Welcome:
Mechanical Codes & Inspection Course

An Overview

- **Introduction of Instructors**
- **Overall Course Objective is to Provide an Understanding of:**
  - General Mechanical Codes and Standards
  - NRC Regulatory Guides associated with Mechanical Codes and Standards
  - NRC SRP for Mechanical Systems
  - 10 CFR Part 50, Appendix A and Appendix B
  - Mechanical Environmental Qualification Testing
  - Case Studies and Other Internationally Recognized Codes and Standards.
- **Two Week Curriculum**

NRC Mechanical Codes and Standards Course

Week 1

	Monday	Tuesday	Wednesday	Thursday	Friday
8:00 to 9:00	10 CFR 50, App A General Design Criteria Review in detail the criteria of 10 CFR 50, Appendix A as it pertains to construction and manufacturing related activities associated with Mechanical SSC(s). **LESSON #1**	ASME NQA-1 General overview of the Basic Requirements and Supplements relevant to NPPs, fuel facility const. and operational programs. Overview of 10 CFR 50.34. NQA-1-1994 - guidance on how to implement the requirements of 10 CFR Part 50, Appendix B. **LESSON #3**	NUREG 0800 Chapters 4.5.1 and 4.5.2 **LESSON #5**	NUREG-0800 Chapter 17, 17.1 **LESSON #11**	ASME O&M Code **LESSON #17**
9:00 to 10:00			NUREG 0800 Chapter 5 5.2.1.1, 5.2.1.2, 5.2.2, 5.2.3, 5.3.1, 5.3.2 and 5.3.3 **LESSON #6**	ASME Sec III Subsection NCA, **LESSON #12**	
10:00 to 11:00				ASME Sec III Subsection NB **LESSON #13**	
11:00 to 12:00			NUREG 0800 Chapter 6 - 6.1.1, 6.1.2, 6.2.4, 6.2.7, and 6.5.1 **LESSON #7**		
Lunch					
1:00 to 2:00	10 CFR 50, App B Provide a general overview of 10 CFR 50, Appendix B and state how its criterion relate to construction and manufacturing activities associated with Mechanical SSC(s). **LESSON #2**	NUREG 0800 Chap 3 Overview of Sections 3.2.1, 3.2.2, 3.3.1, 3.3.2, 3.5.1, 3.5.2, 3.5.3, 3.5.1.4, 3.8.5, 3.9.2, 3.9.3, 3.9.6, 3.10, 3.11, 3.12 and 3.13 **LESSON #4**	NUREG 0800 Chapter 9 - 9.1.4, 9.1.5, 9.3.3, 9.3.4, 9.4.4, and 9.4.5, **LESSON #8**	ASME Sec III Subsections, NC - NH and Code Appendices **LESSON #14**	Discussion of Part 52 and the relationship with the requirements in Part 50. **LESSON #18**
2:00 to 3:00			NUREG 0800 Chapter 10 10.3, and 10.3.6 **LESSON #9**	Discussion of ASME Code Cases, Inquiry Process **LESSON #15**	Discussion of Part 52 ITAAC, ITAAC Closure with Examples from AP-1000 **LESSON #19**
3:00 to 4:00			NUREG 0800 Chapter 14.1 & .2, 14.3.2, &3.3, 14.3.4,.7,&.11 **LESSON #10**	Discussion of 10 CFR 50.55a & Reg Guides 1.84, 147, 192, 193 **LESSON #16**	

NRC Mechanical Codes and Standards Course

Week 2

	Monday	Tuesday	Wednesday	Thursday	Friday
8:00 to 9:00	ASME AG-1, Code on Nuclear Air and Gas Treatment and Stds N509/510/511 **LESSON #20**	ASME Sec II **LESSON #26**	ASME Sec VIII Div 1 and Div 2 **LESSON #33**	IEEE 323, RG 1.89 Environmental Qualification of Equipment **LESSON #39**	Attachment 2 from RFP "Proposed Industry Codes" 1. ANS 56.4 Contmt Transients 2. ISA 67.02.01 **LESSON #42**
9:00 to 10:00	ASME Nuclear Crane Stds, NUM-1 & NOG-1 **LESSON #21**	ASTM Material Specifications **LESSON #27**	ANS 51.1 and ANS 52.1 System Classification **LESSON #34**	Case Studies of Construction Issues Related to Mechanical Components. **LESSON #40**	
10:00 to 11:00	ANSI/ASME (USAS) B31.1, Power Piping **LESSON #22**	ASNT/ANSI-CP189 Certification of Auditors **LESSON #28**	ANSI/ASME N-278 Containment Isolation **LESSON #35**		ITAAC **LESSON #43**
11:00 to 12:00	ANSI/ASME (USAS) B31.3, Process Piping **LESSON #23**	ASME Section XI Overview and focus on Pre-service testing. **LESSON #29**	ASME PTC 32.1 **LESSON #36**		Course Evaluation and Summary **LESSON #44**
Lunch					
1:00 to 2:00	ASME Sec IX & Welding Overview, AWS Standards, Control of Consumables and associated QA requirements **LESSON #24**	ASME Section XI Cont'd **LESSON #29**	Seismic Qualification of Equipment 10 CFR 50.49 overview RGs 1.29, 1.100 and QME-1, RG 1.73 and RG 1.40, IEEE-344 Reg Guides 1.60, 1.61, 1.92 **LESSON #37**	Overview of ISO and other International Standards. **LESSON #41**	
2:00 to 3:00		N-278.1 RVs and PORVs **LESSON #30**			
		ANS 56.3 Overpressure Protection **LESSON #31**			
3:00 to 400	ANSI/ASME B16.5, 16.11, 16.34, 16.47 **LESSON #25**	ASME Section V & Code Case N-307-3 **LESSON #32**	RG 1.75 Separation of Electrical Equipment **LESSON #38**		

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Module 1
10 CFR 50, Appendix A
General Design Criteria

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Module 1

10 CFR 50, Appendix A General Design Criteria

Instructor: Gene Imbro, P.E.

Learning Objectives

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- **Learn about the background of the General Design Criteria**
- **Learn how the General Design Criteria are applied to the design of mechanical systems, structures and components (SSCs)**
- **Learn how the GDC relate to or form the basis for other regulations**

Major References

- **10 CFR 50, Appendix B, *Quality Assurance***
- **10 CFR 50, Appendix J, *Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors***
- **10CFR 50.48, *Fire Protection***
- **10 CFR 50.55a, *Codes and Standards***

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Major References (cont'd)

- **ASME NQA-1, *Quality Assurance Requirements for Nuclear Facility Applications***
- **ASME BPV Code Secs III, XI**
- **ASME *O&M Code***
- **Regulatory Guide 1.7**

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Overview

- **The General Design Criterion were developed in the late 1960s to capture some of the fundamental safety concepts of light water nuclear reactor facility design and codify them in NRC's regulations.**
- **The GDC were issued for Trail Use and Comment in 1967**
  - Some older plants were licensed to the 1967 version of the GDC (Turkey Point)
- **The GDC were finalized and included in the regulations in 1971**

Overview

- **The General Design Criteria are contained in Appendix A to 10 CFR Part 50**
- **Compliance with the GDC is mandatory since they are regulations, unless an exemption is approved by the regulator.**
- **Unusual circumstances would have to exist for an exemption to the GDC to be granted.**

Relation of Part 50 GDC to Part 52

- **Introduction (to 10CFR50, Appendix A)**
  - Under the provisions of § 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. Under the provisions of 10 CFR 52.47, 52.79, 52.137, and 52.157, an application for a design certification, combined license, design approval, or manufacturing license, respectively, must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

Overview

- **There are 55 GDC divided into six categories**
  - Overall requirements
  - Protection by Multiple Fission Product Barriers
  - Protection and Reactivity Control Systems
  - Fluid Systems
  - Reactor Containment
  - Fuel and Radioactivity Control

Overall Requirements

- **Criterion 1–Quality standards and records**
  - Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function.

Overall Requirements

- **Criterion 2–Design bases for protection against natural phenomena**
  - Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

Overall Requirements

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- **Criterion 2–Design bases for protection against natural phenomena (cont'd)**
  - The design bases for these structures, systems, and components shall reflect:
    1. Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated,
    2. appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and
    3. the importance of the safety functions to be performed.

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Overall Requirements

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- **Criterion 3–Fire protection.**
  - SSCs important to safety shall be designed and located to minimize … the probability and effect of fires and explosions.
  - Noncombustible and heat resistant materials shall be used in … locations such as the containment and control room.

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Overall Requirements

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- **Criterion 3–Fire protection (cont'd)**
  - Fire detection and fighting systems…shall be provided and designed to minimize the adverse effects of fires on SSCs important to safety.
  - Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

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Overall Requirements

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- **Criterion 4–Environmental and dynamic effects design bases**
  - Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

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Overall Requirements

- **Criterion 4–Environmental and dynamic effects design bases (cont'd)**
  - These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Overall Requirements

- **Criterion 4–Environmental and dynamic effects design bases (cont'd)**
  - However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Overall Requirements

- **Criterion 5–Sharing of structures, systems, and components.**
  - Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Protection by Multiple Fission Product Barriers

- **Criterion 10–Reactor design.**
  - The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Protection by Multiple Fission Product Barriers

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- **Criterion 11–Reactor inherent protection.**
  - The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

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Protection by Multiple Fission Product Barriers

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- ***Criterion 12–Suppression of reactor power oscillations*.**
  - The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

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Protection by Multiple Fission Product Barriers

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- **Criterion 13–Instrumentation and control.**
  - Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

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Protection by Multiple Fission Product Barriers

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- **Criterion 14–Reactor coolant pressure boundary.**
  - The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

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Protection by Multiple Fission Product Barriers

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- **Criterion 15–Reactor coolant system design.**
  - The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

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- **Criterion 16–Containment design.**
  - Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

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Protection by Multiple Fission Product Barriers

- **Criterion 17, "Electric power systems"**
  - Requires two physically independent offsite electric power supplies
  - Single failure proof onsite system, i.e., two emergency diesels and redundant Class 1E electrical system.
- **Criterion 18, "Inspection and testing of electric power systems"**

Protection by Multiple Fission Product Barriers

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- **Criterion 19–Control room.**
  - A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents.
  - Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Protection by Multiple Fission Product Barriers

- **Criterion 19–Control room (cont'd).**
  - Equipment at appropriate locations outside the control room shall be provided:
    1. with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and
    2. with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

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Protection and Reactivity Control Systems

- **Criterion 20–Protection system functions.**
  - The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

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Protection and Reactivity Control Systems

- **Criterion 22–Protection system independence.**
  - The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

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Protection and Reactivity Control Systems

- **Criterion 23–Protection system failure modes.**
- **Criterion 24–Separation of protection and control systems.**
- **Criterion 25–Protection system requirements for reactivity control malfunctions.**

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- **Criterion 26–Reactivity control system redundancy and capability.**
  - Two independent reactivity control systems of different design principles shall be provided.
  - One of the systems shall use control rods.......
  - The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded.

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Protection and Reactivity Control Systems

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- **Criterion 27–Combined reactivity control systems capability.**
- **Criterion 28–Reactivity limits**
  - The reactivity control systems shall be designed…to assure that…postulated reactivity accidents can not result in any damage to the reactor coolant pressure boundary greater than limited local yielding nor…affect reactor pressure vessel internals, core or core supports…to impair core cooling.
- **Criterion 29–Protection against anticipated operational occurrences.**

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- **Criterion 30–Quality of reactor coolant pressure boundary.**
  - Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

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- **Criterion 31–Fracture prevention of reactor coolant pressure boundary.**
  - The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

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- **Criterion 31–Fracture prevention of reactor coolant pressure boundary. (cont'd)**
  - The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

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- **Criterion 32–Inspection of reactor coolant pressure boundary.**
  - Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

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- **Criterion 33–Reactor coolant makeup.**
  - A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary.

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- **Criterion 33–Reactor coolant makeup (cont'd)**
  - The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Fluid Systems

- **Criterion 34–Residual heat removal.**
  - A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

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- **Criterion 34–Residual heat removal (cont'd)**
  - Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

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- **Criterion 35–Emergency core cooling.**
  - A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

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- **Criterion 35–Emergency core cooling. (cont'd)**
  - Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

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- **Criterion 36–Inspection of emergency core cooling system.**
  - The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

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- **Criterion 37–Testing of emergency core cooling system.**
  - The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and…

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- **Criterion 37–Testing of emergency core cooling system. (cont'd)**
  - (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

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- **Criterion 38–Containment heat removal.**
  - A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

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- **Criterion 38–Containment heat removal (cont'd).**
  - Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure

Fluid Systems

- **Criterion 39–Inspection of containment heat removal system.**
  - The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

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- **Criterion 40–Testing of containment heat removal system.**
  - The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and…

Fluid Systems

- **Criterion 40–Testing of containment heat removal system. (cont'd)**
  - (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system

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- **Criterion 41–Containment atmosphere cleanup.**
  - Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

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- **Criterion 41–Containment atmosphere cleanup (cont'd)**
  - Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

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- **Criterion 42–Inspection of containment atmosphere cleanup systems.**
  - The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

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- **Criterion 43–Testing of containment atmosphere cleanup systems.**
  - The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

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- **Criterion 44–Cooling water.**
  - A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

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- **Criterion 44–Cooling water. (cont'd)**
  - Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

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- **Criterion 45–Inspection of cooling water system. The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.**

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- **Criterion 46–Testing of cooling water system.**
  - The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Reactor Containment

- **Criterion 50–Containment design basis.**
  - The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

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Reactor Containment

- **Criterion 50–Containment design basis.(cont'd)**
  - This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

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Reactor Containment

- **Criterion 51–Fracture prevention of containment pressure boundary.**
  - The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

Reactor Containment

- **Criterion 51–Fracture prevention of containment pressure boundary.(cont'd)**
  - The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.

Reactor Containment

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- **Criterion 52–Capability for containment leakage rate testing.**
  - The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

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- **Criterion 53–Provisions for containment testing and inspection.**
  - The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

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Reactor Containment

- **Criterion 54–Piping systems penetrating containment.**
  - Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Reactor Containment

- **Criterion 55–Reactor coolant pressure boundary penetrating containment.**
  - Lines that are part of the reactor coolant pressure boundary and that penetrate containment
- **Criterion 56–Primary containment isolation.**
  - Lines that connects directly to the containment atmosphere and penetrate containment

Reactor Containment

- **Criterion 57–Closed system**
  - Lines that penetrate containment and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere

Fuel and Radioactivity Control

- **Criterion 60–Control of releases of radioactive materials to the environment.**
  - The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Fuel and Radioactivity Control

- **Criterion 61–Fuel storage and handling and radioactivity control.**
  - The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions

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Fuel and Radioactivity Control

- **Criterion 62–Prevention of criticality in fuel storage and handling.**
  - Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

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Fuel and Radioactivity Control

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- **Criterion 63–Monitoring fuel and waste storage.**
  - Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

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Fuel and Radioactivity Control

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- **Criterion 64–Monitoring radioactivity releases.**
  - Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

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Modules 2 & 3
Nuclear Quality Assurance

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Modules 2 & 3
Nuclear Quality Assurance

Instructors: Gene Imbro & Wes Rowley

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Learning Objectives

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- **Learn about the background of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"**
- **Learn how the 18 Criteria of Appendix B are applied in the siting, design, construction and operation of a nuclear facility**
- **Learn about the relation to of Part 50 Appendix B to Part 52**

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Learning Objectives (cont'd)

U.S.NRC
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- **Learn about ASME NQA-1 and it's relationship to Appendix B and any similarities and differences with Appendix B.**
- **Learn about Regulatory Guide 1.28, "QUALITY ASSURANCE PROGRAM CRITERIA, (DESIGN AND CONSTRUCTION)"**
- **Learn about ISO 9001 and the differences between it and NQA-1**

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Significant Sub Topics

U.S.NRC
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- **10 CFR 50, Appendix B**
  - Development/History
  - Content
  - Application
- **ASME-NQA-1**
  - Content
  - Differences from Appendix B
- **Regulatory Guide 1.28**
- **ISO-9001**

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Major References

- **10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"**
- **ASME NQA-1, *Quality Assurance for Nuclear Facility Applications***
- **ISO 9001**
- **Regulatory Issue Summary (RIS) 2000-18, "Guidance on Managing Quality Assurance Records in Electronic Media," dated October 23, 2000**

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Major References (cont'd)

- **Information Notice (IN) 86-21, "*Recognition of American Society of Mechanical Engineers Accreditation Program for N Stamp Holders,*" dated March 31, 1986**
- **NRC SECY-03-117, "*Approaches for Adopting More Widely Accepted International Quality Standards*"**
- **IAEA Requirements and Guidance**
  - GS-R-3 , "*The Management System for Facilities and Activities*"
  - GS-G-3.1, "*Application of the Management System for Facilities and Activities*"

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Course Objectives

U.S.NRC
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- **Introduction to Nuclear Quality Assurance (QA) Concepts**
- **Provide an understanding of QA requirements and how they are applied in the life-cycle of a nuclear generating facility**
- **Gain familiarity with QA Standards commonly used both in the United States and internationally.**

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Course Outline

U.S.NRC
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Protecting People and the Environment

- **Introduction to Quality Assurance**
- **United States Code of Federal Regulations, 10 CFR 50, Appendix B, "Quality Assurance"**
- **Application of QA concepts throughout the life-cycle of a nuclear generating station, for example**
  - Siting
  - Design
  - Procurement
  - Construction
  - Commissioning and Operation

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Course Outline (cont'd)

- **ASME NQA-1, "Quality Assurance Requirements of Nuclear Facility Applications"**
- **ISO-9001**
- **IAEA Quality Standards**
- **Summary**

Introduction to Nuclear Quality Assurance

- **Fundamental Concepts**
  - What is quality assurance?
    - "All those planned and systematic actions necessary to provide
      adequate confidence that a structure, system, or component will perform satisfactorily in service." As defined in ASME NQA-1.
  - Quality and its Achievement is the Responsibility of Everyone
  - Independence of the Quality Assurance Organization
    - Direct Access to Appropriate Levels of Management
      - Independence from Cost or Schedule
      - Authority to Stop Work
  - Technical Competence of Auditors

Introduction to Nuclear Quality Assurance

U.S.NRC
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- **Fundamental Concepts (cont'd)**
  - Graded Quality Assurance
    - Quality Assurance Commensurate with SSC's Importance to Safety
  - Quality Can Not be Inspected Into an Activity
  - Quality Assurance (QA) vs. Quality Control (QC)
    - Quality Assurance is a program that controls activities affecting quality for SSCs important to safety
    - Quality Control applies to the control of the physical quality of a material or SSC to predetermined requirements

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10 CFR 50, Appendix B "Quality Assurance"

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- **History and Evolution of Nuclear Quality Assurance in the United States**
  - Nuclear Weapons Program during WWII
  - Navy Nuclear Submarine Program, early 1960s
  - ASME Standard for nuclear reactor vessels and piping, 1967
  - Joint AEC (Atomic Energy Commission) NASA (National Astronautics and Space Agency) programs on space applications for nuclear technology, 1968
  - 10 CFR50 Appendix B issued, 1970 after two years of public comment and trial use at the Surry plant.

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10 CFR 50, Appendix A General Design Criteria

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- **Criterion 1–Quality standards and records.**
  - Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

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10 CFR 50, Appendix B "Quality Assurance"

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- **Criterion I Organization**
  - The applicant shall be responsible for the establishment and execution of the quality assurance program.
  - The applicant may delegate to others, the work of establishing and executing the quality assurance program, but shall retain responsibility for the quality assurance program.
  - The authority and duties of persons and organizations performing activities affecting the safety-related functions of structures, systems, and components shall be clearly established and delineated in writing.
  - These activities include both the performing functions of attaining quality objectives and the quality assurance functions. The quality assurance functions are those of (1) assuring that an appropriate quality assurance program is established and effectively executed; and (2) verifying, such as by checking, auditing, and inspecting, that activities affecting the safety-related functions have been correctly performed.

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10 CFR 50, Appendix B "Quality Assurance"

- **Criterion I (cont'd)**
  - The persons and organizations performing quality assurance functions. shall have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions
  - These persons and organizations performing quality assurance functions shall report to a management level so that the required authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations, are provided.

10 CFR 50, Appendix B "Quality Assurance"

- **Criterion II - Quality Assurance Program**
  - The applicant shall establish at the earliest practicable time, a quality assurance program which complies with the requirements of this appendix.
  - This program shall be documented by written policies, procedures, or instructions and shall be carried out throughout plant life.
  - The applicant shall identify the structures, systems, and components to be covered by the quality assurance program and the major organizations participating in the program, together with the designated functions of these organizations.
  - The quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components, to an extent consistent with their importance to safety,

10 CFR 50, Appendix B "Quality Assurance"

- **Criterion II - Quality Assurance Program**
  - Activities affecting quality shall be accomplished under suitably controlled conditions. Controlled conditions include the use of appropriate equipment; suitable environmental conditions for accomplishing the activity, such as adequate cleanness; and assurance that all prerequisites for the given activity have been satisfied.
  - The program shall take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality, and the need for verification of quality by inspection and test.
  - The program shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained.
  - The applicant shall regularly review the status and adequacy of the quality assurance program. Management of other organizations participating in the quality assurance program shall regularly review the status and adequacy of that part of the quality assurance program which they are executing

10 CFR 50, Appendix B "Quality Assurance"

- **Criterion III, "Design Control"**
  - Verifies that the final design of SSCs meets the licensing basis.
  - Provides assurance that regulatory requirements, and design bases have been correctly translated into design, procurement and procedural documents.
  - Scope includes all aspects of the design process such as engineering analyses , field design changes and their disposition, and verification and validation of computer programs used in the design process.
  - Includes verification or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculation methods, or by the performance of a suitable testing program.

