

[Refer sub-section 4B.3.1]

Sl. No.	**Components to be Examined.**	**Methods of Examination**	
1	Piping weld joints Vessel weld joints and nozzles Inspection of core belt region (core shroud) up to the possible extent Core internals	Visual, surface and volumetric Visual, surface and volumetric* Visual, volumetric* Visual	
2	Pumps and valves -Pressure retaining welds -Interior surfaces	 Visual, surface and volumetric Visual, surface#	
3	Piping equipment supports Snubbers Component attachment welds Others	Visual and dimensional@ (wherever is applicable). Functional tests - Free operability over entire stroke length, sensitivity, and drag force measurement in tension as well as in compression Visual and surface# Visual	
4	Rotating Machinery	Visual, surface and volumetric	
5	Mechanical Couplings	Size ≤ 25.4mm	Size > 25.4 mm
	-Bolts - Studs - Nuts - Ligament between threaded stud hole - Bushing	Visual Visual Visual Visual Visual	Visual, surface, volumetric Visual, surface, volumetric Visual, surface, volumetric Visual, surface, volumetric Visual, surface, volumetric
6	Other components.	Visual	Visual, surface, volumetric.
7	All components - Corrosion - Erosion	Dimensional Dimensional Dimensional	
8	Heat Exchangers -Tube side -Channel-side weld joints -Shell-side weld joints including nozzle to vessel welds	 Volumetric Visual, surface and volumetric Visual, surface and volumetric	
9	Steam Generators: - Tube side - Channel-side & shell-side weld joints, including nozzle-to-vessel welds.	 Volumetric Visual, surface and volumetric	

Notes: * To include volumetric examinations using appropriate techniques.
@ Verification of hot-set and cold-set points in case of hanger supports.
# Inspection is required only for material subjected to stress corrosion cracking.

ANNEXURE-III.B

EXTENT OF EXAMINATION FOR PSI AND ISI FOR SYSTEMS AND COMPONENTS

[Refer sub-section 4B.3.2]

Sl. No	Classification of System and Components with Examination Category	Table of the ASME Section XI to be Referred
1	**Class 1 Components**	
1(a)	Pre-Service Examination	Article –IWB-2200
1(b)	Pressure-Retaining Welds in Reactor Vessels. ((B-A)	Table IWB-2500-1 (B-A)
1(c)	Pressure-Retaining Welds in Vessels Other Than Reactor Vessels (B-B)	Table IWB-2500-1 (B-B)
1(d)	Full Penetration Welded Nozzles in Vessels (B-D)	Table IWB-2500-1 (B-D)
1(e)	Pressure-Retaining Dissimilar Metal Welds in Vessel Nozzles(B-F)	Table IWB-2500-1 (B-F)
1(f)	Pressure-Retaining Bolting, Greater Than 2 in. (50 mm) in Diameter (B-G-1)	Table IWB-2500-1 (B-G-1)
1(g)	Pressure-Retaining Bolting, 2 in. (50 mm) and Less in Diameter. (B-G-2)	Table IWB-2500-1 (B-G-2)
1(h)	Pressure-Retaining Welds in Piping. (B-J)	Table IWB-2500-1 (B-J)
1(i)	Welded Attachments for Vessels, Piping, Pumps, and Valves	Table IWB-2500-1 (B-K)
1(j)	Pump Casings; valve bodies (B-L-2, B-M-2).	Table IWB-2500-1 (B-L-2, B-M-2)
1(k)	B-N-1 Interior of Reactor Vessel; B-N-2, Welded Core Support Structures and Interior Attachments to Reactor Vessels; B-N-3, Removable Core Support Structures	Table IWB-2500-1 (B-N-1, B-N-2, B-N-3)
1(l)	Pressure-Retaining Welds in Control Rod Drive and Instrument Nozzle Housings (B-O),	Table IWB-2500-1 (B-O)
1(m	All Pressure retaining components (B-P)	Table IWB-2500-1 (B-P)
1(n)	Successive Inspections	Article IWB-2420
1(o)	Additional Examinations	Article IWB-2430
1(p)	Components exempt from examination	Article IWB-1220
2	**Class 2 Components**	
2(a)	Pre-Service Examination	Article –IWC-2200
2(b)	Pressure-Retaining Welds in Reactor Vessels. ((C-A)	Table IWC-2500-1 (C-A)
2(c)	Pressure-Retaining Nozzle Welds in Pressure Vessels (C-B)	Table IWC-2500-1 (C-B)
2(d)	Welded Attachments for Pressure Vessels, Piping, Pumps, and Valves (C-C)	Table IWC-2500-1 (C-C)
2(e)	Pressure-Retaining Bolting, Greater Than 2 in. (50 mm) in Diameter (C-D)	Table IWC-2500-1 (C-D)
2(f)	Pressure-Retaining Welds in Austenitic Stainless Steel or High Alloy Piping (C-F-1)	Table IWC-2500-1 (C-F-1)
2(g)	Pressure-Retaining Welds in Carbon or Low Alloy Steel Piping (C-F-2)	Table IWC-2500-1 (C-F-2)
2(h)	All Pressure-Retaining Components (C-H)	Table IWC-2500-1 (C-H)
2(i)	Successive Inspections	Article IWC-2420

