

Indian Nuclear Power Program and ISI requirements

By

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Indian Nuclear Power Program

- In India, nuclear energy development began with the objective of peaceful uses of atomic energy in improving the quality of life of the people and to achieve self-reliance in meeting the energy needs.
- The atomic energy program, which was initiated in a modest manner initially, has now grown as a wide spectrum, multi dimensional multidisciplinary with 63 organizations under DAE. The spectrum of these significant activities include R&D in Nuclear Sciences and Engineering, Exploration & Mining of Radioisotopes, Nuclear energy development and implementation, application of Nuclear Energy, Bio-Agricultural Research, Medical Sciences, etc., apart from commercial electrical power generation.
- The Indian nuclear program was conceived based on, unique sequential three-stages and associated technologies essentially to aim at optimum utilization of the indigenous nuclear resource profile of modest Uranium and abundant Thorium resources. This sequential three-stage program is based on a closed fuel cycle, where the spent fuel of one stage is reprocessed to produce fuel for the next stage.
- The commercial nuclear power program of the first stage (comprising of PHWRs and imported LWRs) is being implemented by Nuclear Power Corporation of India Limited (NPCIL), and the second stage (comprising of Fast Breeder Reactors) by Bharatiya Nabhikiya Vidyut Nigam Limited (BHAVINI).
- **STAGE 1: Pressurized Heavy Water Reactor using**
 - ✓ Natural UO₂ as fuel matrix
 - ✓ Heavy water as moderator and coolant
- **STAGE 2: Fast Breeder Reactor**
 - ✓ India's second stage of nuclear power generation envisages the use of Pu-239 obtained from the first stage reactor operation, as the fuel core in fast breeder reactors (FBR)
- **STAGE 3: Breeder Reactor**
 - ✓ The third phase of India's Nuclear Power Generation program is, breeder reactors using U-233 fuel. India's vast thorium deposits permit design and operation of U-233 fuelled breeder reactors

Indian Nuclear Power Program

DAE's own Research & Development wings include

- Bhabha Atomic Research Centre (BARC), Trombay
- Indira Gandhi Centre for Atomic Research (IGCAR)
- The Raja Ramanna Centre for Advanced Technology (RRCAT)
- Atomic Minerals Directorate
- Variable Energy Cyclotron Centre
- Global Centre for Nuclear Energy Partnership, Bahadurgarh in Haryana

Besides carrying out research at its own research centres, the DAE provides full support to seven aided institutions

- Tata Institute of Fundamental Research(TIFR): Main campus is located in Mumbai, but additional campuses are in Pune, Bangalore and Hyderabad.
- Tata Memorial Centre: Several Centers across the Country
- Saha Institute of Nuclear Physics, Bidhannagar, Kolkata
- Institute of Physics, Bhubaneswar
- Institute for Plasma Research, Ahmedabad
- Harish Chandra Research Institute, Allahabad
- Institute of Mathematical Sciences, Chennai

Regulatory Authority - AERB: The AERB reviews the safety and security of the country's Operating Nuclear Power Plants, Nuclear Power Projects, Fuel Cycle Facilities, and Other Nuclear / Radiation Facilities. The regulatory authority of AERB is derived from the rules and notifications promulgated under the Atomic Energy Act, 1962 and the Environmental (Protection) Act, 1986. The mission of the Board is to ensure that the use of Ionising Radiation and Nuclear Power in India does not cause undue risk to health and the Environment

Utility - NPCIL: Nuclear Power Corporation of India Limited (NPCIL) is a Public Sector Enterprise under the administrative control of the Department of Atomic Energy (DAE), Government of India. The Nuclear Power Corporation of India Ltd (NPCIL) is responsible for design, construction, commissioning and operation of thermal nuclear power plants

- NPCIL is presently (Nov-2017) operating 22 nuclear power reactors with an installed capacity of 6780 MW. The reactor fleet comprises two Boiling Water Reactors (BWRs) and 18 Pressurized Heavy Water Reactors (PHWRs) and 2 Pressurized Water Reactors (PWRs)

Indian Nuclear Power Program

Safety standards in Nuclear Power plants

- The performance of Indian nuclear power reactors in respect of safety has been excellent, with about 493 reactor years (till Nov-2018) of safe, reliable and accident-free operation. The releases of radioactivity to the environment have been a small fraction of the limits prescribed by the Atomic Energy Regulatory Board (AERB). The yearly radiation dose around the Indian NPPs, measured over the last many years, is an insignificantly small fraction of natural radiation dose and the stipulated regulatory limits.
- At all nuclear power stations, state of the art safety measures are provided based on principles of redundancy (more numbers than required) and diversity (operating on different principles). These include fail safe shutdown system to safely shutdown the reactor, combination of active and passive (systems working on natural phenomena and not needing motive power or operator action) cooling systems to remove the heat from the core at all times and a robust containment to prevent release of radioactivity in all situations. In addition, all nuclear power plants are designed to withstand extreme natural events like earthquake, flooding, tsunami, etc.
- A multi-tier safety mechanism comprising of safety review committees within Nuclear Power Corporation of India Limited (NPCIL) and safety review committees in the regulatory authority (Atomic Energy Regulatory Board- AERB) is in place to monitor the safety of nuclear power plants. In addition, a framework of periodic safety reviews, audits and inspection is in place.
- Nuclear power stations in coastal areas are designed taking into account the technical parameters related to earthquake, tsunami, storm surges, floods etc. at each site. Appropriate bunds are provided at Tarapur, Kalpakkam and Kudankulam sites for shore protection.
- The shore protection measures are designed and constructed to withstand the possible impact of natural events. Surveillance of these protection measures is carried out periodically. Post Fukushima, the safety review of all nuclear power plants was conducted by task forces of NPCIL and the expert committee of AERB. These safety reviews have found that Indian nuclear power plants are safe and have margins and features in design to withstand extreme events like earthquakes and tsunamis.

Indigenous Technology Development for Indian PHWRs

- The first stage program went through stages of technology demonstration, indigenization, standardization, consolidation and finally commercialization.
- While the first stage began with 220 MWe reactors supplied by AECL, Canada, the subsequent PHWRs have all been indigenous.
- Under international technology denial regime, the Indian scientists and engineers rose to the occasion and with their untiring and innovative efforts, not only RAPS –1 but the design, construction and commissioning of the other unit too (RAPS-2) could be successfully completed.
- Subsequently, MAPS units 1&2 were designed, constructed and commissioned with indigenous efforts. The design of 220 MWe PHWRs was standardized, and NAPS 1&2 & KAPS 1&2 set up. Kaiga 1&2 and RAPS 3&4 were also set up with further improvements in design. The standard 220Mwe design was scaled up to 540 MWe and TAPP 3&4 (2x540Mwe) have been set up. The 700 MWe PHWR design, using the same core of 540MWe, has been developed and construction of such reactors is taken up subsequently.

Additionalities to the Indigenous three-stage program

- For faster nuclear power capacity addition, in parallel to the indigenous three-stage program, additionalities based on imports have been introduced. Two Light Water Reactors (LWRs) of 1000 MWe each are under operation and two more are under construction at Kudankulam in technical cooperation with the Russian Federation. As capacity addition through the indigenous route is guided by the fuel cycle linkages of the sequential three-stage program, faster capacity addition in the near term to meet the electricity needs of the country will be possible through these additionalities.

Indigenous Technology Development for Indian PHWRs

- The country has developed comprehensive capabilities in all aspects of nuclear power from siting, design, construction, operation of nuclear power plants. Comprehensive multidimensional R&D facilities have been set up. Capabilities have also been developed in front and back ends of the fuel cycle, from mining, fuel fabrication, storage of spent fuel, reprocessing and waste management. Infrastructure for other inputs like, heavy water, zirconium components, control and instrumentation etc. has been established.
- Excellent Human Resource and training infrastructure has been developed for the specialized skills needed for nuclear power.

Development of Indian Industry

- At the time of country's independence in 1947 and for several years thereafter, the industry's capability was limited to manufacturing and supply of equipment for cement and sugar industry. The Indian industry exposure, manufacturing and supply of equipment for high technology requirements was quite limited. Whereas other developed countries at that time had well established industrial infrastructure and capability to manufacture equipment for defence and aviation industry. The nuclear industry development in those countries was a spin-off of the well established industry. **The Indian industry development was initiated and achieved maturity with the development of nuclear technology.** Large efforts have been put by DAE and NPCIL to develop the Indian industry to achieve high standards in manufacturing of equipment for nuclear power technology. Currently, the Indian Industry capability in design, engineering and manufacturing of equipment is comparable to the international standards.

Multi-tier Regulatory Review, Consenting and Safety Reviews of Indian NPPs

REGULATORY CONSENTING PROCESS

Consents in the form of Authorization or License at various major stages of establishment of the facility

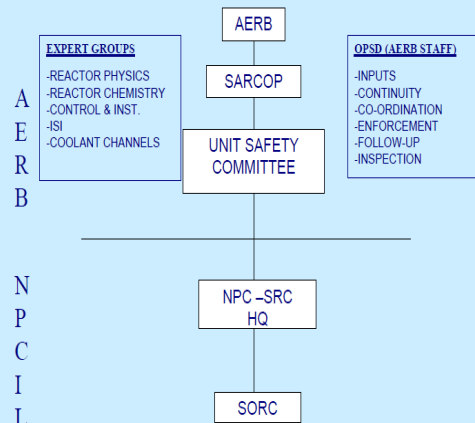
- Siting
- Construction
- Commissioning
- Operation
- Decommissioning

Consents subject to the facility's location, design and operation fulfilling the safety objectives and requirements as specified in the relevant Rules, Codes and Guides and stipulated by AERB

Statutory Provisions

- ✓ Atomic Energy Act 1962
- ✓ Radiation Protection Rules 2004
- ✓ Atomic Energy (Safe Disposal of Radioactive Wastes) Rules 1987
- ✓ Atomic Energy (Factories) Rules 1996
- ✓ Atomic Energy (Working of the Mines, Minerals and Handling of Prescribed Substances) Rules 1984
- ✓ Consents from other statutory agencies such as MOEF and, the Central and State Pollution Control Boards

Safety Review Mechanism for Operating NPPs



Initial Operating Authorization for NPPs

The authorizations for NPPs are issued for,

- Pre-commissioning tests & System Commissioning
- Criticality and Low power tests
- Medium and High power tests
- Stage wise power operation (50%,90% FP)
- Provisional authorization for 90 days at FP
- 3 Year authorization

Initial Authorization is issued after ensuring

- All the pending safety related recommendations are complied with
- Plant fulfils all the consenting conditions specified in regulatory guides,
 - Consenting Process for NPPs & Research Reactors : AERB SG G-1
 - Regulatory Consents for Nuclear and Radiation Facilities: AERB SG G-7
- Final Safety Analysis Report and Technical Specifications for Operation are duly approved by the Regulatory Body
 - Approved Emergency Preparedness Plan is in place

Renewal of Authorization and Periodic Safety Reviews

Application for renewal of Authorization (ARA)

- Utility submits an Application for Renewal of Authorization every three years
- Major elements of ARA are:
 - Safety Performance
 - Operational Performances and Problems
 - Status of implementation of safety related recommendations
 - Compliance with the regulatory inspection findings
 - Radiological status and effluent management
 - Operational Experience Feedback
 - Significant events
 - Unusual Occurrences from other NPPs (India and abroad)
 - Physical Status of Plant
 - ISI programme and findings
 - Major jobs carried out in long outages
 - Public concern in operational safety

Periodic Safety Review (PSR)

- Comprehensive periodic safety reviews (PSR), carried out every 9 years. i.e., after ARA at 3 years and 6 years
- Additional elements of PSR are as per AERB guide on 'Renewal of Authorization for Operation of Nuclear Power Plants' (AERB/SG/0-12),
 - Review of Safety Analysis
 - Equipment Qualification
 - Life Cycle Management
 - Procedures
 - Organizational / Management changes
 - Human factor
 - Emergency Preparedness
 - Environmental Impact
- Integrated review of all safety factors to provide assurance that till the next PSR, the plant can continue to operate with adequate safety margins

Need for ISI

ISI is derived from the following facts;

- a) No Engineering Structure is free from flaw
- b) Degradation of System, Structure and Components is a continuous phenomenon
- c) Rate of degradation depends on operating and environmental conditions

Scope of ISI

ISI should include the fluid boundary portions of components, piping and supports that comprise

- a) Systems containing fluid that directly transport heat from nuclear fuel
- b) Systems essential for safe shut down of reactor, of safe cooling of fuel or both, in the event of process system failure
- c) Other systems or components whose failure could put in jeopardy the integrity of system (a) or (b) or both.

Sample approach for ISI

Samples for ISI are chosen to include areas subject to most extreme conditions as mentioned below:

- a) **Most significant acceptable defects discovered at the time of manufacturing.**
- b) **Areas most subjected to corrosion or erosion.**
- c) **Areas having the severest conditions of service in terms of stresses, particularly cyclic.**
- d) **Areas most subject to creep and radiation**

Systems Included in the ISI programme in Dhruva

- Heavy water and cover gas (coolant, moderator and cover gas)
- Emergency core cooling system
- Emergency Cooling Water (ECW) system or Decay heat removal system.
- Inspection of concrete structures

Considering safety requirements of different systems considered for ISI, inspection level has been categorised as High, Medium and Low. Percentage coverage and type of inspections have been decided based on the inspection categories.

Ageing Management and ISI program of Indian NPPs

- In nuclear power plants, continuous efforts are made to ensure that plants are operated in safe, reliable and economical manner
- While utmost care is taken during design, construction and commissioning of the structures, systems & components (SSCs), continued healthiness is ensured during operation phase in accordance with the design intent through
 - the establishment of a comprehensive life management program of surveillance,
 - condition monitoring,
 - periodic In-service inspections (ISI) and maintenance,the purpose of which is to ensure that required safety margins are maintained for all important SSCs throughout plant service life.
- Ageing management program of Indian PHWR is based on effective maintenance, surveillance and In-service inspection of system, structure & components and systematic & comprehensive approach of plant life management and plant life extension program
- Indian experience has shown that some components like, Pressure Tubes, Feeders & SG tubes failed even before design life is over, whereas some could be used beyond their design life.

Ageing Management and ISI program of Indian NPPs

Major Mechanical systems

- Primary heat transport system
- Reactor components
- Moderator system
- Reactor Auxiliary
- Fueling machine & component
- Secondary cycle piping

Major electrical systems

- Auxiliary transformer
- 415 Volts, switchgear & MCC
- MG sets
- Station Batteries
- DG sets
- Class II, III Relays
- EMTR

Major Civil Structures

- Reactor Buildings, Service Building & Turbine Buildings
- Supplementary control room building
- Ventilation stack , PHT purification building
- D2O upgrading plant, CW Pump house
- NDCT , IDCT & Waste Management Plant

Major C & I systems

- Reactor protective system
- primary shut down system
- Secondary shut-down system
- ALPAS
- ECCS
- Containment system

Ageing management of the majority of the Mechanical systems are covered under the scope of In-service Inspection

Calandria and Endshields see low pressure & low temperature and as a result operate under low stresses. These are made up of SS-304 L . With this and maintenance of close chemistry, no deterioration is anticipated and hence ISI is not envisaged

ISI program of Indian NPPs

➤ **Mandatory inspection of components carried out at INTERVALS after the START-UP of a Plant**

- First ISI interval - 05 Yrs
- Subsequent ISI- 10 Yrs
- Actual ISI interval may be changed based on operational experience and results of previous ISI campaign

➤ **Evolution of ISI program for PHWR is based on:**

- AERB-SG-O2
- CAN-CSA-N285.4-94 , Canadian standard for periodic inspection of CANDU NPP components
- ASME section XI
- IAEA safety guide No.NS-G-2.6, Maintenance , surveillance and In-service Inspection in NPPs

➤ **The In-service inspection program adopted at Indian Nuclear power plant is specific to each plant and include information about:**

- Selection of system and component subject to inspection
- Categorization of the components for inspection
- Selection of inspection methods and procedure
- Selection, location and extent of inspection areas
- Interval of inspection
- Interpretation of In-service Inspection results followed by Codal analysis

ISI program of Indian NPPs

System subjected to Inspections:

- The Reactor coolant pressure boundary and other systems whose failure may result in significant release of radioactive substances
- Systems essential for the safe reactor shutdown or the safe cooling of nuclear fuel or both in the event of process system failure
- Other system and components, the failure or dislodgment of which may jeopardize the integrity of the systems in items mentioned above

Categorization of the systems/components for ISI

- The requirements of inspection, inspection areas and degree of examination are determined based on the matrix consisting of stress intensity ratio and fatigue usage factor based on the size of failure
- Based on above criteria systems/components are categorized as category A, B, C1 & C2 as per AERB-SG-O2 & CAN-CSA-N285.4-94, Canadian standard for periodic inspection of CANDU NPP components

ISI program of Indian NPPs

NDE technique used in ISI are:

- Ultrasonic Examination (Volumetric examination)
- Eddy current Examination (Volumetric examination as far as SG & HX tubes are concerned)
- Liquid Penetrant Examination (Surface examination)
- Magnetic particle Examination (Surface examination)
- Leak Testing
- Visual examination

Systems selected for ISI:

- PHT System (Including PHT relief system)
- Shutdown system
- ECCS system
- Primary shutdown system
- Secondary shutdown system
- ALPAS
- Moderator system
- Main steam lines (Inside RB), Feed water system (Inside RB) & Auxiliary feed water system (Inside & outside RB)
- Process water system
- AGS

ISI program of Indian NPPs

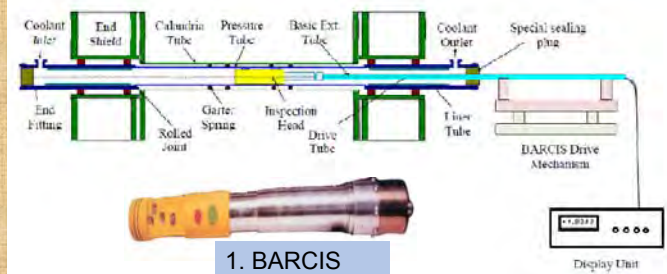
Components subjected for ISI (In case of PHWR)

- **Pressure Tubes**
 - Delayed hydride cracking
 - Irradiation enhanced deformations
 - Changes in pressure tube material propertiesThe basic philosophy of ISI of Pressure tube is to ensure the structural integrity and to demonstrate LBB
- Pressure boundary portion of safety system piping
- **PHT system Feeders**
- **Steam Generators** / Heat Exchangers
- Fueling machine components
- PCP, PCP flywheel, S/D pumps & ABFP
- PHT system boundary valves, SSS shut-off valves, MSL, FW & AFW system valves
- Piping / equipment supports

ISI program of Indian NPPs

ISI Experience : Pressure Tube Inspection

- Special manipulator are being used to detect the deterioration(s) in the pressure tubes
- BARC has developed BARCIS, which addresses all the task using an automated remote inspection system
- BARCIS is capable of examining of pressure tubes without ice-plugging of feeders
- It has eddy current probes for sensing garter spring position and orientation as well as measuring gap between the two concentric tubes
- Ultrasonic probes are used to measure the pressure tube wall thickness and for flaw detection in axial & circumferential directions
- The results of each ISI campaign are evaluated to assess the fitness for service criteria



3. GARTER SPRING REPOSITIONING TOOL

2. The Sliver Sample Scraping Tool (SSST)

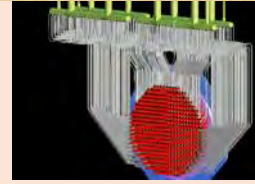
developed for obtaining in-situ scrape samples from dry pressure tube of operating PHWR for measurement of hydrogen concentration



ISI program of Indian NPPs

In-service inspection requirements for PHT feeder pipes

- The selection of outlet feeders for ISI is based on:
 - High velocity criteria
 - High stress intensity criteria
 - High survey factor criteria
 - Thinning experience of other plants
 - High seismic load contribution
- Inlet feeders on sample basis (classified on high velocity, high stress intensity & high survey factor criteria)
- Extent & Interval of ISI on feeders:
 - Thickness measurement of feeder elbows by UT thickness gauging
 - Visual, surface & volumetric examinations of dissimilar metal weld joints
 - Visual examination of readily accessible feeders and their supports
 - Interval of inspection: 1st ISI interval is 05 Years & subsequent ISI interval is 10 Years
- Feeders pipes have complex three dimensional shape, are connected to highly radioactive region of the reactor, are small in diameter
- All these features make feeder pipes highly inaccessible, inspection difficult and time consuming
- PHWR feeders pipe have been degrading through Erosion/corrosion/FAC/SCC
- In order to monitor these degradation UT thickness check of feeder 1st and 2nd elbow is being done in each ISI campaign manually



ISI program of Indian NPPs

In-service Inspection requirements of Steam Generators (SGs):

- As a minimum of 20 % of total number of tubes of each SG are subjected for Eddy-current testing in one ISI interval
- Selection of tubes from specific & random sample
- Visual, surface & volumetric examinations of channel & shell side weld joints
- SG support & fasteners examination.
- Removal of section of one tube in a deposit region for metallurgical examinations
- Interval of inspection : 1st ISI-05 Yrs, subsequent ISI -10 Yrs
- **The common deterioration in SGs tubing is thinning of tubes due to corrosion/erosion, denting, wear, fretting and pitting**
- Eddy current testing (ET) of SG tubes is being extensively carried out as a part of ISI
 - During initial period (late 1970s), the ET was carried out with single /dual frequency equipment
 - There have been considerable improvements in ET techniques. Multi-frequency ET system has been used since early 1990s. Which can enable us detect the defect under the baffle plate
 - Lately use has been made of robotic systems for remote semi-automatic eddy current testing equipment with probe pusher puller
 - This remotely controlled fully automatic device are capable of traversing the probe head parallel to the tubesheet to the next selected tube, pushing the eddy current probe into the full length of the tube and recording information as the probe is subsequently retracted
- No generic degradation in SG tubes has been observed so far in Indian PHWR from NAPS onwards. However few SG tubes (07 Nos.) found leaky due to hitting of foreign material from shell side of the tubes which possibly existed as construction debris

ISI program of Indian NPPs

In-service inspection requirements of piping weld joints:

- Visual, surface and volumetric examinations of minimum of 25 % of total weld joints in each pipe run. UT thickness check on elbows on sample basis
- Selection of weld joints for ISI :
 - The joints having high fatigue usage factor/ high stress intensity ratio
 - Terminal weld joints
 - Dissimilar weld joints
 - Intersection weld joints

In-service inspection requirements of pumps/valves

- Examinations are limited to at least one pump & one valve within each group of pumps/valves performing similar function in the system
- Extent of examination for pump/valves includes: visual, surface & volumetric examinations of casing weld joints (if any) , Visual examination of internal fluid boundary of pump/valve casing and visual & surface examinations of pump/valve fasteners. Volumetric examination of PCP flywheel
- Interval of inspection: 1st ISI interval is 05 Years & subsequent ISI interval is 10 Years

In-Service Inspections of Research Reactors

- In a nuclear reactor, the major concern is accidental release of radioactive products.
- To avoid accidental release of radioactive product into atmosphere defiance in depth philosophy is adapted in reactor design.
- Every reactor follows a surveillance programme to ensure that Structures, Systems and Components (SSCs) are healthy and performing there intended function.
- In-service Inspection (ISI) is necessary components of a surveillance programme.
- ISIs are intended to ensure that an unacceptable degradation in component quality is not occurring and probability of failure remains acceptably low for the life of the plant.
- In-Service Inspections (ISI) are planned inspections to be performed at a definite intervals during the life time of a reactor.

Conclusions

- India's Nuclear Power development is based on indigenous three stage program for peaceful uses of atomic energy
- A robust multi-tier safety mechanism is in force. The effort of Utility and Regulatory in maintenance and In-Service Inspection program resulted lesser downtime of plants
- During Periodic Safety Review conducted by Regulatory Body, the Ageing Management program of Indian PHWR NPPs is monitored
- Based on Ageing Management Program, components like Pressure Tubes & PHT system Feeders have shown deteriorations before their design life is over
- For Pressure Tubes (Zircaloy-2 type) extensive / enhanced ISI and replacement is envisaged
- For PHT system feeders extensive / enhanced ISI program is envisaged

Thank You



ASME Section XI - Rules for Inservice Inspection of Nuclear Power Plants

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*Co-Vice Chair of ASME Section XI
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**Mumbai, India
January 4, 2019**

Objectives

- ◆ Discuss the history of the American Society of Mechanical Engineers (ASME) Code
- ◆ List the available Codes for nuclear applications
- ◆ Describe the Code consensus process
- ◆ Discuss the format, structure and content of ASME Section XI
- ◆ Review examination requirements for various components
- ◆ Describe the new Reliability and Integrity Management (RIM) process

History of ASME Codes and Standards

History of ASME

- ◆ Founded as a technical professional society in 1880 as the American Society of Mechanical Engineers
- ◆ Not-for-profit, unaffiliated with government
- ◆ Engineers convened to foster engineering innovation, education and research
- ◆ First standard issued in 1884: “*Conduct of Trials of Steam Boilers*” (boiler efficiency testing)

History of ASME

- ◆ Circa 1900: Hundreds of boiler explosions each year in U.S.A.
- ◆ States developed regulations, but no standards for construction
- ◆ Manufacturers, users, insurers and regulators approached ASME in 1911
- ◆ ASME served as the neutral convener of a committee to write construction rules

Boiler and Pressure Vessel Code Milestones

- ◆ **1914: First Code edition was approved and published**
- ◆ **Initially only covered Power Boilers, less than 200 pages**
- ◆ **1916: First 'S' Certificate and Stamp issued**
- ◆ **1919: National Board of Boiler and Pressure Vessel Inspectors founded; uniform qualification of third party inspectors**

Boiler and Pressure Vessel Code Milestones

Expansion of BPV scope:

1920's

- **Section III, Locomotive Boilers**
- **Section IV, Heating Boilers**
- **Section V, Miniature Boilers**
- **Section VIII, Unfired Pressure Vessels**

1963

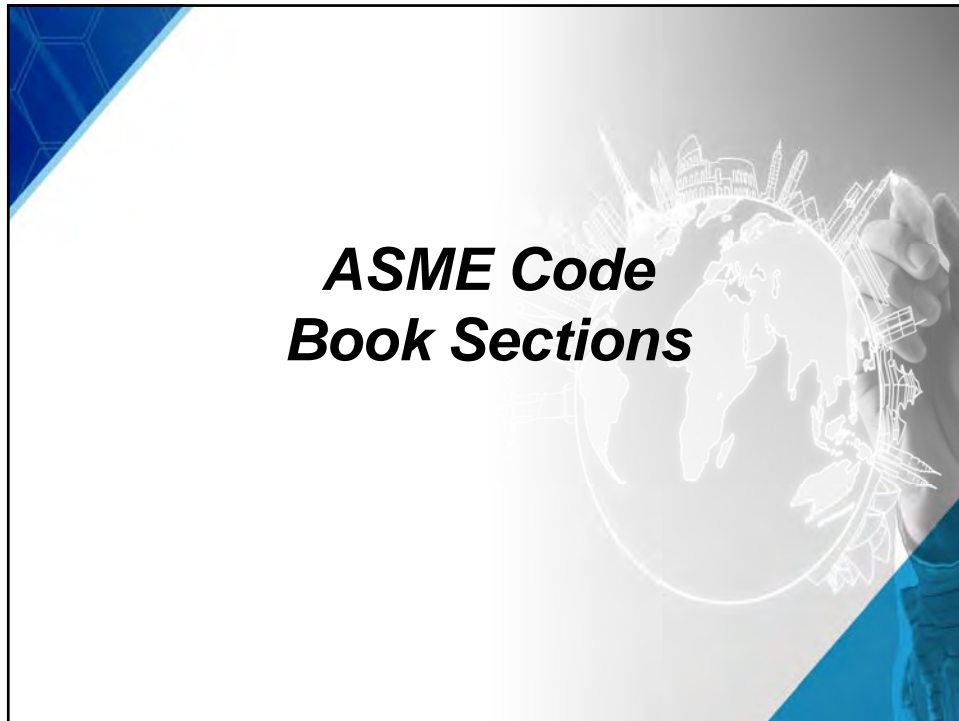
- **Section III Code for Nuclear Vessels issued**

1967

- **Section VIII, Division 2 issued**

1970

- **Section XI issued**



Nuclear Codes and Standards Committees

| | |
|--|---|
| BPV Committee on Construction of Nuclear Facility Components (III) | BPV Committee on Nuclear In-service Inspection (XI) |
| Standards Committee on Nuclear Quality Assurance (NQA) | Standards Committee on Operation and Maintenance of Nuclear Power Plants (O&M) |
| Standards Committee on Qualification of Mechanical Equipment Used in Nuclear Facilities (QME) | Standards Committee on Cranes for Nuclear Facilities (CNF) |
| ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM) | Standards Committee on Nuclear Air and Gas Treatment (CONAGT) |

Boiler and Pressure Vessel Code

| | |
|-------------------|---|
| Section VIII | Pressure Vessels |
| Section IX | Welding and Brazing Qualifications |
| Section X | Fiber-Reinforced Plastic Pressure Vessels |
| Section XI | Rules for Inservice Inspection of Nuclear Power Plant Components |
| Section XII | Rules for the Construction and Continued Service of Transport Tanks |
| Section XIII | Over Pressure Protection |



Codes and Standards Consensus Process

- ◆ Procedures accredited by the American National Standards Institute (ANSI)
- ◆ Decisions are reached through consensus among those affected
- ◆ Participation is open to all affected interests: no membership fees or requirement to be ASME society member
- ◆ Balance is maintained among competing interests
 - Designers, Manufacturers, Users, Regulators, General Interests, etc.
- ◆ The process is transparent - information on the process and progress is directly available

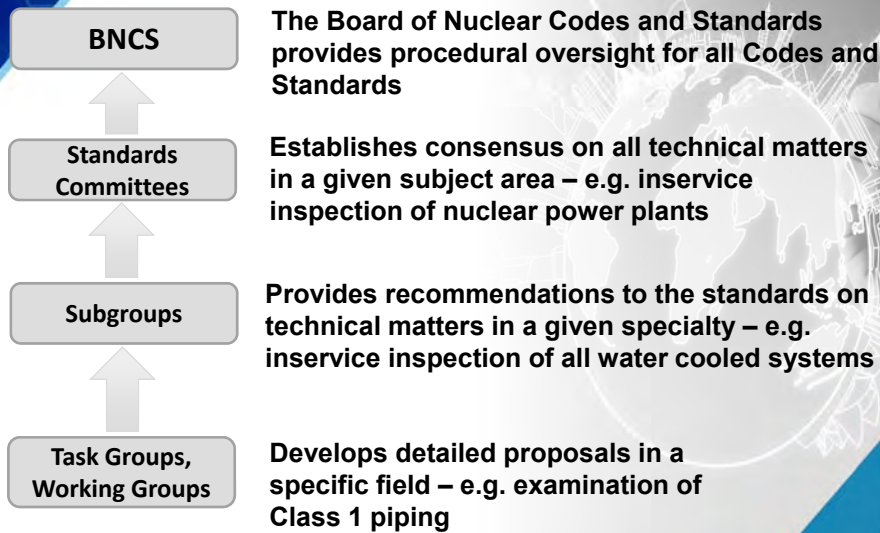
Codes and Standards Consensus Process

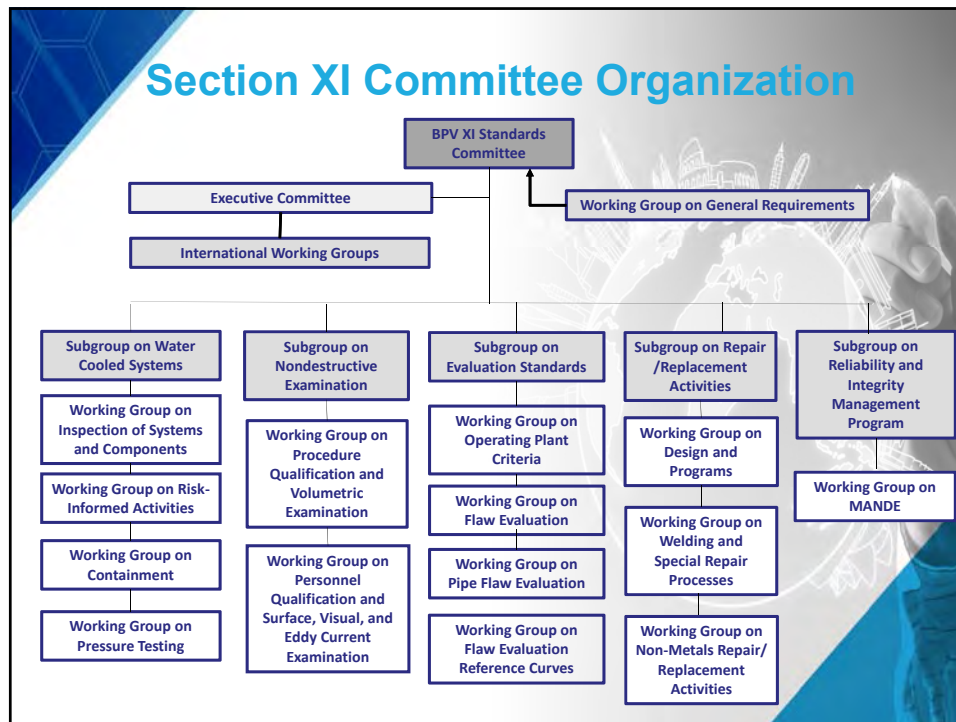
- ◆ The process is robust in that it assures that all views will be considered and that appeals are possible
- ◆ The process is flexible, allowing the use of different methodologies to meet the needs of different technology and product sectors
- ◆ The process is timely; addressing topics that currently impact the industry
- ◆ Standards activities are coordinated, avoiding overlap or conflict
- ◆ The process ensures “Defense-in-Depth”

The Consensus Development Process

- ◆ **Development of standards action**
 - Initiation of Action Item
 - Project Team
 - Review and Comment
 - Subcommittee Vote (sub-tier)
- ◆ **Recorded vote**
 - Project Team considers comments and prepares responses
- ◆ **Public review**
- ◆ **Supervisory Board approval**
- ◆ **Appeals**
- ◆ **ANSI approval**

The Consensus Development Process





BPV Code Week Meetings

- ◆ **Boiler and Pressure Vessel (BPV) Section XI Standards Committee and subordinate groups meet during BPV Code Week. Held four (4) times a year during the months of February, May, August and November**
- ◆ **Location:** BPV Code week is held throughout the USA and sometimes Canada. The location is chosen at least a year in advance and is posted in each Committee web page
- ◆ **Openness of Meetings:** BPV Code meetings are open to the public unless otherwise stated. Everyone can attend and participate in the discussions as well as request interpretations/code cases and/or propose revisions to the Standards.
- ◆ **Sequence of BPV Code Week Meetings:** During Code Weeks, typically working groups and task groups meet earlier in the week with the Standards Committee meetings toward the end of the week, followed by the TOMC meeting. Schedule is made available ahead of time (ASME Hotel Designated Office).