10 CFR 50, Appendix B "Quality Assurance"

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- **Criterion IV - Procurement Document Control**
  - Procurement documents shall be controlled under the quality assurance program established by the applicant
  - Procurement documents shall contain any regulatory requirements, such as quality assurance requirements, material traceability and reportability requirements to be imposed on the vendor or supplier or any specific owner requirements. requirements.
  - The procurement documents shall also specify the documentation to be provided with the item, such as drawings; installation, operation and repair manuals and procedures; lists of parts and ordering information; spare and replacement parts or assemblies to be provided with the item; and special instructions or procedures for shipping, storage, handling, cleaning and preservation of the material or equipment being supplied.

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10 CFR 50, Appendix B "Quality Assurance"

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- **Criterion V - Instructions, Procedures and Drawings**
  - Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, and shall be of a type appropriate to the circumstances. These activities and shall be accomplished in accordance with these instructions, procedures, or drawings.
  - Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that activities important to safety have been satisfactorily accomplished
  - The activity shall be described to a level of detail commensurate with the complexity of the activity and the need to assure consistent and acceptable results. The need for, the level of detail in, written procedures or instructions shall be determined based upon complexity of the task, the significance of the item or activity, work environment, and worker proficiency and capability (education, training, experience).

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- **Criterion VI - Document Control**
  - Document control includes the following:
    - Identification of controlled documents
    - Specified distribution of controlled documents at specified locations
    - Identification of individuals responsible for the preparation, review, approval and distribution of controlled documents
    - Review of controlled documents prior to distribution
    - A method to insure that the latest revision of the documents are being used.

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10 CFR 50, Appendix B "Quality Assurance"

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- **Criterion VII - Control of Purchased Items and Services**
    - Procurement of items and services shall be controlled to ensure conformance with specified requirements
    - Supplier Evaluation and Selection
      – Evaluation of objective evidence of quality furnished by the Supplier,
      – Source inspection or audit

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10 CFR 50, Appendix B "Quality Assurance"

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- **Criterion VII - Control of Purchased Items and Services (cont'd)**
  - Acceptance of Item or Service
    - The extent of the verification activities by the purchaser shall be a function of the relative importance, complexity, and quantity of the item or services procured and the Supplier's quality performance.
  - Methods of acceptance include:
    - Supplier Certificate of Conformance
    - Source verification
    - Receiving inspection
    - Post installation test at the nuclear facility site
    - A combination of these methods

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10 CFR 50, Appendix B "Quality Assurance"

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- **Criterion VIII - Identification and Control of Items**
  - Controls shall be established to assure that only correct and accepted items are used or installed.
  - Identification shall be maintained on the items or in documents traceable to the items, or in a manner that assures that identification is established and maintained.
    - Physical identification shall be used to the extent possible.
    - Items having limited calendar or operating life need to be appropriately controlled and identified to preclude use after the shelf or operating life had expired.

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10 CFR 50, Appendix B "Quality Assurance"

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- **Criterion IX - Control of Special Processes**
  - Special processes that control or verify quality, such as those used in welding, heat treating, and nondestructive examination, shall be performed by qualified personnel using qualified procedures in accordance with specified requirements.
    - Special processes shall be controlled by instructions, procedures, drawings, checklists, travelers, or other appropriate means.
    - Special process instructions shall include or reference procedure, personnel, and equipment qualification requirements.
    - Conditions necessary for accomplishment of the process shall be included. These conditions shall include proper equipment, controlled parameters of the process, specified environment, and calibration requirements
    - Records shall be maintained for currently qualified. personnel, processes and equipment of each process.

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- **Criterion X - Inspection**
  - Inspections required to verify conformance of an item or activity to specified requirements or continued acceptability of items in service shall be planned and executed.
  - Characteristics subject to inspection and inspection methods shall be specified.
  - Inspection results shall be documented.
  - Inspection for acceptance shall be performed by qualified persons other than those who performed or directly supervised the work being inspected.
  - Hold points beyond which work may not proceed must be specified in the appropriate documents.
  - Periodic inspections (e.g., in-service inspections) or surveillances of structures, systems, or components shall be planned and executed to assure the continued performance of their required functions.

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- **Criterion XI - Test Control**
  - Written test procedures should incorporate the requirements and acceptance limits contained in applicable design documents.
  - The test program shall include, as appropriate, proof tests prior to installation, preoperational tests, and operational tests during nuclear power plant operation, of structures, systems, and components.
  - A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed.
  - For those computer programs used in design activities, computer program test procedures shall provide for assuring that the computer program produces correct results.

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- **Criterion XII - Control of Measuring and Test Equipment**
  - Measures shall be established to assure that tools, gages, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within necessary limits.
  - Measuring and test equipment shall be calibrated, at prescribed times or intervals and whenever the accuracy of the measuring and test equipment is suspect.
  - Calibration shall be against and traceable to certified equipment or reference standards
  - Calibration procedures shall identify or reference required accuracy and shall define methods and frequency of checking accuracy.

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10 CFR 50, Appendix B "Quality Assurance"

- **Criterion XIII - Handling, Storage and Shipping**
  - Measures shall be established to control the handling, storage, shipping, cleaning and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration.
  - When necessary for particular products, special protective environments, such as inert gas atmosphere, specific moisture content levels, and temperature levels, shall be specified and provided.

10 CFR 50, Appendix B "Quality Assurance"

- **Criterion XIV - Inspection, Test and Operating Status**
  - Measures shall be established to indicate, by the use of markings such as stamps, tags, labels, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the nuclear power plant.
  - These measures shall provide for the identification of items which have not satisfactorily passed required inspections and tests, to ensure that items that have not passed the required inspections and tests are not inadvertently installed, used, or operated.
  - Measures shall also be established for indicating the operating status of structures, systems, and components of the nuclear power plant, such as by tagging valves and switches, to prevent inadvertent operation.

10 CFR 50, Appendix B "Quality Assurance"

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- **Criterion XV - Nonconforming Materials, Parts, or Components**
  - Items that do not conform to specified requirements shall be controlled to prevent inadvertent installation or use.
  - Nonconforming items shall be identified by legible marking, tagging, or other methods not detrimental to the item and segregated until properly dispositioned.
  - The nonconforming items shall be evaluated and dispositioned and if appropriate reexamined in accordance with the original acceptance criteria.

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- **Criterion XVI - Corrective Action**
  - Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.
  - In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.
  - The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.

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10 CFR 50, Appendix B "Quality Assurance"

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- **Criterion XVII - Quality Assurance Records**
  - Quality assurance records shall furnish documentary evidence that items or activities meet specified quality requirements.
  - Quality assurance records shall be identified, generated, authenticated, and maintained, and their final disposition specified.
  - Records to be generated, supplied, or maintained shall be specified in applicable documents, such as design specifications, procurement documents, test procedures, and operational procedures.
  - Records shall be classified as lifetime or nonpermanent and maintained by the Owner, or authorized agent.

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- **Criterion XVIII - Audits**
  - A comprehensive system of planned and periodic audits shall be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program.
  - The audits shall be performed in accordance with the written procedures or check lists by appropriately trained personnel not having direct responsibilities in the areas being audited.
  - Audit results shall be documented and reviewed by management having responsibility in the area audited.
  - Followup action, including reaudit of deficient areas, shall be taken where indicated

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Quality Assurance During Life Cycle Siting

- **Principle Activities During Site Suitability Characterization**
  - Determination of Soil Characteristics
  - Adequacy of Ultimate Heat Sink
  - Adequacy of Cooling Water Makeup
  - Adequacy, Reliability, and Stability of Offsite Power
  - Evaluation of Human Hazards
    - Transportation of Hazardous Materials
  - Evaluation of Natural Hazards
    - Seismic
    - Strong Wind (Tornado)
    - Flooding, Tsunami, Seiche, etc.
  - Distance and Location of Nearby Population Centers
    - Atmospheric Dispersion
  - Emergency Preparedness Considerations
    - Evacuation Routes, etc.

Quality Assurance During Life Cycle Design and Procurement

- **Principal Activities During the Design and Procurement Process**
  - Performance of analyses by hand calculation or computer
    - Important Aspects of the Design Process
      - – Control of design input data
      - – Control of design outputs
      - – Independent Checking of Analyses
        - » Review and disposition of software error reports
      - – Verification and Validation of Computer Codes
        - » Appropriate Benchmarking Using Verified Examples
      - – Configuration Control, i.e., control of changes
  - Preparation of Purchase Specifications
  - Preparation of Design / Construction Drawings
  - Bid Evaluation
  - Vendor audits
  - Receipt Inspection
  - Storage Requirements

Quality Assurance During Life Cycle Construction

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- **Principal Activities During Construction**
  - Site preparation
    - Excavation and Placement of Engineered Backfill as Necessary
  - Rebar/Concrete Placement
  - Welding
    - Qualification of Welders and Weld Procedures
    - Control of Consumables, e.g., weld rod/filler metal
  - Development of Operating Procedures, AOPs and EOPs
  - Initial Training of Operators and Operator Licensing
  - Installation of Equipment and Piping
    - Equipment placement and alignment
  - Pressure Boundary Integrity and Functional Tests of Equipment and Systems
  - System and Area Turnover to the Owner

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Quality Assurance During Life Cycle Commissioning & Operations

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- **Principal Activities During Commissioning and Operation**
  - Verification of Operability of Systems and Equipment
    - Complete Preservice Inservice Inspection and Inservice Testing
  - Startup Test Program
  - Initial Criticality
  - Power Ascension
  - Plant Shutdown
  - Refueling
  - Response to Plant Transients and Accidents
  - Periodic Test and Surveillances
  - Ongoing Training and Requalification of Operators
  - Maintaining Plant Configuration Control
  - Plant Modifications...Conformance with Licensing and Design Bases

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ASME – NQA 1 - 2008
Quality Requirements for Nuclear Facility Applications

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- **NQA – 1 Overview**
  - Part I - Requirements for Quality Assurance Programs for Nuclear Facilities
  - Part II - Quality Assurance Requirements for Nuclear Facility Applications
  - Part III - Nonmandatory Appendices
  - Part IV - Nonmandatory Appendices: Positions and Applications Matrices

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ASME – NQA 1 - 2008
Quality Requirements for Nuclear Facility Applications

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- **Regulatory Guide 1.28, Rev 4, June 2010**
  - Endorses Parts I and II of NQA-1-2008 and the NQA-1a-2009 Addenda as providing an adequate basis for complying with the requirements of 10 CFR 50, Appendix B for the implementation of a Quality Assurance Program during the design and construction phases of nuclear power plants and fuel reprocessing plants.

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ASME – NQA 1 - 2008 Quality Requirements for Nuclear Facility Applications

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- **Additions and Modifications to NQA-1-2008 and NQA-1a-2009 Addenda imposed by Reg Guide 1.28, Rev 4:**
  - Lifetime and Nonpermanent Records
  - Refers to RIS-2000-18, ""Guidance on Managing Quality Assurance Records in Electronic Media," dated October 23, 2000
  - Internal and External Audits
    - Timing and Frequency
    - Vendor audits

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ASME – NQA 1 - 2008 Quality Requirements for Nuclear Facility Applications

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- **Criterion 3 of Appendix B**
  - Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions.
  - These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled.

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ASME – NQA 1 - 2008
Quality Requirements for Nuclear Facility Applications

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- **Criterion 3 Design Control (cont'd)**
  - Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.
  - Measures shall be established for the identification and control of design interfaces and for coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces.

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Quality Requirements for Nuclear Facility Applications

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- **Criterion 3 Design Control (cont'd)**
  - The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.
  - The verifying or checking process shall be performed by individuals or groups other than those who performed the original design, but who may be from the same organization.

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- **Criterion 3 Design Control (cont'd)**
  - Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it shall include suitable qualifications testing of a prototype unit under the most adverse design conditions
  - Design control measures shall be applied to items such as the following: reactor physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests.

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- **Criterion 3 Design Control (cont'd)**
  - Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design unless the applicant designates another responsible organization.

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- **ASME NQA-1, Requirement 3, Design Control**
  - 100 Basic
  - 200 Design Input
  - 300 Design Process
  - 400 Design Analysis
  - 500 Design Verification
  - 600 Change Control
  - 700 Interface Control
  - 800 Software Design Control
  - 900 Documentation and Records

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- **ASME NQA-1, Requirement 3, Design Control**
  - The design shall be defined, controlled, and verified. Design inputs shall be specified on a timely basis and translated into design documents
  - Design documents shall support facility design, construction, and operation.
  - Design input: those criteria, performance requirements, codes and standards, design bases, regulatory requirements, or other design requirements upon which detailed final design is based.

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- **Appendix B, Criterion 3, Design Control**
  - Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies *are* correctly translated into specifications, drawings, procedures, and instructions.

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- **ASME NQA-1, Requirement 3, Design Control**
  - Appropriate quality standards shall be identified and documented, and their selection shall be reviewed and approved.
  - Design changes shall be governed by control measures commensurate with those specified in the original design
  - The responsible design organization shall prescribe and document design activities to the level of detail…to permit verification …that design requirements are met.

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- **Appendix B, Criterion 3, Design Control**
  - These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled.

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- **ASME NQA-1, Requirement 3, Design Control**
  - The design methods, materials, parts, equipment and process that are essential to the function of the items shall be selected and reviewed for suitability of application.
- **Appendix B, Criterion 3, Design Control**
  - Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.

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- **ASME NQA-1, Requirement 3, Design Control**
  - Interface controls shall include assignment of responsibility and establishment of procedures among participating design organizations for review, approval, release, distribution, and revision of documents involving design interfaces.
- **Appendix B, Criterion 3, Design Control**
  - Measures shall be established for the identification and control of design interfaces and for coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces.

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- **NQA-1 Requirement 3, Design Control**
  - 700 Interface Control

    "Design information transmitted across interfaces shall identify the status of the design information or document provided, and identify incomplete items that require further evaluation, review, or approval. Where it is necessary to initially transmit design information orally or by other informal means, the transmittal shall be confirmed promptly by a controlled document."

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- **Part I - Requirements for Quality Assurance Programs for Nuclear Facilities**
  - NQA-1 Requirement 9, "Control of Special Processes"
    - 100 BASIC
      - Special processes that control or verify quality, such as those used in welding, heat treating, and nondestructive examination, shall be performed by qualified personnel using qualified procedures in accordance with specified requirements.
  - Appendix B Criteria IX, "Control of Special Processes"
    - Measures shall be established to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

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Quality Requirements for Nuclear Facility Applications

- **NQA-1 Requirement 9, "Control of Special Processes"**
  - 200 Process Control
    - 201 Special Processes
      - Special processes shall be controlled by instructions, procedures, drawings, checklists, travelers, or other appropriate means.
      - Special process instructions shall include or reference procedure, personnel, and equipment qualification requirements. Conditions necessary for accomplishment of the process shall be included. These conditions shall include proper equipment, controlled parameters of the process, specified environment, and calibration requirements.

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- **Treatment of Commercial Grade Items in NQA-1-2008**
  - Requirement 3 Design Control
  - 300 Design Control
    - The final design shall identify assemblies and/or components that are part of the item being designed. When such an assembly or component is a *commercial grade item*, the critical characteristics need to be verified for acceptance and the acceptance cri teria for those characteristics shall be documented.
    - Critical characteristics provide reasonable assurance that the component will perform its intended function.

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- **Treatment of Commercial Grade Items in NQA-1-2008 (cont'd)**
  - Requirement 7, Control of Purchased Items
    - 700 Commercial Grade Items and Services
      - Utilization
      - Critical Characteristics
      - Dedication
      - Supplier Deficiency Correction

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- **Treatment of Commercial Grade Items in NQA-1-2008 (cont'd)**
  - SUBPART 2.14, Quality Assurance Requirements for Commercial Grade Items and Services
    - Utilization
    - Technical Evaluations
    - Critical Characteristics
    - Methods of Accepting Commercial Grade Items
    - Commercial Grade Services
    - Documentation

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Quality Requirements for Nuclear Facility Applications - Part II

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- **Part II - Quality Assurance Requirements for Nuclear Facility Application**
  - Part II contains 13 Subparts amplifying quality assurance requirements for certain specific work activities that occur at various stages of a facility.
    - Subparts include amplifying requirements used in the following stages of a facility's life:
      - Siting
      - Construction
      - Maintenance
      - Operations

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ASME – NQA 1 - 2008 Quality Requirements for Nuclear Facility Applications - Part II

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- **Subpart 2.5 - Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete, Structural Steel, Soils. And Foundations for Nuclear Power Plants**
  - 100 GENERAL
    - Provides amplified requirements for installation, inspection, and testing of structural concrete, structural steel, soils, and foundations.
    - Requirements apply to any individual or organization participating in work relating to the production, preparation, placement, installation, inspection and testing of structural concrete, structural steel and foundations.

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- **Subpart 2.5**
  - 200 “GENERAL REQUIREMENTS”
    - Specifies Applicability
      – Formwork
      – Steel Reinforcement
      – Embedded Items
      – Foundation Preparation, etc.
    - 300 “REQUIREMENTS”
      – Measures shall be established and implemented for documenting installation, inspection, and testing activities to verify conformance to specified requirements.

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ASME – NQA 1 - 2008 Quality Requirements for Nuclear Facility Applications - Part II

- **Subpart 2.5**
  - 400 "PRECONSTRUCTION VERIFICATION"
    - Receipt and storage inspections
    - Materials Suitability
      – Concrete
      – Rebar
      – Welding Materials
    - Construction Processes
      – Welding
      – Structural Bolting
      – Splicing of Rebar
      – Concrete Mixing, Measuring. Placement and Curing

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ASME – NQA 1 - 2008 Quality Requirements for Nuclear Facility Applications - Part II

- **Subpart 2.5**
  - 500 "INSPECTIONS OF SOIL AND EARTHWORK"
    - Materials
    - Placing and Compaction Equipment
    - Preplacement Preparations
    - Soil Compaction
    - In-Process Tests on Compacted Fill
  - 600 "INSPECTION OF FOUNDATION PILE and CAISSON CONSTRUCTION"
  - 700 "INSPECTION OF CONCRETE CONSTRUCTION
  - 800 "INSPECTION OF STEEL CONSTRUCTION"

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Quality Requirements for Nuclear Facility Applications - Part III

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- **Part III "NONMANDATORY APPENDICES"**
  - Contains 3 Subparts
    - 3.1 Nonmandatory Guidance on Quality Assurance Programs for Nuclear Applications
    - 3.2 Nonmandatory Guidance on Quality Assurance Programs for Nuclear Facility Applications
    - 3.3 Nonmandatory Guidance on Quality Assurance Program Requirements For Collection of Scientific and Technical Information for Site Characterization of High-Level Nuclear Waste Repositories

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Quality Requirements for Nuclear Facility Applications - Part IV

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- **Part IV "NONMANDATORY APPENDICES - POSITIONS AND APPLICATIONS MATRICES**
  - Contains 6 Subparts
    - 4.1 "Guide on Quality Assurance Requirements for Computer Software"
    - 4.3 "Guide to Modification of an ISO-9001-2008 Quality Program to Meet NQA-1 Requirements"
    - 4.4 "Application Guide for Managing Electronic Information"

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Quality Requirements for Nuclear Facility Applications

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- **NRC Endorsement of NQA-1a-2008 (Addenda to NQA-1-2008)**
  - Draft Guide DG-1215, Proposed Revision 4 to Regulatory Guide 1.28 has been issued.
  - http://adamswebsearch2.nrc.gov/idmws/doccontent.dll?library=PU_ADAMS^PBNTAD01&ID=091870463
  - DG-1215 contains additions and modifications to NQA-1a-2008
  - Includes NRC endorsement of Parts I and II of NQA-1a-2008.
    - Previously NRC endorsement was limited to Part I of NQA-1,

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ISO-9001 - "Quality Management System Requirements"

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- **International Standards Organization**
  - ISO-9001
    - Contract Orientation
    - Not Specific to Nuclear Plants
    - Control of Measuring and Test Equipment
    - Contains Statistical Process Controls
  - NQA-1/Appendix B
    - Focus is on safety of nuclear facilities

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Endorsement Topics for ISO 9001

- **Criteria**
  - Acceptance criteria, design bases, regulatory requirements.
- **Safety Orientation**
  - Safety analyses, corrective action notification requirements, research and development of new technical approaches, environmental qualification, tying quality to safety, applicability of QA program to identified SSCs.
- **Independence**
  - Independence of verifiers and checkers, inspectors, and QA auditors. Authority and freedom from cost and schedule pressures of management responsible for quality activities.

Endorsement Topics for ISO 9001 (cont'd)

- **Testing**
  - Testing environments, procedures for non-conforming SSCs in test and inspection.
- **Control**
  - Applicable codes and standards, personnel qualification, design change control commensurate with control on original design.
- **Documentation**
  - Location, retention, and availability of quality records

Criteria for ISO-9001 Contract Compatibility With 10 CFR 50, Appendix B

- **Acceptance Criteria**
  - Contracts with an ISO supplier must include acceptance criteria that are selected to ensure that SSCs being procured fulfill their safety functions. ISO 9001 does not include specific regulatory safety criteria.
- **Applicability**
  - Contracts with an ISO 9000 supplier should identify structures, systems, and components, major organizations, and the functions of those organizations covered by the QA program. A project QA plan pursuant to ISO 9001 is acceptable and would be required.
- **Authority**
  - Contracts should specify that persons doing QA shall report to managers having sufficient authority and organizational independence from cost and schedule that safety (of SSCs) is not impacted.