Sl. No	Classification of System and Components with Examination Category	Table of the ASME Section XI to be Referred
2(j)	Additional Examinations	Article IWC-2430
2(k)	Components exempt from examination	Article IWC-1220
3	**Class 3 Components**	
3(a)	Pre-Service Examination	Article –IWD-2200
3(b)	Welded Attachments for Pressure Vessels, Piping, Pumps, and Valves (D-A)	Table IWD-2500-1 (D-A)
3(c)	All Pressure-Retaining Components (D-B)	Table IWD-2500-1 (D-B)
3(d)	Components exempt from examination	Article IWD-1220
3(e)	Successive Inspections.	Article IWD-2420
3(f)	Additional Examinations	Article IWD-2430
4	**Class MC and Metallic Liners of Class CC Components**	
4(a)	Pre-Service Examination	Article –IWE-2200
4(b)	Containment Surfaces (E-A)	Table IWE-2500-1 (E-A)
4(c)	Containment Surfaces Requiring Augmented Examination (E-C)	Table IWE-2500-1 (E-C)
4(d)	Pressure-Retaining Bolting (E-G)	Table IWE-2500-1 (E-G)
4(e)	Components exempt from examination	Article IWE-1220
4(f)	Successive Inspections	Article IWE-2420
4(g)	Condition of Surface to be examined	Article IWE-2600
5	**Classes 1, 2, 3, and MC Components' Supports**	
5(a)	Pre-Service Examination	Article –IWF-2200
5(b)	Supports	Table IWF-2500-1 (F-A)
5(c)	Components exempt from examination	Article IWF-1230
5(d)	Support examination boundaries	Article IWF-1300
5(e)	Successive Inspections.	Article IWF-2420
5(f)	Additional Examinations	Article IWF-2430
6	**Class CC Concrete Components**	
6(a)	Pre-Service Examination	Article –IWL-2200
6(b)	Concrete (L-A)	Table IWL-2500-1 (L-A)
6(c)	Components exempt from examination	Article IWL-1220
6(d)	Unbonded Post–tensioning systems	Article IWL-2222
6(e)	In-service –Inspection records	IWA-6340

ANNEXURE-IV.B

COMPONENTS EXEMPT FROM EXAMINATION

[Refer sub-section 4B.5]

IV.B.1 The following components or portions of components are exempted from the volumetric, surface, VT-1 visual, and VT-3 visual examination requirements of IWB-2500:

(a) The components that are connected to the reactor coolant system and are part of the reactor coolant pressure boundary, and that are of such a size and shape so that upon a postulated rupture, the resulting flow of coolant from the reactor coolant system under normal plant operating conditions is within the capacity of the makeup systems and that are operable with on-site emergency power. The emergency core cooling systems are excluded from the calculation of makeup capacity.