Code Committee Activities

◆ Code Revisions

- Could result from a new proposal, incorporation of a Code Case or integration of an Interpretation
- Request for Code revision could be submitted by Committee member or an external source
- Requests for Code revisions result in new action items assigned to technical committees with an individual volunteer Project Manager
- Code revisions published in new Edition of the Code, which comes out every 2 years

Code Committee Activities

◆ Code Cases

- Code Cases represent alternatives or additions to existing Code requirements.
- Code Cases are written as a Question and Reply, and are usually intended to be incorporated into the Code at a later date.
- Code Cases prescribe mandatory requirements in the same sense as the text of the Code.
- Not all regulators, jurisdictions, or Owners automatically accept Code Cases.
- The most common applications for Code Cases are as follows:
 - to permit early implementation of an approved Code revision based on an urgent need
 - to permit use of a new material for Code construction
 - to gain experience with new materials or alternative requirements prior to incorporation directly into the Code

Code Committee Activities – Sample Code Case

ASME BPVC.CC.NC-2017

CASE
N-798

Approval Date: December 20, 2010

Code Cases will remain available for use until annulled by the applicable Standards Committee.

Case N-798
Alternative Pressure Testing Requirements for Class 1 Piping Between the First and Second Vent, Drain, and Test Isolation Devices Section XI, Division 1

Inquiry: What alternative to the extended pressure test boundaries of IWB-5222(b) may be used for the portion of Class 1, vent, drain, and test piping between the first and second isolation devices that normally remain closed during plant operation?

Reply: It is the opinion of the Committee that, for portions of Class 1 vent, drain, and test piping between the first and second isolation devices that normally remain closed during plant operation, only the boundaries of IWB-5222(a) shall apply.

Code Committee Activities

- ◆ **Code Inquiries and Interpretations**
 - Inquirer asks Code Committee its interpretation of published Code criteria
 - Requirements Inquiry
 - Intent Inquiry (Code revision required)
 - Inquiry addressed by applicable Working Group, Subgroup and then Inquiry Panel
 - Inquirer is encouraged to participate in Committee discussions
 - Result of the process is an Interpretation issued by the Code
 - See ASME web site to search for Interpretations:
<https://cstools.asme.org/interpretation/searchinterpretation.cfm>

Code Committee Activities – Sample Inquiries

Question: Is it a requirement of IWA-4540 that a pressure test in accordance with IWA-5000 is required for a repair/replacement activity performed by welding or brazing on a Class 3 component that does not support any of the functions identified in IWD-1210?

Reply: No.

Question: Is it the intent that Appendix E be applied to nozzles in the reactor pressure vessel bellline region?

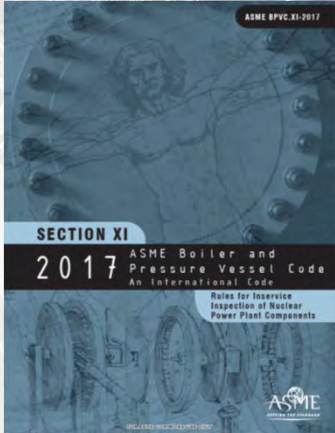
Reply: No.

ASME Section XI Format, Structure and Content

ASME BPVC Section XI

Rules for Inservice Inspection of Nuclear Power Plant Components


- ◆ **Division 1**
 - Rules for Inspection and Testing of Components of **Light-Water-Cooled** Plants
- ◆ **Division 2**
 - Rules for Inspection and Testing of Components of **Gas-Cooled** Plants
 - Currently being revised to Reliability and Integrity Management (RIM) process
- ◆ **Division 3**
 - Rules for Inspection and Testing of Components of **Liquid-Metal-Cooled** Plants



The image shows the cover of the ASME BPVC Section XI 2017 code book. The cover features a blue and white color scheme with technical drawings of nuclear power plant components. The text on the cover includes 'ASME BPVC XI-2017', 'SECTION XI', '2017', 'ASME Boiler and Pressure Vessel Code An International Code', and 'Rules for Inservice Inspection of Nuclear Power Plant Components'. The ASME logo is visible in the bottom right corner.

Section XI, Division 1

- ◆ **Subsections**
 - **IWA** – General Requirements
 - **IWB** – Requirements for Class 1 Components of Light-Water Cooled Plants
 - **IWC** – Requirements for Class 2 Components of Light-Water Cooled Plants
 - **IWD** – Requirements for Class 3 Components of Light-Water Cooled Plants
 - **IWE** – Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants
 - **IWF** – Requirements for Class 1, 2, 3, and MC Components Supports of Light-Water Cooled Plants
 - **IWL** – Requirements for Class CC Concrete Components of Light-Water-Cooled Plants



The background graphic shows a hand holding a globe, with technical drawings of nuclear power plant components overlaid on the globe. The drawings include various parts of a reactor, piping, and structural elements.

Section XI, Division 1

◆ Mandatory Appendices

- **Appendix I** – Ultrasonic Examinations
- **Appendix II** – Owner's Report for Inservice Inspections
- **Appendix III** – Ultrasonic Examination of Vessel and Piping Welds
- **Appendix IV** – Eddy Current Examination
- **Appendix VI** – Qualification of Personnel for Visual Examination
- **Appendix VII** – Qualification of Nondestructive Examination Personnel for Ultrasonic Examination
- **Appendix VIII** – Performance Demonstration for Ultrasonic Examination Systems
- **Appendix IX** – Mechanical Clamping Devices for Class 2 and 3 Piping Pressure Boundary
- **Appendix X** – Standard Units for Use in Equations
- **Appendix XI** – Repair/Replacement Activities for Class 3 Polyethylene Piping

Section XI, Division 1

◆ Non-mandatory Appendices

- **Appendix A** – Analysis of Flaws
- **Appendix C** – Evaluation of Flaws in Piping
- **Appendix D** – Conditioning of Classes 1 and 2 Piping Welds Which Require Examination
- **Appendix E** – Evaluation of Unanticipated Operating Events
- **Appendix G** – Fracture Toughness Criteria for Protection Against Failure
- **Appendix H** – Evaluation Procedures for Flaws in Piping Based on Use of a Failure Assessment Diagram
- **Appendix J** – Guide to Plant Maintenance Activities and Section XI Repair/Replacement Activities
- **Appendix K** – Assessment of Reactor Vessels with Low Upper Shelf Charpy Impact Energy Levels
- **Appendix L** – Operating Plant Fatigue Assessment
- **Appendix M** – Applying Mathematical Modeling to Ultrasonic Examination of Pressure-Retaining Components

Section XI, Division 1

◆ Non-mandatory Appendices (cont.)

- **Appendix N** – Written Practice Development for Qualification and Certification of NDE Personnel
- **Appendix O** – Analytical Evaluation of Flaws in PWR Reactor Vessel Head Penetration Nozzles
- **Appendix P** – Guidance for the Use of U.S. Customary and SI Units in the ASME Boiler and Pressure Vessel Code
- **Appendix Q** – Weld Overlay Repair of Classes 1, 2, and 3 Austenitic Stainless Steel Piping Weldments
- **Appendix R** – Risk-Informed Inspection Requirements for Piping
- **Appendix S** – Evaluating Coverage for Section XI Nondestructive Examination
- **Appendix T** – Reporting of Contracted Repair/Replacement Activities
- **Appendix U** – Analytical Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Piping and Class 2 or 3 Vessels and Tanks
- **Appendix W** – Mechanical Clamping Devices for Class 2 and 3 Piping Pressure Boundary

Structure of Section XI, Division 1 Subsections

◆ Articles

Each Subsection contains Articles with a prefix using the following designators such as IWA, IWB, IWC, or IWD, etc.

- **IWX-1000: Scope and Responsibility** – Includes exemptions from NDE, VT-1 and VT-3 visual examinations as applicable
- **IWX-2000: Examination and Inspection** – Includes Preservice and Inservice Examination requirements in IWB, IWC, IWD, IWE, IWF, and IWL
- **IWX-3000: Acceptance Standards** – Addresses flaws and relevant conditions
- **IWX-4000: Repair/Replacement Activities** – Class 1, 2, 3, MC, and CC items and their supports
- **IWX-5000: System Pressure Tests** – System leakage tests and hydrostatic tests
- **IWX-6000: Records and Reports** – Forms NIS-1, NIS-2, and PSI and ISI Summary Reports
- **IWA-9000: Glossary**

“X” in “IWX” above = A, B, C, D, E, etc.

Content of Subsection IWA: General Requirements

- ◆ **IWA-1000: Scope and Responsibilities**
 - Classifications
 - Owner Responsibilities
- ◆ **IWA-2000: Examination and Inspection**
 - Examination Methods
 - Qualification Requirements
 - Inspection Programs and Plans
 - Inspection Intervals
- ◆ **IWA-3000: Standards for Examination Evaluations**
 - Characterization of Flaws
 - Sizing of Flaws

Content of Subsection IWA: General Requirements

- ◆ **IWA-4000: Repair/Replacement Activities**
 - Applicability
 - Responsibilities
 - Design
 - Welding, Brazing, Metal Removal, Fabrication and Installation Requirements
 - Examination and Testing
 - Alternative Welding Methods (e.g., Temper Bead Welding, Underwater Welding, Heat Exchanger Tube Plugging and Sleeving)
- ◆ **IWA-5000: Pressure Testing**
 - Types of Pressure Tests
 - Pressure Testing of Buried Piping
- ◆ **IWA-6000: Records and Reports**
 - Requirements
 - Types of Reports
- ◆ **IWA-9000: Glossary**

Content of Subsections IWB, IWC, IWD, IWE, IWF and IWL

◆ IWX-1000: Scope and Responsibilities

- Scope
- Exemption Criteria

◆ IWX-2000: Examination and Inspection

- Preservice Examinations
- Inspection Periods
- Inservice Examination Requirements
- Examination Figures
- Successive Examinations
- Additional Examinations

◆ IWX-3000: Acceptance Standards

- Evaluation of Examination Results
- Supplemental Examinations
- Acceptance Standards
- Analytical Evaluation

X" in "IWX" above = B, C, D, E, etc.

Content of Subsections IWB, IWC, IWD, IWE, IWF and IWL

◆ IWX-4000: Repair/Replacement Activities

- Repair/Replacement requirements for all Subsections were combined and moved to IWA-4000 in 1991 Addenda of ASME Section XI

◆ IWX-5000: System Pressure Tests

- System Leakage Tests
- Hydrostatic Tests
- Temperature and Pressure Requirements

X" in "IWX" above = B, C, D, E, etc.

The slide features a background with a hand holding a globe, overlaid with technical drawings of mechanical parts. The title is centered in a bold, italicized font.

ASME Section XI Examination Criteria

The slide features a background with a hand holding a globe, overlaid with technical drawings of mechanical parts. The title is in a blue font.

Examination Requirements

- ◆ **Class 1 Components (Table IWB-2500-1)**
 - Examination Category B-A: Pressure Retaining Welds in Reactor Vessel
 - Examination Category B-B: Pressure Retaining Welds in Vessels Other Than Reactor Vessel
 - Examination Category B-D: Full Penetration Welded Nozzles in Vessels
 - Examination Category B-F: Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles
 - Examination Category B-G-1: Pressure Retaining Bolting, Greater Than 2 in. Diameter
 - Examination Category B-G-2: Pressure Retaining Bolting, 2 in. and Less in Diameter

Examination Requirements

- ◆ **Class 1 Components (Table IWB-2500-1)**
 - Examination Category B-J: Pressure Retaining Welds in Piping
 - Examination Category B-K: Welded Attachments for Vessels, Piping, Pumps and Valves
 - Examination Category B-L-1: Pressure Retaining Welds in Pump Casings (deleted in 2008 Addenda)
 - Examination Category B-L-2: Pump Casings
 - Examination Category B-M-1: Pressure Retaining Welds in Valve Bodies (deleted in 2008 Addenda)
 - Examination Category B-M-2: Valve Bodies
 - Examination Category B-N-1: Interior of Reactor Vessel

Examination Requirements

- ◆ **Class 1 Components (Table IWB-2500-1)**
 - Examination Category B-N-2: Welded Core Support Structures and Interior Attachments to Reactor Vessels
 - Examination Category B-N-3: Removable Core Support Structures
 - Examination Category B-O: Pressure Retaining Welds in Control Rod Housings
 - Examination Category B-P: All Pressure Retaining Components
 - Examination Category B-Q: Steam Generator Tubing

Examination Requirements

- ◆ **Class 2 Components (Table IWC-2500-1)**
 - Examination Category C-A: Pressure Retaining Welds in Pressure Vessels
 - Examination Category C-B: Pressure Retaining Nozzle Welds in Vessels
 - Examination Category C-C: Welded Attachments for Vessels, Piping, Pumps and Valves
 - Examination Category C-D: Pressure Retaining Bolting Greater Than 2 in. Diameter
 - Examination Category C-F-1: Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping
 - Examination Category C-F-2: Pressure Retaining Welds in Carbon or Low Alloy Steel Piping

Examination Requirements

- ◆ **Class 2 Components (Table IWC-2500-1)**
 - Examination Category C-G: Pressure Retaining Welds in Pumps and Valves (deleted in 2008 Addenda)
 - Examination Category C-H: All Pressure Retaining Components
- ◆ **Class 3 Components (Table IWD-2500-1)**
 - Examination Category D-A: Welded Attachments for Vessels, Piping, Pumps, and Valves
 - Examination Category D-B: All Pressure Retaining Components

Examination Requirements

- ◆ **Class MC Components (Table IWE-2500-1)**
 - Examination Category E-A: Containment Surfaces
 - Examination Category E-C: Containment Surfaces Requiring Augmented Examination
 - Examination Category E-G: Pressure Retaining Bolting
- ◆ **Class 1, 2, 3 and MC Supports (Table IWF-2500-1)**
 - Examination Category F-A: Supports
- ◆ **Class CC Components (Table IWL-2500-1)**
 - Examination Category L-A: Concrete
 - Examination Category L-B: Unbonded Post-Tensioning System

Example Examination Requirements: Welds in Class 1 Reactor Vessel

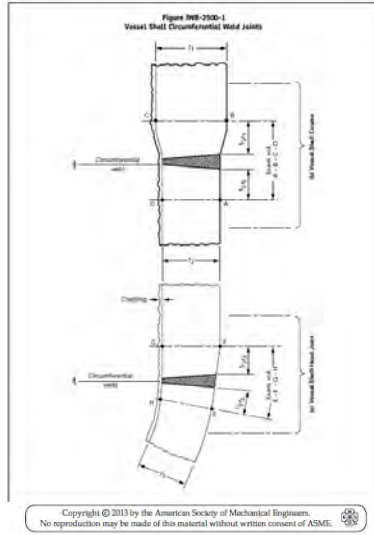
Table IWB-2500-1 (B-A)
Examination Category B-A, Pressure Retaining Welds in Reactor Vessel

| Item No. | Parts Examined | Examination Requirements/ Figure No. | Examination Method | Acceptance Standard | Extent and Frequency of Examination | | Deferral of Examination to End of Interval |
|----------|-------------------------|---|------------------------|---------------------|---|---------------------------------|--|
| | | | | | First Inspection Interval | Successive Inspection Intervals | |
| BI.10 | Shell welds | | | | | | |
| BI.11 | Circumferential | IWB-2500-1 | Volumetric | IWB-3510 | All welds [Note (2)] | Same as for first interval | Permissible |
| BI.12 | Longitudinal | IWB-2500-2 | | | | | |
| BI.20 | Head welds | | | | | | |
| BI.21 | Circumferential | IWB-2500-3 | Volumetric | IWB-3510 | Accessible length of all welds [Note (2)] | Same as for first interval | Permissible |
| BI.22 | Meridional | | | | | | |
| BI.30 | Shell-to-flange weld | IWB-2500-4 | Volumetric | IWB-3510 | Weld [Note (2)] | Same as for first interval | Permissible [Note (3)], [Note (5)] |
| BI.40 | Head-to-flange weld | IWB-2500-5 | Volumetric and surface | IWB-3510 | Weld [Note (2)] | Same as for first interval | Permissible [Note (4)], [Note (5)] |
| BI.50 | Repair welds [Note (1)] | IWB-2500-1 and IWB-2500-2 | Volumetric | IWB-3510 | All weld repair areas | Same as for first interval | Permissible |
| BI.51 | Boltline region | | | | | | |

NOTES:

- (1) Material (base metal) weld repairs where repair depth exceeds 10% nominal of the vessel wall. If the location of the repair is not positively and accurately known, then the individual shell plate, forging, or shell course containing the repair shall be included.
- (2) Includes essentially 100% of the weld length.
- (3) The shell-to-flange weld examination may be performed during the first and third periods, in which case 50% of the shell-to-flange weld shall be examined by the end of the first period, and the remainder by the end of third period. During the first period, the examination need only be performed from the flange face, provided this same portion is examined from the shell during the third period.
- (4) During the first and second periods, the examination may be performed from the flange face, provided these same portions are examined from the head during the third period.
- (5) Deferral in the first inspection interval is not permitted. Deferral in successive inspection intervals is permitted provided that
 - (a) no welded repair/replacement activities have been performed either on the shell-to-flange weld or head-to-flange weld; and
 - (b) neither the shell-to-flange weld nor the head-to-flange weld contains identified flaws or relevant conditions that require successive inspections in accordance with IWB-2420(b).

Example Examination Requirements: Welds in Class 1 Reactor Vessel



Example Examination Requirements: Class 1 Bolting Less Than 2 in.

Table IWB-2500-1 (B-G-2)
Examination Category B-G-2, Pressure Retaining Bolting, 2 in. (50 mm) and Less in Diameter

| Item No. | Parts Examined [Note (1)] | Examination Requirements/ Figure No. | Examination Method | Acceptance Standard | Extent and Frequency of Examination | | Deferral of Examination to End of Interval |
|-------------------------|---------------------------|---|--------------------|---------------------|---------------------------------------|---------------------------------|--|
| | | | | | First Inspection Interval | Successive Inspection Intervals | |
| Reactor Vessel | | | | | | | |
| B7.10 | Bolts, Studs, and Nuts | Surface | Visual, VT-1 | IWB-3517 | All bolts, studs, and nuts [Note (2)] | Same as for 1st interval | Not permissible |
| Pressurizer | | | | | | | |
| B7.20 | Bolts, Studs, and Nuts | Surface | Visual, VT-1 | IWB-3517 | All bolts, studs, and nuts [Note (2)] | Same as for 1st interval | Not permissible |
| Steam Generators | | | | | | | |
| B7.30 | Bolts, Studs, and Nuts | Surface | Visual, VT-1 | IWB-3517 | All bolts, studs, and nuts [Note (2)] | Same as for 1st interval | Not permissible |
| Heat Exchangers | | | | | | | |
| B7.40 | Bolts, Studs, and Nuts | Surface | Visual, VT-1 | IWB-3517 | All bolts, studs, and nuts [Note (2)] | Same as for 1st interval | Not permissible |
| Piping | | | | | | | |
| B7.50 | Bolts, Studs, and Nuts | Surface | Visual, VT-1 | IWB-3517 | All bolts, studs, and nuts [Note (3)] | Same as for 1st interval | Not permissible |
| Pumps | | | | | | | |
| B7.60 | Bolts, Studs, and Nuts | Surface | Visual, VT-1 | IWB-3517 | All bolts, studs, and nuts [Note (2)] | Same as for 1st interval | Not permissible |
| Valves | | | | | | | |
| B7.70 | Bolts, Studs, and Nuts | Surface | Visual, VT-1 | IWB-3517 | All bolts, studs, and nuts [Note (2)] | Same as for 1st interval | Not permissible |

NOTES:

- (1) Bolting is required to be examined only when a connection is disassembled or bolting is removed.
- (2) For components other than piping, examination of bolting is required only when the component is examined under Examination Category B-A, B-B, B-L-2, or B-M-2. Examination of bolted connection is required only once during the interval.
- (3) The examination of flange bolting in piping systems may be limited to one bolted connection among a group of bolted connections that are similar in design, size, function, and service. Examination is required only when a flange is disassembled. Examination of a bolted connection is required only once during the interval.

Example Examination Requirements: Class 1 Piping Welds

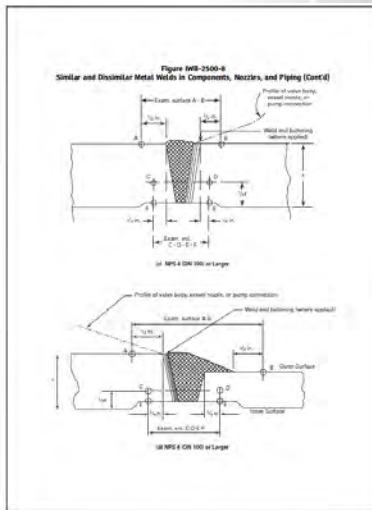
Table IWB-2500-1 (B-J)
Examination Category B-J, Pressure Retaining Welds in Piping

| Item No. | Parts Examined | Examination Requirements/ Figure No. | Examination Method | Acceptance Standard | Extent and Frequency of Examination | | Deferral of Examination to End of Interval |
|----------|---|--|------------------------|--|--|--|--|
| | | | | | First Inspection Interval | Successive Inspection Intervals [Note (1)] | |
| B9.10 | NPS 4 or larger (DN 100) | IWB-2500-8 | Surface and volumetric | IWB-3514 | Welds [Note (2), Note (3)], [Note (4)], [Note (5)], [Note (6)] | Same as for first interval | Not permissible |
| B9.11 | Circumferential welds | | | | | | |
| B9.20 | Less than NPS 4 (DN 100) | | Surface | IWB-3514 | Welds [Note (2), Note (3)], [Note (4)] | Same as for first interval | Not permissible |
| B9.21 | Circumferential welds other than PWR high pressure safety injection systems | | | | | | |
| B9.22 | Circumferential welds of PWR high pressure safety injection systems | Volumetric | | Welds [Note (3), Note (5)], [Note (6)], [Note (7)] | | | |
| B9.30 | Branch pipe connection welds | | | | | | |
| B9.31 | NPS 4 or larger (DN 100) | IWB-2500-9, IWB-2500-10, and IWB-2500-11 | Surface and volumetric | IWB-3514 | Welds [Note (2), Note (3)], [Note (4)], [Note (5)], [Note (6)] | Same as for first interval | Not permissible |
| B9.32 | Less than NPS 4 (DN 100) | | Surface | | | | |
| B9.40 | Socket welds | IWB-2500-8 | Surface | IWB-3514 | Welds [Note (2), Note (3)] | Same as for first interval | Not permissible |

NOTES:
 (1) The initially selected welds are to be examined in the same sequence during successive inspection intervals, to the extent practical.
 (2) Examinations shall include the following:
 (a) All terminal ends in each pipe or branch run connected to vessels.
 (b) All terminal ends and joints in each pipe or branch run connected to other components where the stress levels exceed either of the following limits under loads associated with specific seismic events and operational conditions:
 (i) primary plus secondary stress intensity range of $2.4 S_m$ for ferritic steel and austenitic steel
 (ii) cumulative usage factor U of 0.4
 (c) All dissimilar metal welds not covered under Category B-F.
 (d) Additional piping welds so that the total number of circumferential butt welds (or branch connection or socket welds) selected for examination equals 25% of the circumferential butt welds (or branch connection or socket welds) in the reactor coolant piping system. This total does not include welds exempted by IWB-1220 or welds in Item No. B9.22. These additional welds may be located as follows:
 (1) For PWR plants
 (a) one hot-leg and one cold-leg in one reactor coolant piping loop
 (b) one branch, representative of an essentially symmetric piping configuration among each group of branch runs that are connected to reactor coolant loops and that perform similar system functions
 (c) each piping and branch run exclusive of the categories of loop and runs that are part of system piping of (a) and (b) above.

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Example Examination Requirements: Class 1 Piping Welds



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Example Examination Requirements: Class 1 Welded Attachments

Table IWB-2500-1 (B-K)
Examination Category B-K, Welded Attachments for Vessels, Piping, Pumps, and Valves

| Item No. | Parts Examined [Note (1)] | Examination Requirements/ Figure No. | Examination Method | Acceptance Standard | Extent [Note (2)], [Note (3)] and Frequency [Note (4)] of Examination | | Deferral of Examination to End of Interval |
|----------|---------------------------|---|--------------------|---------------------|---|---------------------------------|--|
| | | | | | First Inspection Interval | Successive Inspection Intervals | |
| B10.10 | Pressure Vessels | | | | | | |
| | Welded attachments | IWB-2500-13, IWB-2500-14, and IWB-2500-15 | Surface [Note (7)] | IWB-3516 | Each welded attachment and each identified occurrence [Note (4)] | Same as for first interval | Not permissible |
| B10.20 | Piping | | | | | | |
| | Welded attachments | IWB-2500-13, IWB-2500-14, and IWB-2500-15 | Surface | IWB-3516 | Each welded attachment and each identified occurrence [Note (5)] | Same as for first interval | Not permissible |
| B10.30 | Pumps | | | | | | |
| | Welded attachments | IWB-2500-13, IWB-2500-14, and IWB-2500-15 | Surface | IWB-3516 | Each welded attachment and each identified occurrence [Note (5)] | Same as for first interval | Not permissible |
| B10.40 | Valves | | | | | | |
| | Welded attachments | IWB-2500-13, IWB-2500-14, and IWB-2500-15 | Surface | IWB-3516 | Each welded attachment and each identified occurrence [Note (5)] | Same as for first interval | Not permissible |

NOTES:

(1) Weld buildup on nozzles that is in compression under normal conditions and provides only component support is excluded from examination. Examination is limited to those welded attachments that meet the following conditions:
 (a) the attachment is on the outside surface of the pressure retaining component;
 (b) the attachment provides component support as defined in NF-1110;
 (c) the attachment weld joins the attachment either directly to the surface of the component or to an integrally cast or forged attachment to the component, and
 (d) the attachment weld is full penetration, fillet, or partial penetration, either continuous or intermittent.

(2) The extent of the examination includes essentially 100% of the length of the attachment weld at each attachment subject to examination, except that, for the configuration shown in Figure IWB-2500-15, examination of surface areas may be limited to the portions of these areas that are accessible without removal of support members.

(3) Selected samples of welded attachments shall be examined each inspection interval.

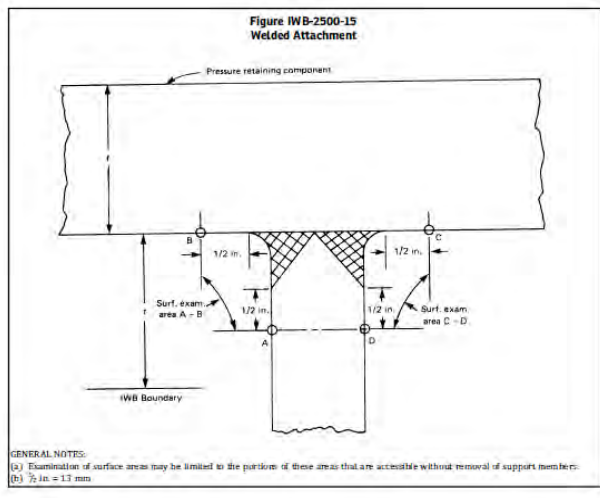
(4) For multiple vessels of similar design, function and service, only one welded attachment of only one of the multiple vessels shall be selected for examination. For single vessels, only one welded attachment shall be selected for examination. The attachment selected for examination on one of the multiple vessels or the single vessel, as applicable, shall be an attachment under continuous load during normal system operation, or an attachment subject to a potential intermittent load (seismic, water hammer, etc.) during normal system operation if an attachment under continuous load does not exist.