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Criteria for ISO-9001 Contract Compatibility With 10 CFR 50, Appendix B (cont'd)

- **Contract**
  - Contracts don't appear in Appendix B, but in ISO-9001 must contain the acceptance criteria and specifications to which the ISO 9000 supplier has agreed to work. Contract review should resolve differences and determine if supplier can meet specs.
- **Control, Change**
  - The contract should specify that changes shall be subject to control measures commensurate with the measures applied to the original design and approved by the original designers or designated successors.
- **Control, Processes**
  - Appendix B is more explicit about qualification of personnel (e.g. welders) to construction standards and codes. Contract may need to specify code qualification if this is necessary to meet regulatory requirements or applicant/licensee commitments.

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Criteria for ISO-9001 Contract Compatibility With 10 CFR 50, Appendix B (cont'd)

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- **Corrective Action**
  - Regulatory notification requirements need to be in the contract or referenced as part of required corrective action procedures.
- **Design Basis**
  - Regulatory requirements and design bases need either be stated or referenced in contracts with ISO 9000 suppliers.
- **Documentation Requirement**
  - Design basis and regulatory requirements are contract provisions. Location of quality records is at the ISO 9000 supplier's discretion unless specified by contract. Contract would have to specify quality record locations matching NRC requirements. Documentation requirements of ISO 9000 otherwise a superset of Appendix B.

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Criteria for ISO-9001 Contract Compatibility With 10 CFR 50, Appendix B (cont'd)

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- **Independence**
  - Contracts should specify that persons doing QA shall report to managers having sufficient authority and organizational independence from cost and schedule that safety (of SSCs) is not impacted.
  - Additional criteria: Verifying and checking designs shall be performed by someone other than the designer; Inspections shall be performed by other than those who did the work; Audits shall be carried out by personnel not directly responsible for the area being audited. Since ISO 9001 uses different terminology, the contract should make clear that Appendix B requirements are met.
- **Regulatory Requirements**
  - Regulatory requirements would need to be referenced in the contract.

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Criteria for ISO-9001 Contract Compatibility With 10 CFR 50, Appendix B (cont'd)

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- **Research and Development**
  - R&D requirements would need to be stated in the contract, because ISO 9000 mentions only R&D to identify and develop necessary measurement techniques: identification of any required measurement technique that exceeds the state of the art and needs to be developed. Appendix B requires prototyping to verify design or construction features whose properties have an uncertain effect on safety.
- **Safety Analysis**
  - The safety analysis approach of the NRC, 10 CFR 50, NUREG-800, and other documents would need to be referenced in the supplier's contract. Specific safety analyses for which the supplier had responsibility would need to be enumerated. The issue of environmental qualification would need to be addressed.
- **Testing**
  - Contract should include most adverse design conditions for test and acceptance criteria, and what to do about non-conformance. Since ISO 9001 uses different terminology, the contact should make it clear that Appendix B requirements are met.

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NRC Comparison of Quality Standards

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- **NRC SECY-03-117, "Approaches for Adopting More Widely Accepted International Quality Standards"**
  - http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2003/secy2003-0117/2003-0117scy.pdf#pagemode=bookmarks
  - NRC's SECY-03-117 Compared the following:
    - 10 CFR 50, Appendix B to ISO-9001-2000, "Quality Management System Requirements
    - ISO-9001-2000, "Quality Management System Requirements to ASME NQA-1, Quality Assurance Requirements for Nuclear Facility Applications"

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IAEA Comparison of IAEA Quality Requirements to ISO-9001

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- **IAEA Safety Reports Series No. 22, "Quality Standards: Comparison Between IAEA 50-C/SG-Q and ISO 9001-2000.**
  - http://www-pub.iaea.org/MTCD/publications/PDF/Pub1127_scr.pdf
    - IAEA 50-C-Q, "Code on the Safety of Nuclear Power Plants: Quality Assurance"

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IAEA Quality Standards

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- **IAEA 50-C-Q revised to provide an Integrated Management System**
  - GS-R-3 , "The Management System for Facilities and Activities"
  - GS-G-3.1, "Application of the Management System for Facilities and Activities"
  - DS 349, "Application of the Management System for Nuclear Facilities" Draft

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IAEA Quality Standards

- GS-R-3 , "The Management System for Facilities and Activities"
- Defines the high-level requirements for establishing, implementing, assessing and continually improving a management system
- Requirements integrate safety, health, environmental, security, quality and economic elements.
- Applicability
  - Nuclear facilities;
  - Activities using sources of ionizing radiation;
  - Radioactive waste management;
  - Transport of radioactive material;
  - Radiation protection activities;
  - The regulation of such facilities and activities

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IAEA Quality Standards

- GS-G-3.1, "Application of the Management System for Facilities and Activities"
  - Supports GS-R-3
  - Identical Scope to GS-R-3
  - Focus on Management Systems
  - Emphasis on "Safety Culture"
    - Safety is a Clearly Recognized Value
    - Leadership for Safety is Clear
    - Accountability for Safety is Clear
    - Safety is Integrated into all activities
    - Safety is Learning Driven

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IAEA Quality Standards

- GS-G-3.1, "Application of the Management System for Facilities and Activities" - (cont'd)
  - Recommends a "Graded Approach" for Management Systems based on significance.
    - Leveraging of resources to focus on the most significant items or activities
  - Recommends Measurement, Assessment and Improvement of Performance by:
    - Critical Self Assessments
    - Independent Assessments
      - » Internal Audits
      - » Surveillance of Work Performance

Summary

- **Course Objectives**
  - Introduction to Nuclear Quality Assurance (QA) Concepts
  - Provide an understanding of QA requirements and how they are applied in the life-cycle of a nuclear generating facility
  - Gain familiarity with QA Standards commonly used both in the United States and internationally.

Module 04

Standard Review Plan (NUREG-0800) Chapter 3

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Module 04

Standard Review Plan (NUREG-0800) Chapter 3

Instructor: Wes Rowley, P.E.

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Learning Objectives

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- **Principal design requirements for mechanical components and design bases**
- **Seismic Classification, Quality Group Classification**
- **Loads and load combinations**
- **Seismic and dynamic analysis, testing, and qualification**
- **Environmental design and qualification**
- **Design of ASME Code Class mechanical components, and component supports**

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Introduction

1. Seismic Classification (SRP 3.2.1)
2. System Quality Group Classification (SRP 3.2.2)
3. Wind Loading (SRP 3.3.1)
4. Tornado Loads (SRP 3.3.2)
5. Generated Missiles (SRP 3.5.1.1 -3.5.1.4)
6. Foundations (SRP 3.8.5)
7. Dynamic Testing and Analysis of Systems, Structures, and Components (SRP 3.9.2)
8. ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures (SRP 3.9.3)

Introduction (cont'd)

9. Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints (SRP 3.9.6)
10. Seismic and Dynamic Qualification of Mechanical and Electrical Equipment (SRP 3.10)
11. Environmental Qualification of Mechanical and Electrical Equipment (SRP 3.11)
12. ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports (SRP 3.12)
13. Threaded Fasteners - ASME Code Class 1, 2, and 3 (SRP 3.13)

References

1. ASME BPV Code, Section III, Division 1
2. Regulatory guides 1.20, 1.26, 1.29, 1.61, 1.76
3. SRP Sections 2.5.2, 3.7.1, 3.7.2, 3.7.3, 3.8.1, 3.8.4

Seismic Classification (SRP 3.2.1)

- Seismic classification - to identify Seismic Category I SSCs, and those SSCs, failure of which will interfere with Category I SSCs fulfilling their safety functions
- Seismic Category I SSCs are those important to safety that must be designed to remain functional if an SSE occurs
- Safety Shutdown Earthquake

Seismic Classification (SRP 3.2.1)

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- **Seismic Category I SSCs are subject to the QA requirements of Appendix B to 10 CFR Part 50**

  - Seismic Category I SSCs include those to assure:
    - The integrity of the reactor coolant pressure boundary
    - The capability to shutdown the reactor and maintain it in a safe shutdown condition

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Seismic Classification (SRP 3.2.1)

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- The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 50.34(a)(1)

- Regulatory Guidance 1.29 “Seismic design classification”
  - Examples of Seismic Category I
    - The reactor coolant pressure boundary
    - The reactor core and reactor vessel internals

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Seismic Classification (SRP 3.2.1)

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- Examples of Seismic Category I (cont'd)
  - Systems or portions thereof that are required for emergency core cooling, post-accident containment heat removal, or post-accident containment atmosphere cleanup
  - Systems or portions thereof that are required for reactor shutdown, residual heat removal, or cooling of the spent fuel storage pool
  - Cooling water and seal water systems or portions thereof that are required for functioning of the RCS components important to safety such the RCPs

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Seismic Classification (SRP 3.2.1)

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- SSCs that are not required to remain functional following a seismic event, but whose failure could reduce the function of any Category I SSCs to an unacceptable safety level, or could result in incapacitating injury to control room occupants, must be analyzed and designed to maintain their integrity under seismic loading from the SSE

- Regulatory Guides 1.143, 1.151 & 1.191 provide seismic design requirements for radwaste management system, instrumentation pipelines, fire protection system respectively

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Quality Group Classification (SRP 3.2.2)

- **Regulatory requirement basis: GDC 1 and 10 CFR 50.55a**
  - Quality Group A – 10 CFR 50.55a Paragraph (c)
  - Regulatory guide 1.26 provides an acceptable approach for identification of Quality Group B, C, and D items on a functional basis
    - Group B – regulatory position 1
    - Group C – regulatory position 2

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Quality Group Classification (SRP 3.2.2)

- Regulatory guide 1.26 (cont’d)
  - Quality Group D – regulatory position
  - Quality standards (shown in the next Slide)

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Quality Group Classification (SRP 3.2.2)

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	QUALITY STANDARDS		
Components	**Quality Group B**	**Quality Group C**	**Quality Group D**
Pressure Vessel	ASME Section III Division 1, Subsection NC	ASME Section III Division 1, Subsection ND	ASME Section VIII Division 1
Piping	As above	As above	ASME B31.1
Pumps	As above	As above	Manufacturers' standards
Valves	As above	As above	ASME B31.1
Atmospheric Storage Tanks	As above	As above	API-650, AWWA D-100, or ASME B96
0–15 psig Storage Tanks	As above	As above	API-620

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Wind Loading (SRP 3.3.1)

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- **GDC 2 requires SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions**

- **Plant Structures must be designed to withstand the effects of the design wind speed specified for the plant**

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Wind Loading (SRP 3.3.1)

- **Design wind speed shall be the most severe wind that has been historically reported for the site and surrounding area**
- **ASCE/SEI 7-05 "Minimum Design Loads for Buildings and Other Structures" – provides acceptable procedures to transform the wind speed into an equivalent pressure to be applied to structures and parts, or portions of structures**

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Wind Loading (SRP 3.3.1)

- **For a design wind speed V, the velocity pressure qz, evaluated at height z:**
- **qz = 0.00256 KzKdtKdV2I (lb/ft2)**
- **Kz, velocity pressure exposure coefficient evaluated at height z**
- **Kdt , topographic factor equal to 1.0**
- **Kd,wind directional factor equal to 1.0**
- **V, design wind speed in miles per hour**
- **I, importance factor equal to 1.5**

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Tornado Loading (SRP 3.3.2)

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- Plant structures shall be designed to withstand the effects of the specified design-basis tornado for the plant
- Any structure or component, if not designed for tornado loads, should be demonstrated its failure will not impact the capability of other structures or components to perform their safety functions

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Tornado Loading (SRP 3.3.2)

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- The tornado wind and associated missiles generated by the tornado wind used in the design shall be the most severe wind that has been historically reported for the site and surrounding area
- Design basis tornado characteristics

Table 1: Design basis tornado characteristics (RG 1.76)

Region	Maximum Wind Speed m/s (mph)	Translational speed m/s (mph)	Rotational maximum speed m/s (mph)	Radius of maximum rotational speed m (ft)	Pressure drop mb (psi)	Rate of pressure drop mb/s (psi/s)
I	103 (230)	21 (46)	82 (184)	45.6 (150)	83 (1.2)	37 (0.5)
II	89 (200)	18 (40)	72 (160)	45.6 (150)	63 (0.9)	25 (0.4)
III	72 (160)	14 (32)	57 (128)	45.6 (150)	40 (0.6)	13 (0.2)

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Tornado Loading (SRP 3.3.2)

- **Transforming tornado parameters into effective loads on structures**
  - Tornado wind effects
  - Atmospheric pressure change effects
    - Open structure
    - Enclosed structure
    - Vented structure
    - Tornado-generated missile impact effects
      - RG 1.76 provides tornado generated missile characteristics and design basis tornado missile spectrum

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Tornado Loading (SRP 3.3.2)

- **Combined tornado effects**

Wt = Wp (1)

Wt = Ww + 0.5 Wp + Wm (2)

Wt = total tornado load

Ww= load from tornado wind

Wp = load from tornado atmospheric pressure change effect

Wm = load from tornado missile impact effect

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Generated Missiles (SRP 3.5.1.1 -3.5.1.4)

- **GDC 4 requires "These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit."**

Generated Missiles (SRP 3.5.1.1 & 3.5.1.2)

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- **Internally generated missiles**
  - Identification of missiles
    - Determination of the potential of pressurized components and systems for generating missiles
    - Determination of the potential of high-speed rotating machinery for generating missiles from component overspeed or failures

Generated Missiles (SRP 3.5.1.1 & 3.5.1.2)

- Statistically significant missiles
  - Determine the probability of missile occurrence (p1); if p1 < 10-7/yr, the missile is not statistically significant. Otherwise,
  - Determine the probability of impact on a significant target (p2); if the product of p1 & p2 < 10-7, the missile is not statistically significant. Otherwise,
  - Determine the probability of significant damage (p3); if the product of p1, p2 & p3 >10-7, missile protection should be provided for SSCs important to safety and nonsafety-related SSCs whose failure could affect safety-related SSCs

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Generated Missiles (SRP 3.5.1.1 & 3.5.1.2)

- Missile protection measures
  - Locating the system or component in a missile proof structure
  - Separating redundant systems or components for missile path or range
  - Providing local shields or barriers
  - Designing equipment to withstand the impact of the most damaging missiles
  - Providing design features to prevent missile from generation
  - Orienting missile sources to prevent missiles from striking equipment important to safety

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Generated Missiles (SRP 3.5.1.3 Turbine Missiles)

- **Turbine missiles (SRP 3.5.1.3)**
  - Safety concern: the large steam turbines have rotors with large masses and rotate at relatively high speeds during normal operation. The failure of rotor may result in the generation of missiles that could affect safety-related SSCs
  - Safety-related plant SSCs must have adequate protection against the effects of potential turbine missiles

Generated Missiles (SRP 3.5.1.3 Turbine Missiles)

- The probability (P4) of unacceptable damage from turbine missiles should be less than or equal to 10^{-7} per year for the plant. P4 can be expressed as

$$P4 = P1 \times P2 \times P3$$

  - P1 = the probability of turbine failure resulting in the ejection of turbine rotor fragments through the turbine casing
  - P2 = the probability of ejected missiles perforating intervening barriers and striking safety-related SSCs
  - P3 = the probability of struck SSCs failing to perform their safety function

Generated Missiles (SRP 3.5.1.4)

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- **Missiles generated by tornados and extreme winds (SRP 3.5.1.4)**
  - Currently, missiles generated by design basis tornados are considered in the plant design bases for all plants
  - Missiles from hurricane and extreme winds are considered on a case-by-case basis when identified

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Foundations (SRP 3.8.5)

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- **Seismic Category I foundations**
  - Containment structure foundation
  - Containment enclosure building foundation
  - Auxiliary building foundations
  - Other Category I foundations, e.g., fuel storage buildings, diesel generator buildings, intake structures, cooling towers

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Foundations (SRP 3.8.5)

- **Codes, Standards, and Specifications**
  - The design, materials, fabrication, erection, inspection, testing, and surveillance, if any, of Seismic Category I foundations are covered by codes, standards, and guides.
- **Loads and load combinations**
  - Those loads encountered during construction
  - Those loads encountered during normal plant startup, operation, and shutdown

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Foundations (SRP 3.8.5)

- **Loads and load combinations (cont'd)**
  - Those loads due to severe environmental conditions, including OBE and design wind speed
  - Those loads due to extreme environmental conditions, including the SSE and design tornado
  - Those loads resulting from abnormal and accident plant conditions
  - Those loads induced by hydrodynamic loads

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Foundations (SRP 3.8.5)

- **Loads and load combinations (cont'd)**
  - Combination of loads
    - The combinations of loads to check against sliding and overturning attributable to earthquakes, winds, tornadoes and against flotation because of floods
      - D + H+ E
      - D + H+ W
      - D + H+ E'
      - D + H+ Wt
      - D + F'

Foundations (SRP 3.8.5)

- **Design and analysis procedures**
  - The design should consider the soil-structure interaction, hydrodynamic effect, and dynamic soil pressure.
  - Design and analysis procedures in the applicable codes should be used
  - The applicability and adequacy of the computer codes employed should be well established

Foundations (SRP 3.8.5)

- **Structural acceptance criteria**
  - For containment foundation, limits in accordance with CC-3400 of the ASME Code and RG 1.136
  - For other foundations: ACI 349 and RG 1.142
  - Factors of safety for the load combinations

	Overturning	Sliding	Flotation
D + H+ E1.5	1.5	1.5	---
D + H+ W	1.5	1.5	---
D + H+ E'	1.1	1.1	---
D + H+ W_t	1.1	1.1	---
D + F'	---	---	1.1

Dynamic Testing and Analysis of SSCs (SRP 3.9.2)

- **Vibration, thermal expansion and dynamic testing during startup testing**
- **Seismic analysis**
- **Dynamic response analysis**
- **Vibration testing during startup**

Dynamic Testing and Analysis of SSCs (SRP 3.9.2)

- **Vibration, thermal expansion and dynamic testing during startup testing**
  - Purpose
  - Scope
    - ASME Class 1, Class 2 and 3 systems
    - Seismic Category I piping systems
    - High-energy portions systems failure of which may impact Seismic I SSCs

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Dynamic Testing and Analysis of SSCs (Seismic Analysis)

- Seismic analysis of Seismic Category I mechanical equipment
- Seismic analysis method
  - Dynamic analysis method (e.g., response spectra, time history method)
  - Equivalent static load method
- Number of earthquake cycles
  - Seismic events
  - Cycles per seismic event

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Dynamic Testing and Analysis of SSCs (Seismic Analysis)

- Selection of frequencies
  - The fundamental frequencies of components and equipment should be preferably < ½ or > twice the dominant frequencies of the support structure
- Combination of modal responses
  - Combining the responses of individual modes (in spectra modal response analysis) to a component of the three orthogonal components of earthquake motion to find the representative response of interest for a SSC
  - Regulatory guide 1.92 "Combining modal responses and spatial components in seismic response analysis"

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Dynamic Testing and Analysis of SSCs Seismic Analysis

- Combining effects caused by the three components of earthquake motion
  - Response spectra method: SRSS combination of the corresponding representative maximum response from the three components calculated separately
  - Time history method:
    - Case 1 : the response to each component of quake motion is calculated separately
    - Case 2: the three components of the earthquake motion are statistically independent

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Dynamic Testing and Analysis of SSCs (Seismic Analysis)

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- Multiply-supported equipment and components with distinct inputs
  - Using an upper-bound envelope of all the individual response spectra for all the support locations
- Use of constant vertical static factors
- Torsional effects of eccentric masses
- Category I buried piping systems
  - The inertial effects due to an earthquake
  - The effects of static resistance of the surrounding soil
  - The effects of local soil settlements, soil arching, etc if applicable

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Dynamic Testing and Analysis of SSCs (Seismic Analysis)

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- Interaction of other piping with Category I piping
- Criteria used for damping
  - Damping
  - Critical damping
  - Regulatory Guide 1.61 “Damping values for seismic design of nuclear power plants”

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Dynamic Testing and Analysis of SSCs
Dynamic Response Analysis of Reactor Internals

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- **Dynamic response analysis of reactor internals**
  - Purpose
  - To assess the vibration behavior of the components, including the definition of the input-forcing functions and estimation of the consequent vibration and stress level
  - Results of vibration and stress calculations
    - Dynamic responses to operating transients at critical locations of internal structures and, in particular at locations where vibration sensors are mounted

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Dynamic Testing and Analysis of SSCs
Dynamic Response Analysis of Reactor Internals

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- Results of vibration and stress calculations (cont'd)
  - Damping factors for different modes
  - Dynamic properties of internal structures
  - Dynamic responses of reactor internals to self-excited flow oscillations
  - Dependence of the dynamic response on hydrodynamic excitation forces like coolant recirculation pump frequencies and the flow path configuration
  - Acceptance criteria for allowable responses and for the locations of vibration sensors

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Dynamic Testing and Analysis of SSCs
Dynamic Response Analysis of Reactor Internals

- Flow induced forcing functions
  - Forcing functions for reactor internals
  - Forcing functions that may be amplified by lock-in with an acoustic and/or structural resonance
  - Methods for specifying forcing functions: analytical method, test-analysis combination method
  - The results of the vibration analysis should be verified by vibration test results

Dynamic Testing and Analysis of SSCs
Vibration Testing of Reactor Internals

- **Vibration testing of reactor internals during startup**
  - Purpose
    - To verify the structural integrity of reactor internals, determine the margin of safety associated with steady-state and anticipated transient conditions, and confirm the results of the vibration analysis

Dynamic Testing and Analysis of SSCs
Vibration Testing of Reactor Internals

- Testing requirements
  - The vibration testing is conducted with the fuel elements or dummy elements in the core
  - Vibration monitoring instrumentation should be provided for critical locations, including those with the most severe vibratory motions and the most effect on safety functions
  - Testing to evaluate potential adverse flow effects should include the steam dryer and MSL valves
  - The duration of the test for the normal operation modes ensures that all critical components are subject to at least 106 cycles of vibration