(b) See (i) through (iii) below -

(i) Components and piping segments NPS 1 (NB 25) and smaller, except for steam generator tubing.
(ii) Components and piping segments which have one inlet and one outlet, both of which are NPS 1 (NB 25) and smaller.
(iii) Components and piping segments which have multiple inlets or multiple outlets whose cumulative pipe cross-sectional area does not exceed the cross-sectional area defined by the OD of NPS 1 (NB 25) pipe.

(c) The reactor vessel head connections and associated piping, NPS 2 (NB 50) and smaller, made inaccessible by control rod drive penetrations.

(d) The welds or portions of welds that are inaccessible due to being encased in concrete, buried underground, located inside a penetration, encapsulated by guard pipe or due to maintenance constraints.

IV.B.2 **Excerpts of IWB-2500 on VT-3 Requirements:**

(a) Components should be examined and tested as specified in Tables as IWB-2500-1 (B-A) through IWB-2500-1 (B-Q). The method of examination for the components and parts of the pressure-retaining boundaries should comply with those tabulated in Tables IWB-2500-1 (B-A) through IWB-2500-1 (B-Q) except where alternate examination methods are used meeting the requirements of IWA-2240.

(b) Tables IWB-2500-1(B-A) through IWB-2500-1(B-Q) are organized as follows:

Sl. No.	Examination Category	Examination Area
1	B-A	Pressure-Retaining Welds in Reactor Vessel
2	B-B	Pressure-Retaining Welds in Vessels other than Reactor Vessels
3	B-D	Full Penetration Welded Nozzles in Vessels
4	B-F	Pressure-Retaining Dissimilar Metal Welds in Vessel Nozzles
5	B-G-1	Pressure-Retaining Bolting, greater than 50 mm (2 in.) in Diameter
6	B-G-2	Pressure-Retaining Bolting, 50 mm (2 in.) and less in diameter
7	B-J	Pressure-Retaining Welds in Piping

Sl. No.	Examination Category	Examination Area
8	B-K	Welded Attachments for Vessels, Piping, Pumps, and Valves
9	B-L-2	Pump Casings
10	B-M-2	Valve Bodies
11	B-N-1	Interior of Reactor Vessel
12	B-N-2	Welded Core Support Structures and Interior Attachments to Reactor Vessels
13	B-N-3	Removable Core Support Structures
14	B-O	Pressure-Retaining Welds in Control Rod Drive and Instrument Nozzle Housings
15	B-P	All Pressure-Retaining Components
16	B-Q	Steam Generator Tubing

ANNEXURE -V.B

TEST REQUIREMENTS FOR PRESSURE RETAINING COMPONENTS

[See subsection 5B.2.3.1]

V.B.1 The pressure-retaining components should be subjected to the following tests:

(a) A system leakage test as a part of ISI.

(b) A system leakage test, undertaken before resuming operation, following each reactor outage where the integrity of the reactor coolant pressure boundary may have been affected.

(c) A system hydrostatic pressure test at or near the end of each inspection interval, if required.

(d) The pressure retaining components should be visually examined to the extent practicable while the system is under the test pressure and temperature. The test pressure and temperature should be maintained for a sufficient period before the examinations, to permit leakage to be identified.

(e) If leakages (other than normal/controlled leakage) are detected during the above test, the source of leakage should be located, and the area examined to the extent necessary to establish if any corrective action is required.

(f) The system leakage test should be performed at the test pressure that is not less than the specified system operating pressure.

(g) The duration of a test performed at a pressure higher than the system design pressure should be limited to prevent excessive stressing of the systems and components.