(5) For piping, pumps, and valves, a sample of 10% of the welded attachments associated with the component supports selected for examination under IWB-2510 shall be examined.

(6) Examination is required whenever component support member deformation, e.g., broken, bent or pulled out parts, is identified during operation, relieving, maintenance, examination, or testing.

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Example Examination Requirements: Class 1 Welded Attachments



Example Examination Requirements: Class 1 Pump Casings and Valve Bodies

Table IWB-2500-1 (B-L-2, B-M-2)
Examination Categories B-L-2, Pump Casings; B-M-2, Valve Bodies

| Item No. | Parts Examined | Examination Requirements/ Figure No. | Examination Method | Acceptance Standard | Extent and Frequency of Examination | | Deferral of Examination to End of Interval |
|---------------|--|---|--------------------|---------------------|-------------------------------------|---------------------------------|--|
| | | | | | First Inspection Interval | Successive Inspection Intervals | |
| Pumps | | | | | | | |
| B12.20 | Pump casing (B-L-2) | Internal surfaces | Visual, VT-3 | IWB-3519 | Internal surface [Note (1)] | Same as for first interval | See [Note (2)] |
| Valves | | | | | | | |
| B12.50 | Valve body, exceeding NPS 4 (DN 100) (B-M-2) | Internal surfaces | Visual, VT-3 | IWB-3519 | Internal surface [Note (3)] | Same as for first interval | See [Note (2)] |

NOTES:
 (1) Examinations are limited to at least one pump in each group of pumps performing similar functions in the system, e.g., recirculating coolant pumps.
 (2) Examination is required only when a pump or valve is disassembled for maintenance, or repair. Examination of the internal pressure boundary shall include the internal pressure retaining surfaces made accessible for examination by disassembly. If a partial examination is performed and a subsequent disassembly of that pump or valve allows a more extensive examination, an examination shall be performed during the subsequent disassembly. A complete examination is required only once during the interval.
 (3) Examinations are limited to at least one valve within each group of valves that are of the same size, constructional design (such as globe, gate, or check valves), and manufacturing method, and that perform similar functions in the system (such as containment isolation and system overpressure protection).

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Example Examination Requirements: Class 1 Periodic Pressure Testing

Table IWB-2500-1 (B-P)
Examination Category B-P, All Pressure Retaining Components

| Item No. | Parts Examined | Examination Requirements/ Figure No. | Examination Method [Note (1)] | Acceptance Standard | Extent and Frequency of Examination | | Deferral of Examination to End of Interval |
|----------|---|---|----------------------------------|---------------------|-------------------------------------|---------------------------------|--|
| | | | | | First Inspection Interval | Successive Inspection Intervals | |
| B15.10 | Pressure retaining components [IWB-5222(a)] | System leakage test [IWB-5220] | Visual, VT-2 | IWB-3522 | Each refueling outage [Note (2)] | Same as for first interval | Not permissible |
| B15.20 | Pressure retaining components [IWB-5222(b)] | System leakage test [IWB-5220] | Visual, VT-2 | IWB-3522 | Once per interval [Note (3)] | Same as for first interval | See [Note (3)] |

NOTES:
 (1) Visual examination of IWA-5240.
 (2) The system leakage test [IWB-5220] shall be conducted prior to plant startup following a reactor refueling outage.
 (3) The system leakage test [IWB-5220] of the boundary of IWB-5222(b) shall be performed at or near the end of the interval.

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Example Examination Requirements: Class 2 Piping Welds

Table IWC-2500-1 (C-F-1)
Examination Category C-F-1, Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping

| Item No. | Parts Examined [Note (1)] | Examination Requirements/ Figure No. | Examination Method | Acceptance Standard | Extent of Examination [Note (2)] | Frequency of Examination [Note (4)] |
|----------|---|---|------------------------|------------------------|--|--|
| CS.10 | Piping Welds $\geq \frac{1}{2}$ in. (13 mm) Nominal Wall Thickness for Piping \geq NPS 4 (DN 100) | | | | | |
| CS.11 | Circumferential Weld | IWC-2500-7 | Surface and Volumetric | IWC-3514 | 100% of each weld requiring examination [Note (3)] | Each inspection interval |
| CS.20 | Piping Welds $\geq \frac{1}{2}$ in. (5 mm) Nominal Wall Thickness for Piping \geq NPS 2 (DN 50) and \leq NPS 4 (DN 100) | | | | | |
| CS.21 | Circumferential Weld | IWC-2500-7 | Surface and volumetric | IWC-3514 | 100% of each weld requiring examination [Note (3)] | Each inspection interval |
| CS.30 | Socket Welds | IWC-2500-7 | Surface | IWC-3514 | 100% of each weld requiring examination | Each inspection interval |
| CS.40 | Pipe Branch Connections of Branch Piping \geq NPS 2 (DN 50) | | | | | |
| CS.41 | Circumferential Weld | IWC-2500-9 to IWC-2500-13, inclusive | Surface | IWC-3514 | 100% of each weld requiring examination [Note (3)] | Each inspection interval |

NOTES:

(1) Requirements for examination of welds in piping \geq NPS 4 (DN 100) apply to PWR high pressure safety injection and auxiliary feedwater systems in accordance with the exemption criteria of IWC-1220.

(2) The welds selected for examination shall include \geq 5%, but not less than 20 welds, of all dissimilar metal, austenitic stainless steel or high alloy welds not exempted by IWC-1220. (Some welds not exempted by IWC-1220 are not required to be nondestructively examined per Examination Category C-F-1. These welds, however, shall be included in the total weld count to which the 7.5% sampling rate is applied.) The examinations shall be distributed as follows:

(a) The examinations shall be distributed among the Class 2 systems provided, to the degree practicable, on the number of nonexempt dissimilar metal, austenitic stainless steel, or high alloy welds in each system (i.e., if a system contains 30% of the nonexempt welds, then 30% of the nondestructive examinations required by Examination Category C-F-1 should be performed on that system).

(b) Within a system, the examinations shall be distributed among terminal ends, dissimilar metal welds, and structural discontinuities [See Note (3)] provided, to the degree practicable, on the number of nonexempt terminal ends, dissimilar metal welds, and structural discontinuities in that system; and

(c) Within each system, examinations shall be distributed between line sizes provided, to the degree practicable.

(3) Structural discontinuities include pipe weld joints to vessel nozzles, valve bodies, pump castings, pipe fittings (such as elbows, tees, reducers, flanges, etc., conforming to ANSI B16.5), and pipe branch connections and fittings.

(4) The welds selected for examination shall be reexamined in the same sequence, during subsequent inspection intervals over the service lifetime of the piping component, to the extent practical.

(5) For circumferential welds with intersecting longitudinal welds, surface examination of the longitudinal piping welds is required for those portions of the welds within the examination boundary of intersecting circumferential welds.

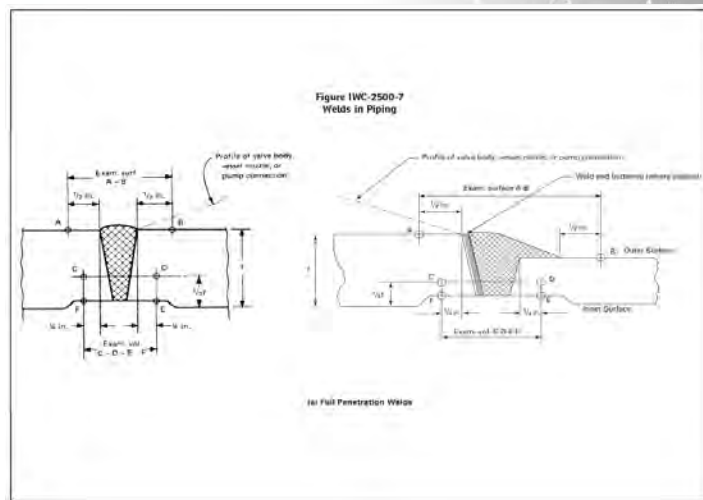
(6) For circumferential welds with intersecting longitudinal welds, volumetric examination of the longitudinal piping welds is required for those portions of the welds within the examination boundary of intersecting circumferential welds. The following requirements shall also be met:

(a) When longitudinal welds are specified and locations are known, examination requirements shall be met for both transverse and parallel flaws at the intersection of the welds and for that length of longitudinal weld within the circumferential weld examination volume.

(b) When longitudinal welds are specified but locations are unknown, or the existence of longitudinal welds is uncertain, the examination requirements shall be met for both transverse and parallel flaws within the entire examination volume of intersecting circumferential welds.

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Example Examination Requirements: Class 2 Piping Welds



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Example Examination Requirements: Class MC Containment Surfaces

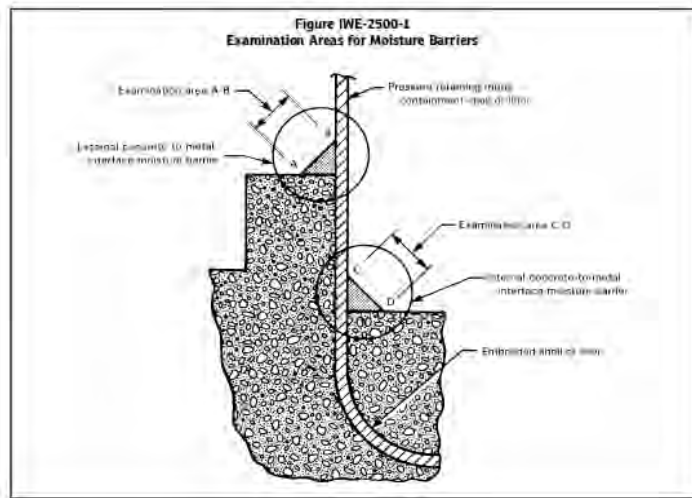
Table IWE-2500-1 (E-A)
Examination Category E-A, Containment Surfaces

| Item No. | Parts Examined | Examination Requirements/ Fig. No. (Note 1) | Examination Method | Acceptance Standard | Extent and Frequency of Examination | | Deferral of Inspection to End of Interval |
|----------|---|---|--------------------|---------------------|-------------------------------------|------------------------------------|---|
| | | | | | 1st Inspection Interval | Successive Inspection Intervals | |
| EL 10 | Containment Shell Pressure Retaining Boundary (Note 1) | IWE-2310 | General visual | IWE-3510 | 100% During each Inspection Period | 100% During each Inspection Period | N/A |
| EL 11 | Accessible Surface Areas | | | | | | |
| EL 12 | Wetted Surfaces of Submerged Areas | IWE-2310 | VT-3 | IWE-3510 | 100% | 100% | See (Note 2) |
| EL 20 | DWV Vent System Accessible Surface Areas (Note 1) (Note 1a) | IWE-2310 | VT-3 | IWE-3510 | 100% | 100% | See (Note 2) |
| EL 30 | Moisture Barriers (Note 1a) | IWE-2310, Figure IWE-2500-1 | General visual | IWE-3510 | 100% During each Inspection Period | 100% During each Inspection Period | N/A |

NOTES:
 (1) Examination shall include all accessible interior and exterior surfaces of Class MC components, parts, and appurtenances, and metallic shell and penetration liners of Class CC components. The following items shall be examined:
 (a) Integral attachments and structures that are parts of reinforcing structure, such as stiffening rings, manhole frames, and reinforcement around openings.
 (b) Surfaces of attachment welds between structural attachments and the pressure retaining boundary or reinforcing structure, except for non-structural or temporary attachments as defined in NS-4435, and minor permanent attachments as defined in CC-454.4.
 (c) Surfaces of containment structural and pressure boundary welds, including longitudinal welds (Category A), circumferential welds (Category B), flange welds (Category C), and nozzle-to-shell welds (Category D) as defined in NS-3351 for Class MC and CC-3040 for Class CC, and surfaces of Flued Head and Bellows Seal Circumferential Welds joined to the Penetration.
 (d) Pressure retaining bolted connections, including bolts, studs, nuts, lockings, washers, and threads in base material and flange ligaments between flange holes. Bolted connections need not be disassembled for performance of examinations.
 (2) Examinations may be performed at any time during the interval, provided successive examinations are performed no less frequently than every third period.
 (3) Includes flow channeling devices within containment vessels.
 (4) Examination shall include moisture barrier materials intended to prevent intrusion of moisture against inaccessible areas of the pressure retaining metal containment shell or liner at concrete-to-metal interfaces and at metal-to-metal interfaces which are not seal-welded. Containment moisture barrier materials include caulking, flashing, and other sealants used for this application.

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Example Examination Requirements: Class MC Containment Surfaces



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Example Examination Requirements: Class 1, 2, 3 and MC Supports

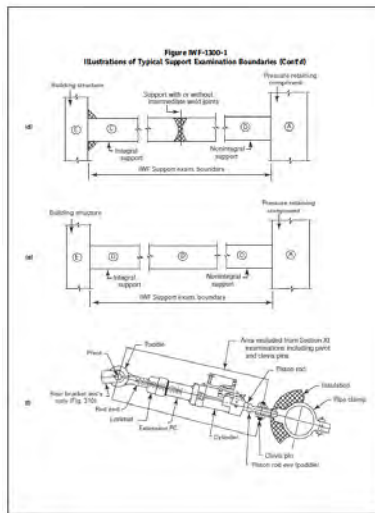
**Table IWF-2500-1 (F-A)
Examination Category F-A, Supports**

| Item No. [Note (1)] | Support Types Examined | Examination Requirements [Note (2)] [C]1) Fig. No. | Examination Method | Acceptance Standard | Extent and Examination | Frequency of Examination [Note (3)] |
|------------------------|---|---|--------------------|---------------------|----------------------------------|-------------------------------------|
| F1.10 | Class 1 Piping Supports | IWF-1300-1 | Visual, VT-3 | IOW-3410 | 25% of Class 1 [Notes (4)] | Each inspection interval |
| F1.20 | Class 2 Piping Supports | IWF-1300-1 | Visual, VT-3 | IOW-3410 | 15% of Class 2 [Notes (4)] | Each inspection interval |
| F1.30 | Class 3 Piping Supports | IWF-1300-1 | Visual, VT-3 | IOW-3410 | 10% of Class 3 [Notes (4)] | Each inspection interval |
| F1.40 | Supports Other Than Piping Supports (Class 1, 2, 3, and MC) | IWF-1300-1 | Visual, VT-3 | IOW-3410 | 100% of the supports [Notes (5)] | Each inspection interval |

NOTES:
 (1) Item numbers shall be categorized to identify support types by component support function (e.g., A = supports such as one directional rod hangers; B = supports such as multi-directional restraints; and C = supports that allow thermal movement, such as springs).
 (2) Examination may be limited to portions of supports that are accessible for examination without disassembly or removal of support members.
 (3) To the extent practical, the same supports selected for examination during the first inspection interval shall be examined during each successive inspection interval.
 (4) The total percentage sample shall be comprised of supports from each system (such as Main Steam, Feedwater, or RHR), where the individual sample sizes are proportional to the total number of non-energized supports of each type and function within each system.
 (5) For multiple components other than piping, within a system of similar design, function, and service, the supports of only one of the multiple components are required to be examined.

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Example Examination Requirements: Class 1, 2, 3 and MC Supports



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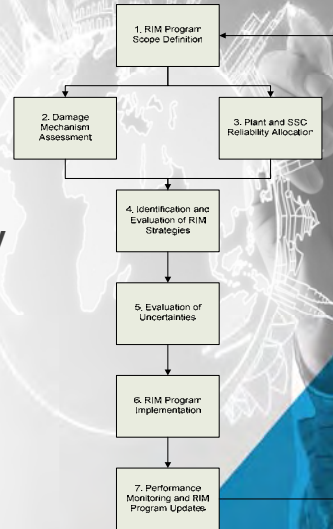
Reliability and Integrity Management Initiative

Revised Section XI, Division 2 Reliability and Integrity Management (RIM)

- ◆ **Written for all reactor types**
- ◆ **Specific reactor requirements will be included in the appendices**
 - light-water-cooled, gas-cooled, liquid-metal-cooled, molten salt reactors, generation 2 LWRs
- ◆ **Expected to be published in the 2019 Edition**
- ◆ **Considers plant design, inservice inspection, online monitoring, testing**
- ◆ **Determines appropriate level of reliability of structures, systems and components (SSCs)**
- ◆ **Provides continuing assurance over the life of the plant that such reliability is maintained**
- ◆ **Considers design and inspection or monitoring features important to reliability**
- ◆ **Monitoring and Nondestructive Examination Expert Panel considers numerous factors (e.g., materials, fabrication, reliability targets, environment, design margins, etc.)**

Revised Section XI, Division 2 Reliability and Integrity Management (RIM)

- ◆ The RIM Process consists of the following steps:
- ◆ Determine Scope for RIM Program
- ◆ Evaluate Degradation Mechanisms
- ◆ Determine Plant and Reliability and Capability Requirements
- ◆ Evaluate RIM Strategies to Achieve Reliability Targets
- ◆ Evaluate Uncertainties in Reliability Performance
- ◆ RIM Program Implementation
- ◆ Performance Monitoring and RIM Program Updates



Revised Section XI, Division 2 Reliability and Integrity Management (RIM)

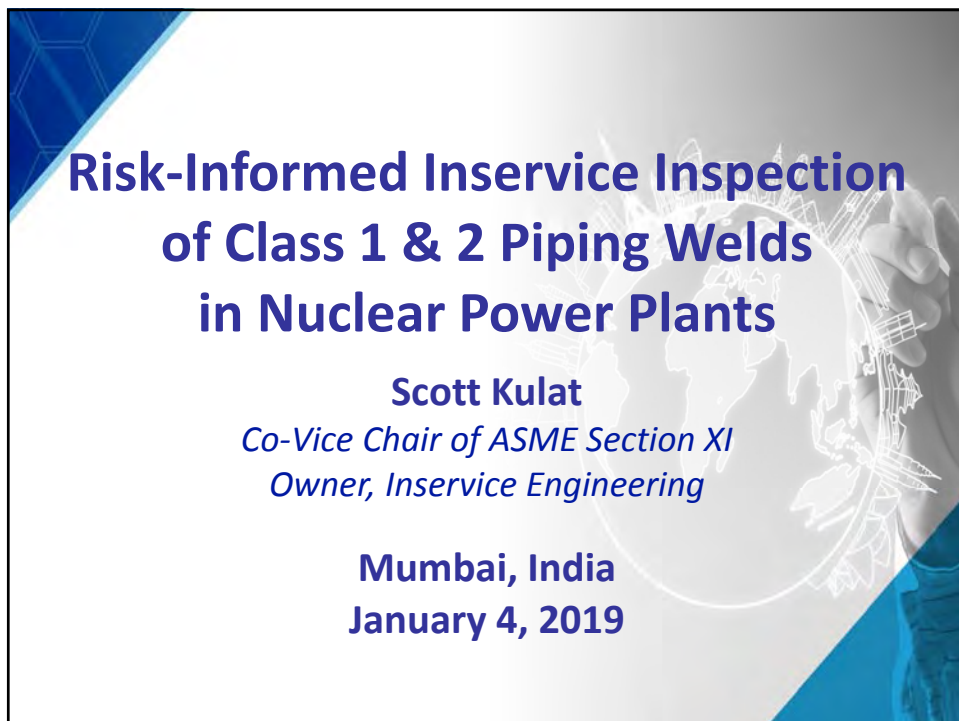
- ◆ RIM-1000: Scope and Responsibility
- ◆ RIM-2000: Reliability and Integrity Management (RIM) Program
- ◆ RIM-3000: Acceptance Standards
- ◆ RIM-4000: Repair/Replacements Activities
- ◆ RIM-5000: System Leak Monitoring and Pressure Tests
- ◆ RIM-6000: Records and Reports
- ◆ RIM-9000: Glossary

Conclusions on ASME Section XI

- ◆ ASME has a long history of providing Codes and Standards that establish criteria to ensure the safe and reliable operation of nuclear facilities
- ◆ ASME publishes numerous Codes, including Section XI which establishes the rules for operating nuclear power plants
- ◆ The ASME Codes are written and updated using a consensus process that is robust, open and balanced
- ◆ ASME Section XI provides established criteria for examinations, flaw evaluations, repair/replacement activities and pressure testing
- ◆ ASME Section XI provides detailed criteria for the performance of inservice inspections
- ◆ A new Reliability and Integrity Management (RIM) initiative has been generated to provide criteria for all reactor types

QUESTIONS





Objectives

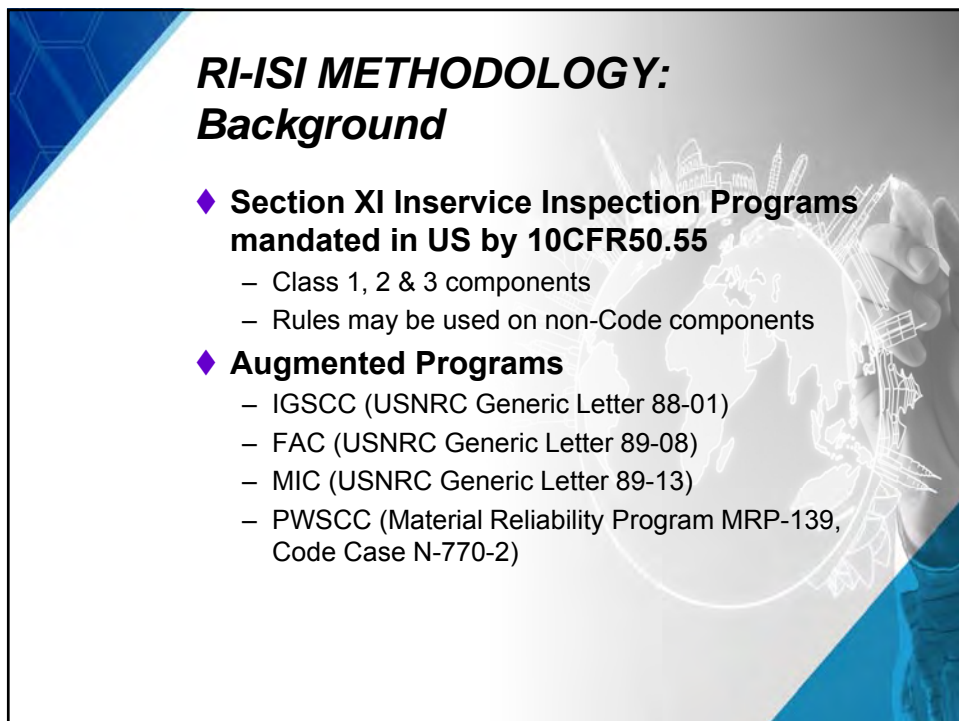
- ◆ Provide a background on risk-informed inservice inspection (RI-ISI) applications for Class 1 and 2 piping welds and describe how they interface with standard inservice inspection programs
- ◆ Define risk
- ◆ Describe the steps involved in a standard risk-informed methodology application
- ◆ Discuss one of the two risk-informed inservice inspection methodologies and its corresponding Code Cases
- ◆ Discuss the beneficial reduction in dose, radiation and examination costs associated with a risk-informed inservice inspection application

Overview

- ◆ Background
- ◆ Application of EPRI RI-ISI Methodology on Class 1 & 2 Piping Welds
- ◆ Cost and Radiation Reduction
- ◆ Living Program Requirements



Background



RI-ISI METHODOLOGY: Background

- ◆ **Section XI Inservice Inspection Programs mandated in US by 10CFR50.55**
 - Class 1, 2 & 3 components
 - Rules may be used on non-Code components
- ◆ **Augmented Programs**
 - IGSCC (USNRC Generic Letter 88-01)
 - FAC (USNRC Generic Letter 89-08)
 - MIC (USNRC Generic Letter 89-13)
 - PWSCC (Material Reliability Program MRP-139, Code Case N-770-2)

Original ASME Section XI Inspection Requirements

- ◆ **ASME Code required inspection of 100% of B-F (dissimilar metal) Class 1 welds, 25% of B-J (similar metal) Class 1 welds, and 7½% of C-F-1 and C-F-2 Class 2 welds**
- ◆ **Welds selected based on “high stress/high fatigue” locations**

What Were the Plants Telling Us?

- ◆ **Inservice failures (cracks, leaks, or breaks) are caused by corrosion or fatigue**
 - Thermal Fatigue (Thermal Transient & TASCs)
 - Stress Corrosion Cracking (IGSCC, TGSCC, PWSCC, ECSCC)
 - Localized Corrosion (MIC, Pitting, Crevice Corrosion)
 - Flow Sensitive Attack (FAC, Erosion/Cavitation)
 - High Cycle Mechanical Vibration Fatigue
- ◆ **Failures do not correlate with stress or fatigue usage factor values contained in Design Reports**
- ◆ **Failures do not always occur in welds**

Effectiveness of Original Section XI Examination Programs

- ◆ In a survey of 733 years of reactor operation only 156 of 37,332 B-J welds were found to contain flaws, of which 151 were due to IGSCC*
- ◆ 0.6% of the welds inspected were found to contain flaws by ASME Section XI examinations*

**ASME Task Group on ISI Optimization
(ASME 1995)*

Basis for Risk-Informed Inservice Inspection

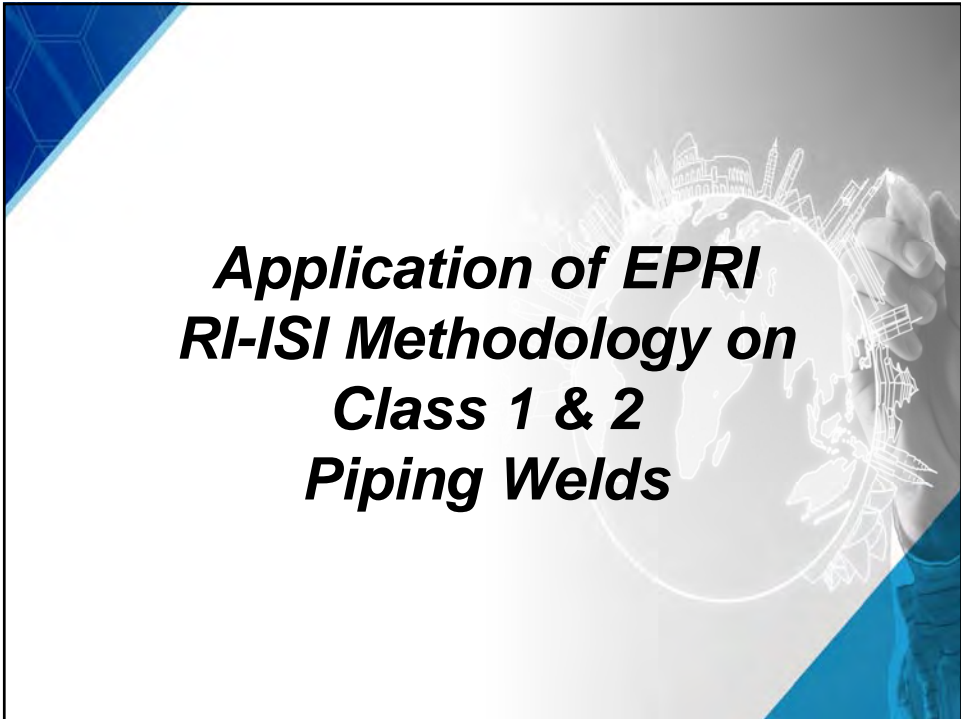
- ◆ All US nuclear power plants were required to perform an Individual Plant Examination (Probabilistic Safety Analysis) per USNRC Generic Letter 88-20
 - Determine plant vulnerabilities to:
Core Damage Frequency (CDF) and
Large Early Release Frequency (LERF)
- ◆ CDF and LERF can be used to determine an optimum inservice inspection scheme

Definition of Risk

- ◆ **Risk is a function of consequence and probability**
 - Consequence is measured by CDF and LERF
 - Probability is a function of potential degradation modes as determined by physical characteristics and operational parameters

USNRC Regulatory Guides

- ◆ **USNRC RG 1.174 – An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant Specific Changes to the Licensing Basis**
- ◆ **USNRC RG 1.178 – An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping**
- ◆ **Additional Regulatory Guides for other Risk-Informed applications (RI-IST, etc.)**



Application of EPRI RI-ISI Methodology on Class 1 & 2 Piping Welds



EPRI Code Cases

◆ **N-560**

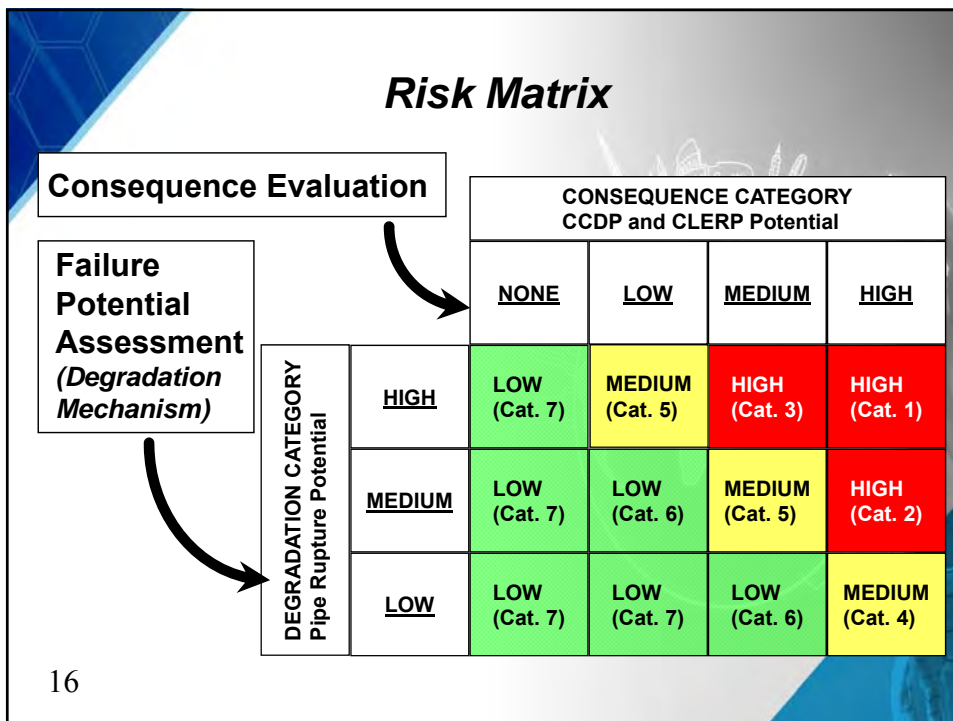
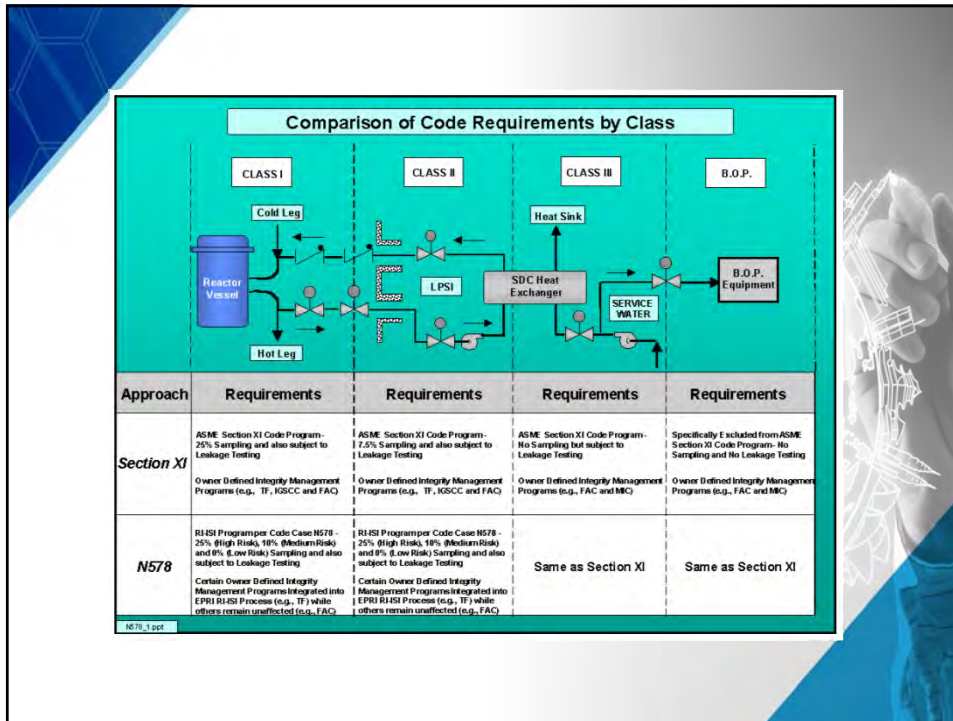
- Approved in 1996
- Applies to B-J welds, excluding socket weld
- Inspect 10% of B-J welds selected by risk
- Code Case N-560-2 published