Dynamic Testing and Analysis of SSCs
Vibration Testing of Reactor Internals

- Testing requirements (continued)
  - Walkdown inspections should be conducted during the testing and visual and NDE surface inspections be conducted after completion of the test
  - A summary evaluation of plant startup and power ascension should be provided to NRC within 90days of the plant startup. If full licensed power is not achieved in that time period, a supplemental report should be provided within 30days after achieving full licensed power

Dynamic Testing and Analysis of SSCs
Vibration Testing of Reactor Internals

- Testing requirements (continued)
  - Testing should include all of the flow modes and anticipated operational transients
  - The methods and procedures for data acquisition, reduction and processing should provide meaningful interpretation of the vibration behavior of various components, e.g., amplitude, frequency content, stress state, and possible effects on safety function, and detailed analysis of bias errors and uncertainties from various sources
  - Vibration predictions, test acceptance criteria and basis, and permissible deviations from the criteria should be provided before the test

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ASME Code Class 1, 2 and 3 Components, Supports and Core Support Structures (SRP 3.9.3)

- **Loading combinations, system operating transients, and stress limits**
- **Design and installation of pressure relief devices**
- **Component supports**

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ASME Code Class 1, 2 and 3 Components, Supports and Core Support Structures (SRP 3.9.3)

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- **Loading combinations, system operating transients, and stress limits**
  - Design loadings and design limits
    - Design loadings shall be established in the Design Specification. The design limits of the appropriate subsection of the Code shall not be exceeded for the design loadings specified
    - Fatigue evaluations

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ASME Code Class 1, 2 and 3 Components, Supports and Core Support Structures (SRP 3.9.3)

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- Service limit A
  - All classes of components, component supports, and core support structures shall meet a service limit not greater than Level A when subjected to sustained loads resulting from normal plant/system operation

- Service limit B
  - All classes of components, component supports, and core support structures shall meet a service limit not greater than Level B when subjected to the appropriate combination of loadings resulting from
    1. sustained loads,
    2. specified plant/system operating transients (SOT), and
    3. the OBE

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ASME Code Class 1, 2 and 3 Components, Supports and Core Support Structures (SRP 3.9.3)

- Service limit C
  - All classes of components, component supports, and core support structures shall meet a service limit not greater than Level C when subjected the appropriate combination of loadings resulting from 1) sustained loads, and 2) design basis pipe break (DBPB)
  - The DBPB includes loads from the postulated pipe break, itself, and also any associated system transients or dynamic effects resulting from the postulated pipe break

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ASME Code Class 1, 2 and 3 Components, Supports and Core Support Structures (SRP 3.9.3)

- Service limit D
  - All classes of components, component supports, and core support structures shall meet a service limit not greater than Level D when subjected the appropriate combination of loadings resulting from 1) sustained loads, and 2) either the DBPB, MS/FWPB, or LOCA, and 3) SSE
  - The DBPB, MS/FWPB, and LOCA, includes loads from the postulated pipe breaks, themselves, and also any associated system transients or dynamic effects resulting from the postulated pipe breaks

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ASME Code Class 1, 2 and 3 Components, Supports and Core Support Structures (SRP 3.9.3)

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Service Stress Limits for Specified Service Loading Combinations

Plant Event	System Operating Conditions	Service Loading Combinations	Service Stress Limits
1. Normal operation	Normal	Sustained loads	A
2. SOT + OBE	Upset	Sustained loads+SOT +OBE	B
3. DBPB	Emergency	Sustained loads + DBPB	C
4. MS/FWBP	Faulted	Sustained loads + MS/FWBP	D
5. DBPB or MS/FWBP + SSE	Faulted	Sustained loads + DBPB or MS/FWBP +SSE	D
6. LOCA	Faulted	Sustained loads + LOCA	D
7. LOCA + SSE	Faulted	Sustained loads +LOCA+SSE	D

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ASME Code Class 1, 2 and 3 Components, Supports and Core Support Structures (SRP 3.9.3)

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- **Design and installation of pressure relief devices**
  - Appendix O, ASME Code, Section III Division 1, “Rules for the design of safety valve installations”
  - Additional requirements for the case where more than one valve is installed on the same pipe run

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ASME Code Class 1, 2 and 3 Components, Supports and Core Support Structures (SRP 3.9.3)

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- **Component supports**
  - The component support designs should provide adequate safety margins under all combinations of loadings
  - Deformation of limits for components supports of active pumps and valves
  - Functionality assurance for snubbers

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Inservice Testing Programs (SRP 3.9.6)

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- **10 CFR 50.55a requirements for IST programs**
- **An overview of organization of ASME OM Code**
- **Inservice testing program for pumps**
- **Inservice testing program for Valves**
- **Inservice testing program for dynamic restraint (snubbers)**

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Inservice Testing Programs (SRP 3.9.6)

- **Organization of ASME OM Code**
  - Four Subsections
    - Subsection ISTA, General Requirements
    - Subsection ISTB, Inservice testing of Pumps
    - Subsection ISTC, Inservice testing of Valves
    - Subsection ISTD, Inservice testing of Dynamic Restraints (Snubbers)
  - Two Mandatory Appendices
    - Appendix I, Inservice testing of Pressure Relief Devices
    - Appendix II, Check Valve Condition Monitoring Program

Inservice Testing Programs (SRP 3.9.6)

- **Inservice testing program for pumps**
  - Scope of the test program:
    - All Class 1, 2 and 3 pumps
    - Safety related pumps that are not classified as ASME Code Class 1, 2, or 3 pumps
  - ISTB-3000
    - Establishing reference values
    - Frequency of ISTs
    - Test parameters

Inservice Testing Programs (SRP 3.9.6)

- ISTB-5000
  - measurement methods
  - testing duration requirements
- ISTB-3500
  - Accuracy of instruments

Inservice Testing Programs (SRP 3.9.6)

- **Inservice testing program for Valves**
  - Scope of the test program:
    - All safety related Class 1, 2 and 3 valves
    - Safety related valves that are not classified as ASME Code Class 1, 2, or 3 valves
  - IST program shall meet the requirements of Subsection ISTC of the ASME OM Code
  - Specific valve or actuator types should meet the requirements provided in related NRC regulatory guidance documents

Inservice Testing Programs (SRP 3.9.6)

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- **Inservice testing program for dynamic restraint (snubbers)**
  - Meeting requirements of 10 CFR 50.55a Paragraph (b)(3)(v), by complying with
    - IST programs for dynamic restraints must meet the provisions of Section XI, IWF-5200(a)&(b), and IWF-5300(a)&(b); or
    - Subsection *ISTD* of ASME OM Code

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Seismic and Dynamic Qualification of Mechanical and Electrical Equipment (SRP 3.10)

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- **Purpose**
  - Seismic and dynamic testing and analysis of mechanical and electrical equipment, and associated supports to ensure such equipment will withstand seismic and dynamic loads, as a result of, or in combination with, other environmental conditions

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Seismic and Dynamic Qualification of Mechanical and Electrical Equipment (SRP 3.10)

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- **Scope**
  - Emergency reactor shutdown
  - Containment isolation
  - Reactor core cooling and residual heat removal
  - Containment heat removal
  - Containment and mitigation of radioactive material release
  - Actuation systems
  - Essential supporting functions
  - Equipment for accident condition monitoring

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Seismic and Dynamic Qualification of Mechanical and Electrical Equipment (SRP 3.10)

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- **Acceptable testing and analysis procedures**
  - IEEE Std 344-1987 as endorsed by RG 1.100 for seismic qualification of mechanical and electrical equipment
  - SRP Section 3.10 provides supplemental guidance with respect to combining seismic loads with other loads and conditions, modeling of supports, documentation, etc

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Seismic and Dynamic Qualification of Mechanical and Electrical Equipment (SRP 3.10)

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- Qualification for equipment functionality
  - Combination of seismic loads with other loads
  - Equipment should be tested in the operational condition
  - Seismic and dynamic input motions
  - Consideration of dynamic coupling
  - Consideration of the loads imposed by the attached piping for pumps and valves
  - Static testing of a pump or valve in lieu of the dynamic testing
  - In situ testing
  - The test program may be based on a selective number of components

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Seismic and Dynamic Qualification of Mechanical and Electrical Equipment (SRP 3.10)

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- Qualification for equipment functionality (cont'd)
  - Damping values
  - Functionality assurance program
- Experience-based qualification approach

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Environmental Qualification of Mechanical and Electrical Equipment (SRP 3.11)

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- **General requirement for environmental design and qualification**
  - Equipment important to safety shall be designed to have the capability of performing its design safety functions under all anticipated operational occurrences and normal, accident, and post accident environment, and for the length of time for which its function is required

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Environmental Qualification of Mechanical and Electrical Equipment (SRP 3.11)

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- **General requirement for environmental design and qualification (cont'd)**
  - Environmental qualification of equipment located in harsh environment shall be demonstrated by appropriate testing and analyses
  - A QA program meeting 10 CFR 50, Appendix B, shall be established and implemented to provide the assurance that all requirements have been satisfactorily accomplished

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Environmental Qualification of Mechanical and Electrical Equipment (SRP 3.11)

- **Scope**
  - Equipment associated with systems that are essential for emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise are essential in preventing significant release of radioactive material to the environment
  - Equipment that initiates or is used to manually the above functions automatically

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Environmental Qualification of Mechanical and Electrical Equipment (SRP 3.11)

- **Scope (cont'd)**
  - Equipment, though not safety related, whose failure can prevent the satisfactory accomplishment of one or more of the above safety functions
  - Certain post-accident monitoring equipment
- **Acceptance criteria**
  - Effect of chemical exposure must be addressed

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Environmental Qualification of Mechanical and Electrical Equipment (SRP 3.11)

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- **Acceptance criteria (cont'd)**
  - Mechanical components must be designed to be compatible with postulated environmental conditions, including those associated with loss-of-coolant accidents (LOCAs).
  - For electrical and mechanical equipment located in a mild environment, acceptable environmental design can be demonstrated by the "design / purchase" specifications for the equipment.

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Environmental Qualification of Mechanical and Electrical Equipment (SRP 3.11)

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- **Acceptance criteria (cont'd)**
  - Electrical equipment located in harsh environment
    - I. NUREG-0588 & RG 1.86
    - II. IEEE Std 323 "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations" as endorsed by RG 1.89
    - III. IEEE Std 334 as endorsed by RG 1.40 for design and environmental qualification of continuous-duty Class 1E Motors
    - IV. IEEE Std 317 as endorsed by RG 1.63 for electrical penetrations

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Environmental Qualification of Mechanical and Electrical Equipment (SRP 3.11)

v. IEEE Std 382, as endorsed by RG 1.73 for Class 1E electrical valve operators
vi. Regulatory Guide 1.97 provides guidance for the environmental qualification of the post-accident monitoring equipment
vii. IEEE Std 383 for Class 1E cable and field splices
viii. IEEE Std 572as endorsed by RG 1.156 for Class 1E connection assemblies
ix. IEEE Std 535 as endorsed by RG 1.158 for lead batteries
x. Regulatory Guide 1.180 provides guidance for determining electromagnetic compatibility for I&C equipment during service

ASME Code Class 1, 2 &3 Piping Systems (SRP 3.12)

- **Scope**
  - Seismic Category I, II and non-safety related piping systems
- **Acceptance criteria for**
  - Piping analysis methods
  - Piping modeling techniques
  - Piping stress analyses
  - Piping support design

Thread Fasteners (SRP 3.13)

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- **Scope**
  - ASME Code Class 1, 2 and 3 Thread Fasteners
- **Compliance with ASME Codes**
  - Compliance with the construction rules for Class 1, 2 and 3 components of ASME Code Section III, Division
  - Compliance with the pre- and in-service inspection requirements of ASME Code Section XI

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Thread Fasteners Table 13.1 ASME Code Section III Criteria for Selection and Testing of Bolting Materials

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		ASME Criteria
Materials selection (NX is NB, NC, and ND for Class 1, 2 and 3 fasteners respectively)		NCA-1220 NX-2128
Materials test coupons and Specimens for ferritic steels (tensile test)	Heat treatment criteria	NX-2210
	Test coupons requirements bolting/stub materials	NX-2221 NX-2224.3
Fracture toughness requirements	Material to be Impact Tested	NX-2311
	Types of Impact Test	NX-2321
	Test Coupons	NX-2322
	Acceptance Standards	NX-2333
	Number of Impact Tests Necessary	NX-2345
	Retesting	NX-2350
	Calibration of Test Equipment	NX-2360
Examination Criteria for Bolts, Studs, and Nuts		NX-2580

Thread Fasteners
Table 13.2 ASME Section XI Examination Categories

Examination Type	ASME Class 1 Criteria	ASME Class 2 Criteria	ASME Class 3 Criteria
Specific Bolting Inspections	Table IWB-2500-1, Exam. Cat. B-G-1 for bolting greater than 2 inches in diameter Table IWB-2500-1, Exam. Cat. B-G-1 for bolting less than or equal to 2 inches in diameter	Table IWC-2500-1, Exam. Cat. C-D for bolting greater than 2 inches in diameter	Not Applicable - Currently there are no examination categories that correspond to those that exist for ASME Class 1 and 2 bolting.
System Pressure Tests	Table IWB-2500-1, Exam. Cat. B-P	Table IWC-2500-1, Exam. Cat. C-H	Table IWD-2500-1, Exam. Cat. D-B

Module 05 & 06

Standard Review Plan (NUREG-0800) Chapters 4 and 5

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Module 05 & 06

Standard Review Plan (NUREG-0800) Chapters 4 and 5

Instructor: Gene Imbro, P.E.

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Learning Objectives

- **Materials specifications for control rod drive mechanism and reactor internal structures**
- **Processes and controls for fabrication and handling of control rod drive mechanism and reactor internal structures**
- **The construction rules for the aforementioned components and structures**

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Learning Objectives (cont'd)

- **The ASME Code and 10 CFR 50.55a**
- **Provisions for ensuring the integrity of the RCPB including the reactor vessel**

SRP Chapter 4

- **Title: Reactor**
  - Selected Subsections:
    - SRP 4.5.1 Control rod drive structural materials
    - SRP 4.5.2 Reactor internal and core support structure materials

SRP Chapters 5

- **Title: Reactor Coolant System and Connected Systems**
  - Compliance with ASME Code and 10CFR50.55a (SRP 5.2.1.1);
  - Applicable Code Cases (SRP 5.2.1.2)
  - Overpressure protection (SRP 5.2.2)
  - Reactor Coolant System Materials (SRP 5.2.3)
  - Reactor Vessel Materials (SRP 5.3.1)
  - Pressure -Temperature Limits, Upper-Shelf Energy and Pressurized Thermal Shock (SRP 5.3.2)
  - Reactor Vessel Integrity (SRP 5.3.3)

Major Sub-topics

- Material specifications
- Processes and controls for fabrication of control rod drive mechanisms and reactor internal structures
- Article NG of the ASME Code Section III, Division 1
- Cleaning and cleanness control;
- Regulatory Guide 1.44, Control of Sensitized Stainless Steel
- 10 CFR 50.55a

Major Sub-topics (cont'd)

U.S.NRC
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Protecting People and the Environment

- Overpressure protection for the reactor coolant system
- Material specifications for the RCPB components
- Special controls and processes for fabrication of ferritic and austenitic stainless steel components
- PTS rule, 10 CFR 50.61
- Fracture toughness requirements, Appendix G to 10 CFR 50
- Reactor vessel materials surveillance requirements, Appendix H to 10 CFR 50

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References

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- **ASME BPV Code, Section III, Division 1**
- **Article NG of ASME Code Section III, Division 1**
- **ASME BPV Code Section III, NB-7000, NB-2300**
- **SRP Sections 5.2**
- **10 CFR Part 50.55a, 50.61, Appendix G, Appendix H**
- **Regulatory guides 1.31, 1.34, 1.37, 1.43, 1.44, 1.50, 1.71**
- **NUREG-0313 Revision 2**

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Control Rod Drive Structural Materials (SRP 4.5.1)

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- **Scope**
- **Materials specifications**
- **Stainless steels**
- **Austenitic stainless steel components**
- **Other materials**
- **Cleaning and cleanliness control**

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Control Rod Drive Structural Materials (SRP 4.5.1)

U.S.NRC
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- **Scope**
  - Including the control rod drive mechanism (CRDM) and extending only to the coupling interface with the reactivity control (poison) elements
  - Not including the electrical and hydraulic systems necessary to actuate the CRDMs

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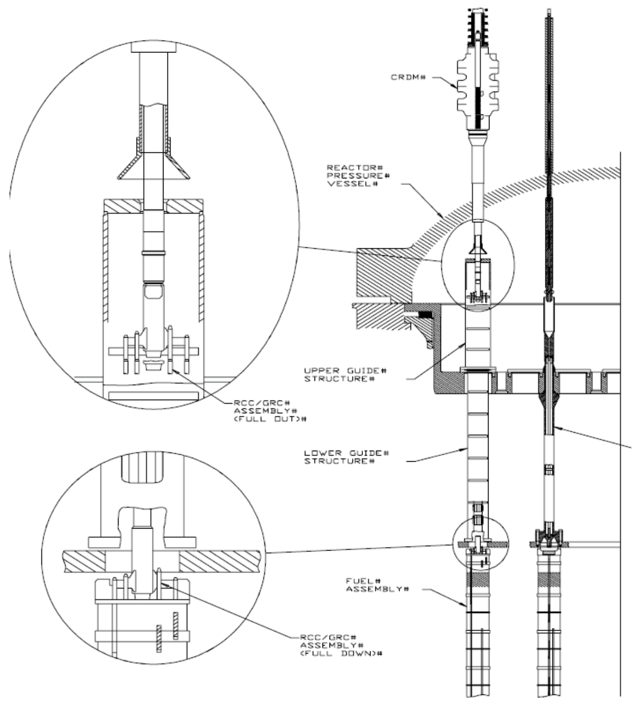

Rod Cluster Control and Drive Rod Assembly with Interfacing Components

(Source: AP1000 DCD)

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Control Rod Drive Structural Materials (SRP 4.5.1)

- **Materials specifications**
  - ASME Code Section III, Appendix I " Design Fatigue Curves"
  - ASME Code Section II "Materials, Part A, B, and C"
  - RG 1.85, "Materials Code Case Acceptability ASME Section III Division 1"

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Control Rod Drive Structural Materials (SRP 4.5.1)

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- **Stainless Steels**
  - All true stainless steels contain a minimum of about 12% Cr, permitting a thin protective surface layer of chromium oxide to form when the steel is exposed to oxygen
  - Ferritic stainless: less than 0.12% C and up to 30% Cr; solution strengthening and strain hardening
  - Martensitic stainless steels: 0.1%-1.0% C and typically <17% Cr
  - Austenitic stainless steels

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Control Rod Drive Structural Materials (SRP 4.5.1)

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- **Precipitation hardening (PH) stainless steels**

	Steel	%C	% Cr	% Ni	Others
Austenitic	201	0.15	16-18	3.5-5.5	5.5-7.5%Mn
	304	0.08	18-20	8.0-10.5	
	304L	0.03	18-20	8-12	
Ferritic	430	0.12	16-18		
	442	0.12	18-23		
Martensitic	416	0.15	12-14		0.60%Mn
	431	0.20	15-17	1.25-2.50	
PH	17-4	0.07	16-18	3-5	0.15-0.45%Nb
	17-7	0.09	16-18	6.5-7.8	0.75-1.25%AL

Control Rod Drive Structural Materials (SRP 4.5.1)

- **Austenitic Stainless Steel Components**
  - RG 1.44, "Control of the Use of Sensitized Stainless Steel", describes accepted methods for preventing intergranular corrosion of stainless steel components
  - Controls for abrasive work on austenitic stainless steel surfaces to preventing contamination that promotes stress corrosion cracking

Control Rod Drive Structural Materials (SRP 4.5.1)

- **Other materials**
  - Compatibility with the reactor coolant
  - Heat treatment of precipitation hardening stainless steels and martensitic stainless steels

- **Cleaning and cleanliness control**

Reactor Internal and Core Support Structure Materials (SRP 4.5.2)

U.S.NRC
United States Nuclear Regulatory Commission
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- **Material specifications**
- **Control of welding**
- **Nondestructive examination**
- **Austenitic stainless steels**
- **Other materials**

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Reactor Internal and Core Support Structure Materials (SRP 4.5.2)

U.S.NRC
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- **Material specifications**
  - ASME Code Section III, Appendix I " Design Fatigue Curves"
  - ASME Code Section II "Materials, Part A, B, and C"
  - Regulatory guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME III"

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Reactor Internal and Core Support Structure Materials (SRP 4.5.2)

- **Control of welding**
  - NG-4000 of ASME Code, Section III, Division 1
  - NG-5000 of ASME Code, Section III, Division 1
  - Nondestructive examination
    - NG-2500
    - NG-5300

Reactor Internal and Core Support Structure Materials (SRP 4.5.2)

- Austenitic stainless steel
- Other materials

Compliance with 10 CFR 50.55a (SRP 5.2.1.1 & 5.2.1.2)

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Protecting People and the Environment

- **Overview of 10 CFR 50.55a**
- **Approved ASME Code Cases**
- **Component Code for RCPB components**

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Compliance with 10 CFR 50.55a Overview of 10 CFR 50.55a

U.S.NRC
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- **Overview of 10 CFR 50.55a**
  - The Codes and Standard Rule, 10 CFR 50.55a, establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components of BWR and PWR power plants by requiring conformance with appropriate editions of specified published industry codes and standards