ABBREVIATIONS

AERB	Atomic Energy Regulatory Board
ALARA	As Low As Reasonably Achievable
ARJ	Adjacent to Rolled Joint
ASME B & PVC	American Society of Mechanical Engineers Boiler and Pressure Vessel Code
BARCIS	Bhabha Atomic Research Centre Inspection System
BSD	Biennial Shut Down
CC	Concrete Component
CCL	Critical Crack Length
CT	Calandria Tube
DCR	Design Concession Report
DHC	Delayed Hydride Crack
DHCV	Delayed Hydride Crack Velocity
ECT	Eddy Current Test
EFPY	Effective Full Power Year
EL	Elongation
EMCCR	En-Masse Coolant Channel Replacement
FAC	Flow/Assisted Corrosion
FPY	Full Power Year
H_{eq}	Equivalent Hydrogen Concentration
HFU	Horizontal Flux Unit
HOY	Hot Operating Year
HPFC	High Pressure Feeder Coupling
ID	Inside Diameter/Internal Diameter
IGSCC	Inter Granular Stress Corrosion Cracking
ISI	In-Service Inspection
K_{IC}	Plane Strain Fracture Toughness
K_{IH}	Threshold Stress Intensity Factor for Delayed Hydride Cracking
LBB	Leak Before Break
MC	Metallic Component
NCR	Non-Conformance Report
NDE	Non-Destructive Examination
NDT	Non-Destructive Testing
NPP	Nuclear Power Plant
NWT	Nominal Wall Thickness
OD	Outside Diameter/Outer Diameter
OO	Operating Organisation
PCP	Primary Coolant Pump
PHT	Primary Heat Transport System
PIE	Post Irradiation Examination
PIU	Poison Injection Unit
PSD	Periodic Shut Down/Planned Shut Down
PSI	Pre-Service Inspection
PT	Pressure Tube
QA	Quality Assurance
RJ	Rolled Joint
RO	Responsible Organisation
SAT	Site Acceptance Tests
SCP	Secondary Cycle Piping
SSC	Structure, Systems and Components
TOFD	Time of Flight Diffraction
UT	Ultrasonic Test
UTS	Ultimate Tensile Strength
YS	Yield Strength

BIBLIOGRAPHY

1. ATOMIC ENERGY REGULATORY BOARD, 'Glossary of Terms for Nuclear and Radiation Facilities and Associated Activities,' AERB/GLO, Rev.1, Mumbai, India (2022).

2. ATOMIC ENERGY REGULATORY BOARD, 'Code on Design of Heavy Water Reactor Based Nuclear Power Plants,' No. AERB/NPP-PHWR/SC/D, Rev.1, Mumbai, India, (2009).

3. ATOMIC ENERGY REGULATORY BOARD 'Code on the Nuclear Power Plant Operation,' No. AERB/SC/O (Rev. 1), Mumbai, India (2008).

4. ATOMIC ENERGY REGULATORY BOARD 'Code on Quality Assurance for Safety in Nuclear Power Plants,' AERB/SC/QA (Rev.1), Mumbai, India (2009).

5. BUREAU OF INDIAN STANDARDS, 'General Standard for Qualification and Certification of NDT Personnel,' IS: 13805, India. (2006).

6. ATOMIC ENERGY OF CANADA LIMITED 'Canadian Standard for Periodic Inspection of Nuclear Power Plant Components,' CAN CSA N- 285.4-14, Rev.1, (2016).

7. AMERICAN SOCIETY OF MECHANICAL ENGINEERS 'Boiler & Pressure Vessel Code on In-Service Inspection,' Section-XI, (2012).

8. AMERICAN SOCIETY OF MECHANICAL ENGINEERS, 'Code of Operation and Maintenance of Nuclear Power Plants,'. (2009 & 2014).

9. AMERICAN SOCIETY OF MECHANICAL ENGINEERS, 'Boiler & Pressure Vessel Code on Non-Destructive Examination,' Section-V, (2017).

10. AMERICAN SOCIETY OF MECHANICAL ENGINEERS, 'Code Comparison Report for Class-1 Nuclear Power Plant Components,' STP-NU-051-1, (2012).

11. INTERNATIONAL ATOMIC ENERGY AGENCY, 'Maintenance, Surveillance and ISI in Nuclear Power Plants,' Specific Safety Guide No. SSG-774, Vienna, (2022).

12. ATOMIC ENERGY REGULATORY BOARD, 'Life Management of Nuclear Power Plants,' AERB Safety Guide, AERB/NPP/SG/O-14, Mumbai, India, (2005).

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June 17, 2016	June 29, 2016
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December 09, 2016	January 05, 2017
February 16, 2017	March 9, 2018
May 7&8, 2018	September 18 &19, 2018
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