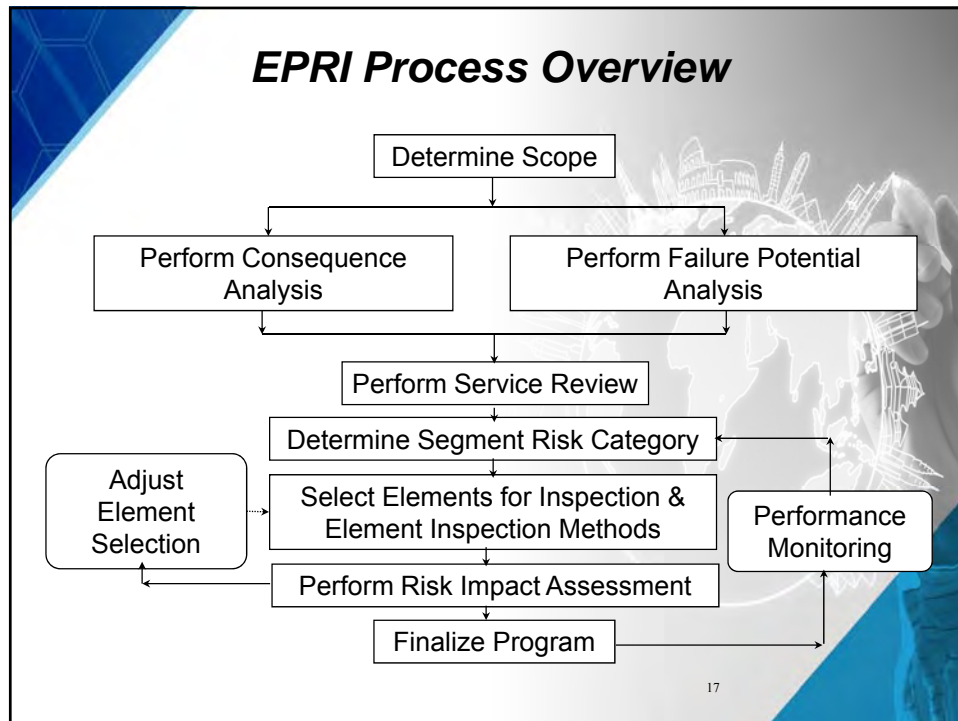
◆ **N-578**

- Approved in 1997
- Can be applied to class 1, 2, &/or 3 (& may be used for non-code welds)
- Inspect 25% of high risk, 10% of medium risk
- Code Case N-578-1 published

◆ **Risk assessment process is same--element selection process is slightly different**

◆ **Incorporated into ASME Section XI Nonmandatory Appendix R**





- ### **Scope Determinations**
- ◆ Determine systems in RI-ISI application
 - ◆ Determine boundaries of application (i.e., typically Class 1 and 2)
 - ◆ Consider former and current exemptions



Consequence Assessment



Consequence Assessment

◆ Goal

- To assign a consequence rank to each location within the piping system.

◆ Parameters

- Break size (small, large, worst case)
- Isolability of the break (success and failure)
- Direct effects (flow diversion)
- Indirect effects (spatial effects including walkdown findings, inventory loss, flooding)
- Containment performance
- Recovery

Consequence Assessment (cont'd)

◆ Consequence Evaluation consists of four major steps

1. Plant PSA models, systems, and initiators are evaluated. The initial consequence rank is established based on the Pressure Boundary Failure's (PBFs) impact on Core Damage Frequency (CDF), by estimating Conditional Core Damage Probability (CCDP) values.
2. Containment performance is evaluated. The consequence rank is reviewed and adjusted to reflect the PBFs impact on Large Early Release Frequency (LERF), by estimating Conditional Large Early Release Probability (CLERP) values, or by evaluating the likelihood of containment bypass.

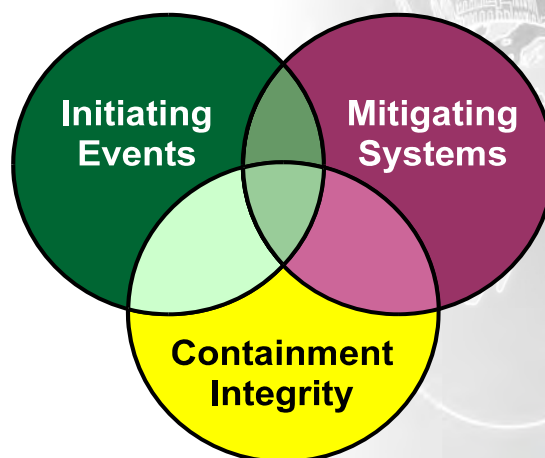
Consequence Assessment (cont'd)

3. Shutdown operation is evaluated. The consequence rank is reviewed and adjusted to reflect the PBFs impact on the plant operation during shutdown.
4. External events are evaluated. The consequence rank is reviewed and adjusted to reflect the PBFs impact on the mitigation of external events.

Consequence Considerations

- ◆ **Initiating events**
- ◆ **Mitigating ability**
 - Loss of system(s) or train(s)
 - Degradation of system(s) or train(s)
- ◆ **Containment effects**
 - Loss of containment integrity
 - Degradation of containment integrity
- ◆ **Combination event**

Consequence Evaluation Factors



Pressure Boundary Failures (PBFs) Could Occur in Three Different Configurations:

1. Operating :

A PBF occurs in an operating (pressurized) system, and usually results in an initiating event. Exposure time is equal to all year.

2. Standby:

A PBF occurs in a standby system, and it is detected either by instrumentation, test or visual inspection. After the failure is detected, the plant enters the Allowed Outage Time (AOT). Exposure time is equal to the AOT plus detection time.

PBFs Configurations (cont'd)

3. Demand:

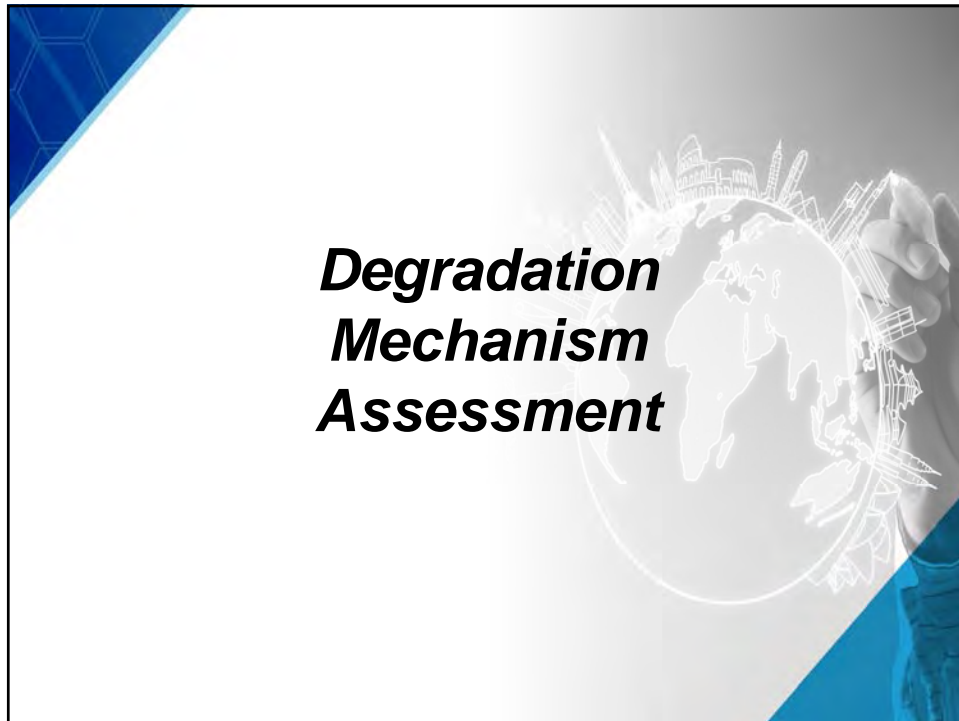
A PBF occurs when system operation is required by an independent demand. Exposure time, in this case is equal to the test interval, or to all year if the system is not tested. Test pressures or flows are credited as equivalent to demand conditions, thereby reducing exposure time in the analysis of demands for piping that experiences testing.

Basic Consequences Ranking

- ◆ **High**
 - Pressure boundary failures resulting in events which are important contributors to the plant risk.
 - Pressure boundary failures which significantly degrade plant mitigating ability.
- ◆ **Low**
 - Pressure boundary failures resulting in anticipated operational events or events which are not important contributors to the plant risk.
 - Pressure boundary failures which do not significantly impact plant mitigating ability.
- ◆ **Medium**
 - “Something Between”

Quantitative Ranking Criteria

| <u>Consequence Category</u> | <u>Corresponding CCDP Range</u> | <u>Corresponding CLERP Range</u> |
|-----------------------------|---------------------------------|----------------------------------|
| High | $CCDP > 1E-4$ | $CLERP > 1E-5$ |
| Medium | $1E-6 < CCDP \leq 1E-4$ | $1E-7 < CLERP \leq 1E-5$ |
| Low | $CCDP \leq 1E-6$ | $CLERP \leq 1E-7$ |



Degradation Mechanism Category

| Large Pipe Break Potential | Leak Conditions | Degradation Mechanism |
|----------------------------|-----------------|--|
| HIGH | Large | Flow Accelerated Corrosion (FAC) |
| MEDIUM | Small | Thermal Fatigue Stress Corrosion Cracking (IGSCC, TGSCC, PWSCC, ECSCC) Localized Corrosion (MIC, Pitting, Crevice Corrosion) Erosion/Cavitation |
| LOW | None | No Degradation Mechanisms |

Overview of Degradation Mechanisms

- ◆ **Thermal Fatigue**
 - Thermal Stratification, Cycling and Striping (TASCS)
 - Thermal Transients (TT)
- ◆ **Stress Corrosion Cracking**
 - Intergranular Stress Corrosion Cracking (IGSCC)
 - Transgranular Stress Corrosion Cracking (TGSCC)
 - External Chloride Stress Corrosion Cracking (ECSCC)
 - Primary Water Stress Corrosion Cracking (PWSCC)

Overview of Degradation Mechanisms (cont'd)

- ◆ **Localized Corrosion**
 - Microbiologically Influenced Corrosion (MIC)
 - Pitting (PIT)
 - Crevice Corrosion (CC)
- ◆ **Flow Sensitive**
 - Erosion-Cavitation (E-C)
 - Flow Assisted Corrosion (FAC)
- ◆ **Vibrational Fatigue**
- ◆ **Water Hammer**

Inputs for Degradation Mechanisms Assessment

◆ Design characteristics

- Materials
- Pipe size
- Component type
- Configuration layout

◆ Fabrication practices

- Welding processes and materials
- Heat treatment, stress improvement, etc.
- Crevices

Inputs for Degradation Mechanisms Assessment

◆ Operating conditions

- Pressure and temperature
- Flow conditions (stagnant, laminar, turbulent)
- Chemistry controls
- Service environment (humidity, radiation, etc.)

◆ Service experience

Degradation Mechanisms Assessment - Thermal Fatigue

- ◆ **Caused by alternating stresses due to thermal cycling**
- ◆ **Only transients which occur under normal operating and upset conditions need to be considered**
- ◆ **There are two broad categories of Thermal Fatigue**
 - Thermal Transients (TT)
 - Thermal Stratifification, Cycling and Stripping (TASCS)

Degradation Mechanisms Assessment - TASCS Criteria

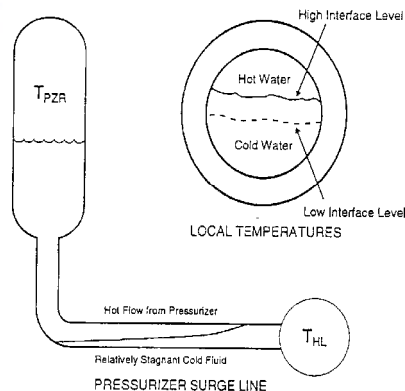
- ◆ **TASCS is Thermal Fatigue related to local stratified flow conditions and other localized cyclic conditions such as valve leakage or fluid mixing**
- ◆ **The basis for the criteria for TASCS is provided in the EPRI Fatigue Management handbook, EPRI TR-104534**
 - Criteria***
 - $NPS > 1$
 - Pipe segment has slope $< 45^\circ$
 - Temperature between mixing fluids $> 50^\circ\text{F}$
 - Richardson number > 4.0

Degradation Mechanisms Assessment - TASCs Criteria

◆ Sources of hot/cold fluid mixing:

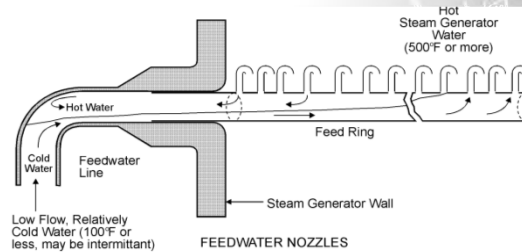
- Low flow in a pipe section connected to a component allowing mixing of hot and cold fluids
- Leakage past a valve (in-leakage, out-leakage, cross-leakage)
- Convection heating in dead end pipe sections connected to a source of hot fluid
- Two phase (steam/water) flow
- Turbulent penetration in a branch pipe

Degradation Mechanisms Assessment - Low Flow TASCs



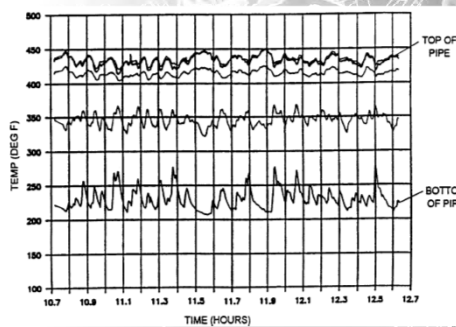
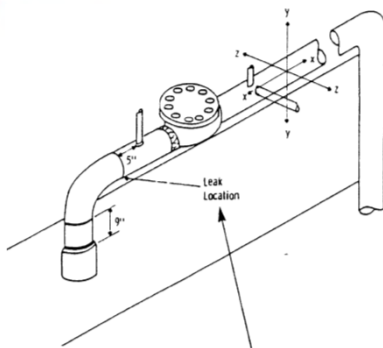
- ◆ Stratified conditions exist in surge lines (ΔT higher during startup & shutdown)
- ◆ Cyclic conditions occur during in-surge and out-surge
- ◆ Stratified conditions cause additional stresses and displacements not considered in original design
- ◆ No leakage has occurred
- ◆ Re-evaluation required by USNRC Bulletin 88-11

Degradation Mechanisms Assessment - Low Flow TASCS

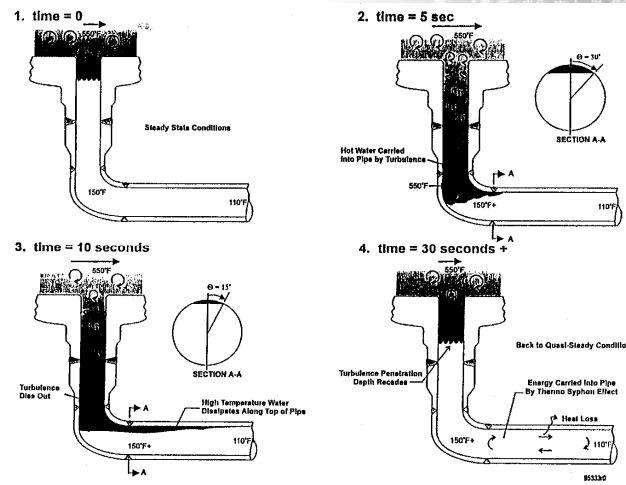


- ◆ Original problem identified prior to 1980, (USNRC Bulletin 79-13)
- ◆ During low-flow conditions with low temperature auxiliary feedwater, stratified flow occurs in nozzles
- ◆ Thermal cycling occurs due to variation of flow rate
- ◆ Thermal striping occurs at interface
- ◆ Fatigue crack propagation has caused leakage - generally originating at counterbores

Degradation Mechanisms Assessment Valve Leakage TASCS (cont'd)



Degradation Mechanisms Assessment Turbulence Penetration TASCs



Degradation Mechanisms Assessment Thermal Transient Criteria

- ◆ Occurs in systems that experience significant changes in fluid temperature conditions during normal operation
- ◆ The basis for criteria is provide in EPRI TR-104534

Degradation Mechanisms Assessment Thermal Transient Criteria (cont'd)

◆ Criteria

- Operating temp
 - > 270°F for stainless steel
 - > 220°F for carbon steel
- Rapid temperature changes
 - cold fluid injection into hot pipe
 - hot fluid injection into cold pipe
- $|\Delta T| > 200^\circ\text{F}$ for stainless steel
- $|\Delta T| > 150^\circ\text{F}$ for carbon steel
- $|\Delta T| > \Delta T_{\text{allow}}$ (from EPRI TR-10534)
 - ΔT_{allow} based on pipe material, diameter, thickness, injection flowrate

Degradation Mechanisms Assessment Stress Corrosion Cracking (SCC)

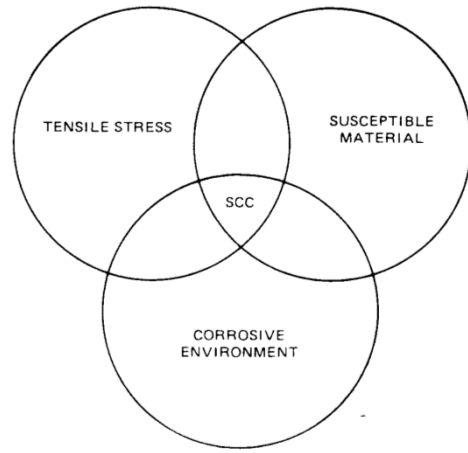
◆ Requires a combination of factors

- Corrosive or oxygenated environment
- Susceptible material
- Tensile stresses
- Temperature

◆ Applies to austenitic stainless steel welds and heat affected zone (HAZ)

◆ Also applies to Inconel (Alloy 600/82/182) piping in case of Primary Water Stress Corrosion Cracking (PWSCC)

Degradation Mechanisms Assessment Stress Corrosion Cracking (SCC) (cont'd)



Degradation Mechanisms Assessment Stress Corrosion Cracking (SCC) (cont'd)

Intergranular Stress Corrosion Cracking (IGSCC)

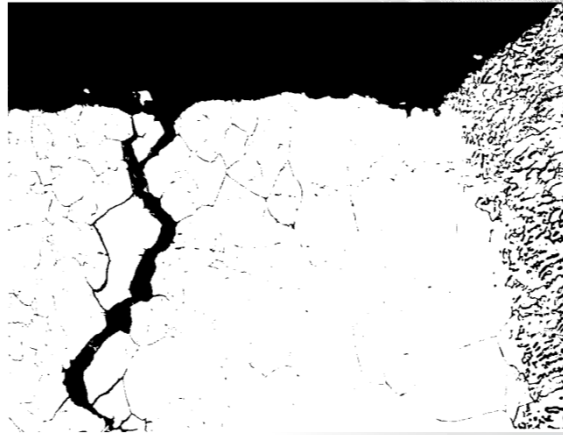
- ◆ **Stress corrosion cracking which occurs along the grain boundaries of the material due to formation of Cr_{23}C_6**

BWRS

Criteria

- Currently, IGSCC in BWRS is evaluated in accordance with existing plant IGSCC program per NRC Generic Letter 88-01 and NUREG-0313, Rev. 2
- The industry has made a request to the NRC for changes in the requirements of GL 88-01 based on plants' experience with IGSCC over the last several years (BWRVIP-75-A)

Degradation Mechanisms Assessment Stress Corrosion Cracking (SCC) (cont'd)



Degradation Mechanisms Assessment Stress Corrosion Cracking (SCC) (cont'd)

IGSCC - PWRS

Criteria

- Operating temperature > 200°F
 - Susceptible material (Carbon content $\geq 0.035\%$)
 - Tensile stress
 - Oxygen or oxidizing species
- Operating temperature < 200°F
 - Susceptible material, tensile stress (as for T > 200°F)
 - Initiating contaminants
 - thiosulfate, caustic--w/o O₂
 - fluoride, chlorides--w/ O₂

Degradation Mechanisms Assessment Stress Corrosion Cracking (SCC) (cont'd)

Transgranular Stress Corrosion Cracking (TGSCC)

- ◆ **TGSCC is stress corrosion cracking which occurs through the grains of the material**

Criteria

- Operating temperature > 150°F
- For stainless steels:
 - halides (fluoride, chloride)
- Oxygen or oxidizing species (with halides)

Degradation Mechanisms Assessment Stress Corrosion Cracking (SCC) (cont'd)



Degradation Mechanisms Assessment Stress Corrosion Cracking (SCC) (cont'd)

External Chloride Stress Corrosion Cracking (ECSCC)

◆ Stress corrosion cracking due to chloride intrusion on the outside surface of the pipe

Criteria

- Operating temperature > 150°F
- Tensile stress
- Outside pipe diameter exposure
 - Non-metallic insulation that is not in compliance to Reg. Guide 1.36 and in proximity of a probable leak
 - Wetting from chloride bearing environments

Degradation Mechanisms Assessment Stress Corrosion Cracking (SCC) (cont'd)

Primary Water Stress Corrosion Cracking (PWSCC)

◆ Only occurs in PWRs

Criteria

- Operating temperature > 570°F
- Primary water
- Piping material is Inconel (Alloy 600)
 - Mill-annealed and cold-worked
 - Cold-worked and welded without stress relief
- Weld material is Inconel (Alloy 82/182)

Degradation Mechanisms Assessment Localized Corrosion

- ◆ **Produces localized degradation in piping components**
- ◆ **Typically associated with low flow or “hideout” regions**
- ◆ **Three common forms of Localized Corrosion**
 - Microbiologically Influenced Corrosion (MIC)
 - Pitting (PIT)
 - Crevice Corrosion (CC)

Degradation Mechanisms Assessment Localized Corrosion (cont'd)

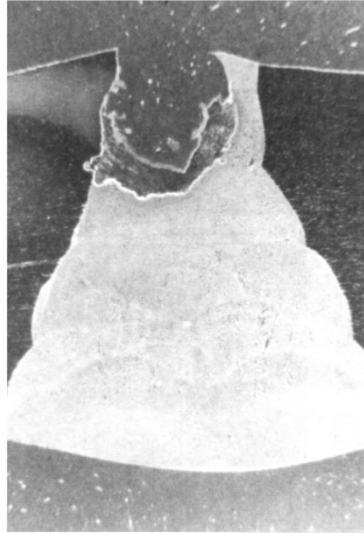
Microbiologically Influenced Corrosion (MIC)

- ◆ **Interaction of microbes' metabolism with corrosion processes to produce pits, often at very high rates**

Criteria

- Operating temp < 150°F
- pH < 10
- Potential for low flow
 - Presence of organic material (e.g., raw water system)
 - Water source not treated with biocides

Degradation Mechanisms Assessment Localized Corrosion (cont'd)



Degradation Mechanisms Assessment Localized Corrosion (cont'd)

Pitting

- ◆ **Initiation and propagation of localized attack on boldly exposed surfaces**
- ◆ **All materials are susceptible, corrosion resistant alloys like stainless steels can be affected, producing unanticipated degradation and failures**

Criteria

- Potential exists for low flow
- Oxygen or oxidizing species
- Initiating contaminants (e.g. fluoride, chloride)

Degradation Mechanisms Assessment Localized Corrosion (cont'd)

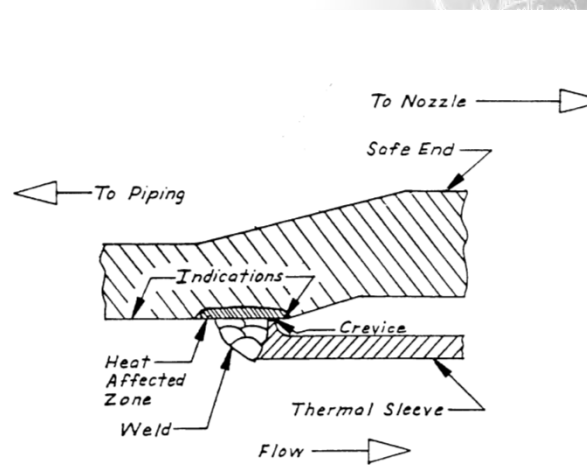
Crevice Corrosion

- ◆ Localized corrosion that occurs in the presence of a crevice (e.g. thermal sleeve)

Criteria

- Crevice condition exists
- Operating temperature > 150°F
- Oxygen or oxidizing species

Degradation Mechanisms Assessment Localized Corrosion (cont'd)



Degradation Mechanisms Assessment Flow Sensitive

- ◆ **High fluid velocity when combined with other factors result in erosion and/or corrosion of piping wall**
- ◆ **Two mechanisms are classified under flow sensitive**
 - Erosion-Cavitation (E-C)
 - Flow Accelerated Corrosion (FAC)

Degradation Mechanisms Assessment Flow Sensitive (cont'd)

Erosion – Cavitation (E-C)

- ◆ **Caused by turbulent flow conditions, which erode the pipe wall by cavitation**
- ◆ **Cavitation damage is the result of the formation and instantaneous collapse of small voids in a fluid subjected to rapid pressure and velocity changes**
- ◆ **Occurs immediately downstream of flow restrictors (e.g. valves, orifices)**

Degradation Mechanisms Assessment Flow Sensitive (cont'd)

Flow-Accelerated Corrosion (FAC)

- ◆ **Complex phenomenon that exhibits attributes of erosion and corrosion in combination**
- ◆ **Evaluated in accordance with plant FAC program**
- ◆ **More details documented in EPRI NP-3944**

Degradation Mechanisms Assessment Flow Sensitive (cont'd)

Criteria (general)

- Carbon steel piping < 1% chromium
- High flow velocity
- Low dissolved oxygen (< 15 ppb)
- Relatively low pH
- High moisture content of steam
- Flow path geometry (discontinuities such as elbows, branch connections, reducers, etc.)