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Compliance with 10 CFR 50.55a Overview of 10 CFR 50.55a

- 10 CFR 50.55a specifies editions, addenda, and code cases of the industrial codes and standards required by NRC for construction, inservice inspection (ISI), inservice testing (IST) and safety systems:
  - 50.55a (a) requires SSCs be designed, constructed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed

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Compliance with 10 CFR 50.55a Overview of 10 CFR 50.55a

  - 50.55a (b) incorporates by reference ASME BPV Code Sections III and XI and ASME OM Code; specifies approved code editions, addenda and code cases as well as limitations and modifications.
  - 50.55a (c) – (d) require that RCPB, Quality Group B and Quality Group C components must meet the requirements for Class 1, Class 2 and Class 3 components of ASME RPV Code Section III respectively

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Compliance with 10 CFR 50.55a
Overview of 10 CFR 50.55a

- 50.55a (f) establishes requirements for inservice testing of ASME Code Class 1, 2 and 3 pumps and valves, including timing and frequency of inservice testing.
- 50.55a (g) establishes requirements for preservice and inservice inspection of ASME Code Class 1, 2, 3 components and their support, including timing and frequency of inservice inspection.
- 50.55a (h) establishes requirements for protection and safety systems by incorporation of IEEE Std. 603-1991 and IEEE Std. 279

Compliance with 10 CFR 50.55a
Code Cases

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- Approved Code Cases
  - Code Cases are documents issued by the ASME to clarify the intent of existing Code requirements or to provide alternative rules
  - Code Cases, if adopted by NRC, become approved alternatives to the Codes: Regulatory Guides 1.84, 1.147 & 1.192

Compliance with 10 CFR 50.55a Component Codes

- Component Codes for RCPB Components
  - RCPB include all those pressure retaining components that are:
    - Part of the reactor coolant system, or
    - Connected to the reactor coolant system, up to and including any and all of the following
      - i. The outermost containment isolation valve in system piping which penetrates primary reactor containment

Compliance with 10 CFR 50.55a Component Codes

- ii. The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment
- iii. The reactor coolant system safety and relief valves

– For nuclear power reactors of the direct cycle boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valve in the main steam and feedwater piping

Compliance with 10 CFR 50.55a Component Codes

- 10 CFR 50.55a (c) – Reactor Coolant Pressure Boundary
  - RCPB pressure-retaining components must meet requirements for ASME Code Class 1 components and be constructed in accordance with the rules of the ASME BPV Code, Section III, Division 1
  - Other RCPB components, failure of which will not prevent an orderly shutdown and cool down of the reactor, assuming makeup is provided by the reactor coolant makeup system only, may be constructed in accordance with the rules for Class 2 components

Compliance with 10 CFR 50.55a Component Codes

- Overview of organization of ASME BPV Code Section III, Division 1
- 8 Subsections
  - NCA: General Requirements
  - NB: Class 1 Components
  - NC: Class 2 Components
  - ND: Class 3 Components
  - NE: Class MC Components
  - NF: Supports
  - NG: Core Support Structures
  - NH: Class 1 Components in Elevated Temp. Service
- Appendices
- Code Cases

Compliance with 10 CFR 50.55a Component Codes

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Organization of Subsection NB	
Topic	Code Section
Introduction	NB-1000
Materials	NB-2000
Design	NB-3000
Fabrication and installation	NB-4000
Examination	NB-5000
Testing	NB-6000
Overpressure protection	NB-7000
Certification/Name Plate	NB-8000

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RCPB Overpressure Protection (SRP 5.2.2)

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- **Introduction**
  - Material specification for pressure relief devices
  - Overpressure protection system design requirements for BWRs
  - Overpressure protection system design requirements for PWRs
  - Case studies

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RCPB Overpressure Protection Materials Specification

- Material specification for pressure relief devices
- ASME Code Section III, Appendix I " Design Fatigue Curves"
- ASME Code Section II "Materials, Part A, B, and C"
- Regulatory Guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME III"

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RCPB Overpressure Protection
Design Requirements for BWRs

- Overpressure protection system design requirements for BWRs operating at power
  - The BWR reactor and associated systems shall be designed such that safety valves are not actuated during normal operational transients permitted by TS.
  - Safety valves should have sufficient capacity to limit the pressure to less than 110% of the RCPB design pressure as specified by ASME Code Article NB-7000, during the most severe AOO with reactor scram.
  - Meeting the single failure criterion

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RCPB Overpressure Protection
Design Requirements for PWRs

- Overpressure protection system design requirements for PWRs operating at power
  - The PORVs or the pressurizer should have sufficient capacity to preclude actuation of safety valves during normal operational transients permitted by TS.
  - Safety valves should have sufficient capacity to limit the pressure to less than 110% of the RCPB design pressure as specified by ASME Code Article NB-7000, during the most severe AOO with reactor scram.
  - Meeting the single failure criterion

RCPB Overpressure Protection
Design Requirements for PWRs

- Overpressure protection system design requirements for PWRs operating at low temperature
  - Protection against brittle facture for operations at low temperature or water-solid conditions
  - Low temperature overpressure protection (LTOP) should meet requirements in Branch Technical Position (BTP) 5-2
  - LTOP design should meet the single failure criterion

RCPB Overpressure Protection

Design Requirements for PWRs

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Overpressure protection system design requirements for PWRs operating at low temperature
  - The LTOP system should have sufficient pressure relieving capacity to satisfy P-T limits of the TS while operating at low temperature or water solid conditions
  - The LTOP should be operable during startup and shutdown conditions below the enable temperature
    - The enable temperature means a metal temperature of ≥ RTNDT + 50°C (90°F) at the beltline location

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RCPB Overpressure Protection

Case Study

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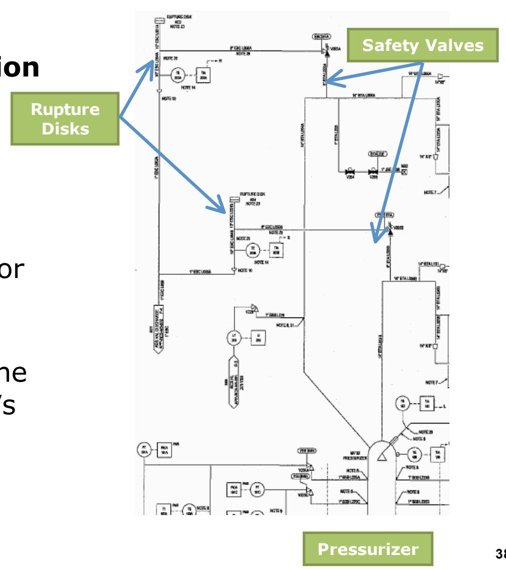
- **Overpressure protection in AP1000 for power operation**
  - Two Pressurizer SVs, each sized to maintain pressure <110% in conjunction with reactor scram during AOOs
  - Coolant is discharged into containment via the rupture disks when SVs actuated



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RCPB Overpressure Protection

Case Study – LTOP in AP1000

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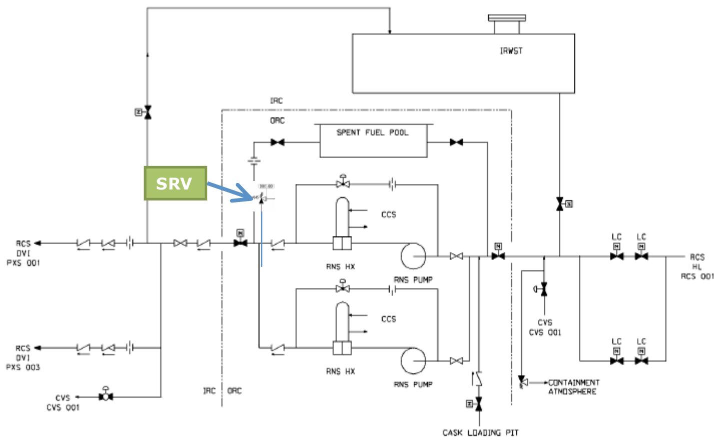

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RCPB Overpressure Protection Case Study

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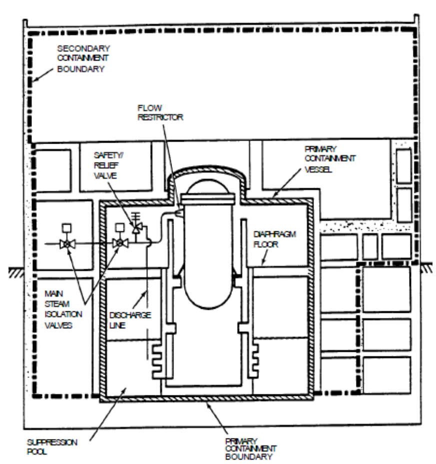
- **Overpressure protection in the ABWR design**
  - Four groups of SRVs
  - When functioning as SVs, they are self-actuated by inlet steam pressure
  - Steam being discharged to the suppression pool



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RCPB Materials (SRP 5.2.3)

- **Introduction**
- **Materials specification**
- **Compatibility of materials with reactor coolant**
- **Fabrication and processing of ferritic materials**
- **Fabrication and processing of austenitic stainless steel**

RCPB Materials
Materials Specifications

- **Material specifications**
  - ASME Code Section III, Appendix I " Design Fatigue Curves"
  - ASME Code Section II "Materials, Part A, B, and C"
  - Regulatory guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME III"

RCPB Materials

Compatibility of Materials with the Reactor Coolant

- **Compatibility of materials with the reactor coolant**
  - Compatibility with the coolant, contaminants, radiolytic products to which the materials may be exposed
  - Compatibility with external insulation and with the environment in the event of reactor coolant leakage
  - Fabrication and cleaning controls for stainless steel components to minimize contamination with chloride and fluoride ions

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RCPB Materials

Fabrication and Processing of Ferritic Materials

- **Fabrication and processing of ferritic materials**
  - Ferritic materials: carbon and low carbons steels, higher allow steels, etc.
  - Fracture Toughness
    - Appendix G to 10 CFR 50 requires that the pressure retaining components of the RCPB made of ferritic materials meet the requirements of the ASME Code for fracture toughness during hydrostatic tests and any condition of normal operation including AOOs

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RCPB Materials

Fabrication and Processing of Ferritic Materials

- Fracture Toughness
  - ASME Code Section III, NB-2300: Fracture toughness requirements for materials
  - NB-2330 provides the acceptance standards for fracture toughness for vessel material, materials for piping, pumps and valves, and bolting materials
  - Materials for vessels (NB-2331): establishing the material reference temperature (RTNDT) that at RTNDT + 33°C (60°F) Charpy V-notch tests shows at least 35 mils (0.89mm) lateral expansion and not less than 50 ft-lb(68J) absorbed energy.

RCPB Materials

Fabrication and Processing of Ferritic Materials

- Fracture Toughness
  - Piping, pumps and valves: the lowest temperature shall not be lower than RTNDT+100°F (56°C) for wall thickness greater than 2½ inches (64mm); for others:

Table NB-2332(a)-1

Required Cv Values for Piping, Pumps and Valves

Nominal Wall Thickness, in (mm)	Lateral Expansion, mils (mm)
⅝ (16) or less	No test required
Over ⅝ to 3/4 (16 to 19), incl.	20 (0.50)
Over 3/4 to 1½ (19 to 38), incl.	25 (0.64)
Over 1½ to 2½ (38 to 64), incl.	40 (1.0)

- Subsection NB-2333 and Table NB-2333-1for bolting materials

RCPB Materials

Fabrication and Processing of Ferritic Materials

- **Control of welding in ferritic materials**
  - Control of preheat temperature to minimize cold cracks or reheat cracking forming in underbead areas and heat affected zones:
    1. The amount of specified preheat must be in accordance with Appendix D, Paragraph D-1210 of ASME Code Section III
    2. The welding procedure should be qualified and specify the minimum preheat and maximum interpass temperatures; the temperature limits be monitored in production welding
    3. For production welds the preheat temperature should be maintained until a post-weld heat treatment has been performed

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RCPB Materials

Fabrication and Processing of Ferritic Materials

- **Control of welding in ferritic materials**
  - Control of electroslag weld properties
    - Maintaining a weld solidification pattern characterized by a strong intergranular bond in the center of the weld to assure the quality of electroslag welds
    - Welding variables affecting the weld solidification pattern, i.e., slag pool depth, electrode feed speed rate and oscillation, current, voltage, slag conductivity, should be properly controlled to produced acceptable solidification pattern.

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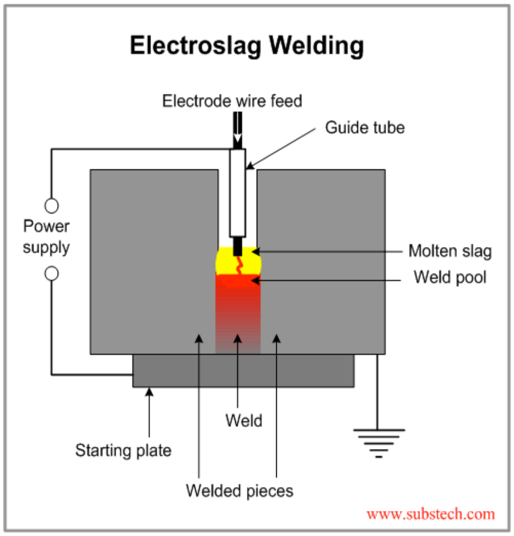
Electroslag Welding

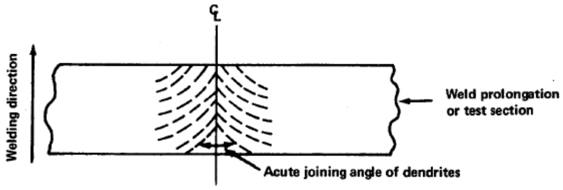


RCPB Materials

Fabrication and Processing of Ferritic Materials

- Control of welding in ferritic materials
  - Control of electroslag weld properties

    iii. Acceptable welds should show a dendritic freezing pattern with a joining angle of less than 90°.



Longitudinal section showing solidification pattern (Source: RG 1.34)

iv. RG 1.34, "Control of electroslag weld properties"

RCPB Materials
Fabrication and Processing of Ferritic Materials

- Control of welding in ferritic materials
  - Control of stainless steel weld cladding of low-alloy steel components to prevent underclad cracking
    - Underclad cracking
    - Welding processes that induce underclad cracking by generating excessive heating and promoting grain coarsing in the base metal should be not used for cladding any grade of materials that has a known susceptibility to underclad cracking.
    - Welding procedures for cladding these grades of materials should be qualified to demonstrate that underclad cracking is not induced.

RCPB Materials
Fabrication and Processing of Ferritic Materials

- Control of welding in ferritic materials
  - Welder qualification for areas of limited accessibility (Reg Guide 1.71)
  - Nondestructive examination shall meet requirements in Paragraphs NB-2550, NB-2560 and NB-2570.

RCPB Materials

Fabrication and Processing of Austenitic Stainless Steel

- **Fabrication and Processing of Austenitic Stainless Steel**
  - Control to avoid severe sensitization
    - The steel is said to be sensitized when the steel cools down slowly from 870°C to 425 °C, chromium carbides precipitate at the grain boundaries which may reduce chromium in the austenite below 12%.
    - As a result, sensitized steel is much more susceptible to stress corrosion cracking than non-sensitized and stabilized sensitized austenitic stainless steels

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RCPB Materials

Fabrication and Processing of Austenitic Stainless Steel

- Control to avoid severe sensitization
  - Solution heat treatment and testing should normally performed at the appropriate stage of fabrication when sensitized steels are used for RCPB components
  - Welding should be controlled to avoid excessive sensitization of base metal heat-affected zone.
  - Controls should be applied to avoid intergranular stress corrosion in and near welds in BWR austenitic stainless steel piping in accordance with NUREG – 0313 or Attachment to Generic Letter 88-01

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RCPB Materials

Fabrication and Processing of Austenitic Stainless Steel

- **Protecting the material from contaminants capable of causing stress corrosion cracking**
- **Cold-worked austenitic stainless steels used in RCBP should have an upper limit on the yield strength of 620 Mpa (90,000psi)**
- **Compatibility of austenitic stainless steel with insulation**

RCPB Materials

Fabrication and Processing of Austenitic Stainless Steel

- **Control of welding**
  - Control of electroslag weld quality
  - Welder requalification for welding operation in areas of limited accessibility
  - Weld procedures, weld metal composition and delta ferrite percentages should be specified to minimize the possibility of hot cracking during welding austenitic stainless steels.
- **Nondestructive examination: ASME Code Section III, Paragraphs NB-2550, NB-2560 and NB-2570**

Reactor Vessel Materials (SRP 5.3.1)

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- **Introduction**
- **Materials specifications**
- **Fracture toughness (10 CFR 50, Appendix G)**
- **Material surveillance program (10 CFR 50, Appendix H)**

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Reactor Vessel Materials Material Specification

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- **Material Specifications**
  - ASME Code Section III, Appendix I
  - Acceptability of materials not specified in the ASME Code is evaluated with respect to mechanical properties, weldability, and physical changes of the material
  - Meeting the requirements of Appendix G to 10 CFR Part 50

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Reactor Vessel Materials Fracture Toughness

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- **Fracture Toughness**
  - ASME Code Section III, Sub Article NB-2300
    - Ferritic materials shall be tested in accordance with NB-2300
    - The fracture toughness is characterized by the reference temperature (RTNDT). At RTNDT + 33°C (60°F), Charpy V-notch tests exhibit not less than 0.89mm (35mils) lateral expansion and not less than 68 J (50ft-lbs) of absorbed energy

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Reactor Vessel Materials Fracture Toughness

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- ASME Code Section III, Sub Article NB-2300
  - RTNPT is determined in accordance with Paragraph NB-2330, as summarized below
    - i. Determine a temperature TNDT that at or above the nil-ductility temperature by drop weight tests
    - ii. At a temperature not greater than TNDT + 33°C, if each Charpy V-notch (Cv) specimen tested exhibit at least 0.89mm lateral expansion and not less than 68J of absorbed energy, TNDT is defined as RTNDT. If the above requirement is not met.

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Reactor Vessel Materials Fracture Toughness

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- RTNPT is determined in accordance with Paragraph NB-2330, as summarized below:
  iii. Additional Cv impact tests are performed to determine a temperature Tcv at which the requirements are met. In this case, RTNDT is equal to Tcv – 33°C (60°F)
  iv. Alternatively, a temperature representing a minimum of 68J and 0.89mm lateral expansion may be obtained from a full Cv impact curve developed from the minimum data points of all the Cv impact tests performed

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Reactor Vessel Materials Fracture Toughness

U.S.NRC
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- Supplemented requirements in Paragraph IV of 10 CFR 50, Appendix G
  - Reactor vessel beltline materials shall have a minimum upper shelf energy of 102J (75ft-lbs) as determined from Cv impact tests on unirradiated specimens according to NB-2331(a) of the Code Section III; and maintain an upper shelf energy no less than 68J (50ft-lbs) through the life of the vessel
  - Pressure-temperature limits and minimum temperature requirements

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Reactor Vessel Materials
Material Surveillance Program

- Reactor Material Surveillance Program
- Objective
  - Monitoring changes in the fracture toughness properties of ferritic materials in the reactor beltline region due to exposure to neutron irradiation and the thermal environment
  - Material specimens withdrawn periodically from the reactor vessel are used to obtain fracture toughness test data, which are used to determine the conditions under which the vessel can be operated with adequate margins of safety against fracture during its service life

Reactor Vessel Materials
Material Surveillance Program

- 10 CFR 50, Appendix H “Reactor Vessel Material Surveillance Program Requirements”
  - No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods that the peak neutron fluence (E>1Mev) will not exceed 1.0E17 n/cm2 at the end of the design life of the vessel; Otherwise,
  - Reactor vessels shall have their beltline materials monitored by a surveillance program complying with ASTM E-185 as modified by Appendix H to 10 CFR 50

Reactor Vessel Materials
Material Surveillance Program

- Surveillance specimen capsules must be located near the inside vessel wall in the beltline region so as to duplicate the neutron spectrum, temperature history, and maximum neutron fluence experienced by the vessel wall to the extent practical
- For multiple reactors located at a single site, Appendix H provides for an integrated surveillance program maybe authorized by the NRC

Pressure-Temperature Limits, Upper Shelf Energy and PTS (SRP 5.3.2)

- **Pressure-Temperature limits**
  - P-T limits are imposed on the RCPB components to ensure adequate safety margins of structural integrity for the ferritic components of the RCPB
  - P-T limits by Appendix G to 10 CFR 50
  - Appendix G to 10 CFR 50 requires that P-T limits must be at least as conservative as limits obtained by following Appendix G to ASME Section XI and additional safety margins when the core is critical

Pressure-Temperature Limits

Appendix G of Section XI limits:

- **i. Preservice hydrotests**
  $K_{applied} = K_I(pressure) < K_{Ic}$
- **ii. Inservice leak & hydrotests**
  $K_{applied} = 1.5K_I(pressure) < K_{Ic}$
- **iii. Heatup & cooldown operations**
  $K_{applied} = 2K_I(pressure) + K_I(Temperature) < K_{Ic}$
- **iv. Core critical operation: the temperature must be higher than required in ii and 40°F over required iii**

Operating Condition	Vessel Pressure	Requirements for P-T limits
1. Hydrostatic pressure and leak tests (core is not critical)		
1.a Fuel in the vessel	≤20%	ASME Appendix G limits
1.b Fuel in the vessel	>20%	ASME Appendix G limits
1.c No fuel in the vessel	all	Not Applicable
2. Normal operations, including AOOs		
2.a Core not critical	≤20%	ASME Appendix G limits
2.b Core not critical	>20%	ASME Appendix G limits
2.c Core critical	≤20%	ASME Appendix G limits + 40°F
2.d Core critical	>20%	ASME Appendix G limits + 40°F
2.e Core critical for BWR	≤20%	ASME Appendix G limits + 40°F

Upper Shelf Energy

- **Upper shelf energy (USE) requirements**
  - Initially, the USE value in the transverse direction for base material and along the weld must not be less than 102 J (75 ft-lb)
  - Charpy USE throughout the life of the vessel must be maintained at no less than 68 J (50 ft-lb), unless it is demonstrated that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

Pressure Thermal Shock (PTS)

- **Pressure thermal shock**
  - Pressure shock event
  - PTS requirements (10 CFR 50.61)
    - Projected RTPTS not exceed the screening criteria of 132°C (270°F) for plates, forgings, and axial weld materials, and 149 °C (300 °F) for circumferential weld materials
    - For RTPTS values projected to exceed the screening criteria, measures/programs being implemented to prevent potential PTS-related failure of the reactor vessel if continued plant operation beyond the screening criterion is allowed

Reactor Vessel Integrity (SRP 5.3.3)

- **Reactor vessel integrity**
  - The SAR Section 5.3.3 summarizes information relevant to the integrity of the reactor vessel, including,
    - Design in accordance with the ASME Code
    - Construction materials
    - Fabrication methods, i.e., welding
    - Inspection
    - Shipment and Installation
    - Operating condition (i.e., Appendix G, 50.61)
    - Inservice surveillance (i.e., Appendix H of 10 CFR 50)

Module 07
Standard Review Plan (NUREG-0800) Chapter 6

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Module 07
Standard Review Plan (NUREG-0800) Chapter 6

Instructor: Gene Imbro, P.E.

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Learning Objectives

- **Compatibility of ESF fluids**
- **Acceptable coating systems inside containment**
- **Fracture toughness testing for the containment pressure boundary**
- **Design requirements for ESF atmosphere cleanup systems**
- **Design requirements for containment isolation systems**

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SRP Chapter 6

- **Title: Engineered Safety Features**
  - Selected Subsections:
    1. EFS materials (SRP 6.1.1)
    2. Protective coating systems (paints) – organic materials (SRP 6.1.2)
    3. Containment isolation system (SRP 6.2.4)
    4. Fracture prevention of containment pressure boundary (SRP 6.2.7)
    5. ESF Atmosphere cleanup systems (SRP 6.5.1)

Major Sub-topics

- EFS component materials and fabrication; composition of ESF fluids and compatibility with materials; nonmetallic thermal insulation
- Coating systems applied inside a containment
- Containment isolations system, e.g., safety function, general design requirements
- Facture toughness testing requirements for containment pressure boundary ferritic materials
- ESF atmosphere cleanup system, e.g., safety function, design criteria, functional and qualification testing
- Containment isolation systems

References

1. **Appendix A to 10 CFR 50**

2. **Regulatory guides 1.36, 1.44, 1.31,1.37,1.52, 1.54**

Engineered Safety Features Materials (SRP 6.1.1)

- **Engineered safety features**
  - Those safety-related systems designed to mitigate the consequence of design basis accidents, typically including:
    - containment systems
    - emergency core cooling systems
    - residual heat removal system
    - containment heat removal systems
    - containment atmosphere cleanup systems
    - other ESFs described in the SAR

Engineered Safety Features Materials (SRP 6.1.1)

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- **Materials and fabrication**
  - Austenitic stainless steels
    - RG 1.44 provides for acceptable criteria for preventing intergranular corrosion of stainless steel components
    - RG 1.31 provides acceptable criteria for assuring the integrity of welds in austenitic stainless steel components
    - RG 1.37 provides guidance on the controls for abrasive work one austenitic stainless steel surfaces to prevent contamination

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Engineered Safety Features Materials (SRP 6.1.1)

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- Austenitic stainless steels (cont'd)
  - NUREG-0313 and GL 88-01 describe criteria to assure adequate resistance to intergranular stress corrosion cracking for susceptible boiling water austenitic stainless steel piping
- Ferritic Steel Welding
  - Amount of specified preheat
  - Moisture control
  - Areas of limited access

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Engineered Safety Features Materials (SRP 6.1.1)

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- **Thermal insulation**
  - Safety concern
  - Regulatory guide 1.36 , “Nonmetallic thermal insulation for austenitic stainless steel”

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Engineered Safety Features Materials (SRP 6.1.1)

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- **Composition and compatibility of ESF fluids**
  - For PWRs, the composition of containment spray and core cooling water should be controlled to ensure a minimum pH of 7.0
  - For BWRs, water used for emergency core cooling systems and spray systems should be controlled to ensure: Conductivity ≤ 0.5 mS/m at 25°C; Cl- < 0.20 ppm; pH = 5.3 to 8.6 at 25 °C

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Protective Coating Systems (SRP 6.1.2)

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- **Use of protective coatings**
- **Deterioration of protective coating materials**
- **Safety concern with deterioration of coating materials inside the containment**
  - Detached coatings from a substrate that are subsequently transported to the intake structures of emergency core cooling systems (ECCSs) may have a negative impact on the performance of the long-term cooling

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Protective Coating Systems (SRP 6.1.2)

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- **A coating system to be applied inside containment should meet the guidance in RG 1.54 and the standards**
  - ASTM D5144-00
  - ASTM D3911-03

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Containment Isolation System (SRP 6.2.4)

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- **Containment isolation system**
  - The containment isolation system allows the normal or emergency passage of fluids through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products from postulated accidents
  - Isolation barriers include valves, closed piping systems, and blind flanges
  - Safety-related

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Containment Isolation System (SRP 6.2.4)

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- **General requirements for containment isolation system**
  - Comply with requirements in GDC 1, GDC 2 and GDC 4 with regard to quality standards, design for natural phenomena and environmental design
  - GDC 16 as related to the containment function
  - GDC 54 – overall requirements for containment isolation system design
  - GDC 55 , 56 and GDC 57

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Fracture Prevention of Containment Pressure Boundary (SRP 6.2.7)

- **Primary safety function of the containment system**
  - To provide a final physical barrier, by enclosing the reactor system, against the release of radioactive fission products from the reactor core to the environment in the event of a DBA including LOCA events

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Fracture Prevention of Containment Pressure Boundary (SRP 6.2.7)

- **Containment pressure boundary**
  - Consisting of ferritic steel parts of the containment system to sustain loading and provide a pressure boundary in the performance of the containment function under the operating, maintenance, testing and postulated accident conditions
  - Containment pressure boundary components

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Fracture Prevention of Containment Pressure Boundary (SRP 6.2.7)

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- **GDC 51 of Appendix A to 10 CFR 50**
  - GDC 51 requires, as it relates to the containment pressure boundary integrity, its ferritic materials behave in a non-brittle manner and the probability of rapidly propagating fractural minimized
  - Acceptance criteria
  - Article NE-2300 of Section III, Division 1 or Article CC-2520 of Section III, Division 2 of the ASME Code

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EFS Atmospheric Cleanup Systems (SRP 6.5.1)

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- **EFS Atmospheric Cleanup Systems**
  - Those systems that are designed for fission product removal in post accident environment and are credited in the design basis accident analysis in mitigating radiological consequences
  - Primarily for removal of radioactive iodine and particular matter that may be released into the building or containment during or after an accident

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EFS Atmospheric Cleanup Systems (SRP 6.5.1)

- **EFS Atmospheric Cleanup Systems (cont'd)**
  - Including containment atmospheric cleanup system (or the primary system), and secondary systems such as post-accident air-cleaning systems for fuel handling, control room, shield building, and areas containing ESF components.
  - Two ESF atmospheric cleanup systems for illustrative purpose (source: RG 1.52)

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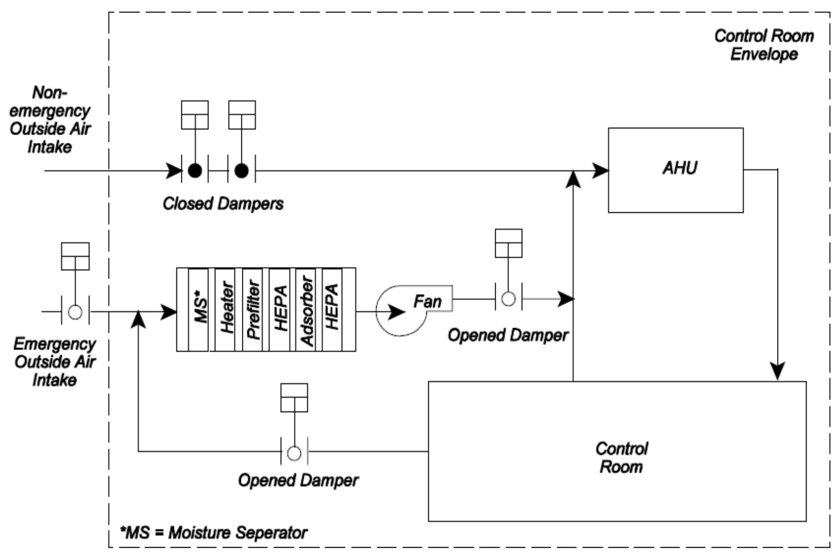
EFS Atmosphere Cleanup Systems (SRP 6.5.1)



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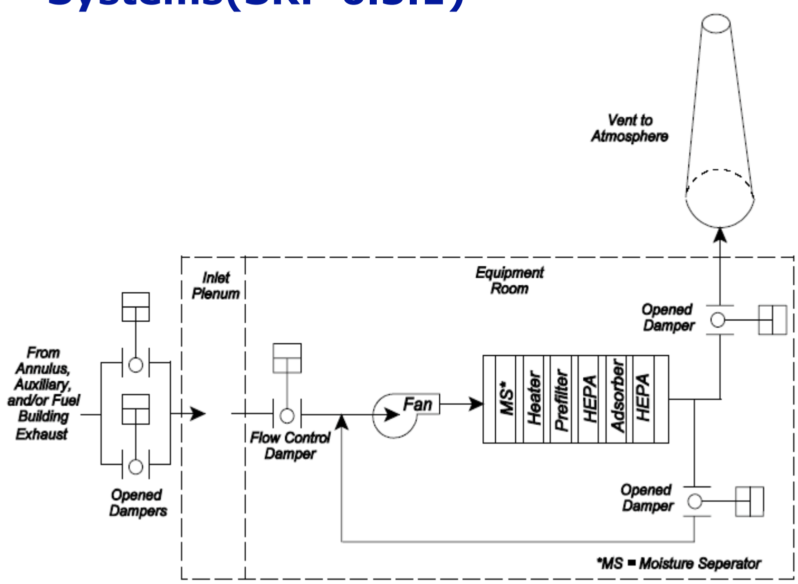
EFS Atmosphere Cleanup Systems(SRP 6.5.1)



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EFS Atmosphere Cleanup Systems (SRP 6.5.1)

- **Design acceptance criteria (or requirements) for EFS Atmosphere Cleanup Systems**
  - RG 1.52 provides detailed acceptance criteria
  - Some general requirements and guidance
    - ESF atmosphere cleanup systems should be designed to operate under the environmental conditions that would be generated during and after design basis accidents
    - Source terms, RG 1.183
    - Should have appropriate redundancy, testability, and provisions facilitating operation and maintenance

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EFS Atmosphere Cleanup Systems (SRP 6.5.1)

- General requirements and guidance (cont'd)
  - Should be designed to withstand dynamic and seismic loads as appropriate
  - Should be equipped with appropriate instrument for signal, measurement, alarm, etc
  - Should be activated automatically when an accident occur
  - Components and equipment should be designed, tested and qualified to meet the applicable code and standards

Module 8
Standard Review Plan Chapter 9

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 8
Standard Review Plan Chapter 9

Instructor: C. Wesley Rowley, P.E.

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Learning Objectives

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- **Learn what is considered an "Auxiliary System" in SRP Chapter 9.**
- **Learn which references, including Codes & Standards, are used as a vehicle to meet SRP requirements.**

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Significant Sub-topics

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- Definition of "Auxiliary" Systems
- Matrix of NRC Regulations and RGs for Various Auxiliary Systems
- Identification of Identified Codes & Standards
- Summary of Each Identified Code / Standard
- Specific Use of Code / Standard to Meet SRP Requirement

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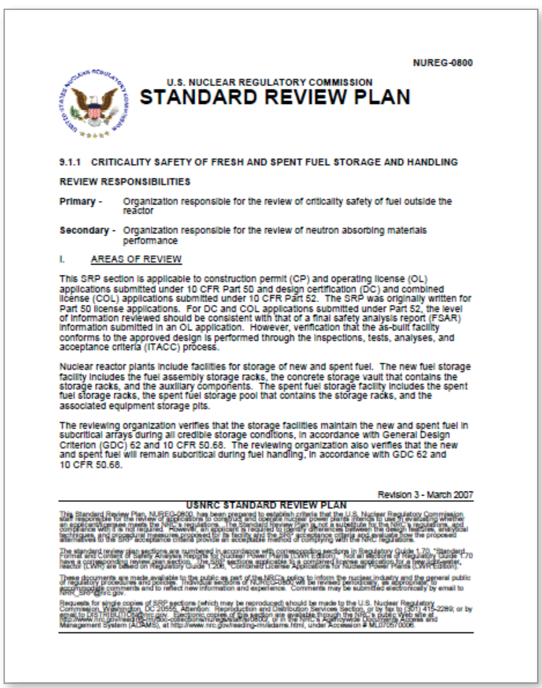

NUREG-0800

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN

9.1.1 CRITICALITY SAFETY OF FRESH AND SPENT FUEL STORAGE AND HANDLING

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of criticality safety of fuel outside the reactor

Secondary - Organization responsible for the review of neutron absorbing materials performance

I. AREAS OF REVIEW

This SRP section is applicable to construction permit (CP) and operating license (OL) applications submitted under 10 CFR Part 50 and design certification (DC) and combined license (COL) applications submitted under 10 CFR Part 52. The SRP was originally written for Part 50 license applications. For DC and COL applications submitted under Part 52, the level of information reviewed should be consistent with that of a final safety analysis report (FSAR) information submitted in an OL application. However, verification that the as-built facility conforms to the approved design is performed through the inspections, tests, analyses, and acceptance criteria (ITACC) process.

Nuclear reactor plants include facilities for storage of new and spent fuel. The new fuel storage facility includes the fuel assembly storage racks, the concrete storage vault that contains the storage racks, and the auxiliary components. The spent fuel storage facility includes the spent fuel storage racks, the spent fuel storage pool that contains the storage racks, and the associated equipment storage pits.

The reviewing organization verifies that the storage facilities maintain the new and spent fuel in subcritical arrays during all credible storage conditions, in accordance with General Design Criterion (GDC) 62 and 10 CFR 50.68. The reviewing organization also verifies that the new and spent fuel will remain subcritical during fuel handling, in accordance with GDC 62 and 10 CFR 50.68.

Revision 3 - March 2007

USNRC STANDARD REVIEW PLAN

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NUREG-0800
Standard Review Plan
Chapter 9 Auxiliary
Systems

CONTENTS

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Nuclear Fuel Criticality
SRP Chapter 9.1

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9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling
9.1.2 New and Spent Fuel Storage
9.1.3 Spent Fuel Pool Storage and Cleanup System
9.1.4 Light Load Handling System (Related to Refueling)
9.1.5 Overhead Heavy Load Handling Systems

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Chapter 9.1.1

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- **Criticality safety of fresh and spent fuel storage and handling**
  - New fuel storage facility includes:
    - Fuel assembly storage racks
    - Concrete storage vault containing the storage racks
    - Auxiliary components
  - Spent fuel storage facility includes:
    - Spent fuel storage racks
    - Spent fuel storage fuel that contains the storage racks
    - Associate equipment storage pits
  - New and spent fuel need to remain sub-critical during all credible storage conditions
  - Will also remain sub-critical during fuel handling

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Chapter 9.1.1 (cont'd)

- **References**
  - 10 CFR 50 Appendix A GDC 62, *Prevention of Criticality in Fuel Storage and Handling*
  - 10 CFR 50.68 *Criticality Accident Requirements*
  - ANSI / ANS 57.1 - 1992, *Design Requirements for Light Water Reactor Fuel Handling Systems* (reaffirmed in 2005)
  - ANSI / ANS 57.2 - 1983, *Design Requirements for Light Water Reactor Spent Fuel Storage Facilities in Nuclear Power Plants* (withdrawn in 1999 / reaffirmation in process)
  - ANSI / ANS 57.3 - 1983, *Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants* (withdrawn in 1993 / reaffirmation in process)
  - NRC Regulatory Guide 1.13, Revision 2, *Spent Fuel Storage Facility Design Basis*, March 2007

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Chapter 9.1.2

- **New and spent fuel storage**
  - Maintain fuel assemblies in safe and subcritical array during all credible storage conditions
  - Provide safe means of loading the spent fuel assemblies into shipping or storage casks
  - Addresses new and spent fuel storage facilities
  - Addresses all anticipated operating and accident conditions

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Chapter 9.1.2 (cont'd)

- **References**
  - 10 CFR 50 Appendix A GDC 61, *Fuel Storage and Handling and Radioactivity Control*
  - 10 CFR 50 Appendix A GDC 63, *Monitoring Fuel and Waste Storage*
  - 10 CFR 50.68, *Criticality Accident Requirements*
  - 10 CFR 70.24, *Criticality Accident Requirements*
  - NRC Regulatory Guide 1.13, Revision 2, *Spent Fuel Storage Facility Design Basis*, March 2007
  - NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants*
  - ANSI / ANS 57.2 - 1983, *Design Requirements for Light Water Reactor Spent Fuel Storage Facilities in Nuclear Power* Plants (withdrawn in 1999 / reaffirmation in process)
  - ANSI / ANS 57.3 - 1983, *Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants* (withdrawn in 1993 / reaffirmation in process)

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Chapter 9.1.3

- **Spent fuel pool cooling and cleanup system**
  - Spent fuel assemblies must be cooled and must remain covered with water (removal of decay heat).
  - Cleanup of the spent fuel pool, refueling canal, refueling water storage tank, and other equipment storage pools
  - Means for filling and draining the refueling canal and other storage pools
  - Surface skimming to provide clear water in the spent fuel pool
  - Seismic Category I water source
  - Clean-up system filter-demineralizers
  - Regenerative process to the point of discharge to the Rad-waste System

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Chapter 9.1.3 (cont'd)

- **References**
  - 10 CFR 50 Appendix A GDC 62, *Prevention of Criticality in Fuel Storage and Handling*
  - 10 CFR 50.68 Criticality Accident Requirements
  - ANSI / ANS 57.1 - 1992, *Design Requirements for Light Water Reactor Fuel Handling Systems* (reaffirmed in 2005)
  - ANSI / ANS 57.2 - 1983, *Design Requirements for Light Water Reactor Spent Fuel Storage Facilities in Nuclear Power Plants* (withdrawn in 1999 / reaffirmation in process)
  - ANSI / ANS 57.3 - 1983, *Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants* (withdrawn in 1993 / reaffirmation in process)
  - NRC Regulatory Guide 1.13, Revision 2, *Spent Fuel Storage Facility Design Basis*, March 2007

Chapter 9.1.4

- **Light Load Handling System (Related to Refueling)**
  - Consists of all components and equipment used for handling new fuel from the receiving station to loading spent fuel into the shipping cask
  - Heavy crane is used for moving spent fuel casks.
  - Objective is to avaoid criticality accidents, radioactivity releases from irradiated fuel, and unacceptable personnel radiation exposures.

Chapter 9.1.4 (cont'd)

- **References**
  - 10 CFR 50 Appendix A GDC 61, *Fuel Storage and Handling and Radioactivity Control*
  - 10 CFR 50 Appendix A GDC 62, *Prevention of Criticality in Fuel Storage and Handling*
  - ANSI / ANS 57.1 - 1992, *Design Requirements for LWR Fuel Handling Systems*

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Chapter 9.1.5

- **Overhead heavy load handling system**
  - A heavy load is defined as weighing more than one fuel assembly and its handling device.
  - Emphasis is on heavy load handling.
  - Objective is to avoid inadvertent operations or equipment malfunctions, separately or in combination, could cause a release of radioactivity, a criticality accident, inability to cool fuel within the reactor vessel or spent fuel pool or could prevent safe shutdown of the reactor.

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Chapter 9.1.5 (cont'd)

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- **References**
  - Regulatory Guide 1.13, *Spent Fuel Storage Facility Design Basis*
  - *NUREG-0554, Single-Failure-Proof Cranes For Nuclear Power Plants*
  - NUREG-0612, Control of Heavy Loads At Nuclear Power Plants
  - NRC Inspection Manual Chapter IMC 2504, *Construction Inspection Program – Non-ITAAC Inspections,* issued April 25, 2006
  - ANSI N14.6-1993, *Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10,000 lbs (4500 kg) or More*
  - ASME B30.2-2005, *Overhead and Gantry Cranes – Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist*
  - ASME B30.9-2003, *Slings*
  - ASME NOG-1-2004, *Rules for Construction of Overhead and Gantry Cranes*

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Water Systems
SRP Chapter 9.2

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9.2.1 Station Service Water Systems

9.2.2 Reactor Auxiliary Cooling Water Systems

9.2.3 Demineralized Water Makeup System

9.2.4 Potable and Sanitary Water Systems

9.2.5 Ultimate Heat Sink

9.2.6 Condensate Storage Facilities

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Chapter 9.2.1

- **Station Service Water System**
  - Provides essential cooling to safety related equipment
  - May also cool non-safety related auxiliary components used for normal plant operation
  - The ultimate heat sink (SRP Chapter 9.2.5) is the intake for the SWS.
  - The SWS pump performance characteristics are compared to the high and low water levels of the ultimate heat sink.
  - Other special equipment required to prevent or mitigate postulated accidents is a SWS interface system.