Degradation Mechanisms Assessment Vibrational Fatigue

- ◆ **Vibration is not specifically part of the Risk-Informed ISI process**
- ◆ **Generally occurs only in socket welds in small bore piping**
- ◆ **Most damage occurs in the initiation phase and crack propagation occurs at a rapid rate**
- ◆ **Does not lend itself to ISI exams for management of this mechanism**
- ◆ **Best to manage it taking guidance from EPRI fatigue management handbook (EPRI TR-104534)**

Degradation Mechanisms Assessment Water Hammer

- ◆ **Water hammer is a rare, severe loading condition as opposed to a degradation mechanism**
- ◆ **Water hammer occurs in systems where there is potential for fluid voiding and relief valve discharge**
- ◆ **The potential for water hammer at a location in conjunction with one or more of the listed mechanisms could be cause for a higher rupture potential**
- ◆ **The potential for water hammer is evaluated on plant by plant basis**
- ◆ **More details on water hammer occurrences and mitigation is provided in EPRI TR-106438**

Typical Mechanisms for PWR Systems

| <u>System</u> | <u>Typical Material</u> | <u>Typical Mechanisms</u> |
|-------------------------------------|--------------------------------|----------------------------------|
| Reactor Coolant | Stainless, Carbon | Thermal Fatigue |
| Safety Injection / Shutdown Cooling | Stainless | Thermal Fatigue, ISCC |
| Feedwater | Carbon | IASCS, FAC |
| Main Steam | Carbon | FAC |
| Chemical & Volume Control | Stainless | Thermal Treatment |
| Service Water | Carbon, Stainless | MIC, Pitting, Erosion-Cavitation |
| Containment Spray | Stainless | None |
| Emergency Feedwater / Aux Feedwater | Carbon | IASCS, Thermal Treatment |

Performance of a Risk-Informed Inservice Inspection Service History Review

Service History Review

- ◆ **Plant specific review of plant and industry databases required for each system within scope**
- ◆ **Identify any mechanisms or events potentially resulting in piping failures**
- ◆ **May influence degradation mechanism category and/ or inspection element selection**
- ◆ **Particularly appropriate for water hammer, which is a strong function of plant's unique system configurations and operational and maintenance practices**

Service History Review

- ◆ **Examples: Degradation Mechanisms**
 - External Chloride Stress Corrosion Cracking (ECSCC) on stainless steel piping
 - Chlorinated service water
 - Leaking Containment Fan Coil Units (CFCUs) effects piping inside containment near penetrations
 - Leaking service water valves effects containment spray piping outside containment
 - Numerous indications documented in Outage Summary Reports that were “Acceptable As Is”
 - Don't effect degradation mechanism evaluation or risk ranking
 - Considered during element selection meeting
 - Linear indications on feedwater nozzles that have since been replaced
 - Intergranular Stress Corrosion Cracking (IGSCC) in “flame bent” containment spray piping

Service History Review


◆ Water Hammer Events

- Water hammer events identified and described
- Cause of water hammer events identified
- Steps taken to mitigate future water hammer events
- Conclusions presented

Service History Review

◆ Conclusions

- Information on degradation mechanisms identified during the service history review are incorporated into the degradation mechanism evaluation and used during the element selection meeting
- Water hammer events are evaluated to determine if they have an effect on the RI-ISI application or if steps have been taken to mitigate the reoccurrence of events
- Results are documented in the Service History Review document



Performance of a Risk-Informed Risk Ranking



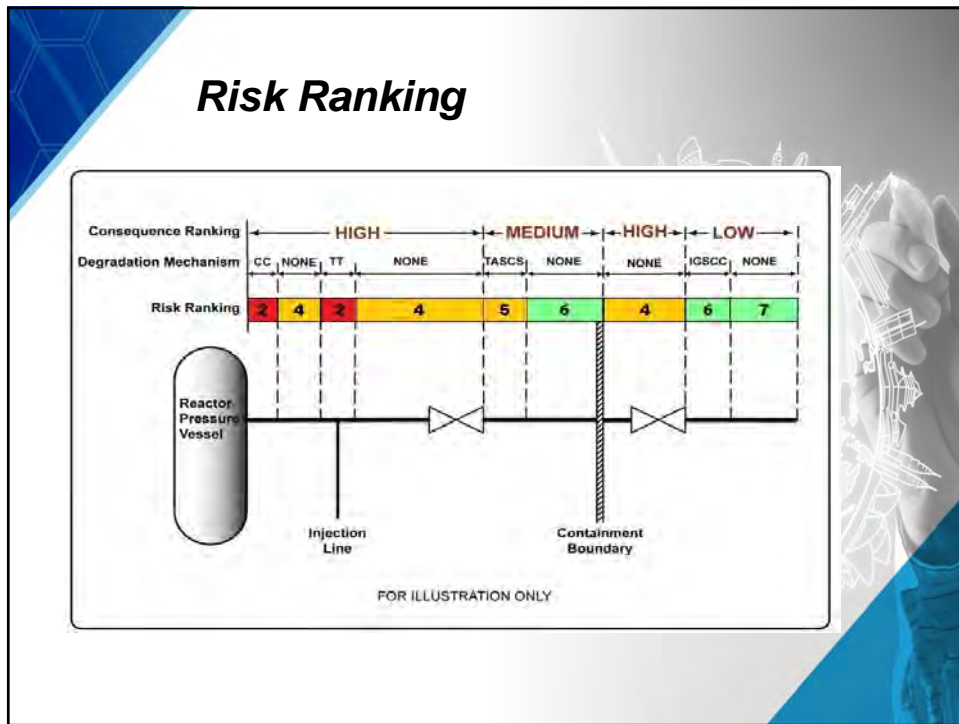
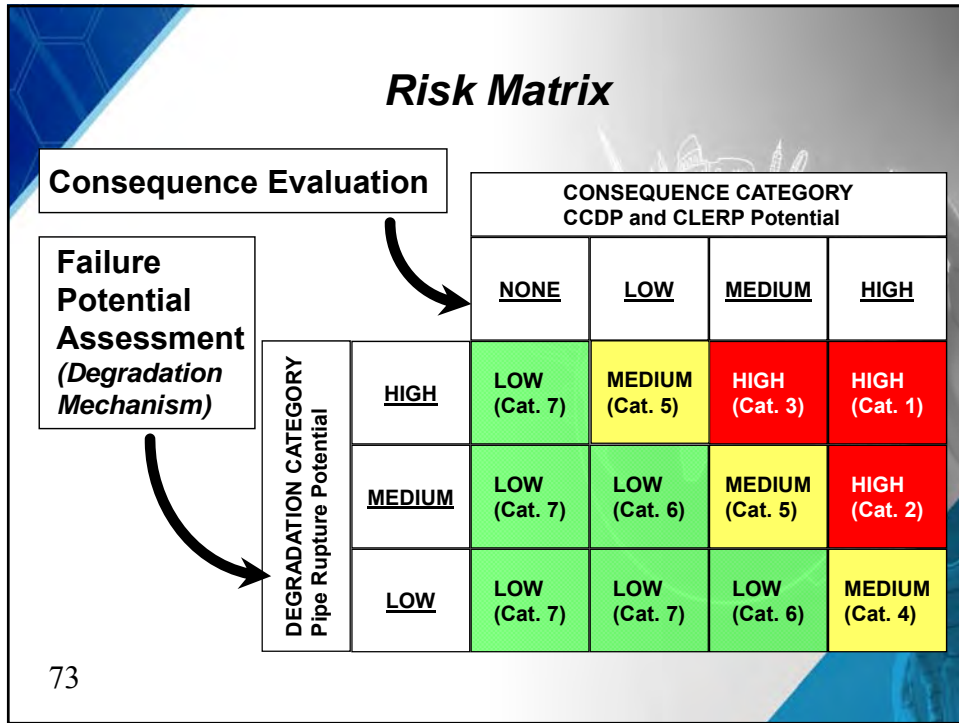
Risk Ranking

◆ Design Inputs

- Consequence Evaluation results
- Degradation Mechanism Calculation results
- Service History Review results

◆ Analysis

- Risk segments formed
- Risk categories assigned



Risk Ranking Spreadsheets

◆ Results of Analysis

- Results summarized in three tables
 - Risk ranking summary table
 - Risk ranking matrix table
 - Risk ranking report table
- Risk ranking performed both with and without FAC considered

Risk Ranking Report

- ◆ Summary Listing of Risk Ranking**
- ◆ Breakdown of Elements into Risk Categories**
- ◆ Details of Risk Ranking Results**
 - System
 - Risk Group
 - Risk Category
 - Consequence ID and Ranking
 - Degradation Mechanism and Ranking
 - Line Numbers
 - Weld Numbers
 - Drawing Numbers

Risk Ranking Matrix

| Unit 1 Risk Ranking Matrix | | Consequence Evaluation <i>Conditional Core Melt Potential</i> | | | |
|---|--------------------|---|---|---|--|
| | | NONE | LOW | MEDIUM | HIGH |
| Degradation Mechanism Assessment <i>Pipe Rupture Potential</i> | H I G H | Category 7b - Low Risk RCS - 0 CVCS - 0 SIS - 0 RHRS - 0 CSS - 0 FWS - 0 MSS - 0 Total - 0 Elements | Category 5b - Medium Risk RCS - 0 CVCS - 0 SIS - 0 RHRS - 0 CSS - 0 FWS - 0 MSS - 0 Total - 0 Elements | Category 3 - High Risk RCS - 0 CVCS - 0 SIS - 0 RHRS - 0 CSS - 0 FWS - 0 MSS - 0 Total - 0 Elements | Category 1 - High Risk RCS - 0 CVCS - 0 SIS - 0 RHRS - 0 CSS - 0 FWS - 0 MSS - 0 Total - 0 Elements |
| | M E D I U M | Category 7c - Low Risk RCS - 0 CVCS - 0 SIS - 0 RHRS - 0 CSS - 0 FWS - 0 MSS - 0 Total - 0 Elements | Category 6b - Low Risk RCS - 0 CVCS - 8 SIS - 22 RHRS - 0 CSS - 0 FWS - 0 MSS - 0 Total - 30 Elements | Category 5a - Medium Risk RCS - 65 CVCS - 0 SIS - 12 RHRS - 0 CSS - 0 FWS - 8 MSS - 0 Total - 85 Elements | Category 2 - High Risk RCS - 32 CVCS - 0 SIS - 0 RHRS - 0 CSS - 0 FWS - 0 MSS - 0 Total - 32 Elements |
| | L O W | Category 7d - Low Risk RCS - 0 CVCS - 0 SIS - 0 RHRS - 0 CSS - 0 FWS - 0 MSS - 0 Total - 0 Elements | Category 7a - Low Risk RCS - 15 CVCS - 30 SIS - 225 RHRS - 0 CSS - 234 FWS - 0 MSS - 0 Total - 504 Elements | Category 6a - Low Risk RCS - 61 CVCS - 278 SIS - 691 RHRS - 8 CSS - 178 FWS - 435 MSS - 165 Total - 1816 Elements | Category 4 - Medium Risk RCS - 224 CVCS - 0 SIS - 177 RHRS - 258 CSS - 10 FWS - 0 MSS - 0 Total - 669 Elements |

Risk-Informed Inservice Inspection Element Selection Process

Element Selection Process

- ◆ **Consequence and degradation mechanism evaluations complete**
- ◆ **Segment risk ranking complete**
- ◆ **List of elements* compiled by system, identifying:**
 - risk category
 - consequence category
 - applicable degradation mechanism(s)
 - whether element was in prior Section XI or augmented program
 - additional considerations (ALARA, access, previous repairs, etc.)

*Note: Elements are typically welds, but it is occasionally deemed appropriate to select some base material locations for examination, depending on the nature of the degradation mechanism

Element Selection Meeting

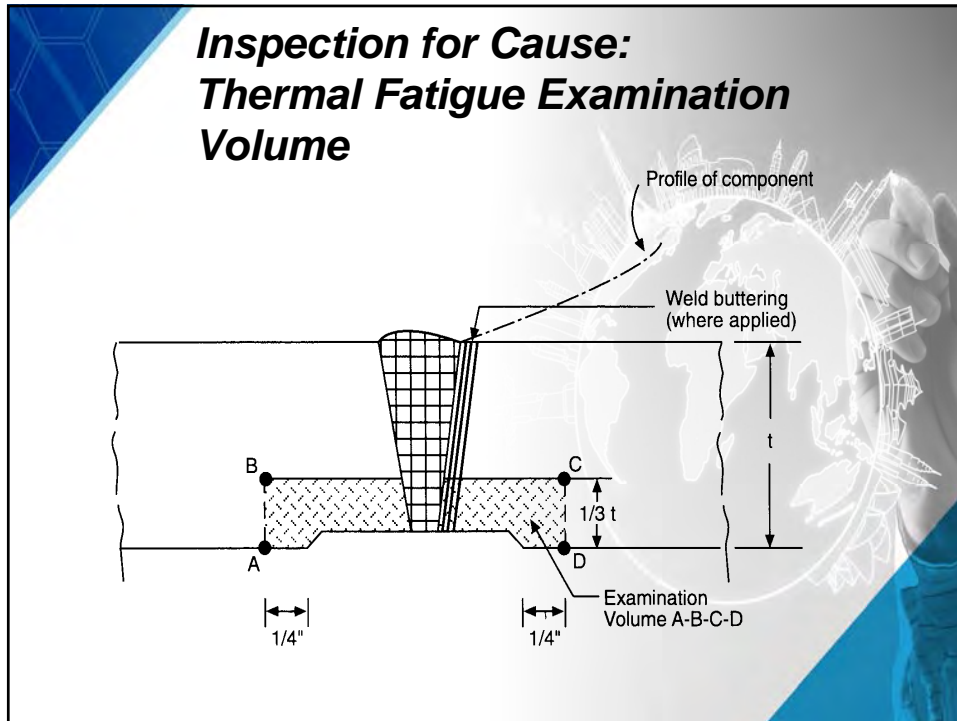
- ◆ **Assemble team to select elements for examination**
 - ISI Coordinator
 - ISI Supervisor/ Manager
 - Systems engineers responsible for systems included in program
 - Operations representative
 - PRA representative
 - NDE representative
 - HP representative
 - Individuals knowledgeable on scaffolding/ insulation/ craft requirements
 - Members of RI-ISI project team

Element Selection Process

- ◆ **Select 25% of high risk and 10% of medium risk locations, based on risk ranking**
- ◆ **Additional considerations for elements with the same risk ranking**
 - Previous examination results
 - Prior repairs or grinding
 - Design irregularities
 - Accessibility
 - ALARA
 - Representation of all systems and degradation mechanisms in selection

Examination Volumes and Methods

- ◆ **ASME Section XI identifies locations and methods for examination without identifying what, if any, degradation mechanism exists**
- ◆ **RI-ISI is a procedure for selecting and examining locations based on their risk significance**
- ◆ **RI-ISI methodology selects examination volumes and techniques based on identified degradation mechanism(s)**
- ◆ **Surface examinations are not required for most elements**
- ◆ **Appropriate examinations and associated examination volumes/ areas specified in Code Case N-578-1 and EPRI Topical Report TR-112657**
- ◆ **Code Case N-711, “Alternative Examination Coverage Requirements for Examination Categories B-F, B-J, C-F-1, C-F-2 and R-A”**



Risk-Informed Inservice Inspection Risk Impact Analysis

Change in Risk Assessment

- ◆ **A “Risk Impact” calculation is performed to demonstrate that revision to ISI Program meets Regulatory Guide 1.174 guidelines**
 - Risk decrease
 - Risk neutral
 - Insignificant risk increase
- ◆ **Options available for satisfying this requirement**
 - Quantitative
 - Boundary
 - Simplified
 - Complex

Risk Impact Analysis

- ◆ **The impact of the proposed change in risk is due to two opposing factors:**
 - The number of inspection locations is reduced
 - Most reductions occur in Low and Medium risk areas
 - Reductions in High risk areas are minimal.
 - The inspection techniques have improved
 - An inspection for cause approach is employed, that utilizes examination methods and volumes designed for specific degradation mechanisms
 - In many cases the Probability of Detection (POD) has increased

Risk Impact Analysis

- ◆ The change in core damage frequency (CDF) due to application of the RI-ISI process is estimated based on the following equation:

$$\Delta R_{CDF} = CCDP * RF * [(POD_S * N_S) - (POD_R * N_R)]$$

- ◆ Where:

CCDP = Conditional Core Damage Probability

RF = Rupture Frequency

POD_S = Probability of Detection associated with the ASME Section XI Code Program

POD_R = Probability of Detection associated with the EPRI TR-112657 RI-ISI Program

N_S = Number of Inspection Locations in the ASME Section XI Code Program

N_R = Number of Inspection Locations in the EPRI TR-112657 RI-ISI Program

Risk Impact Analysis

- ◆ Additional considerations in conducting the quantitative analysis
 - The EPRI criterion requires that the cumulative change in core damage frequency (CDF) and large early release frequency (LERF) be less than 1E-07 and 1E-08 per year per system, respectively
 - The analysis was performed both with and without taking credit for enhanced inspection effectiveness due to an increased POD from application of the RI-ISI process.
 - In general, only those ASME Section XI Code inspection locations that received a volumetric examination in addition to a surface examination are included in the count. Inspection locations previously subjected to a surface examination only are not considered in accordance with Section 3.7.1 of EPRI TR-112657, with the exception of those inspection locations considered potentially susceptible to external chloride stress corrosion cracking.

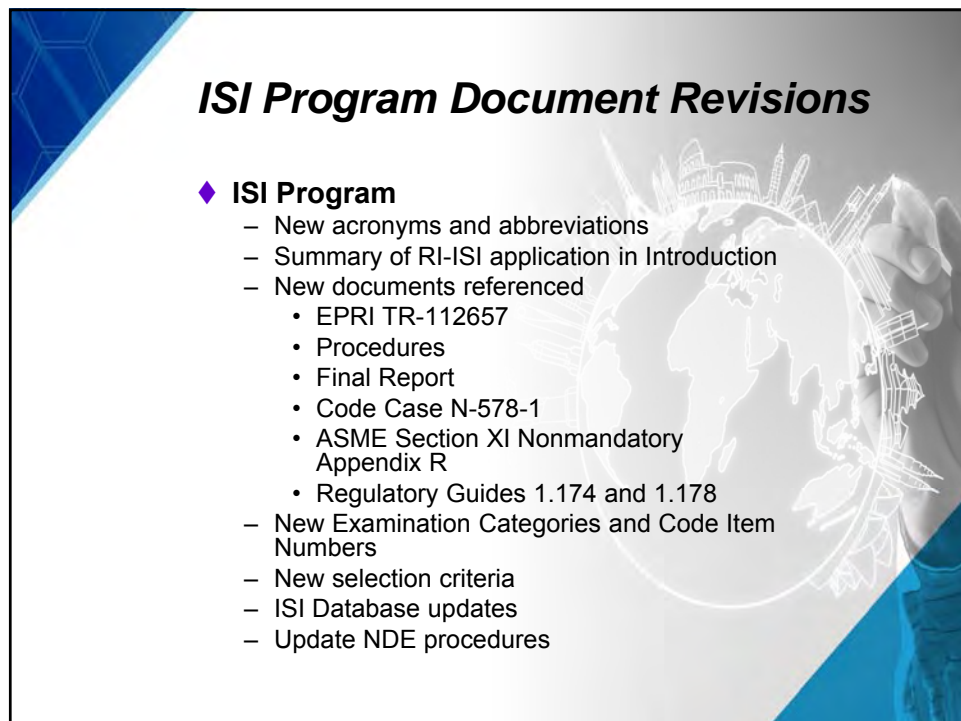
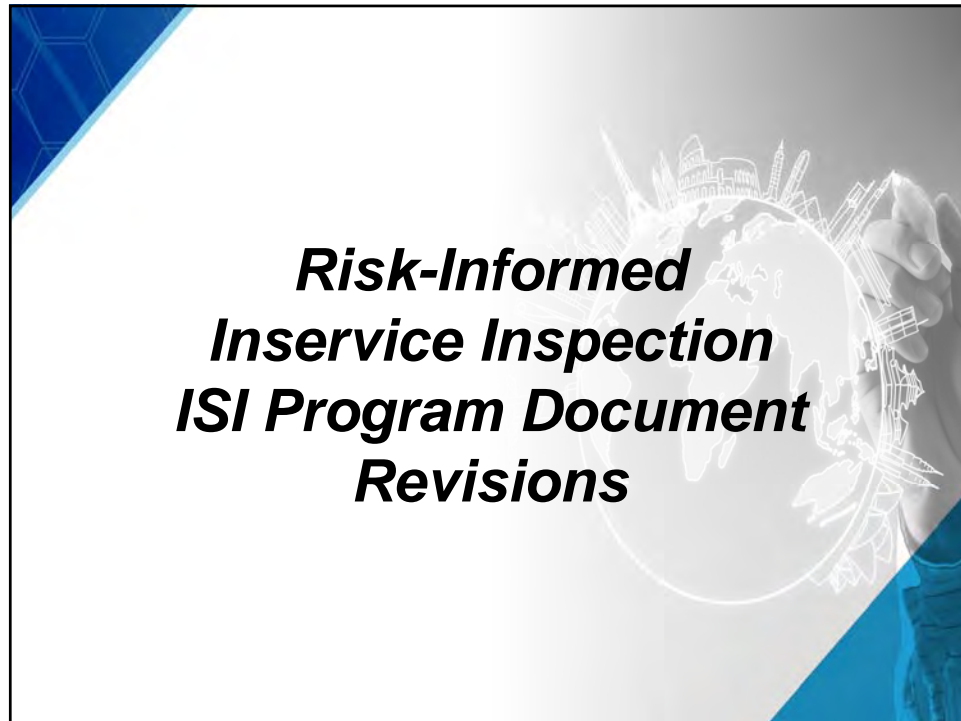


Generation of a Risk-Informed Inservice Inspection Template



RI-ISI Template Generation

- ◆ **Section 1: Introduction**
- ◆ **Section 2: Proposed Alternative to Current Inservice Inspection Programs**
- ◆ **Section 3: Risk-Informed ISI Process**
- ◆ **Section 4: Implementation and Monitoring Program**
- ◆ **Section 5: Proposed ISI Program Plan Change**
- ◆ **Section 6: References/Documentation**
- ◆ **Tables**



ISI Program Document Revisions

◆ Scheduling

- Must still meet period percentage requirements of IWB-2412 and IWC-2412
- Schedule examinations on welds that are near each other during the same outage
- Schedule examinations on welds that are near other components (e.g., supports, valves) that are being examined during the same outage
- Consider radiation exposure
- Consider "pre-outage" examinations
- Distribute examinations evenly between outages
- Consider when a weld was previously examined (e.g., which period was the weld examined during the previous interval)

Radiation Reduction and Cost Benefit

Cost and Radiation Reduction

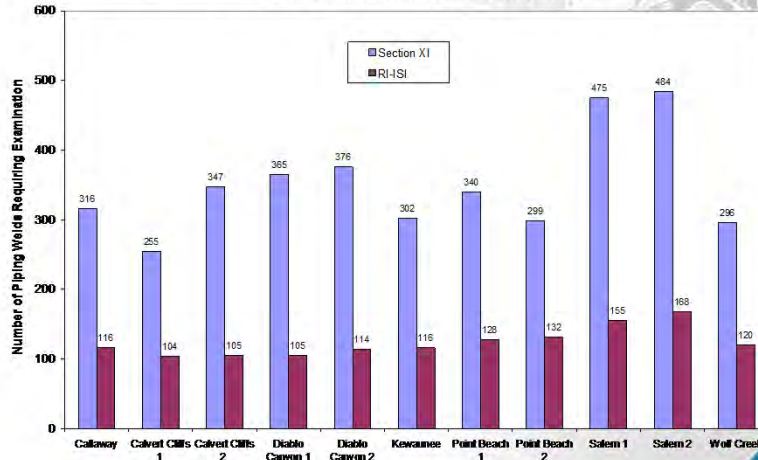
- ◆ **Estimated cost savings between US \$1M and US \$3.6M per unit per ten year interval for a Class 1 & 2 RI-ISI application**
- ◆ **Average cost savings estimated at US \$1.6M per unit per interval for a Class 1 & 2 application**
- ◆ **Cost savings based on average cost per weld of US \$7600 for examination, including the following activities:**
 - Erection and removal of scaffolding
 - Removal and replacement of insulation
 - Removal and replacement of interferences
 - Weld preparation
 - Examination
 - Documentation
 - Craft support
 - \$7600 per weld does not include “ALARA Penalty”

Cost and Radiation Reduction

- ◆ **Estimated reduction in radiation exposure of 75% to 90% for a Class 1 & 2 RI-ISI application**
 - Number of welds selected for examination decreases by 60% to 75%
 - Surface examinations essentially eliminated
 - Within a given risk category, welds can be selected for examination based on additional factors such as the minimization of radiation exposure
- ◆ **Cost and radiation exposure reduction figures similar for both BWRs and PWRs**

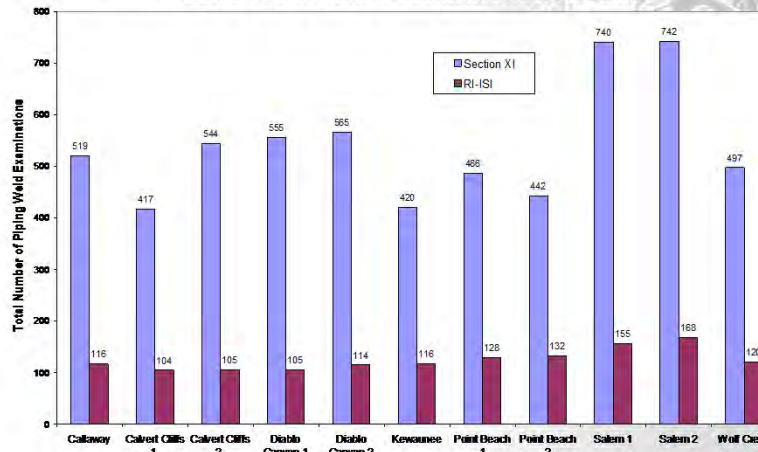
Reduction in Welds Examined

REDUCTION IN CLASS 1 AND 2 PIPING WELDS REQUIRING EXAMINATION EVERY TEN-YEAR INTERVAL AFTER A RISK-INFORMED APPLICATION



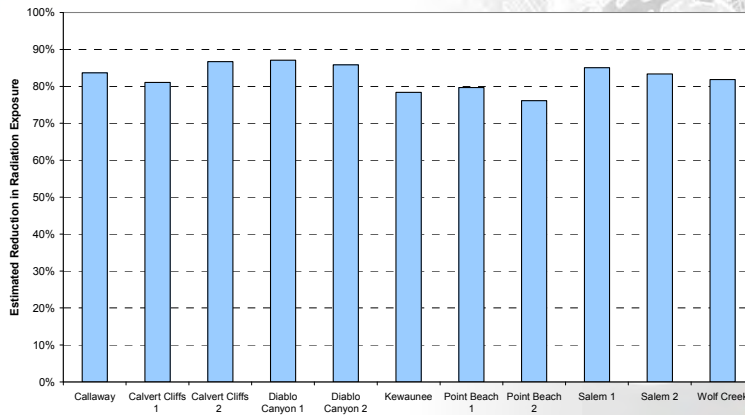
Reduction in Total Weld Examinations

REDUCTION IN CLASS 1 AND 2 PIPING WELD EXAMINATIONS (SURFACE AND VOLUMETRIC) EVERY TEN-YEAR INTERVAL AFTER A RISK-INFORMED APPLICATION



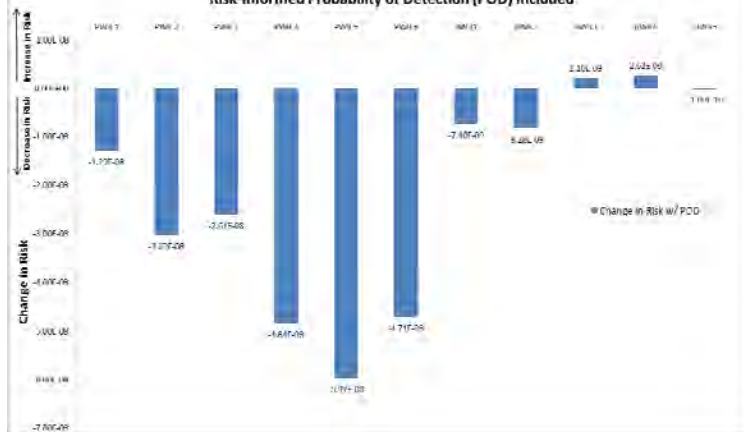
Reduction in Radiation Exposure

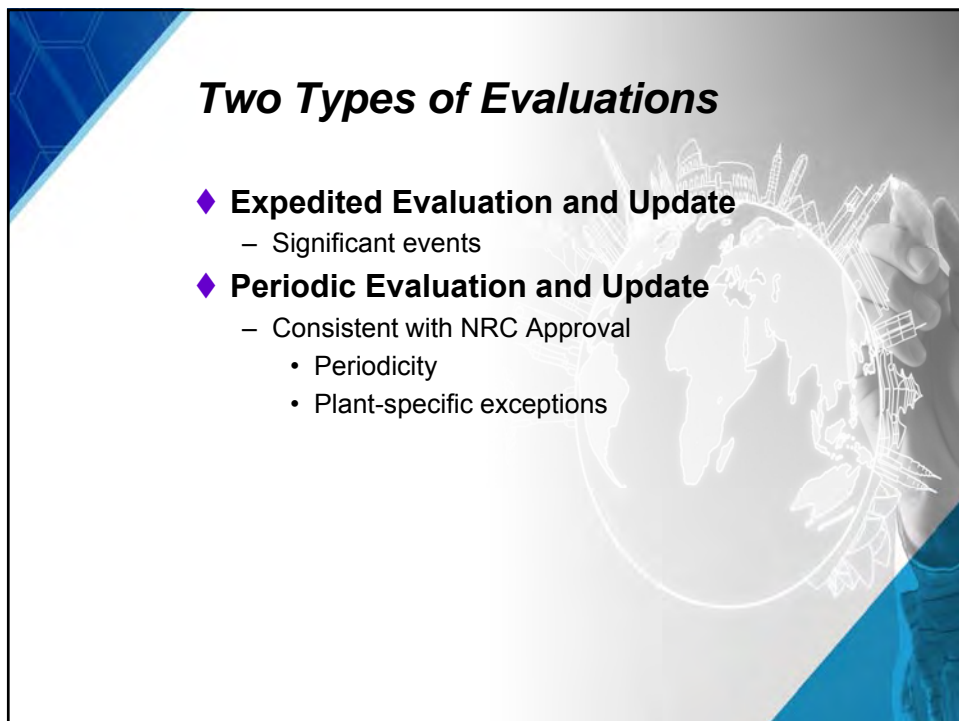
ESTIMATED REDUCTION IN RADIATION EXPOSURE FOR CLASS 1 AND 2 PIPING WELD EXAMINATIONS RESULTING FROM RISK-INFORMED APPLICATION



Change in Risk

Change in Risk Based on Core Damage Frequency (CDF) - Risk-Informed Probability of Detection (POD) Included





Components of a Living Program

- ◆ **Monitoring**
- ◆ **Evaluation**
 - **Plant Examination Results**
 - **Plant and Industry Piping Failures**
 - **Probabilistic Risk Assessment (PRA) Model Updates**
 - **Plant Design Changes**
 - **Plant Programmatic Changes**
 - **Changes in Postulated Conditions or Operations**
- ◆ **Update**
- ◆ **Notification**

Conclusions on RI-ISI Applications

- ◆ **Risk-informed methodology is a practical engineering approach for selecting locations for piping inservice inspections**
- ◆ **The methodology is highly reliable and considers both the consequences and potential for failures**
- ◆ **USNRC has approved numerous plant-specific applications as well as EPRI's generic Topical Report**
- ◆ **Application of the risk-informed methodology results in major reductions in inspections, radiation exposure and associated costs**
- ◆ **Requirements are in place to keep the RI-ISI Program up to date as a "Living Program"**



In-Service Inspection & Testing Practices in PHWR NPPs






By
Braham Parkash
NPCIL

Outline of presentation

- Introduction to Indian PHWRs
 - Introduction to In-Service Inspection Program
 - Various Degradation Mechanisms in PHWRs
 - Philosophy for ISI
 - ISI Interval
 - Systems Subject to Inspection
 - Magnitude of ISI for Identical Component for Inspection
 - Extend of System Subject to Inspection
 - Inspection Categories
 - Equipment, Inspection Methods, Techniques and Procedures
 - Acceptance Criteria
 - Inspection Requirements for Different Components
 - Challenges in ISI
 - Optimization of ISI Requirements
-