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Chapter 9.2.1 (cont'd)

- **References**
  - 10 CFR 50 Appendix A GDC 44, *Cooling Water*
  - 10 CFR 50 Appendix A GDC 45, *Inspection of Cooling Water Systems*
  - 10 CFR 50 Appendix A GDC 46, *Testing of Cooling Water Systems*
  - Generic Letter 89-13, *Service Water System Problems Affecting Safety Related Equipment*
  - Generic Letter 91-13, Request for Information Related to Resolution of Generic Issue 130 "*Essential Water Service Water System Failures at Multi-Unit Sites*"
  - NUREG-0927, *Evaluation of Water Hammer in Nuclear Power Plants*
  - NUREG-1461, Regulatory Analysis for the Resolution Of Generic Issue 133 "*Loss of Essential Water Service in LWRs*"

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Chapter 9.2.2

- **Reactor Auxiliary Cooling Water Systems**
  - Provides a closed loop of cooling water for reactor system components, reactor shutdown equipment, ventilation equipment, and components of ECCS.
  - These system components are required for safe shutdown during normal operations, anticipated operational occurrences, and accident conditions.
  - These system components are also required to prevent or mitigate the consequences of an accident.

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Chapter 9.2.2 (cont'd)

- **References**
  - 10 CFR 50 Appendix A GDC 44, *Cooling Water*
  - 10 CFR 50 Appendix A GDC 45, *Inspection of Cooling Water Systems*
  - 10 CFR 50 Appendix A GDC 46, *Testing of Cooling Water Systems*
  - IEEE Std 603-1980, *Standard Criteria for Safety Systems for Nuclear Generating Stations*

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Chapter 9.2.3

- **Demineralized Water Makeup System**
  - This system provides an adequate supply of treated water of reactor coolant purity to other systems as makeup.
  - This system is generally not safety related, but failure or malfunction of the system might adversely affect safety related systems.
  - This system treats raw water to provide reactor coolant purity water.

Chapter 9.2.3 (cont'd)

- **References**
  - 10 CFR 50 Appendix A GDC 2, *Design Basis for Protection Against Natural Phenomena*
  - 10 CFR 50 Appendix A GDC 5, *Sharing of Structures, Systems, and Components*

Chapter 9.2.4

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- **Potable and Sanitary Water Systems**
  - Prevent connection to systems with potential for containing radioactive materials.
  - Evaluate potential radioactive contamination, including accidental, and safety implications of sharing systems on a multi-unit site.
  - The potable water system is protected by an air gap, where necessary.

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Chapter 9.2.4 (cont'd)

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- **References**
  - 10 CFR 50 Appendix A GDC 60, *Control of Releases of Radioactive Materials to the Environment*

    GDC 60 – The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

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Chapter 9.2.5

- **Ultimate Heat Sink**
  - Typically consists of an assured supply of water that is credited for dissipating reactor decay heat and essential station heat loads after a normal reactor shutdown or a shutdown following an accident or transient (including LOCA).
  - Many NPPs rely on the atmosphere for all or a portion of this Ultimate Heat Sink.
  - Potential issues are size & type of cooling water supply, makeup sources, and the capability of the Ultimate Heat Sink to deliver the required flow of cooling water at appropriate temperatures.

Chapter 9.2.5 (cont'd)

- **References**
  - 10 CFR 50 Appendix A GDC 44, *Cooling Water*
  - 10 CFR 50 Appendix A GDC 45, *Inspection of Cooling Water Systems*
  - 10 CFR 50 Appendix A GDC 46, *Testing of Cooling Water Systems*
  - RG 1.27, *Ultimate Heat Sink for Nuclear Power Plants*
  - RG 1.72, *Spray Pond Piping Made From Fiberglass-Reinforced Thermosetting Resin*
  - ANS 5.1, *Decay Heat Removal for Light Water Reactors,* October 1979

Chapter 9.2.6

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- **Condensate Storage Facilities**
  - It acts as a receiver for excess water from other systems.
  - It also acts as a source of water for various auxiliary systems.
  - Due to the interfacing system requirements, the volume of this tank is typically quite large (about 100,000 gal of water).

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Chapter 9.2.6 (cont'd)

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- **References**
  - 10 CFR 50 Appendix A GDC 44, *Cooling Water*
  - 10 CFR 50 Appendix A GDC 45, *Inspection of Cooling Water Systems*
  - 10 CFR 50 Appendix A GDC 46, *Testing of Cooling Water Systems*
  - 10 CFR 50 Appendix A GDC 60, *Control of Releases of Radioactive Materials to Environment*
  - RG 1.143, *Design Guidance for Radioactive Waste Management of Structures, Systems, and Components Installed in Light-Water-Cooled Nuclear Power Plants*
  - RG 1.155, *Station Blackout*

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Fluid Control Systems
SRP Chapter 9.3

9.3.1 Compressed Air System

9.3.2 Process and Post-Accident Sampling System

9.3.3 Equipment and Floor Drainage Systems

9.3.4 Chemical and Volume Control System (PWR) (including boron recovery system)

9.3.5 Standby Liquid Control System (BWR)

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Chapter 9.3.1

- **Compressed Air System**
  - Provides compressed air to:
    - Instrument air (safety related)
    - Service air (non-safety related)
  - The two compressed air systems may be cross connected.
  - Quality of instrument air is very important.
  - Service air is frequently a backup source of air for Instrument air.

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Chapter 9.3.1 (cont'd)

- **References**
  - NUREG-1275, Volume 2, *Operating Experience Feedback Report – Air Systems Problems*
  - NRC Generic Letter 88-14, *Instrument Air Supply System Problems Affecting Safety Related Equipment*
  - ANSI / ISA-S7.3-1976 (1981), *Quality Standard for Instrument Air*

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Chapter 9.3.2

- **Process and Post-Accident Sampling System**
  - Identify the process streams to be sampled and the parameters to be determined through sampling (i.e., gross beta-gamma concentration, boric acid concentration).
  - Provision for representative sampling (i.e., purge times).
  - Location of sampling points and sampling stations.

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Chapter 9.3.2 (cont'd)

- **References**
  - 10 CFR 50.34(f)(2)(xxvi) *Additional TMI-related Requirements*
  - ANSI / HPS N13.1-1999, Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities
  - NUREG-1793, *FSER Related to Certification of AP1000 Standard*, Vol 1, September 2004
  - EPRI Report TR-103515-R2, *BWR Water Chemistry Guidelines*, Rev 2
  - ERPI Report TR-1008224-R6, *PWR Secondary Water Chemistry Guidelines:* Rev 6
  - EPRI Report TR-1002884-V1R5, *PWR Primary Chemistry Guideline:* Vol 1 Rev 5
  - EPRI Report TR-105714-V2R4, *PWR Primary Chemistry Guidelines:* Vol 2 Rev 4

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Chapter 9.3.3

- **Equipment and Floor Drainage Systems**
  - Designed to ensure that waste liquids, valve and pump leak-offs, and tank drains are directed to proper areas for processing or disposal.
  - Designed so that excessive water accumulation and flooding is prevented.
  - Designed so that collecting, routing, and disposing of fluids from inside containment.
  - Piping and pumps from equipment or floor drains to the sumps, drain tanks, devices that are credited with preventing reverse flow, and any additional equipment necessary to route effluents to the drain tanks and then liquid waste management systems or other points of discharge.

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Chapter 9.3.3 (cont'd)

- **References**
  - 10 CFR 50 Appendix A GDC 2, *Design Basis for Protection Against Natural Phenomena*
  - 10 CFR 50 Appendix A GDC 4, *Environmental and Dynamic Effects Design Basis*
  - 10 CFR 50 Appendix A GDC 60, *Control of Releases of Radioactive Materials to the Environment*

Chapter 9.3.4

- **Chemical and Volume Control System (PWR) (including boron recovery system)**
  - This system:
    - Maintains required water inventory and quality in the RCS
    - Provides seal-water flow to RCS pumps and pressurizer auxiliary spray
    - Controls the boron neutron absorber concentration in the reactor coolant
    - Controls the primary water chemistry
    - Reduces reactor coolant reactivity level
  - Provides recycled reactor coolant for demineralized water makeup for normal operation
  - May provide high pressure injection for ECCS.

Chapter 9.3.4 (cont'd)

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- **References**
  - 10 CFR 50 Appendix A GDC 33, *Reactor Coolant Makeup*
  - 10 CFR 50 Appendix A GDC 35, *Emergency Core Cooling*
  - 10 CFR 50 Appendix A GDC 60, *Control of Releases of Radioactive Materials to the Environment*
  - NUREG-0718, *Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses*
  - NRC Generic Letter 80-21, *Vacuum Condition Resulting in Damage to CVCS Holdup Tanks (Sometimes Called Clean Waste Receiver Tanks)*
  - NRC Generic Letter 89-04, *Guidance on Developing Acceptable Inservice Testing Programs*

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Chapter 9.3.5

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- **Standby Liquid Control System (BWR)**
  - May be part of ECCS
  - Provides backup capability for reactivity control independent of the CRD system
  - Functions by injecting a boron solution into the reactor to effect shutdown
  - Can control reactivity difference between steady-state operating condition (at any time during core life) and cold shutdown condition.

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Chapter 9.3.5 (cont'd)

- **References**
  - NRC Generic Letter 85-03, *Clarification of Equivalent Control Capacity for Standby Liquid Control Systems*
  - NRC Information Notice 1991-012, *Potential Loss of NPSH of Standby Liquid Control System Pumps*
  - NRC Information Notice 2001-013, *Inadequate Standby Liquid Control System Relief Valve Margin*

Ventilation Systems
SRP Chapter 9.4

9.4.1 Control Room Area Ventilation System

9.4.2 Spent Fuel Pool Area Ventilation System

9.4.3 Auxiliary and Radwaste Area Ventilation System

9.4.4 Turbine Area Ventilation System

9.4.5 Engineered Safety Feature Ventilation System

Chapter 9.4.1

- **Control Room Area Ventilation System**
  - Provides a controlled environment for the comfort and safety of control room personnel
  - Assures the operability of the control room components during normal operating, anticipated operational transient, and design basis accident conditions
  - Portions of the system may also be relied upon to ensure coping with and recovering from a station blackout event

Chapter 9.4.1 (cont'd)

- **References**
  - 10 CFR 50 Appendix A GDC 19, *Control Room*
  - NRC RG 1.52, *Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light Water Cooled NPPs*
  - NRC RG 1.78, *Evaluating the Habitability of a NPP Control Room During a Postulated Hazardous Chemical Release*
  - NRC RG 1.140, *Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light Water Cooled NPPs*
  - ASME AG-1 Code, *Code for Nuclear Air and Gas Treatment*, 1991

Chapter 9.4.2

U.S.NRC United States Nuclear Regulatory Commission Protecting People and the Environment

- **Spent Fuel Pool Area Ventilation System**
  - Function is to:
    - Maintain ventilation
    - Permit personnel access
    - Control air-borne radioactivity
  - Functions:
    - During normal plant operation
    - During anticipated operational occurrences
    - Following postulated fuel handling accidents
  - Discharges to the gaseous cleanup and treatment system or station vents

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Chapter 9.4.2 (cont'd)

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- **References**
  - 10 CFR 50 Appendix A GDC 61, *Fuel Storage and Handling and Radioactivity Control*
  - NRC RG 1.13, *Fuel Storage Facility Design Basis*
  - NRC RG 1.52, *Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light Water Cooled NPPs*
  - NRC RG 1.140, *Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light Water Cooled NPPs*

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Chapter 9.4.3

- **Auxiliary and Radwaste Area Ventilation System**
  - Function is to:
    - Maintain ventilation
    - Permit personnel access
    - Control the concentration of airborne radioactive material
  - Functions:
    - During normal operation
    - During anticipated operational occurrences
    - After postulated accidents
  - System includes air intakes, ducts, air conditioning units, filters, blowers, isolation dampers, and exhaust fans.

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Chapter 9.4.3 (cont'd)

- **References**
  - 10 CFR 50 Appendix A GDC 60, *Control of Releases of Radioactive Materials to the Environment*
  - NRC RG 1.52, *Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light Water Cooled NPPs*
  - NRC RG 1.140, *Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light Water Cooled NPPs*

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Chapter 9.4.4

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- **Turbine Area Ventilation System**
  - Function is to:
    - Maintain ventilation
    - Permit personnel access
    - Control the concentration of airborne radioactive material
  - Functions:
    - During normal operation
    - During anticipated operational occurrences
    - After any accident that releases radioactive material
  - System includes air intakes, ducts, air conditioning units, filters, blowers, isolation dampers, and exhaust fans.

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Chapter 9.4.4 (cont'd)

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- **References**
  - 10 CFR 50 Appendix A GDC 60, *Control of Releases of Radioactive Materials to the Environment*
  - NRC RG 1.52, *Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light Water Cooled NPPs*
  - NRC RG 1.140, *Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light Water Cooled NPPs*

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Chapter 9.4.5

- **Engineered Safety Feature Ventilation System**
  - Purpose of this system is to provide a suitable and controlled environment for the ESF components during certain transients and design basis accidents.
  - The areas include the service water pump house, diesel-generator areas, ECCS pump rooms, component cooling water pump room, aux feedwater pump area, and other areas containing equipment necessary for safe shutdown and accident mitigation.

Chapter 9.4.5 (cont'd)

- **References**
  - NRC RG 1.52, *Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light Water Cooled NPPs*
  - NRC RG 1.140, *Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light Water Cooled NPPs*
  - NRC RG 1.55, *Station Blackout*
  - NUREG/CR-660*, Enhancement of Onsite Emergency Diesel Generator Reliability*

Electrical Support Systems
SRP Chapter 9.5

9.5.1 Fire Protection Program
9.5.2 Communications System
9.5.3 Lighting System
9.5.4 Emergency Diesel Engine Fuel Oil Storage and Transfer System
9.5.5 Emergency Diesel Engine Cooling Water System
9.5.6 Emergency Diesel Engine Starting System
9.5.7 Emergency Diesel Engine Lubricating System
9.5.8 Emergency Diesel Engine Combustion Air Intake and Exhaust System

Chapter 9.5.1

- **Fire Protection Program**
  - Purpose of the FP program is to provide defense-in-depth that the NRC FP objectives are satisfied:
    - Prevent fires from starting
    - To detect rapidly, control, and extinguish those fires that do occur
    - To provide protection for SSCs important to safety so that a fire that is not promptly extinguished by fire suppression activities will not prevent the safe shutdown of the plant.
  - In addition, FP systems must be designed such that their failure or inadvertent operation will not adversely impact SSCs important to safety to perform their function.

Chapter 9.5.1 (cont'd)

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- **References**
  - 10 CFR 50.48, *Fire Protection*
  - 10 CFR 50 Appendix R, *Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979*
  - NFPA 804*, Standard for Fire Protection for Advanced LWR Electric Generating Plants*
  - NFPA 805, *Performance-Based Standard for Fire Protection for LWR Electric Generating Plants*
  - Draft NUREG-1824 / EPRI 1011999, *Verification and Validation of Selected Fire Models for NPP Applications*
  - NRC RG 1.189, *Fire Protection for NPPs*
  - NRC RG 1.191, *Fire Protection Program for NPPs During Decommissioning and Permanent Shutdown*

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Chapter 9.5.2

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- **Communications System**
  - Limited to intraplant and plant-to-offsite communications
  - During normal operations, transients, fire, accidents, off-normal phenomena, and security related events
  - Between the plant and the NRC Incident Response Center and local authorities

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Chapter 9.5.2 (cont'd)

- **References**
  - 10 CFR 73.55(f), *Communications Subsystems*
  - 10 CFR 50 Appendix A GDC 19, *Control Room*
  - NRC RG 1.180, *Guidelines for Evaluating Electro-magnetic and Radio-frequency Interference in Safety Related Instrumentation and Control Systems*
  - EPRI Topical Report TR-106439, *Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications*, October 1996

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Chapter 9.5.3

- **Lighting System**
  - The capability of the normal lighting system(s) to provide adequate lighting during all plant operating conditions
  - The capability of the emergency lighting system to provide adequate lighting during all plant operating conditions, including fire, transient, and accident conditions
  - The effect of the loss of AC power (e.g., station blackout) on emergency lighting systems
  - The failure analysis of normal and emergency lighting systems

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Chapter 9.5.3 (cont'd)

- **References**
  - NUREG-1793, *FSER Related to Certification of the AP1000 Standard Design*
  - NRC RG 1.206, *Combined License Applications for NPPs (LWR Edition)*
  - *Illuminating Engineering Society of North America Lighting Handbook*

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Chapter 9.5.4

- **Emergency Diesel Engine Fuel Oil Storage and Transfer System**
  - Covers the system up to the engine housing
  - Compliance with GDC 2, 4, 5, and 17 by piping up to:
    - Engine interface
    - Fuel oil storage tanks
    - Fuel oil transfer pumps
    - Day tanks
    - Tank storage vaults
  - Quality and quantity of fuel oil stored onsite
  - Availability and procurement of additional fuel oil from offsite sources

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Chapter 9.5.4 (cont'd)

- **References**
  - 10 CFR 50 Appendix A GDC 17, *Electrical Power Systems*
  - NRC RG 1.137, *Diesel Generator Fuel Oil Systems*
  - NUREG/CR-0660, *Enhancement of Onsite Emergency Diesel Generator Reliability,* University of Dayton Research Institute, UDR-TR-79-07, February 1979
  - ANSI / ANS-59.51-1997, *Fuel Oil Systems for Safety Related Emergency Diesel Generators*
  - Diesel Engine Manufacturers Association (DEMA) Standard, 1974

Chapter 9.5.5

- **Emergency Diesel Engine Cooling Water System**
  - Compliance with GDC 2, 4, 5, 17, 44, 45, and 46
  - Portions of the diesel engine receiving heat from components essential for proper operation
  - Parts of system transferring engine heat to heat sink
  - All valves, heat exchangers, pumps, and piping up to engine interface

Chapter 9.5.5 (cont'd)

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- **References**
  - 10 CFR 50 Appendix A GDC 44, *Cooling Water*
  - 10 CFR 50 Appendix A GDC 45, *Inspection of Cooling Water Systems*
  - 10 CFR 50 Appendix A GDC 46, *Testing of Cooling Water Systems*
  - NUREG/CR-0660*, Enhancement of Onsite Emergency Diesel Generator Reliability,* University of Dayton Research Institute, UDR-TR-79-07, February 1979
  - Diesel Engine Manufacturers Association (DEMA) Standard, 1974

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Chapter 9.5.6

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- **Emergency Diesel Engine Starting System**
  - Reliable engine starting following loss of off-site power
  - Compliance with GDC 2, 4, 5, and 17
  - System components:
    - Air compressors
    - Air dryers
    - Air receivers
    - Devices to crank the diesel engine
    - Valves
    - Piping to engine interfaces
    - Filters
    - Ancillary instrumentation and controls

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Chapter 9.5.6 (cont'd)

- **References**
  - 10 CFR 50 Appendix A GDC 5, *Sharing of SSCs*
  - 10 CFR 50 Appendix A GDC 17, *Electrical Power Systems*
  - NUREG/CR-0660, *Enhancement of Onsite Emergency Diesel Generator Reliability,* University of Dayton Research Institute, UDR-TR-79-07, February 1979
  - Diesel Engine Manufacturers Association (DEMA) Standard, 1974

Chapter 9.5.7

- **Emergency Diesel Engine Lubricating System**
  - Provides essential lubrication to engine components
  - Compliance with GDC 2, 4, 5, and 17
  - Includes the following portions of the system:
    - Piping
    - Pumps
    - Components
    - Auxiliary equipment

Chapter 9.5.7 (cont'd)

- **References**
  - 10 CFR 50 Appendix A GDC 5, *Sharing of SSCs*
  - NUREG/CR-0660, *Enhancement of Onsite Emergency Diesel Generator Reliability*, University of Dayton Research Institute, UDR-TR-79-07, February 1979
  - ANSI / ANS-59.52-1998, Lubricating Oil Systems for Safety Related Diesel Generators
  - Diesel Engine Manufacturers Association (DEMA) Standard, 1974

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Chapter 9.5.8

- **Emergency Diesel Engine Combustion Air Intake and Exhaust System**
  - Supplies combustion air of reliable quality to diesel engines
  - Exhausts combustion products from the diesel engines to the atmosphere
  - System extends from outside air intakes to combustion air supply lines to the engine interface
  - System extends from the exhaust connections at the engine interface to the discharge point outside the building

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Chapter 9.5.8 (cont'd)

- **References**
  - 10 CFR 50 Appendix A GDC 5, *Sharing of SSCs*
  - NUREG/CR-0660, *Enhancement of Onsite Emergency Diesel Generator Reliability*, University of Dayton Research Institute, UDR-TR-79-07, February 1979
  - Diesel Engine Manufacturers Association (DEMA) Standard, 1974

Learning Questions

- **What is considered an "Auxiliary System" in SRP Chapter 9?**
- **Which references, including Codes & Standards, are used as a vehicle to meet SRP requirements?**
- **Can auxiliary system features be important in the construction of a nuclear power plant?**

Module 9

Standard Review Plan Chapter 10

Module 9
Standard Review Plan Chapter 10

Instructor: C. Wesley Rowley, P.E.