INDIAN PHWR PROGRAM

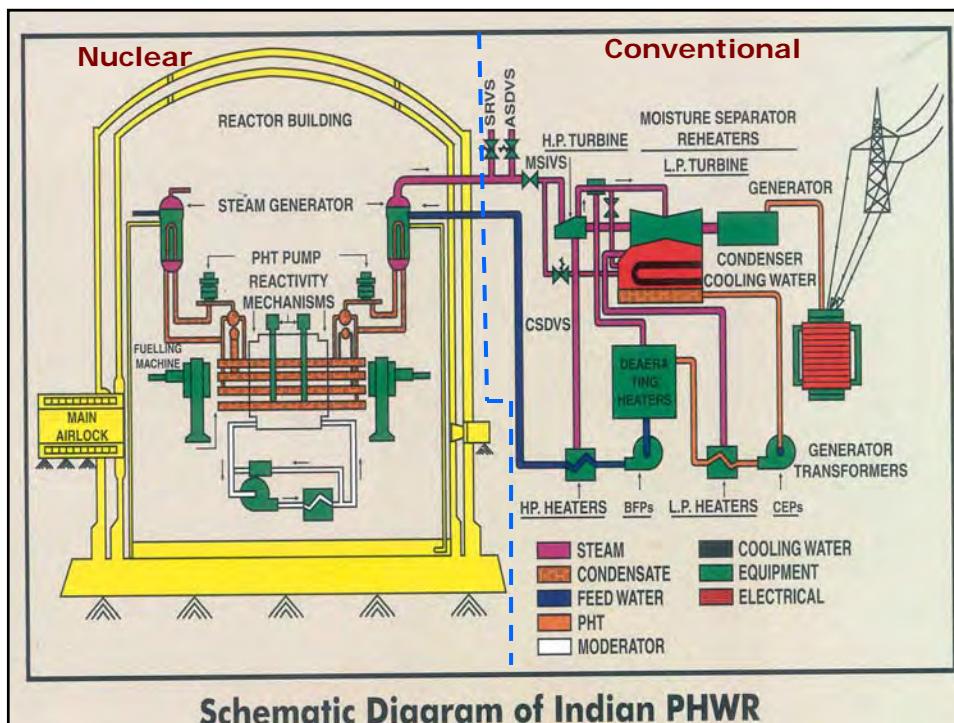
Operating PHWRs in India

| | | |
|---|---|--|
|  |  |  |
| RAPS-2 to 6 (220MWe) | MAPS-1&2 (220MWe) | NAPS-1&2 (220MWe) |
|  |  |  |
| KAPS-1&2 (220MWe) | KGS-1&4 (220MWe) | TAPS-3&4 (540MWe) |

PHWRs under Construction

- 2x 700MWe KAPP-3&4
- 2x 700MWe RAPP-7&8
- 2x 700MWe GHAVP-1&2

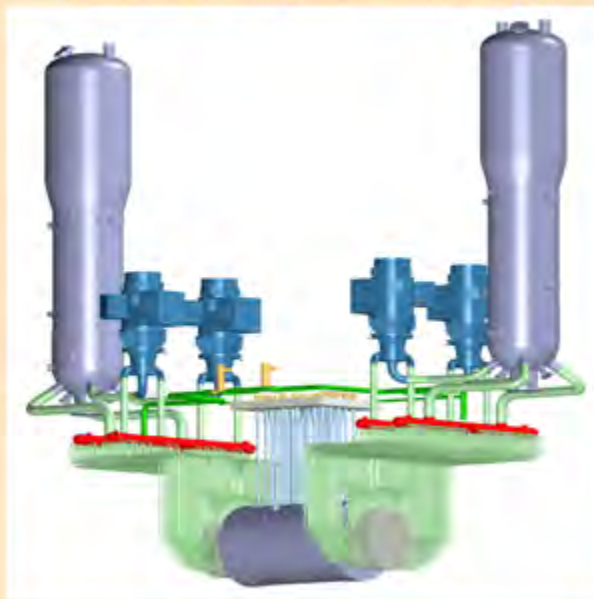
Ten 700MWe PHWRs under Fleet Mode

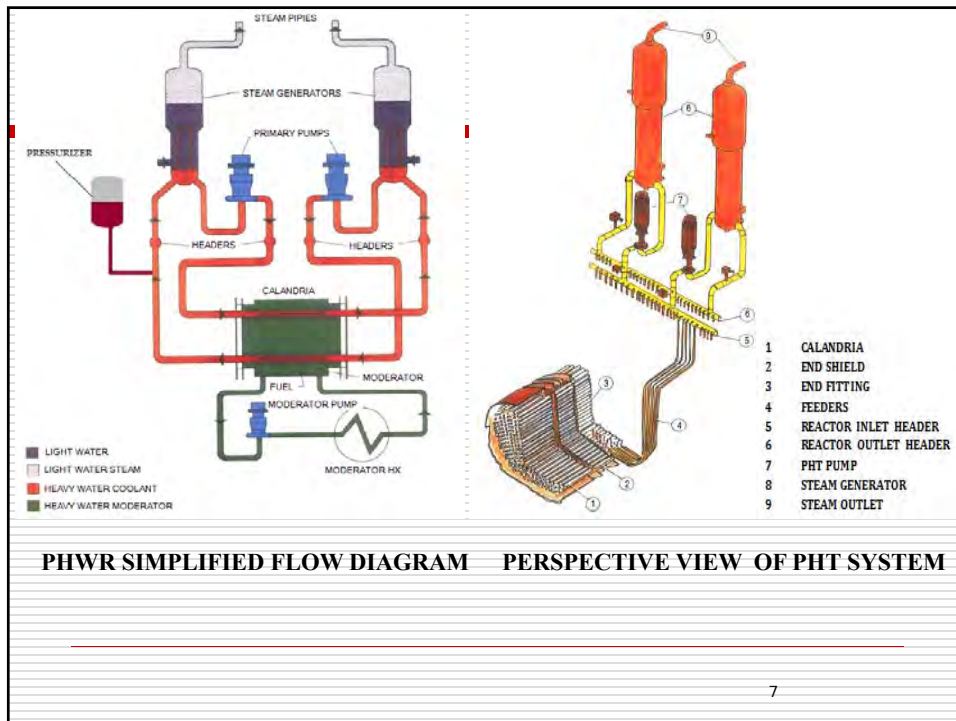


Introduction to Indian PHWRs : Design Features

- Horizontal Reactor Vessel – Calandria
- Pressure Tube Concept (306/392 channels)
- Natural Uranium Fuelled (Fuel pins; Bundles)
- Heavy Water Cooled and Moderated
- Calandria surrounded by water enclosed in a concrete structure –Calandria Vault
- Two Independent, Diverse and Fast-Acting Shut Down Systems
- Emergency Core Cooling through Passive Accumulators and Active Pumping System
- On-power Refueling
- Double Containment
- Suppression Pool/Containment Spray System

Pressurized Heavy Water Reactor System





Introduction to Indian PHWRs : Major Plant Systems

- Reactor Assembly (Core Components)
- Primary Heat Transport System
- Moderator System
- Reactivity Control & Shutdown Systems
- Reactor Auxiliaries
- Process Water Systems
- Process Control Systems
- Fuel Handling System
- TG & Secondary Cycle Systems
- Electrical System

Introduction to In-Service Inspection Program

In-Service Inspection (ISI)

The periodic inspection is intended primarily to ascertain any change in the structural integrity of Systems Structures and Components (SSCs) during the service life of the plant.

The objective of ISI is:

- to ensure SSCs are safe.
- to ensure their functionality whenever required.
- to ensure the safety of personnel and plant.
- to gain sufficient assurance against Unplanned Outages leading to Economic Loss and Un-planned Radiation Dose Consumption.

Pre-Service Inspection (PSI)

Inspection performed on SSCs before plant start-up for all areas intended to undergo subsequent inspection during ISI, is called Pre-Service Inspection (PSI). The result of the PSI of the components prior to commencement of operation establishes [the base line data](#), required for comparison during subsequent ISI.

Introduction to In-Service Inspection Program

Defence-in-Depth (DiD) Philosophy

DiD applied to all safety activities, provides protection against a wide variety of events resulting from equipment failure or human action, originating internally within the plant or externally

Level-1 – **Prevent** deviation from normal operation.

Level-2– **Detect and intercept** deviations from normal operational states to prevent AOOs from escalating to accident conditions

Level-3– **Control accidents** within the design basis. (Engineered safety features, maintaining at least one barrier for confinement)

Level-4– **Control severe accidents** (protection of confinement function)

Level-5– **Mitigate radiological consequences** (on-site and off-site emergency response)

ISI is under Level-I of Defence-in-Depth

Various Degradation Mechanisms in PHWRs

- Fatigue Damage
 - Irradiation damage to Material
 - Corrosion & Erosion
 - Flow Assisted Corrosion (FAC)
 - Vibration
 - Fretting Damage
 - Irradiation enhance damage
 - Axial Creep
 - Diametral Creep
 - Creep Change
 - Delay Hydride Cracking (DHC)
-

Philosophy for ISI

- The SSCs should be examined for possible deterioration so as to assess whether they are acceptable for continued safe operation of the plant or whether remedial measures should be taken.
 - The objective is to draw up on a logical basis detailed ISI Program covering entire service life of the plant to monitor the quality condition of various components.
 - While formulating the ISI program, limitations have been recognized with respect to
 - Present state-of-the-art of inspection in terms of instruments/equipment availability.
 - Radiation exposure to plant personnel: ALARA Principle
 - Most essential requirement for carrying out successfully any examination is **provision of adequate access and proper configuration**, which have to be taken care of right at the Design stage.
-

Philosophy for ISI

ISI Program in Indian PHWRs followed following Codes & Guide:

- AERB-SG-02, In-Service Inspection of NPPs
- CAN-CSA-N285.4, Canadian standard for Periodic Inspection of CANDU NPP Components.
- ASME BPV Section XI.

For any defect/ flaw found during ISI, the structural integrity assessment of the component/piping is carried out as per ASME-Section XI

Philosophy for ISI

- ISI Interval/Inspection period
 - Systems/Components Subject to Inspection
 - Magnitude of ISI for Identical Component for inspection
 - Extend of System Subject to Inspection
 - Inspection categories
 - Equipment, Inspection Methods, Techniques and procedures
 - Inspection Requirements for Different Components-Examples
 - PHT Piping
 - Steam Generator
 - Pressure Tube
 - Feeders
 - Pump & Valves
 - Piping & Equipment Supports
 - Snubbers
 - Dormant System
 - Acceptance criteria
-

Philosophy for ISI: ISI Interval

- The interval of ISI for each component is determined on a logical basis and not by a fixed time cycle and may vary from plant to plant based on operating history/ material characteristic etc.
- The approach has been taken that the initial period or 1st ISI interval may be of short duration in order to ensure conservatism and that if any, faster deterioration than expected is taking place, it would come to notice.
- Subsequent ISI interval may be varied by severity of operating conditions, results from earlier inspections, and any known abnormality observed during operation.
- Accordingly, the first ISI interval is 5 year period, commencing 1 year after the first synchronization of the unit. The inspection should be distributed evenly as far as practical.
- The subsequent periodic inspections shall be planned for time intervals that do not exceed 1/3rd of the design operational life of the plant or 10 years which is shorter.

Philosophy for ISI: ISI Interval

INSPECTION SCHEDULE FOR PRESSURIZED HEAVY WATER

| Inspection Interval | Inspection period indicated as calendar year of plant service from commencement of operation | Minimum percentage of examinations required to be completed | Maximum percentage of examinations credited |
|---------------------|--|---|---|
| 1st (5 years) | 0-2 | 16 | 34 |
| | 2-5 | 100 | 100 |
| 2nd (10 years) | 5-8 | 16 | 34 |
| | 8-12 | 50 | 67 |
| | 12-15 | 100 | 100 |
| 3rd (10 years) | 15-18 | 16 | 34 |
| | 18-22 | 50 | 67 |
| | 22-25 | 100 | 100 |
| 4th (10 years) | 25-28 | 16 | 34 |
| | 28-32 | 50 | 100 |
| | 32-35 | 100 | 100 |

Philosophy for ISI: Systems Subject to Inspection

SSCs subject to PSI/ISI include the following or portions thereof :

- a) The reactor coolant pressure boundary and other systems whose failure may result in a significant release of radioactive substances
 - b) Systems essential for the safe reactor shutdown and/or the safe cooling of the nuclear fuel or both in the event of process system failure; and
 - c) Other systems and components whose failure or dislodgment could jeopardize the integrity of systems as mentioned in item (a) or (b) above, or both (such as Process water system, flywheels, Snubbers etc.)
 - d) SSCs required for handling design extension conditions.
-

Philosophy for ISI: Systems Subject to Inspection

Various systems subjected to ISI

- Primary Heat Transport (PHT) System
 - Moderator system.
 - Shutdown Cooling system
 - Emergency Core Cooling System (ECCS)
 - Passive Decay Heat Removal System (PDHRS)
 - Primary Shutdown System/Shutdown System -1 (SDS-1)
 - Secondary shutdown system/Shutdown System -2 (SDS-2)
 - ALPAS/MLPAS
 - GRAB/LPIS
 - Main steam lines(Inside RB) , Feed water system (Inside RB)
 - Process and service water system
 - Fire water system
 - Diesel Generator (DG) fuel oil system
 - Annulus Gas Monitoring System (AGMS)
 - Containment Spray System (CSS)
 - Containment Filtration and venting system (CFVS)
 - Primary Containment Controlled Discharge (PCCD) System
 - Passive Accident Hydrogen Management System (PAHMS)
-

Philosophy for ISI: Components Subject to Inspection

- PHT System Piping
 - PHT Feeders
 - Steam Generators and Heat Exchangers
 - Pressure Tubes
 - PCP, PCP flywheel, Shutdown Cooling pumps
 - PHT System Boundary Valves
 - Fuelling Machine Components
 - Reactivity Devices
 - Piping/Equipment Supports.
-

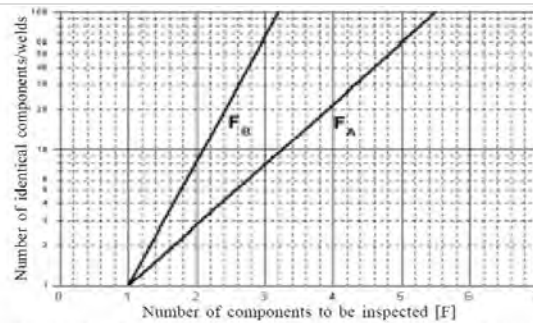
Philosophy for ISI: Magnitude of ISI for Identical Component for inspection

- ISI philosophy gives due consideration to the fact that **Similar Components** subject to essentially **Similar Operating conditions** will behave in **Similar manner**. Hence Representative sampling, which is rotated through similar components is used to ensure safety of the components.
 - This will further minimize the radiation exposure to inspection personnel. For identical welds of identical components, the number of welds to be inspected are reduced suitably
-

Philosophy for ISI: Magnitude of ISI for Identical Component for inspection

For identical components that are operated under similar conditions, following sampling requirements may be acceptable:

- For identical components in inspection category A areas, the number of identical components to be inspected should not be less than F_A .
- For identical components in inspection category B areas, the number of identical components to be inspected should not be less than F_B .
- Where the number determined by item (a) or (b) above includes a fraction greater than one-third, it shall be rounded up to the next integer.



INSPECTION SAMPLE FOR IDENTICAL COMPONENTS/WELDS

Philosophy for ISI: Extent of Systems/Components subjected to Inspection

- The inspection requirement applies to the fluid boundary portions of all components & piping and to their supports.
- All accessible supports falling under purview of PSI/ISI are visually inspected for any corrosion/damage during PSI and at least once in each ISI interval.

Philosophy for ISI: Inspection Categories

- The requirements of inspection, inspection areas and degrees of examination are determined by matrix consisting stress intensity ratio and fatigue usage factor based on size of failure.
- The two major factors that determine the category are stress intensity ratio and fatigue usage factor.

DETERMINATION OF INSPECTION CATEGORIES (A, B, C1, and C2)
FOR LARGE FAILURE SIZE.

| Stress Intensity Ratio | Low fatigue factor | Medium fatigue factor | High fatigue factor | |
|------------------------|--------------------|-----------------------|---------------------|-------------------------|
| | 0 | 0.01 | 0.1 | |
| $> 2/3$ | C1 | A | A | High stress intensity |
| $\leq 2/3$ | C1 | B | A | Medium stress intensity |
| $\leq 1/3$ | C2 | C1 | C1 | Low stress intensity |

Fatigue usage factor

Philosophy for ISI: Equipment, Inspection methods, Testing and procedures

Equipment

All equipment used for ISI are of acceptable quality, range, performance characteristics and accuracy in accordance with applicable codes/standards/approved procedures.

Inspection methods, Techniques and Procedures

- The inspection is carried out with written and approved qualified procedures.
- The methods and techniques employed are capable of yielding results to an acceptable Codes /Standard.
- For PSI/ISI of all critical components with complex geometry mock up trials is carried out to qualify the PSI/ISI procedures, inspection personnel and inspection equipment.

Philosophy for ISI: Equipment, Inspection methods, Testing and procedures

The different inspection methods employed during ISI are as follows:

- **Dimensional examinations**
 - **Visual Examination (VT)**

 - **Surface Examination:** A Surface examination is undertaken to determine discontinuities (i.e., surface, subsurface) by methods such as
 - Liquid Penetration (PT)
 - Magnetic Particle (MT) and
 - Eddy Current Examination (ET)

 - **Volumetric Examination:** A volumetric examination is undertaken to cover entire volume for flaw or discontinuities and usually involves
 - Radiography Examination (RT)
 - Ultrasonic Examination (UT) and
 - Eddy Current Examination (ECT) (for tubing).
-

Philosophy for ISI: Acceptance Criteria

- Acceptance standards for visual, surface and volumetric examinations are established before the start of the programme.
 - As-manufactured specific standards are taken as the basis for arriving at acceptable standards
 - For cases where the acceptance standards are not in existence or are not relevant to the situation, acceptance standards are established in consultation with the regulatory body.

 - **Additional Examinations:**
 - When a flaw exceeding the acceptance standards is found in a sample, additional examinations are performed to include the specific problem area in an additional number of analogous components (or areas) approximately equal to the number of components (or areas) examined in the sample.
 - For multi-unit stations with identically designed plants, the area corresponding to that having the indication are inspected on the identical component in each reactor unit, immediately following the next reactor cool down.
-

Philosophy for ISI: Inspection Requirements for Different Components

Inspection requirements for different inspection categories :

1) Category -A -Inspection Requirements

a) Piping weld joints:

- i. At least one joint in each pipe run. The joint having the highest fatigue usage factor is selected.
- ii. Where fatigue usage factors are not calculated, the weld joint having highest stress ratio is selected.
- iii. Representative weld joints between dissimilar materials, if any.
- iv. Terminal weld joints.
- v. In addition to this, where any recordable indication obtained during /PSI/ earlier ISI campaign.
- vi. Minimum Quantum - 25% of Total welds

b) Vessels:

All pressure retaining welds are inspected except that the number of identical welds to be inspected may be suitably reduced.

Philosophy for ISI: Inspection Requirements for Different Components

C) Mechanical Couplings:

- i. All bolting
- ii. All ligaments between threaded stud holes.
- iii. Visual examination of size $\leq 1''$ & Visual and surface examination for size $>1''$

d) Supports:

- i. All supports.
- ii. All components attachment welds.
- iii. All snubber.

e) Rotating Machinery (e.g. Flywheels): All regions shall be inspected.

j) Pumps & Valves : All pressure-retaining welds are inspected. Visual and surface examination of internal surface of pump (at least one pump of each type). Surface examination is to be carried out, if material is susceptible to Stress Corrosion Cracking (SCC). All fasteners shall be examined.

Philosophy for ISI: Inspection Requirements for Different Components

2) Category B: Inspection Requirements:

Piping:

- i. For each pipe run having one or more category-B regions, the joint having the highest fatigue usage factor are inspected, if the pipe run has a category- A region, no further inspection will be required.
- ii. Where fatigue usage factors are not calculated, the joint having the highest stress ratio is selected for inspection.
- iii. Representative weld joints between dissimilar materials.
- iv. Terminal weld joints.
- v. In addition to this, where any recordable indication obtained during /PSI/earlier ISI campaign is additionally inspected along with scheduled quantum for next three successive ISI campaigns and if no growth of the indications observed, the original ISI program is followed.
- vi. Minimum Quantum - 7.5 % of Total welds

Vessels: All pressure retaining welds shall be inspected except that the number of identical welds to be inspected is suitably reduced.

Philosophy for ISI: Inspection Requirements for Different Components

Mechanical Couplings:

- i. Bolting - 10% of the total number of fasteners in the joint, to the next higher integer, is inspected;
- ii. Flange Ligaments - 10% of the flange ligaments between threaded stud holes, to the next higher integer, is inspected; and
- iii. Visual examination of size $\leq 1"$. Visual and surface examination for size $> 1"$

Supports:

- i. All supports
- ii. Where a supports has one or more component attachment welds, at least one weld that having the highest fatigue usage factor is inspected. Where fatigue usage factor is not calculated, the component attachment weld having highest stress is selected.

Rotating Machinery:

The region that is defined as category B and has the highest stress in this category is inspected, except that if the component has a category A region, no further inspection is required.

Philosophy for ISI: Inspection Requirements for Different Components

Pumps & Valves : Pressure-retaining welds having highest stress ratio is inspected.(If applicable). Visual and surface examination of internal surface of pump (at least one pump of each type). Surface examination is required only for material subject to stress corrosion cracking. All fasteners shall be examined.

3) Category C1:

No ISI are required for material that is in this category, provided that it is not composed of dissimilar metals.

4) Category C2:

No ISI is required for material that is in this category. However piping between 1 “ to 4” which are connected to reactor coolant pressure boundary though fall under category C2, one representative weld joint shall be included in the scope of ISI and subjected for visual and surface examinations.

Philosophy for ISI: Inspection Requirements- Examples

ISI Requirements for PHT system Piping

| S No | Description (Line Nos.) | Parts to be examined | | Insp Cat. | Insp. Area | Insp. method | Insp. Interval 1 st /subsequent ISI | Total No. of weld joints/Unit Quantum. | | Minimum quantum in the scope of ISI. | |
|------|---|-----------------------------|----------------------------|-----------|--|--------------|--|--|-----------------|--------------------------------------|-----------------|
| | | From | To | | | | | TAPS-3 | TAPS-4 | TAPS-3 | TAPS-4 |
| 1 | Steam Generator (SG) inlet pipe. 3311-1,2,6,7, 11, 12,16,17 | Reactor Outlet Header (ROH) | SG inlet | A | SG inlet nozzle weld joint. And terminal weld joint | 1,3A,3B | 05/10 Yrs | 48 weld joints. | 48 weld joints. | 24 weld joints | 26 weld joints |
| 2 | Pump suction piping. 3311-3, 3311-8 3311-13, 3311-18 | SG outlet | Pump suction | A | PCP suction nozzle weld joint. And terminal weld joint | 1,3A,3B | 05/10 Yrs | 20 weld joints. | 20 weld joints. | 16 weld joints | 16 weld joints |
| 3 | Pump discharge piping. 3311-4,5,9,10,14,15, 19,20 | Pump discharge | Reactor inlet header (RIH) | A | PCP discharge nozzle weld joint. And terminal weld joint | 1,3A,3B | 05/10 Yrs | 24 weld joints. | 24 weld joints. | 16 weld joints. | 16 weld joints. |
| 4 | Shut down (S/D) cooling piping. 3341-1,4,7,10 | ROH | 1 st Tee | A | Terminal weld joints. | 1,3A,3B | 05/10 Yrs | 08 weld joints | 08 weld joints | 04 weld joints | 04 weld joints |
| | | 1 st tee | S/D cooling pump suction. | | weld joint of 3 rd elbow with pipe joining isolation valve. And terminal weld joints. | | | 28 weld joints | 28 weld joints | 12 weld joints | 12 weld joints |

1- Visual, 3A- Surface Examination (LPT,MPT), 3B- Volumetric examination (UT)

Philosophy for ISI: Inspection Requirements- Examples

Steam Generator:

Inspection- category: A



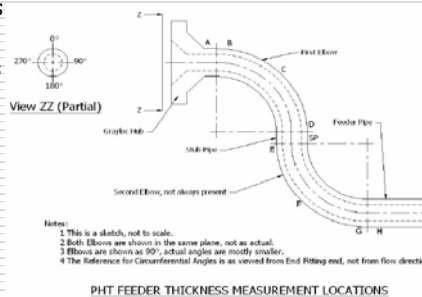
| Inspection Area | Inspection Method |
|---|---|
| SG Tubes | <ul style="list-style-type: none"> ECT of tubes using ID probe, As a minimum of 20% of each SG shall be subjected for ECT in one ISI interval. Selection of tubes from specific & random sample |
| Channel and shell side weld joints including Nozzle to vessel weld joints | Visual Examination + Surface Examination + Volumetric Examination |
| Equipment Fasteners | Visual Examination + Surface Examination |
| Support component | Visual Examination + Dimensional Examination |
| Support component attachment weld | Visual Examination + Surface Examination |

SG Tube Removal for Metallurgical Examination

Philosophy for ISI: Inspection Requirements- Examples

PHT Feeders: Based on size of failure criteria, feeders fall under category C2. For category C2 no ISI is called for. However based on degradation observed, following examinations are carried out:

- UTG of representative selected feeder elbows
- Visual, surface and volumetric examination on representative selected feeders & Weld Joints.
- Visual, surface and volumetric examination of weld joints at header (Header to Feeder stub Weld Joints) on representative selected feeders.
- UT thickness gauging of pipe portion minimum of 300mm downstream of orifice/ventures
- UT Thickness gauging of 1st bend from header end on representative selected feeders.



Philosophy for ISI: Inspection Requirements- Examples

Pump & Valves:

- Examinations are limited to at least one pump & one valve within each group of pumps/valves performing similar function in the system.
 - Extent of examination for pump/valves includes:
 - Visual , Surface Examination & Volumetric Examinations of Casing Weld Joints (if any)
 - Visual Examination of Internal Fluid Boundary of pump/valve casing . Surface examination is required only for material subject to stress corrosion cracking.
 - Visual & surface examinations of pump/valve fasteners.
 - Volumetric examination of PCP flywheel.
-

Philosophy for ISI: Inspection Requirements- Examples

ISI requirements for Equipment & Piping Supports:

- The inspection area of a support shall include the whole support and shall include the support settings of constant and variable spring-type hangers, Snubbers, and shock absorbers;
- Component Attachment Welds

Inspection Method:

- Visual + Dimensional (Wherever Applicable)
 - Surface examination
-

Philosophy for ISI: Inspection Requirements- Examples

Snubber:

- i. Visual Examination test
- ii. Free operability test
- iii. Drag Force Measurement
- iv. Sensitivity and lock test

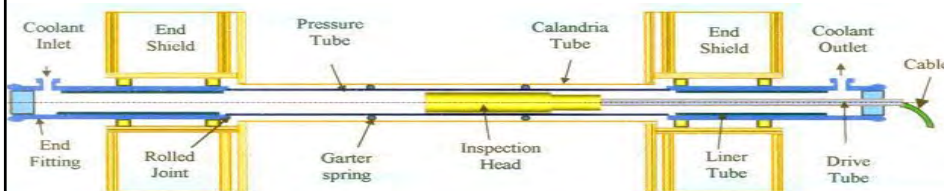
Snubber support structure:

- a) Visual examination of all snubber support structure.
- b) Visual and surface examination of all welds connecting snubber bracket to snubber support structure.

Philosophy for ISI: Inspection Requirements- Examples

Pressure Tubes

- Various possible Degradation Mechanisms in Pressure tubes are
 - Creep: Axial and Diametrical
 - Hydride Blister
 - Material properties degradation due to Irradiation.
- Pressure tubes need to be monitored at various stages during reactor operation for their ID, thickness, presence of flaws, garter spring position etc.
- These inspections are carried out using a ingeniously developed tool- BARCIS (BARC Channel Inspection System) which is a semi-automated remote controlled channel inspection system for ISI of pressure tubes.



Philosophy for ISI: Inspection Requirements - Examples

ISI requirements for Dormant systems:

Dormant Systems are those system which are normally not functioning and required to operate on demand.

- Safety related Dormant systems are subjected to ISI.
 - The inspection interval for such systems is less than the interval corresponding to inspection interval of the non-passive systems.
 - Example of such systems is ECCS, GRAB, CSS, CFVS, PCCD, PCRD and PAHMS.
 - For such systems 1st ISI interval is 04 years and subsequent ISI interval shall be 8 Years.
-

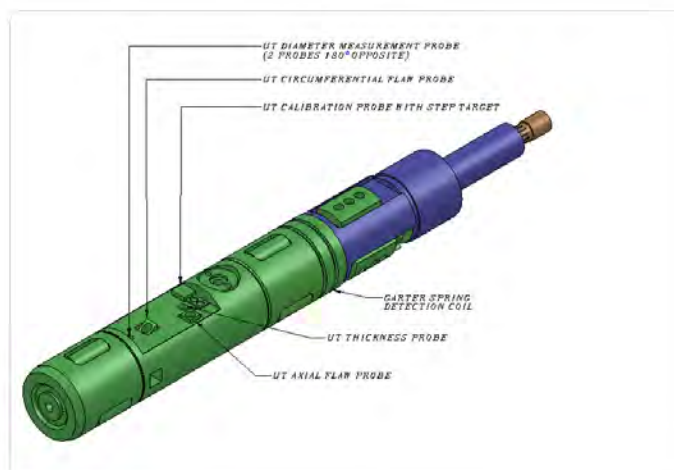
Challenges in ISI

- Radiation Environment-Man-Rem Consumption
 - Provisions to enable examinations remotely to reduce radiation exposure : Special Tooling/Manipulators requirement
 - e.g.
 - BARCIS For Pressure Tube Inspection
 - ManPos for ECT of SG Tubes
 - Access Limitation: Accessibility to areas and feasibility of the examination of components
 - Radiation Shielding Considerations: to meet ALARA requirements
 - Adequate space in the plant layout for installation, handling, removal, disassembly, reassembly, placing and mounting of inspection equipment and/or probes
 - Provision of Decontamination Facilities
 - Provision of test coupons for assessing ageing effects of various operating conditions such as load, temperature, radiation etc., on material properties e.g. NDTT, strength, fracture toughness, creep rate, and corrosion rate.
-

BARCIS Tool

- Pressure tubes need to be monitored at various stages during reactor operation for their ID, thickness, presence of flaws, garter spring position etc. These inspections are carried out using a indigenously developed tool-BARCIS (BARC Channel Inspection System)
 - BARCIS inspection head contain various NDE sensors :
 - Ultrasonic sensors for measurement of dimensions of pressure tube- ID and Thickness
 - Ultrasonic detection of flaws in longitudinal and circumferential directions on ID as well as OD
 - ECT to detect garter spring positions
 - ECT to estimate annular gap between Pressure Tube and Calandria Tube
-

BARCIS Tool



Steam Generator

ECT of SG Tubes: ManPOS



Optimization of ISI Requirements

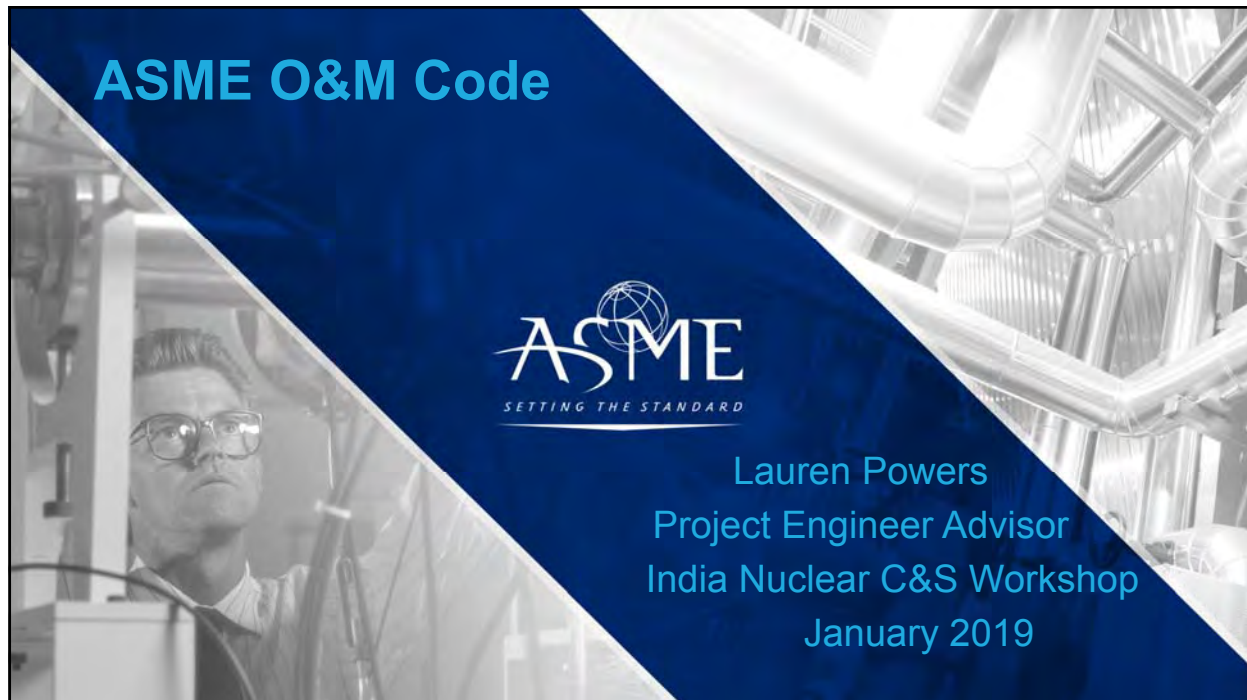
- Suitable Provisions in Design
 - Reduction in nos. of Weld Joints
 - Reduction of Active Components viz., Snubbers etc.

- Operation Feed Back

“ The optimization of ISI requirements has to be a continuously evolving process taking into account the operational experience of reactors worldwide as well as within the country without jeopardising safety “

Thank you

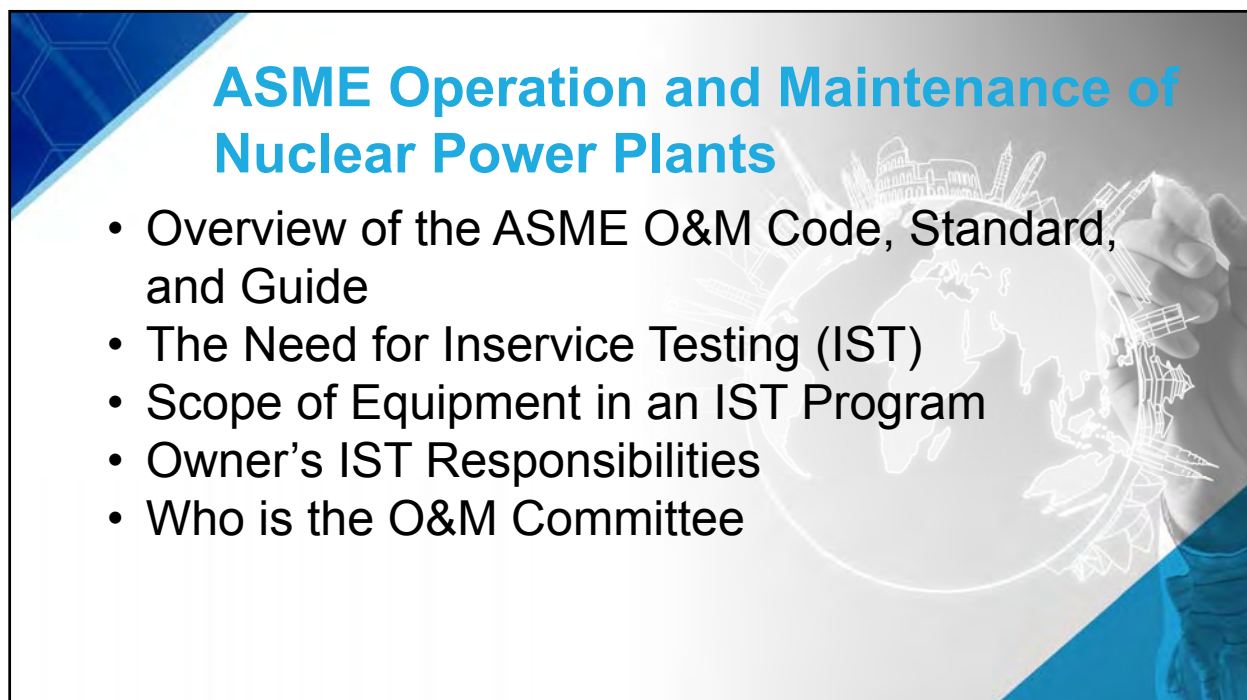


A presentation slide for the ASME O&M Code. The slide features a dark blue background with a diagonal split. The left side shows a grayscale image of a man wearing glasses and a white lab coat, looking upwards. The right side shows a grayscale image of a complex industrial piping system. The ASME logo, consisting of a globe and the text 'ASME SETTING THE STANDARD', is centered. Below the logo, the text 'Lauren Powers Project Engineer Advisor India Nuclear C&S Workshop January 2019' is displayed in a light blue font.

ASME O&M Code

ASME
SETTING THE STANDARD

Lauren Powers
Project Engineer Advisor
India Nuclear C&S Workshop
January 2019

A presentation slide titled 'ASME Operation and Maintenance of Nuclear Power Plants'. The slide has a light gray background with a blue geometric pattern in the top-left corner. The title is in a large, bold, blue font. Below the title is a bulleted list of five items. In the background, there is a faint image of a hand holding a pen over a globe with technical drawings overlaid on it.

ASME Operation and Maintenance of Nuclear Power Plants

- Overview of the ASME O&M Code, Standard, and Guide
- The Need for Inservice Testing (IST)
- Scope of Equipment in an IST Program
- Owner's IST Responsibilities
- Who is the O&M Committee

ASME OM Divided into Three Divisions

- Division 1 – Code
- Division 2 – Standard
- Division 3 – Guide

Division 1: OM Code Section IST

Subsections:

- ISTA General Requirements
- ISTB Pump IST Requirements (before 2000)
- ISTC Valve IST Requirements
- ISTD Snubber IST Requirements
- ISTE Risk-Informed IST Requirements
- ISTF Pump IST Requirements (after 2000)

Division 1: OM Code Section IST

Mandatory Appendices:

- I Inservice Testing of Pressure Relief Devices
- II Check Valve Condition Monitoring Program
- III Preservice and Inservice Testing of Active Electric Motor-Operated Valve Assemblies
- IV Preservice and Inservice Testing of Pneumatically Operated Valve Assemblies
- V Pump Periodic Verification Test Program

Division 2: OM Standard

- Part 3 Vibration Monitoring of Piping Systems
- Part 12 Loose Parts Monitoring
- Part 16 Performance Testing and Inspection of Diesel Drive Assemblies
- Part 21 Inservice Performance Testing of Heat Exchangers
- Part 24 Reactor Coolant and Recirculation Pump Condition Monitoring
- Part 26 Determination of Reactor Coolant Temperature from Diverse Measurements
- Part 28 Standard for Performance Testing of Systems
- Part 29 Alternative Treatment Requirements for RISC-3 Pumps and Valves

Division 3: OM Guide

- Part 5 Inservice Monitoring of Core Support Barrel Axial Preloads
- Part 7 Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems
- Part 11 Vibration Testing and Assessment of Heat Exchangers
- Part 14 Vibration Monitoring of Rotating Equipment
- Part 19 Preservice and Periodic Performance Testing of Pneumatically and Hydraulic Operated Valve Assemblies
- Part 23 Inservice Monitoring of Reactor Internals Vibration

Preventive Maintenance Process

- Scheduled Maintenance
- Condition Maintenance
- Overhaul Maintenance
- Predictive Maintenance
- ***Inservice Testing***
- Inservice Inspection

Important to Keep in Mind the Functions of Safety Systems

- Shutting down the reactor to the safe shutdown condition
- Maintain the safe shutdown condition
- Mitigate the consequences of an accident

Intent or Rationale for Inservice Testing (IST)

Ensure operational readiness of Safety Systems by periodically testing the functionality of the active components (pumps and valves) of these systems.

ASME OM Code Definitions

- Inservice Test:

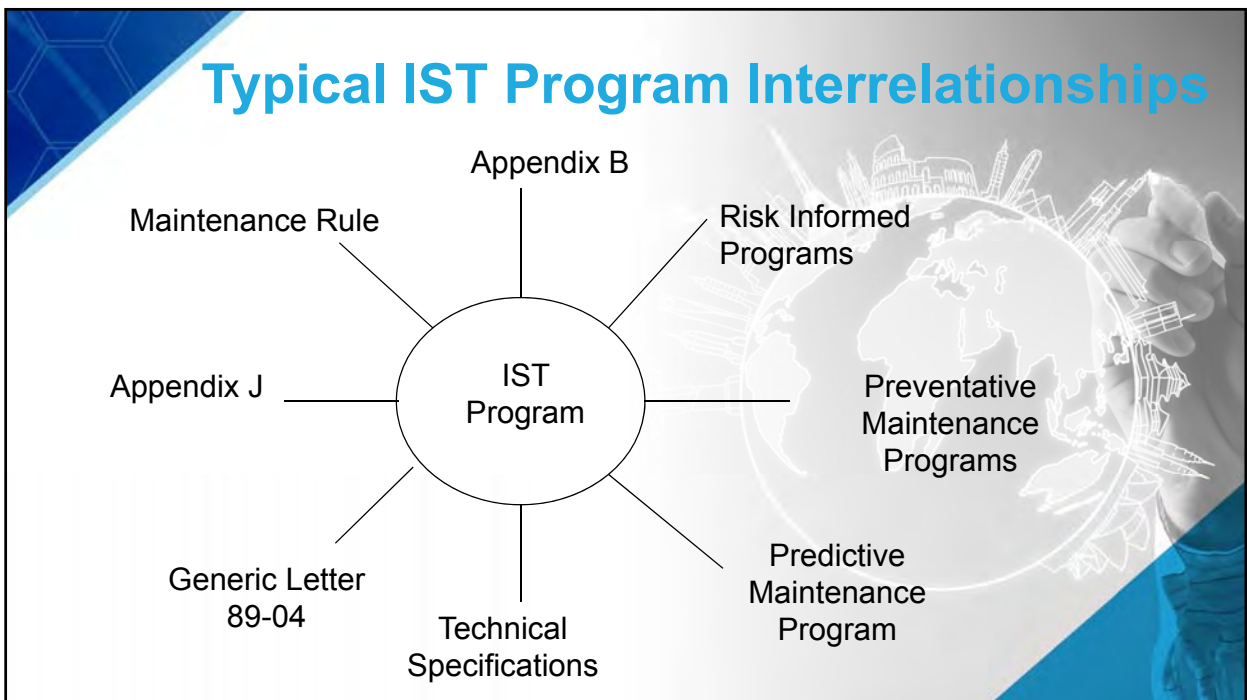
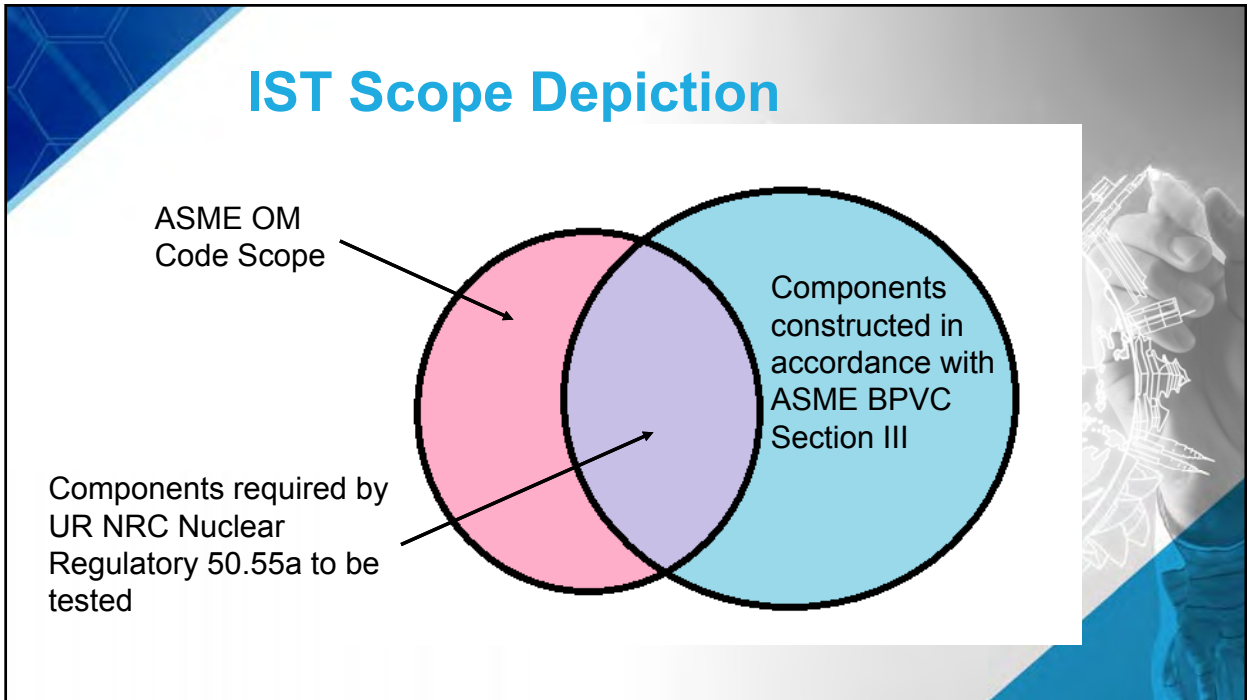
“A test to determine the operational readiness of a system, structure, or component after first electrical generation by nuclear heat.”

- Operational Readiness:

“The ability of a component to perform its specified functions.”

Establishing IST Program Scope

The scope of the IST program must include ASME Code Class 1, 2 and 3 components provided with an emergency power source, that are required in shutting down a reactor to the safe shutdown condition, maintaining the safe shutdown, or mitigating the consequences of an accident.



ASME OM Code ISTA-1100: Scope

- Section IST establishes the requirements for preservice and inservice testing and examination of certain components to assess their operational readiness in light-water reactor nuclear power plants.
- Identifies the components subject to test or examination, responsibilities, methods, intervals, parameters to be measured and evaluated, criteria for evaluating the results, corrective action, personnel qualification, and record keeping.

ISTA-1100: Scope (cont.)

These requirements apply to:

- (a) Pumps and valves that are required to perform a specific function in shutting down a reactor to the safe shutdown condition, in maintaining the safe shutdown condition, or in mitigating the consequences of an accident;
- (b) Pressure relief devices that protect systems or portions of system that perform one or more of these three functions; and
- (c) Dynamic restraints (snubbers) used in systems that perform one or more of these three functions

ISTA-1500: Owner's Responsibilities

The responsibilities of the Owner of the nuclear power plant shall include the following:

- (a) Determination of the appropriate Code Class for each component of the plant, identification of the system boundaries for each class of components subject to test or examination, and the components exempt from testing or examination requirements;
- (b) Design and arrangement of system components to include allowance for adequate access and clearance for conduct of tests and examinations;

ISTA-1500: Owner's Responsibilities (cont.)

- (c) Preparation of plans and schedules;
- (d) Preparation of written test and examination instructions and procedures;
- (e) Qualification of personnel who perform and evaluate examinations and tests in accordance with the Owner's quality assurance program;
- (f) Performance of required tests and examinations;
- (g) Recording of required test and examination results that provide a basis for evaluation and facilitate comparison with the results of subsequent tests or examinations;

ISTA-1500: Owner's Responsibilities (cont.)

- (h) Evaluation of tests and examination results;
- (i) Maintenance of adequate test and examination records such as test and examination data and description of procedures used; and
- (j) Retention of all test and examination records for the service lifetime of the component of system

Who is the O&M Committee?

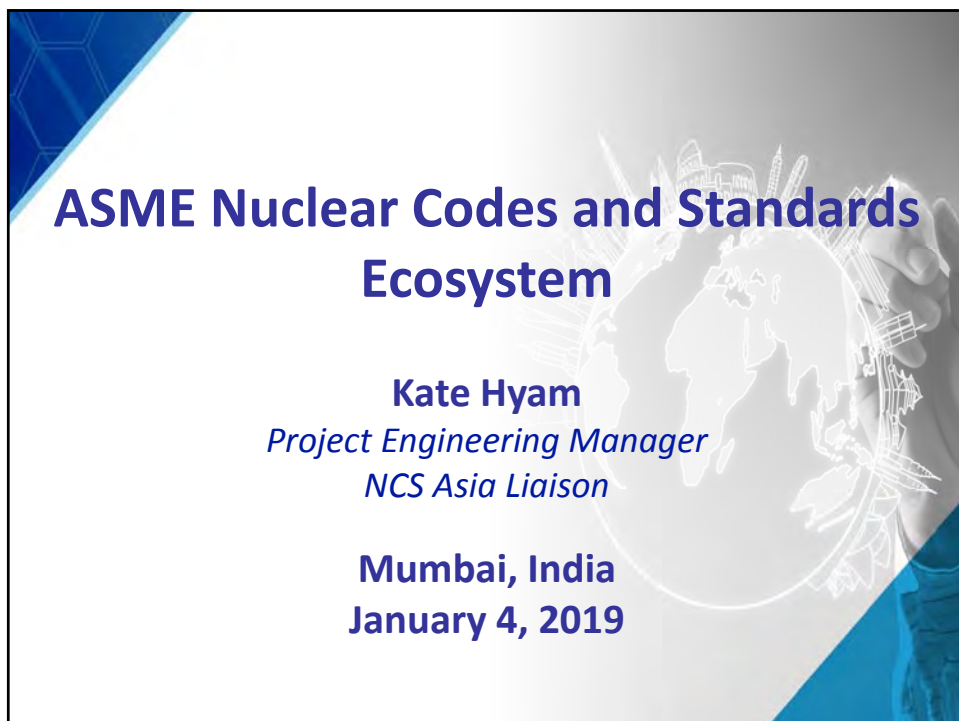
- The Main Standard Committee consists of 30 members
- 100+ volunteers who help develop the O&M Code
- Every Utility and NPP in the US
- Westinghouse, NuScale Power, and NTS (formerly Wyle Laboratories)
- International participants from Japan, Korea, and Spain
- NRC representative on the standards committee and all subcommittees and subgroups

O&M Committee Interaction

- ISTOG (IST Owner's Group)
- SNUG (Snubber's Users Group)
- MUG/AUG (MOV/AOV User's Group)
- NEI (Nuclear Energy Institute)
- INPO (Institute Nuclear Power Operations)

Questions?

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PowersL@asme.org
(212) 591-7008



Overview

- I. ASME Nuclear Codes and Standards
- II. Nuclear Certification Programs
- III. Learning and Development
- IV. Conferences/Symposia
- V. Other Publications
- VI. References

I. ASME Nuclear Codes and Standards

Nuclear Codes and Standards Committees

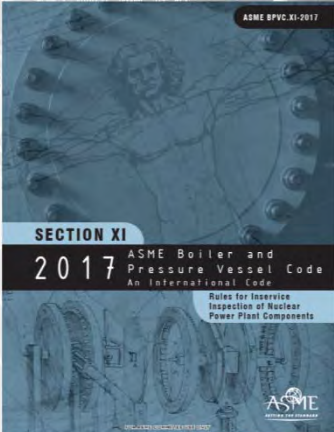
| | |
|---|--|
| BPV Committee on Construction of Nuclear Facility Components (III) | BPV Committee on Nuclear In-service Inspection (XI) |
| Standards Committee on Nuclear Quality Assurance (NQA) | Standards Committee on Operation and Maintenance of Nuclear Power Plants (O&M) |
| Standards Committee on Qualification of Mechanical Equipment Used in Nuclear Facilities (QME) | Standards Committee on Cranes for Nuclear Facilities (CNF) |
| ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM) | Standards Committee on Nuclear Air and Gas Treatment (CONAGT) |

5

ASME BPVC Section XI

Rules for Inservice Inspection of Nuclear Power Plant Components

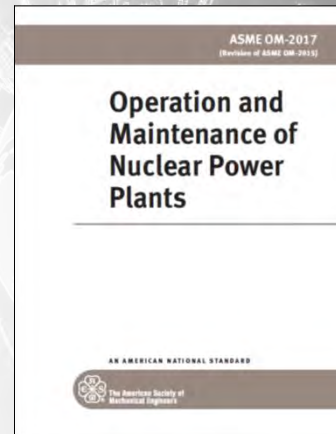
- **Division 1**
 - Rules for Inspection and Testing of Components of **Light-Water-Cooled** Plants
- **Division 2**
 - Rules for Inspection and Testing of Components of **Gas-Cooled** Plants
 - Currently being revised
- **Division 3**
 - Rules for Inspection and Testing of Components of **Liquid-Metal-Cooled** Plants



The image shows the cover of the ASME BPVC Section XI 2017 code book. The cover features a technical drawing of a nuclear reactor component. Text on the cover includes 'ASME BPVC XI-2017', 'SECTION XI', '2017', 'ASME Boiler and Pressure Vessel Code An International Code', and 'Rules for Inservice Inspection of Nuclear Power Plant Components'. The ASME logo is visible in the bottom right corner.

Code for Operation and Maintenance of Nuclear Power Plants (OM-2017)

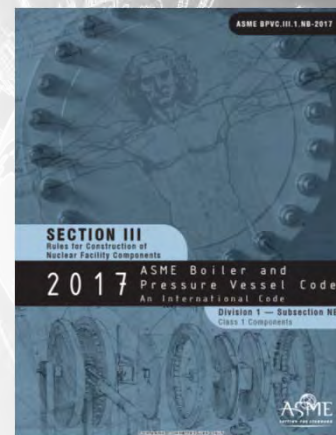
- 3 Divisions
- Establishes the requirements for preservice and inservice testing and examination of certain components to assess their operational readiness in light-water reactor power plants.
- Identifies the components subject to test or examination, responsibilities, methods, intervals, parameters to be measured and evaluated, criteria for evaluating the results, corrective action, personnel qualification, and record keeping.



ASME BPVC Section III *Rules for Construction of Nuclear Facility Components*

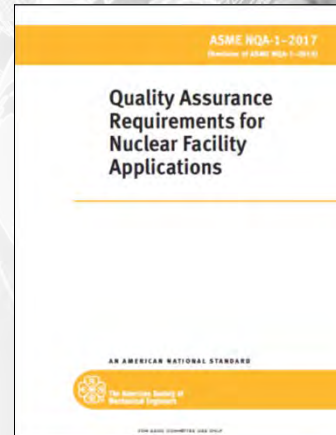
Section III establishes rules of safety relating only to **pressure integrity**, which governs the **construction** of boilers, pressure vessels, transport tanks, nuclear components **and their supports**.

This will be covered in detail during tomorrow's Workshop.



Quality Assurance Requirements for Nuclear Facility Applications (NQA-1)

This Standard provides requirements and guidelines for the establishment and execution of quality assurance programs during siting, design, construction, operation and decommissioning of nuclear facilities. This Standard reflects industry experience and current understanding of the quality assurance requirements necessary to achieve safe, reliable, and efficient utilization of nuclear energy, and management and processing of radioactive materials.



Qualification of Active Mechanical Equipment Used in Nuclear Facilities (QME-1 - 2017)

- Describes the requirements and guidelines for qualifying mechanical equipment, such as pumps, valves, and dynamic restraints, used in nuclear facilities. The requirements and guidelines presented include the principles, procedures, and methods of qualification.



QME-1 – 2017 Sections

- **Section QR:** General Requirements
- **Section QDR:** Qualification of Dynamic Restraints
- **Section QP:** Qualification of Active Pump Assemblies
- **Section QV:** Qualification Requirements for Active Valve Assemblies for Nuclear Facilities
- **Section QVG:** Guide to Section QV: Determination of Valve Assembly Performance Characteristics

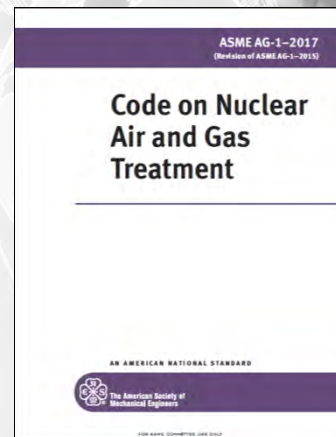
Standards Committee Qualification of Mechanical Equipment Used in Nuclear Facilities (QME)

- **QME Subcommittees:**
- SC on General Requirements
- SC on Qualification of Active Dynamic Restraints
- SC on Qualification of Pump Assemblies
- SC on Qualification of Valve Assemblies
- **QME China International Working Group**



Code on Nuclear Air and Gas Treatment (AG-1 – 2017)

Provides requirements for the performance, design, construction, acceptance testing, and quality assurance of equipment used as components in nuclear safety-related air and gas treatment systems in nuclear facilities.

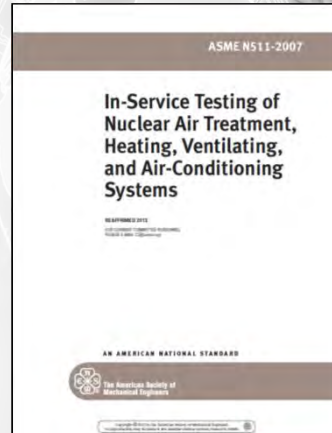


Code on Nuclear Air and Gas Treatment (AG-1 – 2017)

- **Division I:** General Requirements
 - Section AA: Common Articles
- **Division II:** Ventilation Air Cleaning and Ventilation Air Conditioning
 - 20 Sections covering components such as fans, blowers, ductwork, refrigeration equipment, adsorbers, filters, instrumentation and controls, etc.
- **Division III:** Process Gas Treatment
 - 13 Sections covering items like heat exchangers, scrubbers, mist eliminators, etc.
- **Division IV:** Testing Procedures
 - Section TA: Field Testing of Air Treatment Systems
 - Section TB: Field Testing of Gas-Processing Systems

In-Service Testing of Nuclear Air Treatment, Heating, Ventilating, and Air-Conditioning Systems (N511-2007)

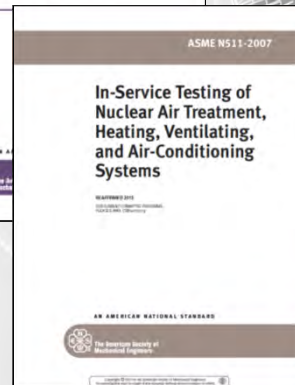
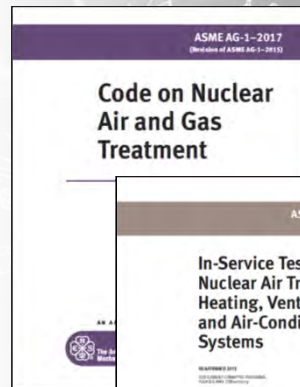
This standard covers the requirements for in-service testing of nuclear safety-related air treatment, heating, ventilating, and air conditioning systems in nuclear facilities.



Standards Committee on Nuclear Air and Gas Treatment (CONAGT)

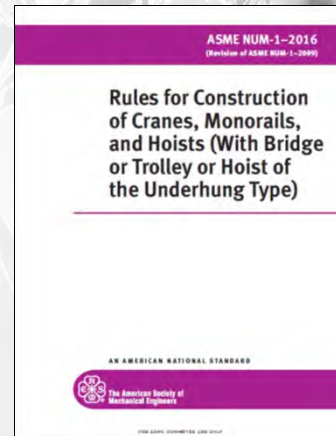
CONAGT Subcommittees:

- SC Filtration
- SC Technology
- SC Adsorption
- SC Common Equipment
- SC Testing & Inspection
- SC Ventilation & Air Conditioning
- SC Gas Process Treatment
- SC General Support Services



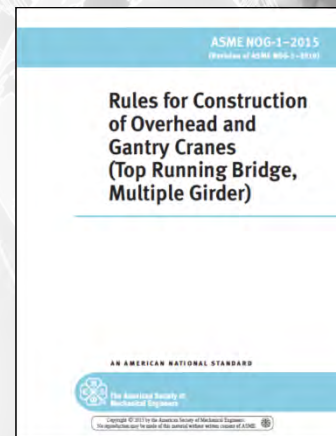
Rules for Construction of Cranes, Monorails, and Hoists (with Bridge or Trolley or Hoist of the Underhung Type)

- This Standard covers underhung cranes, top-running bridge and gantry cranes with underhung trolleys, traveling wall cranes, jib cranes, monorail systems, overhead hoists, and hoists with integral trolleys used in nuclear facilities. All of the above cranes, whether single or multiple girder, are covered by this Standard with the exception of multiple girder cranes with both top-running bridge and trolley, which are covered by ASME NOG-1.



Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)

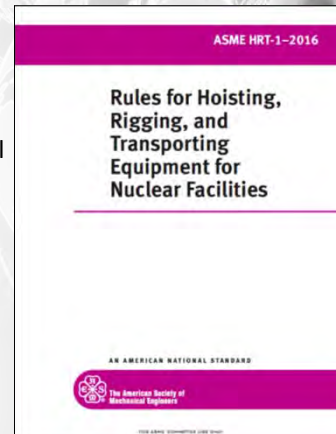
This Standard covers electric overhead and gantry multiple girder cranes with top running bridge and trolley used at nuclear facilities and components of cranes at nuclear facilities.



Rules for Hoisting, Rigging and Transporting Equipment for Nuclear Facilities (HRT-1 – 2016)

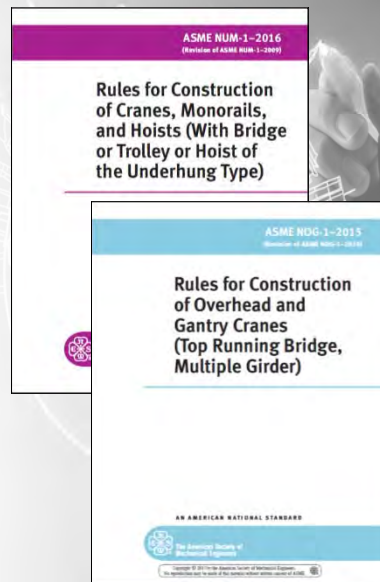
This Standard provides requirements for the design and use of hoisting, rigging, and transporting equipment used in the delivery of nuclear facility components to a nuclear facility's point of receipt and the handling of such components until the start of the facility's operating phase, defined as the point of initial fuel load.

The requirements of this Standard are also applicable to the design and use of hoisting, rigging, and transporting equipment for modifications at operating nuclear facilities when such equipment is not already controlled by existing facility procedures



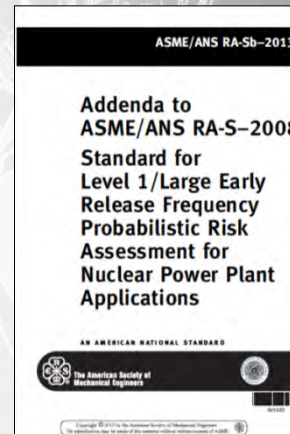
Standards Committee on Cranes for Nuclear Facilities (CNF)

- **CNF Subcommittees:**
- CNF Engineering Support Subcommittee
- Subcommittee Operation & Maintenance for Cranes



Probabilistic Risk Assessment for Nuclear Power Plant Applications (PRA) (RA-S)

This Standard sets forth requirements for probabilistic risk assessments (PRAs) used to support risk-informed decisions for commercial nuclear power plants, and prescribes a method for applying these requirements for specific applications.



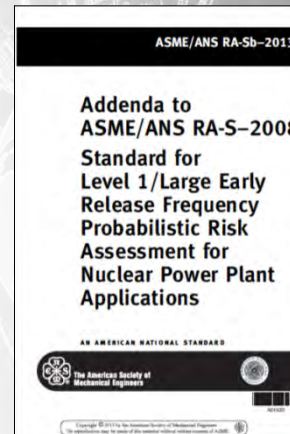
ASME/ANS – RA-S-1.2-2014 and RA-S-1.4-2013 (Draft Standards for Trial Use)

- **RA-S-1.2-2014 – Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs)**
- This standard sets forth the requirements for probabilistic risk assessments (PRAs) used to support risk-informed decisions for commercial light water reactor (LWR) nuclear power plants. Unique requirements are specified as needed for specific reactor designs.
- **RA-S-1.4-2013 – Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants**
- This standard sets forth the requirements for probabilistic risk assessments (PRAs) used to support risk-informed decisions for advanced non-light water reactor (non-LWR) nuclear power plants (NPPs) and prescribes a method for applying these requirements for specific applications.

ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM)

- **JCNRM Subcommittees**

- Subcommittee on Risk Application
- Subcommittee On Standards Maintenance
- Subcommittee on Standards Development
- **JCNRM China International Working Group**
- **JCNRM Japan International Working Group**



Other Codes and Standards for the Nuclear Industry

- **ANDE-1 – 2015: ASME Nondestructive Examination and Quality Control Central Qualification and Certification Program**

This Standard includes both performance-based and prescriptive requirements to be used for an ASME Nondestructive Examination and Quality Control Central Qualification and Certification Program that applies to NDE personnel and QC Inspection personnel.

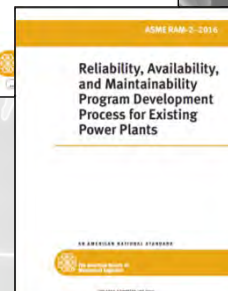
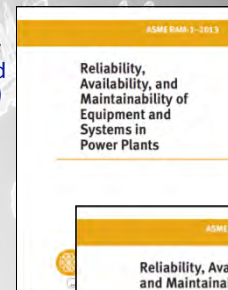
- **Part 1:** General Requirements
- **Part 2:** NDE Personnel Qualification and Certification Requirements
- **Part 3:** QC Personnel Qualification and Certification Requirements

Other Codes and Standards for the Nuclear Industry

- **QAI-1 – 2016: *Qualifications for Authorized Inspection Nuclear Related Portions:***
 - **Part 1:** Qualifications and Duties for Authorized Inspection Agencies, Nuclear Inspectors, and Nuclear Inspector Supervisors (BPVC Section III, Divisions 1, 3, and 5)
 - **Part 2:** Qualifications and Duties for Authorized Inspection Agencies, Nuclear Inservice Inspectors, and Nuclear Inservice Inspector Supervisors (BPVC Section XI)
 - **Part 3:** Qualifications and Duties for Authorized Inspection Agencies, Nuclear Inspectors (Concrete), and Nuclear Inspector Supervisors (Concrete) (BPVC)
 - **Part 4:** Accreditation of Authorized Inspection Agencies
 - **Parts 5-8:** Not applicable for use with Nuclear Codes and Standards

Other Codes and Standards for the Nuclear Industry

- **RAM-1 – 2013: *Reliability, Availability, and Maintainability of Equipment and Systems in Power Plants***
 - A RAM program is a structured methodology to identify and deliver the reliability, availability, and maintainability (RAM) requirements of a power plant in the most cost-effective manner.
 - This is an assurance standard to govern the planning process for a RAM program. It is intended to provide a methodology to develop and implement a comprehensive availability assurance program for the design, construction, and operation phases of the RAM project.
 - For new and existing plants
- **RAM-2 – 2016: *Reliability, Availability, and Maintainability Program Development Process for Existing Power Plants***
 - This Standard provides guidance for the program implementation portion of the RAM process described in ASME RAM-1. It is intended to implement a comprehensive availability assurance program
 - For Existing Plants only



Other Codes and Standards for the Nuclear Industry

V&V 30 Nuclear Reactor Design

- Will provide the practices and procedures for **verification** and **validation** of software used to calculate nuclear system thermal fluids behavior. The software includes system analysis and computational fluid dynamics, including the coupling of this software.
 - Initial focus -- high temperature gas-cooled reactors in response to U.S. Department of Energy activities
 - Now – May be applied to various reactors
- Responsive to changes in industry to promote best V&V practices within the community

V&V 50 – Computational Modeling for Advanced Manufacturing

- Will provide procedures for verification, validation, and uncertainty quantification in modeling and computational simulation for advanced manufacturing.
- Subcommittee was formed in March 2016 and currently about 33 members including members from FDA, FAA, and NASA, as well as major National Labs

Joint Efforts in Standards and Certification

- API 579-1/ASME FFS-1- Fitness-for Service
- ASME/ANS RA-S - Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications (JCNRM)
- ISO/ASME 14414 - Pump system energy assessment
- Joint ACI-ASME Committee on Concrete Components for Nuclear Service
- ASTM material specifications for ASME Section II

- Standard Developer Organization (SDO) Convergence Board
- Input to MDEP CSWG and Code Comparison Project
- NRC Standards Forum
- WNA Cooperation in Reactor Design Evaluation and Licensing (CORDEL)



II. Nuclear Certification Programs



Nuclear Component Certification Program

- **N-type Certificates of Authorization** issued by ASME signifies that a Certificate Holder has been through a rigorous survey to verify the adequacy and effective implementation of the quality assurance program.
- **N-type Certificates of Authorization** allow Certificate Holders to certify and stamp newly constructed components, parts, and appurtenances used at a nuclear facility with the Certification Mark in accordance with Section III of the ASME BPVC.
 - **N** – Vessels, pumps, valves, piping systems, storage tanks, core support structures, concrete containments, and transport packaging
 - **NA** – Field installation and shop assembly of all items
 - **NPT** – Parts, appurtenances, welded tubular products, and piping subassemblies
 - **NS** – Supports
 - **NV** – Pressure relief valves
 - **N3** – Transportation containments and storage containments
 - **OWN** – Nuclear power plant owner

Nuclear Material Organization Certification

- The Nuclear Material Organization Certification Program certifies organizations that provide materials and services to the nuclear industry.
- **Quality System Certificates (QSC)** issued by ASME verify the adequacy of a Material Organization's quality system program.
- The quality system program provides assurance that the organization's **operations, processes, and services** related to the **procurement, manufacture, and supply of material, source material, and unqualified source material** are performed in accordance with the requirements of the ASME BPVC, Section III, NCA-3800 and NCA-3900.

Nuclear Quality Assurance (NQA-1) Certification

- Provides **centralized, independent, third-party certification** for quality assurance programs in conformance with the ASME NQA-1 standard, "Quality Assurance Requirements for Nuclear Facility Applications".
- Entails a **full audit of the Quality Assurance Program** performed by trained ASME auditors with an extensive background in quality assurance. A successful audit will yield a NQA-1 Quality Program Certificate.
- Seeks to meet the needs of the nuclear industry by expanding the supply chain with organizations who are committed to understanding quality and providing high quality products and services.

ASME Nondestructive Examination and Quality Control Inspection Personnel Certification (ANDE)

- The ANDE program recognizes an individual's ability to perform nondestructive examination and quality control inspection in accordance with the ANDE-1 Standard.
- Provides independent, third-party centralized certification for NDE & QC inspection personnel as an alternate option to the historical, employer-based NDE & QC certification systems.
- At the moment, ANDE focuses on nuclear in-service inspection and new nuclear construction.
- There are three ANDE certification, levels I, II or III. To successfully gain an ANDE Certification, the individual must successfully pass the written examination and, practical demonstration.

III. Learning and Development

ASME Learning and Development

- Most Code related courses are taught by ASME Code Committee members who develop the codes and standards
 - **Public Courses:** More than 150 courses ranging from fundamental to advanced levels
 - **Corporate Programs:** Any of the public courses can be customized for training at a company's headquarters, anywhere in the world
 - **E-Learning Courses:** Both instructor-led and self-study courses
 - **Master Class Series:** Practical, case study-driven training sessions that emphasize learning through discussion of real world applications.

Nuclear Power Industry Learning Matrix

| | Fundamental | Intermediate | Advanced |
|------------------------------|---|---|---|
| Facility Construction | BPV Code, Section III: Introduction (IS) – EL509 BPVC Code, Section III, Division 1: Rules for Construction of Nuclear Facility Components (LC) – PD184 Overview of Codes & Standards for Nuclear Power Plant Construction (LC) – PD633 Design in Codes, Standards and Regulations for Nuclear Power Plant Construction (LC) – PD632 | BPV Code, Section III, Division 1: Class 1, 2, & 3 Piping Design (LC) – PD615 Design of Buried High Density Polyethylene (HDPE) Piping Systems (LC) – PD617 Also available as online course EL544 (IS) | Advanced Design & Construction of Nuclear Facility Components per BPVC Code, Section III (LC) – PD644 Also available as online course EL524 Design Basis vs. Beyond Design Basis (IS) Considerations in Nuclear Plants (LC) – MCT20 |
| Quality Assurance | NQA-1 Quality Assurance for Nuclear Facility Applications (LC) – PD635 Add NQA-1 Quality Assurance Requirements for Nuclear Facility Applications (IS) - EL520 NQA-1 Requirements for Computer Software Used in Nuclear Facilities (LC) – PD606 | QA Considerations for New Nuclear Facility Construction (LC) – PD523 ASME NQA-1 Lead Auditor Training (LC) – PD675 Comparison of Global Quality Assurance & Management System Standards Used for Nuclear Applications (IS) – EL526 (Online course Z1360) ASME NQA-1 and DOE Quality Assurance Rule 10 CFR 830 (LC) – PD711 | Real World Application of Commercial Grade Dedication (LC) – MC102 Identifying and Preventing the Use of Counterfeit, Fraudulent, and Suspect Items (LC) – MC103 Software Dedication Training on Use of Commercial Grade Computer Programs for Design and Analysis in Nuclear Applications (LC) – MC105 |

Nuclear Power Industry Learning Matrix

| | Fundamental | Intermediate | Advanced |
|-------------------------|---|--|--|
| Balance of Plant | <p>BPVC Code, Section VIII, Division 1: Design & Fabrication of Pressure Vessels (IS) – PD442 Also available as online course EL501</p> <p>BPVC Code, Section VIII, Division 2: Pressure Vessels (IS) – PD448 Also available as online course EL502</p> <p>Non-Destructive Examination – Applying ASME Code Requirements (BPV Code, Section V) (LC) – PD389</p> <p>Pressure Relief Devices: Design, Sizing, Construction, Inspection & Maintenance (LC) – PD583</p> <p>B31.1 Power Piping Code (LC) – PD013</p> | <p>Bases and Application of Design Requirements for High Pressure Vessels in Section VIII, Division 3 of the ASME Boiler and Pressure Vessel Code (MC127)</p> <p>How to Predict Thermal-Hydraulic Loads on Pressure Vessels & Piping (LC) – PD382</p> <p>Seismic Design and Retrofit of Equipment and Piping (LC) – PD394</p> <p>Inspection, Repairs and Alterations of Pressure Vessels (IS) – PD441 Also available as online course EL503</p> <p>Flow-Induced Vibration with Applications to Failure Analysis (LC) – PD146</p> | <p>Bases and Application of Heat Exchanger Mechanical Design Rules in Section VIII of the ASME Boiler and Pressure Vessel Code (LC) – MC104</p> <p>Bases and Application of Piping Flexibility Analysis to ASME B31 Codes (LC) – MC110</p> <p>Piping Failures - Causes and Prevention (LC) – MC117</p> |
| Inservice | <p>Developing a 10-Year Pump Inservice Test Program (IS) – PD595 Also available as online course EL523</p> <p>Developing a 10-Year Valve IST Program (LC) – PD596 Also available as online course EL521 (IS)</p> | <p>BPV Code, Section XI: Inservice Inspection of Nuclear Power Plant Components (LC) – PD192</p> <p>Risk-Informed Inservice Testing Program (LC) – PD597 Also available as online course EL527 (IS)</p> | <p>Run-or-Repair Operability Decisions for Pressure Equipment and Piping Systems in Nuclear Plants (LC) – MC115</p> <p>Life Cycle Management of Pressure Equipment and Piping Integrity (LC) – MC116</p> <p>Environmentally-Assisted Fatigue Analysis, Monitoring and Management of Nuclear Plant Components (LC) – MC118</p> <p>Corrosion and its Mitigation in Light Water Reactors (LC) – MC119</p> |



Pressure Vessel and Piping Conference



- **July 14-19, 2018 in San Antonio, Texas USA**
<https://www.asme.org/events/pvp/>
- Global conference promoting excellence in the global pressure vessel and piping industry.
- Over 180 planned sessions including workshops and tutorials, a Technology Demonstration Forum and the 26th Rudy Scavuzzo Student Paper Symposium and Competition.
- **Topics:**
 - Codes & Standards
 - Computer Technology & Bolted Joints
 - Design & Analysis
 - Fluid-Structure Interaction
 - High Pressure Technology
 - Materials & Fabrication
 - Operations, Applications, & Components
 - Seismic Engineering
 - Non-Destructive Examination
 - Rudy Scavuzzo Student Paper Competition and Symposium

International Conference on Nuclear Engineering



- **May 19-24, 2019 in Tsukuba, Japan**
 - <https://www.asme.org/events/icone/>
- Global conference on nuclear reactor technology
- Co-sponsored by ASME, JSME and CNS
- Industry Forums, Technical, Keynote, Plenary and Poster Sessions, plus Workshops
- **Technical Tracks:**
 - Track 1** Operations & Maintenance, Engineering, Modifications, Life extension, Life Cycle and Balance of Plant
 - Track 2** Nuclear Fuel and Material, Reactor Physics and Transport Theory
 - Track 3** Plant Systems, Structures, Components and Materials
 - Track 4** Instrumentation and Control (I&C) and Influence of Human Factors
 - Track 5** Advanced Reactors and Fusion Technologies
 - Track 6** Nuclear Safety, Security, and Cyber Security
 - Track 7** Codes, Standards, Licensing, and Regulatory Issues
 - Track 8** Thermal-Hydraulics and Safety Analyses
 - Track 9** Computational Fluid Dynamics (CFD)
 - Track 10** Decontamination & Decommissioning, Radiation Protection, and Waste Management
 - Track 11** Mitigation Strategies for Beyond Design Basis Events
 - Track 12** Nuclear Education and Public Acceptance
 - Track 13** Innovative Nuclear Power Plant Design and SMRs
 - Track 14** Risk Assessments and Management
 - Track 15** Computer Code Verification and Validation
 - Track 16** Student Paper Competition
 - Track 17** Keynote, Plenary, Panels Sessions
 - Track 18** Workshops

POWER & ENERGY Conference & Exhibition



- **July 14-18, 2019 in Salt Lake City, Utah, USA**
 - <https://www.asme.org/events/power-energy>
- Two of ASME's major conferences and two forums come together to create an event of major impact for the Power and Energy sectors:
 - ASME Power Conference
 - ASME Energy Sustainability Conference
 - Fuel Cell and Energy Storage topics are included as part of the Energy Sustainability Conference
- **ASME Nuclear Forum**
 - Presents the most recent developments in the Nuclear Power Industry
 - **Track 1-1** Codes, Standards, Licensing and Regulatory Compliance
 - **Track 1-2** Plant Construction Issues and Supply Chain Management
 - **Track 1-3** Structures, Components and Materials
 - **Track 1-4** I&C, Digital Controls, and Influence of Human Factors
 - **Track 1-5** Plant Operations, Maintenance, Aging Management, Reliability and Performance
 - **Track 1-6** Thermal Hydraulics and Computational Fluid Dynamics
 - **Track 1-7** Posters



ASME/NRC Pump & Valve Symposium



- Biannual symposium, last held: **July 16-19, 2017**
 - <https://www.asme.org/events/nrc-pump-valve-symposium>
- Discusses the latest issues, technology, developments and research in the pre-service and in-service testing of nuclear power plants and components.
- Sessions include important topics in pre-service and in-service testing from the perspective of industry best practices and ASME codes
 - O&M Scope and Philosophy
 - New Reactors
 - Valves, pumps, snubbers
 - Rulemaking
 - Risk insight activities



V. Other Publications



ASME Standards Technology, LLC *Nuclear Project Work*

- Provides research and technology development in order to establish and maintain the technical relevance of codes and standards, especially in regards to emerging and newly commercialized technologies
- Aims to meet the needs of industry and government by providing new standards-related products and services, which advance the application of scientific and technological development.
- **Examples of Published Work:**
 - **STP-NU-072:** Small Modular Reactors (SMR) Roadmap.
 - **STP-NU-057:** ASME Code Development Roadmap for HDPE Pipe in Nuclear Service.
 - **STP-NU-051-1:** Code Comparison Report for Class 1 Nuclear Power Plant Components.
 - **STP-NU-045-1:** Roadmap to Develop ASME Code Rules for the Construction of HTGRs.

NOTE: The full list of published work for the nuclear industry can be found in the Nuclear Resources Brochure

Journals

- **Journal of Nuclear Engineering and Radiation Science**
 - For specialists in the nuclear/power engineering areas of industry, academia, and government.
- **Journal of Pressure Vessel Technology**
 - Publishes quality research on design, analysis, materials, fabrication, construction, inspection, operation, failure prevention, and NDE of pressure vessels and their components.
- **Journal of Verification, Validation and Uncertainty Quantification (VVUQ)**
 - Disseminates original and applied research, illustrative examples, and high quality validation experiment data from leaders in the field of VVUQ of Computation Models.
- **Journal of Engineering Materials and Technology**
 - Peer-reviewed research papers on engineering materials and technology and covers a broad spectrum of issues regarding experimental, computational and theoretical studies of mechanical properties of materials.

<http://asmedigitalcollection.asme.org/journals.aspx>

Conference Proceedings

- ASME sponsors more than 30 conferences per year and publishes approximately 100 proceedings volumes annually. Conference topics encompass the entire spectrum of subject areas of interest to mechanical engineers and associated disciplines. ASME Conference Proceedings are available in print and also in digital format through the ASME Digital Collection. Individual conference papers from 2002 to the present are currently available through the ASME Digital Collection.

<http://proceedings.asmedigitalcollection.asme.org/conferenceproceedings.aspx>

Companion Guides/Handbooks

- **BPVC Companion Guide – 2017**
 - Two-volume publication that focuses on all twelve sections of the ASME Boiler and Pressure Vessel Code, as well as relevant piping codes
- **ASME Decommissioning Handbook - 2004**
 - Serves as an introduction to those new to the field, as well as a reference on regulations and resources for experienced practitioners.
 - Provides a forum building a network of **consistent approaches, practices, and results**. Covering both NRC and DOE approaches, this book applies not only to decommissioning existing nuclear facilities, but by crossing the traditional lines between operations and reuse, this will also allow us to rethink the construction of new facilities

VI. References

References

- **Join an ASME Standards Development committee**
 - <https://www.asme.org/about-asme/get-involved/standards-certification-development-committees/join-a-c-s-committee.aspx>
- **ASME Product Catalog**
 - <https://www.asme.org/shop/standards>
- **Learning and Development**
 - <https://www.asme.org/shop/courses/asme-training-development>
- **ASME STLLC**
 - <http://asmestllc.org/>
- **ASME Digital Collection**
 - <https://asmedigitalcollection.asme.org/>

References

- **ASME Certification and Accreditation**
 - <https://www.asme.org/shop/certification-accreditation>
- **Nuclear Component Certification**
 - <https://www.asme.org/shop/certification-accreditation/nuclear-component-certification>
- **Personnel Certification**
 - <https://www.asme.org/shop/certification-accreditation/personnel-certification>



ASME Section XI Standards Committee Activities and IWG Opportunities

Kate Hyam
Project Engineering Manager
NCS Asia Liaison

Mumbai, India
January 4, 2019

Agenda/Topics


- BPVC and BPV XI Committee Organization and Committee Activities
- ASME Committee Membership Options
- BPV XI IWG Benefits
- How to Join a Committee

ASME BPV Structure / Process Overview


| | |
|-----------------------------|---|
| BNCS / BPTCS | Provides procedural oversight for all NCS / PTCS activities |
| TOMC | Technical Oversight Management Committee Advises on consistency, strategy, and R&D |
| Standards Committees | Establishes consensus on all technical matters in their Committee Charter |
| Subgroups | Provides recommendations on technical matters in a given specialty – e.g., design |
| WGs, TGs, PTs | Develops detailed proposals in a specific Field, such as valve design |

Boiler and Pressure Vessel Code


Construction Sections



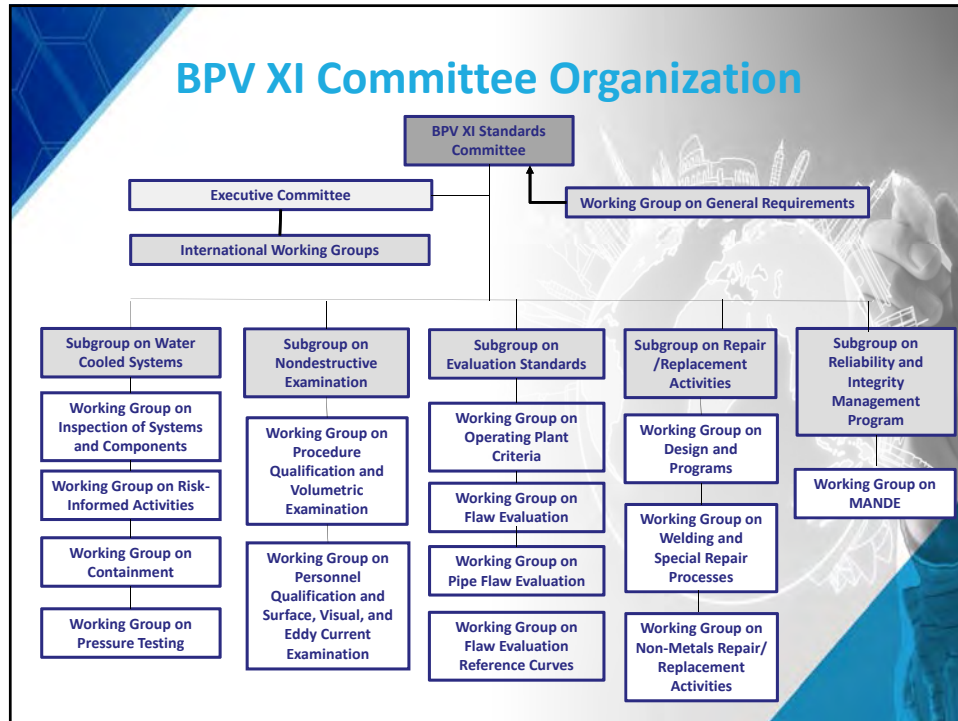
Service Sections



Nuclear Sections



**2017 Edition:
12 Sections
32 Volumes
~18,5000 pages
Updated every 2 years**



Committee Meeting Activities

- Committee meetings are held 4 times a year to discuss:
 - Interpretations, revisions, code cases, etc. can originate internally or could be submitted from external requests from users.
 - Requests for Code revisions result in new projects assigned to technical committees with an individual volunteer Project Manager
 - Inquirers are encouraged to participate in committee discussion, meetings are open to the public
go.asme.org/inquiry

2 Year BPV Code Publication Cycle

- Prior to the 2013 Edition the Code was published as a new edition every three years along with a yearly addenda service.
- 2013 Edition
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- Next edition will be the released on July 1, 2019

Committee Membership

- All ASME committee members are volunteers
- Committee participation is free, open to anyone with an interest in the subject area and the requisite technical expertise.
 - Note that committee members are always viewed as individuals, not as representatives of their employers or other organizations.
- Criteria for membership appointments include:
 - Your experience and technical qualifications
 - Your ability to participate in committee activities
 - The business interest of the organization, if any, that supports your committee participation (interest classification)
 - Committee size limits

Membership Participation Options

- **Member**
 - Required to actively participate in committee discussions, work assignments, vote on ballots, and attend meetings.
 - Each member may appoint an alternate to represent their position in their absence.
- **Contributing Member**
 - A classification for individuals interested in contributing to standards development activities, but unable to actively/constantly participate.
 - The contributing member is required to participate in committee discussions and work assignments, but attendance at meetings is optional they and only can only review and comment on ballots.

Membership Participation Options

- **Delegate**
 - An individual representing a group of experts from outside of the U.S. and Canada. Delegates must have fluency in the English language and a working knowledge of the technical aspects of the committees work.
 - Delegates, with the support of their group, are expected to provide collective group comments on committee standards actions, comment on proposed revisions, vote on first consideration ballots, and participate on work assignments and committee discussions.
- **International Working Group (IWG)**
 - Designed for when a diverse group of individuals in a country or region outside the U.S. and Canada express a desire to contribute to the work of a standards development committee..
 - IWG's function like any other subgroup that reports to the Executive Committee, but meet regularly in their own country.

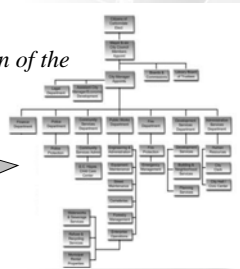
International Working Group Concept

- An IWG fits into the “family” of committees under the Standards Committee. These committees provide essential expertise in the development of new and revised code rules.

This is only an illustration of the relationship.



IWG



International Working Groups (IWGs)

- Developed to gain input from international stakeholders who have experience, technical expertise, and distinct perspectives that enhance the global relevance of ASME’s standards
- Currently, IWGs are populated by virtue of a common geographic location, rather than a common engineering discipline or expertise (e.g., a working group on “Design”).
 - Members provide expertise across a diverse mixture of disciplines (e.g., design, materials, etc.)
- IWGs play a similar roles as subordinate groups in that they should be expected to both develop and review proposed standards actions for subsequent consideration by their respective standards committees
- IWGs typically conduct all of their meetings outside of the U.S. and Canada.
 - IWGs may choose to conduct their meetings in a language other than English.

Benefits to Members

- Collaborate with technical subject matter experts from around the world
- Create personal network of contacts for technical advice on the standard
- Knowledge of impending Code revisions
- Develop leadership skills in teamwork and running meetings
- Career development
- Roster of Committee members in the Code book
- Complimentary PDF copies of the Code for use in committee work

Benefits to Supporting Organizations

- Ensures that the interests, practices and experience of the organization are considered in developing and updating code requirements
- Early access to upcoming revisions – possible economic advantage
- Continuously aware of technical problems in their industry and how others are dealing with them
- Participants become subject matter experts in the standard that affects their business
- <https://www.asme.org/about-asme/get-involved/standards-certification-development-committees/get-involved>

ASME BPV Code International Working Groups

India: BPV Sections I, III, BPV VIII, B31.3 and B31.8 and currently soliciting interested NQA and BPV XI IWG Participants!

Argentina: BPV Sections III and **XI**

China: BPV Sections II, III, VIII and **XI**, O&M, QME, NQA, JCNRM

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Korea: BPV Sections III and *XI and NQA* are under consideration

JOINING A COMMITTEE

- Let us know if you are interested in any level of membership!
- We are looking for interested parties for NQA, BPV XI or BPV III IWGs – see sign up sheet at back of room.
- Details will be sent to you regarding how to apply.
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**ASME Section XI Standards Committee
Activities and IWG Opportunities**

Kate Hyam
*Project Engineering Manager
NCS Asia Liaison*

**Mumbai, India
January 4, 2019**

The text is centered on a white background with blue geometric patterns in the corners. The background also features a faint, white line-art illustration of a hand holding a globe, surrounded by architectural and engineering symbols, similar to the ASME logo image above.

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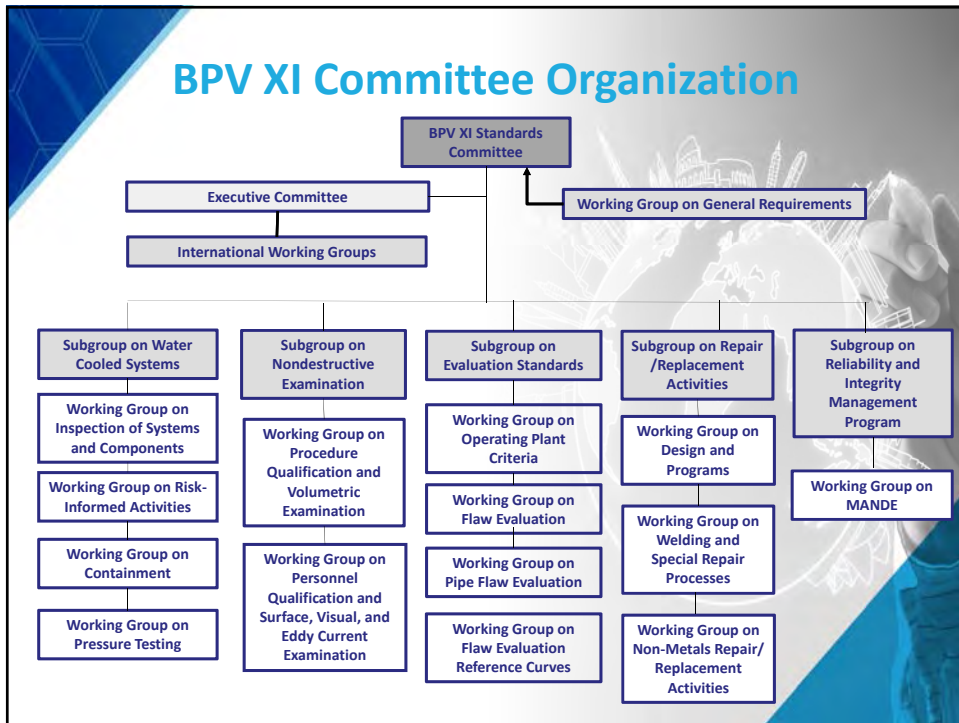
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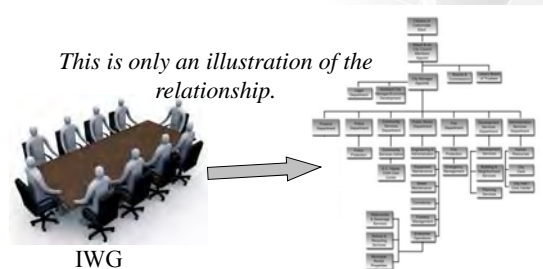
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QUESTIONS?

Kate Hyam
hyamk@asme.org