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Learning Objectives

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- **Learn what is considered a "Steam & Power Conversion System" in SRP Chapter 10**
- **Learn which Codes & Standards are used as a vehicle to meet SRP requirements.**

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Significant Sub-topics

U.S.NRC
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- Definition of "Steam & Power Conversion" Systems
- Matrix of NRC Regulations and RGs for Various Aspects of the Steam & Power Conversion System
- Description of Each Referenced Code / Standard
- Summary of Each Identified Code / Standard
- Specific Use of Code / Standard to Meet SRP Requirement

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NUREG-0800

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN

10.2 TURBINE GENERATOR

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of power conversion systems

Secondary - None

I. <u>AREAS OF REVIEW</u>

Nuclear reactor plants include a turbine generator system (TGS) to convert the energy in steam from the nuclear steam supply system into electrical energy. The TGS consists essentially of (1) the turbine unit and the automatic devices, alarms, and trips that control and regulate turbine action and (2) the generator unit and its controls. The turbine control system, steam inlet stop and control valves, low-pressure turbine steam intercept and inlet control valves, and extraction steam control valves control the speed of the turbine under normal and abnormal conditions and are thus related to the overall safe operation of the plant.

The TGS installed in a nuclear plant is typically equipped with redundant overspeed protection instrumentation and controls. The main steam and reheat steam control and stop valving arrangements typically provide redundancy in the valves essential for overspeed protection. The intent of the review under this SRP section is to verify that such redundancy, in conjunction with inservice inspection and testing of the essential valves, makes a turbine overspeed condition that exceeds the design overspeed very unlikely and to ensure conformance with General Design Criterion (GDC) 4.

Revision 3 - March 2007

USNRC STANDARD REVIEW PLAN

This Standard Review Plan, NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC's regulations. The Standard Review Plan is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide an acceptable method of complying with the NRC regulations.

The standard review plan sections are numbered in accordance with corresponding sections in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of Regulatory Guide 1.70 have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) are based on Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

These documents are made available to the public as part of the NRC's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Individual sections of NUREG-0800 will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments may be submitted electronically by email to NRR_SRP@nrc.gov.

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SRP Chapter 10
Steam and Power Conversion
CONTENTS

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SRP Chapters

- SRP 10.2 Turbine Generator
- SRP 10.2.3 Turbine Rotor Integrity
- SRP 10.3 Main Steam Supply System
- SRP 10.3.6 Steam and FW System Materials
- SRP 10.4.1 Main Condensers
- SRP 10.4.2 Main Condenser Evacuation
- SRP 10.4.3 Turbine Gland Sealing
- SRP 10.4.4 Turbine Bypass
- SRP 10.4.5 Circulating Water System
- SRP 10.4.6 Condensate Cleanup
- SRP 10.4.7 Condensate and Feedwater
- SRP 10.4.8 Steam Generator Blowdown
- SRP 10.4.9 Auxiliary Feedwater System

Applicable Regulations

- Turbine Generator
- Turbine Rotor Integrity
- Main Steam Supply System – 10 CFR 50.63
- Steam and FW System Materials – 10 CFR 50.55a, 52.47, and 52.97
- Main Condensers – 10 CFR 50 Appendix I and 10 CFR 52
- Main Condenser Evacuation – 10 CFR 50 Appendix I and 10 CFR 52
- Turbine Gland Sealing – 10 CFR 52
- Turbine Bypass – 10 CFR 52.47 and 10 CFR 52.80
- Circulating Water System – 10 CFR 52.47 and 10 CFR 52.80
- Condensate Cleanup – 10 CFR 52.47 and 10 CFR 52.80
- Condensate and Feedwater
- Steam Generator Blowdown – 10 CFR 52.47 and 52.97
- Auxiliary Feedwater System – 10 CFR 50.62, 50.63

Regulation Titles

- **10 CFR 50.55a *Codes and Standards***
- **10 CFR 50.62 *Requirements for Reduction of Risk from ATWS Events for Light-Water NPPs***
- **10 CFR 50.63 *Loss of All AC Power***
- **10 CFR 50 Appendix B, *Quality Assurance Criteria for NPPs and Fuel Processing Plants***
- **10 CFR 50 Appendix I, *Numerical Guides for Design Objectives and Limiting Conditions for Operation To Meet the Criterion "As Low As Reasonably Achievable" for Radioactive Material in Light Water Cooled Nuclear Power Reactor Effluents***
- **10 CFR 52 *Early Site Permits; Standard Design Certification; and Combined Licenses for NPPs***
- **10 CFR 52.47 *ITAAC for Design Certification (Contents of Applications)***
- **10 CFR 52.80 *Issuance of Combined Licenses***
- **10 CFR 52.97 *ITAAC for Combined Licenses***

Applicable Appendix A GDCs

- **Turbine Generator – GDC 4**
- **Turbine Rotor Integrity – GDC 4**
- **Main Steam Supply System – GDCs 2, 4, 5, and 34**
- **Steam and FW System Materials – GDC 1 and 35**
- **Main Condensers – GDC 60**
- **Main Condenser Evacuation – GDC 60**
- **Turbine Gland Sealing – GDC 60**
- **Turbine Bypass – GDC 4 and 34**
- **Circulating Water System – GDC 4**
- **Condensate Cleanup – GDC 14**
- **Condensate and Feedwater – GDCs 2, 4, 5, 44, 45, and 46**
- **Steam Generator Blowdown – GDCs 1, 2, 13, and 14**
- **Auxiliary Feedwater System – GDCs 2, 4, 5, 19, 34, 44, 45, and 46**

GDC Titles

U.S.NRC United States Nuclear Regulatory Commission *Protecting People and the Environment*

- GDC 1 - Quality standards and records
- GDC 2 - Design bases for protection against natural phenomena
- GDC 4 - Environmental and dynamic effects design bases
- GDC 5 - Sharing of structures, systems, and components
- GDC 13 - Instrumentation and control
- GDC 14 - Reactor coolant pressure boundary
- GDC 19 - Control room
- GDC 34 - Residual heat removal
- GDC 35 - Emergency core cooling
- GDC 44 - Cooling water
- GDC 45 - Inspection of cooling water system
- GDC 46 - Testing of cooling water system
- GDC 60 - Control of releases of radioactive materials to the environment

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Applicable RGs

U.S.NRC United States Nuclear Regulatory Commission *Protecting People and the Environment*

- Turbine Generator – RG 1.68
- Turbine Rotor Integrity
- Main Steam Supply System – RGs 1.29, 1.115, 1.117, and 1.155
- Steam and FW System Materials – RGs 1.37, 1.50, 1.71, and 1.85
- Main Condensers
- Main Condenser Evacuation
- Turbine Gland Sealing
- Turbine Bypass
- Circulating Water System
- Condensate Cleanup
- Condensate and Feedwater – RG 1.29
- Steam Generator Blowdown – RGs 1.26, 1.29, and 1.143
- Auxiliary Feedwater System – RG 1.29, 1.59, 1.62, 1.76, 1.102, 1.117, and 1.155

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RG Titles

- RG 1.26 Quality Group Classifications for Water-, Steam-, and Radioactive-Waste Containing Components in NPPs
- RG 1.29 Seismic Design Classification
- RG 1.37 Quality Assurance Requirements for Cleaning of Fluid Systemsand Associated Components of Water-Cooled NPPs
- RG 1.50 Control of Preheat Temperature for Welding of Low-Alloy Steel
- RG 1.59 Design Basis Floods for NPPs
- RG 1.62 Manual Initiation of Protective Actions
- RG 1.68 Initial Test Programs for Water-Cooled Reactor Power Plants
- RG 1.71 Welder Qualification for Areas of Limited Accessibility
- RG 1.76 Design Basis Tornado for NPPs
- RG 1.85 Design, Fabrication, and Materials Code Case Acceptability, ASME Section III
- RG 1.102 Flood Protection for NPPs
- RG 1.115 Protection Against Low-Trajectory Turbine Missiles
- RG 1.117 Tornado Design Classification
- RG 1.143 Design Guidance for Radioactive Waste Management SSCs in LWR Reactor NPPs
- RG 1.155 Station Blackout

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Applicable BTPs

- Turbine Generator – BTP 3-3 and BTP 3-4
- Turbine Rotor Integrity
- Main Steam Supply System – BTP 3-1, BTP 3-2, and BTP 5-4
- Steam and FW System Materials
- Main Condensers
- Main Condenser Evacuation
- Turbine Gland Sealing
- Turbine Bypass – BTP 3-1, BTP 3-3, BTP 3-4, and BTP 10-2
- Circulating Water System
- Condensate Cleanup
- Condensate and Feedwater – BTP 10-2
- Steam Generator Blowdown – BTP 5-3
- Auxiliary Feedwater System – BTP 5-4 and BTP 10-1

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BTP Titles

- **BTP 3-1** ***Classification of Main Steam Components Other Than the Reactor Coolant Pressure Boundary for BWR Plants***
- **BTP 3-2** ***Classification of BWR/6 Main Steam and Feedwater Components Other Than Reactor Coolant Pressure Boundary***
- **BTP 3-3** ***Protection Against Postulated Piping Failures in Fluid Systems Outside Containment***
- **BTP 3-4** ***Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment***
- **BTP 5-3** ***Monitoring of Secondary Side Water Chemistry in PWR Steam Generators***
- **BTP 5-4** ***Design Requirements of the Residual Heat Removal System***
- **BTP 10-1** ***Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for PWRs***
- **BTP 10-2** ***Design Guidelines for Avoiding Water Hammers in Steam Generators***

Applicable NUREGs

- **Turbine Generator**
- **Turbine Rotor Integrity**
- **Main Steam Supply System – NUREG-0138**
- **Steam and FW System Materials – NUREG-1344**
- **Main Condensers**
- **Main Condenser Evacuation**
- **Turbine Gland Sealing**
- **Turbine Bypass**
- **Circulating Water System**
- **Condensate Cleanup**
- **Condensate and Feedwater**
- **Steam Generator Blowdown**
- **Auxiliary Feedwater System – NUREG-0611, NUREG/CR-2300, and NUREG/CR-2815**

NUREG Titles

- NUREG-0138 *Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3 1976 Memorandum from Director NRR to NRR Staff*
- NUREG-0611 *Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse - Designed Operating Plants*
- NUREG-0619 *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*
- *NUREG-0635 Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering - Designed Operating Plants*
- NUREG-0737 *Clarification of TMI Action Plan Requirements*
- NUREG-0927 *Evaluation of Water Hammer Occurrences in NPPs*
- NUREG-1344
- NUREG/CR-2300 *PRA Procedures Guide,* January 1983
- NUREG/CR-2815 *PSA Procedures Guide,* January 1984

Applicable Codes & Standards

- Turbine Generator – ASME BPV Code Section XI
- Turbine Rotor Integrity – ASME BPV Code Sections III, V, & XI; ASTM A370 & ASTM E208
- Main Steam Supply System
- Steam and FW System Materials – ANSI N45.2.1, ASME BPV Code Sections II and IX
- Main Condensers
- Main Condenser Evacuation
- Turbine Gland Sealing
- Turbine Bypass
- Circulating Water System
- Condensate Cleanup
- Condensate and Feedwater
- Steam Generator Blowdown
- Auxiliary Feedwater System

Codes & Standards Titles

- **ANSI N45.2.1-1973, Cleaning of Fluid Systems and Associated Components During Construction Phase of NPPs**
- **ASME BPV Code, *Boiler and Pressure Vessel Code***
  - Section II Materials
  - Section III Nuclear Construction
  - Section V Nondestructive Examination
  - Section IX Welding and Brazing
  - Section XI Inservice Inspection
- **ASTM A370-05, Standard Test Methods and Definitions for Mechanical Testing of Steel Products**
- **ASTM E208-95a (2000), Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels**

Applicable Other References

- **Turbine Generator –**
- **Turbine Rotor Integrity – Westinghouse Scientific Paper and ORNL Report**
- **Main Steam Supply System – SECY 93-07, GL 86-09**
- **Steam and FW System Materials – Appendix B, NSAC-202L-R2, GL 89-08**
- **Main Condensers – SECY 93-087**
- **Main Condenser Evacuation**
- **Turbine Gland Sealing**
- **Turbine Bypass – SECY 93-087**
- **Circulating Water System**
- **Condensate Cleanup – EPRI Reports**
- **Condensate and Feedwater – GL 80-95, GL 81-11, and GL 89-08**
- **Steam Generator Blowdown**
- **Auxiliary Feedwater System – IE Bulletin 85-01 and GL 88-14**

Other Reference Titles

- **EPRI Reports**
  - NP-3944 *Erosion / Corrosion in Nuclear Plant Steam Piping: Causes and Inspection Guid*elines
  - NSAC 202L-R2 *Recommendations for an Effective Flow Accelerated Corrosion Program*, April 8, 1999
  - BWR Water Chemistry Guidelines
  - PWR Water Chemistry Guidelines
- **NRC**
  - GL 80-95 *Final Edition of NUREG-0619, BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*
  - GL 81-11 *BWR Feedwater Nozzle Cracking*
  - GL 86-09 *Technical Resolution of Generic Issue No. B-59, (N-1) Loop Operation in BWRs and PWRs*
  - GL 88-14 *Instrument Air Supply System Problems Affecting Safety Related Equipment*
  - GL 89-08 *Erosion / Corrosion – Induced Pipe Wall Thinning in US NPPs*
  - IE Bulletin 85-01 *Steam Binding of AFW Pumps*
  - SECY 93-087 *Policy, Technical, and Licensing Issues Pertaining to Evolutionary and ALWR Designs (Paragraph II.E, Classification of Main Steamlines in BWRs)*
- **ORNL-TM-3894 Report, *A Procedure for Determining Bounding Values on Fracture Toughness K_{Ic} at Any Temperature***
- **Westinghouse Scientific Paper 71-1E7-MSLRF-P1, *Correlation of Fracture Toughness and Charpy Properties for Rotor Steels***

Case Study

- **Case Study on Auxiliary Feedwater System**
  - What is the purpose of the AFW System?
  - What kind of diversity is utilized to improve the reliability of the AFW System?
  - How important is the AFW System?
  - What is the source of water for the AFW System?

Learning Questions

- **What is considered a "Steam & Power Conversion System" in SRP Chapter 10?**
- **List some example Codes & Standards that are used as a vehicle to meet SRP requirements in Chapter 10.**
- **What would be some additional Codes & Standards used for these SSCs that are not in the SRP chapters?**
- **Where would one look to identify the particular standards for a particular portion of the steam system?**

UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 10
Standard Review Plan Chapter 14

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 10
Standard Review Plan Chapter 14

Instructor: C. Wesley Rowley, PE

AdSTM Advanced Systems Technology and Management, Inc.

Learning Objectives

U.S.NRC United States Nuclear Regulatory Commission *Protecting People and the Environment*

- **Learn what is considered an "Initial Test Program" in SRP Chapter 14**
- **Learn how SRP Chapter 14 is used to review the acceptability of the Design Certification ITAACs**
- **Learn how Codes & Standards are used as a vehicle to meet regulatory requirements.**

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Significant Sub-topics

- Initial Test Program content
- Design Certification and COL ITAAC
- New reactor First of a Kind tests
- Specific Use of Codes/ Standards

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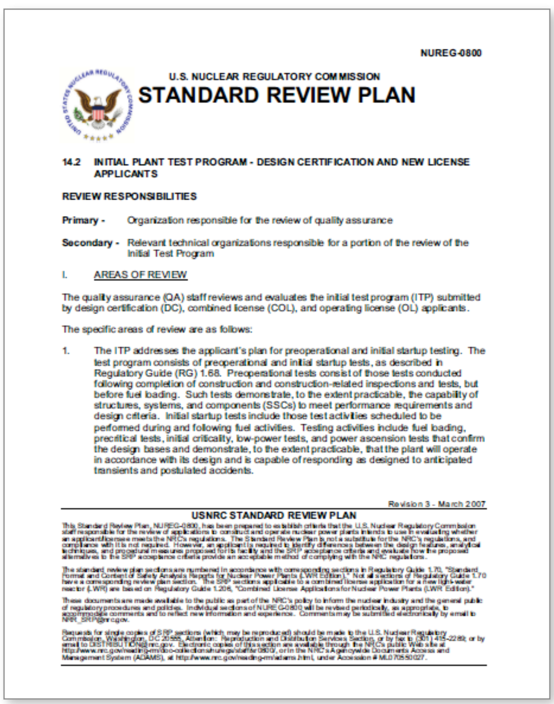

NUREG-0800

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN

14.2 INITIAL PLANT TEST PROGRAM - DESIGN CERTIFICATION AND NEW LICENSE APPLICANTS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of quality assurance

Secondary - Relevant technical organizations responsible for a portion of the review of the Initial Test Program

I. AREAS OF REVIEW

The quality assurance (QA) staff reviews and evaluates the initial test program (ITP) submitted by design certification (DC), combined license (COL), and operating license (OL) applicants.

The specific areas of review are as follows:

1. The ITP addresses the applicant's plan for preoperational and initial startup testing. The test program consists of preoperational and initial startup tests, as described in Regulatory Guide (RG) 1.68. Preoperational tests consist of those tests conducted following completion of construction and construction-related inspections and tests, but before fuel loading. Such tests demonstrate, to the extent practicable, the capability of structures, systems, and components (SSCs) to meet performance requirements and design criteria. Initial startup tests include those test activities scheduled to be performed during and following fuel activities. Testing activities include fuel loading, precritical tests, initial criticality, low-power tests, and power ascension tests that confirm the design bases and demonstrate, to the extent practicable, that the plant will operate in accordance with its design and is capable of responding as designed to anticipated transients and postulated accidents.

Revision 3 - March 2007

USNRC STANDARD REVIEW PLAN

This Standard Review Plan, NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC's regulations. The Standard Review Plan is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide an acceptable method of complying with the NRC regulations.

The standard review plan sections are numbered in accordance with corresponding sections in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of Regulatory Guide 1.70 have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) are based on Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

These documents are made available to the public as part of the NRC's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Individual sections of NUREG-0800 will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments may be submitted electronically by email to NRR_SRP@nrc.gov.

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SRP Chapter 14
Initial Test Program
CONTENTS

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Initial Test Program

- **The ITP identifies the applicant's plan for preoperational and initial startup testing.**
- **The NRC guidance for what constitutes an acceptable ITP guidance is described in RG 1.68.**
- **Standard Review Plan 14.2 outlines in detail the staff review of the applicants proposed ITP. This review includes the design certification and COL programs.**
- **Preoperational tests:**
  - are conducted after construction but before fuel loading.
  - demonstrate that SSCs meet performance requirements and design criteria,
- **Initial startup tests:**
  - are conducted during and after fuel loading.
  - consists of fuel loading, precritical tests, initial criticality, low power tests, and power ascension tests.
  - confirm the design basis.
  - Demonstrate, to the extent practicable, that plant will operate in accordance with its design and that it is capable of responding as designed to anticipated transients and postulated accidents.

NRC Staff Review

- **For the AP 1000, a structured review plan was developed and implemented by the NRC staff. It was consisted of the following activities:**
  - Verify that the AP1000 initial test program adequately demonstrates the performance of SSCs important to safety which are as follows:
    - Safety-related.
    - Within the scope of Regulatory Treatment of Non-Safety Systems (RTNSS).
    - Within the scope of the Design Reliability Assurance Program (D-RAP) as important to safety.
  - Verify that test abstracts included in DCD Section 14.2 adequately describe the required testing.
  - Verify that the preoperational test program was consistent with the system-based Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) described in DCD
  - Tier 1 information. This review ensured that the preoperational testing abstracts contained in DCD Tier 2, Chapter 14.2, were consistent with ITAAC requirements.
  - Additionally, the staff verified that the preoperational test program included all initial test program activities associated with the ITAAC and that they would be accomplished prior to initial fuel loading.

NRC Staff Review (cont'd)

- **A test by test and RG by RG comparison by the staff identified a number of tests needing clarification or in some cases, exceptions. These included:**
  - Exceptions:
    - DC power test-alternative test approved for the ADS squib valves so they would not have to be fired.
    - Safe shutdown criteria for the passive PRHX differs from typical PWR RHR.
  - Clarifications:
    - Natural circulation testing test and basis-this capability is not part of the safety case but will be tested.
    - Battery test criteria: the NRC staff considered alternative IEEE battery test standards.
    - Remote shutdown panel testing-clarification was provided by the applicant.

Operating Experience

- **SRP 14.2 also includes the need to consider operating experience in the test program. This is another important part of the staff review. Incorporation of operating experience into safety programs is not just and NRC position, but an international one, endorsed by the IAEA in their safety standards.**
- **DCD Section 14.2.5,"Utilization of Reactor Operating and Testing Experiences in the Development of the Test Program," states the following:**

> *The design, testing, startup, and operating experience from previous pressurized water reactor plants is utilized in the development of the initial preoperational and startup test program for the AP1000 plant.*
>
> *Other sources of experience reported and described in various documents such as NRC reports, including NRC bulletins, and Institute of Nuclear Power Operations (INPO) reports including Significant Operating Event Reports (SOERs), are also utilized in the AP1000 initial preoperational and startup test program.*

First of a Kind Tests

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- **Operating and regulatory experience, as well as new design features, create the need for first of a kind or special tests.**
- **For the AP 1000, for example, these tests include:**
  - In containment Refueling Water Storage Tank Heat up tests
  - Pressurizer Surge Line Stratification Tests
  - Reactor Vessel Internals Vibration Tests
  - Natural Circulation Tests

(this is by no means a complete list)

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NRC Staff Review (cont'd)

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- The staff determined that the applicant adequately addressed the methods and guidance in SRP Section 14.2, Revision 2, and all the applicable RGs (e.g., RG 1.68) referenced in SRP Section 14.2 in developing the AP1000 initial test program. **"The AP1000 initial test program will demonstrate, with reasonable assurance, that the SSCs important to safety will adequately perform their intended function"**

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SRP Section 14.3

- This section of the SRP is needed to provide review guidance to the staff for review of all Tier 1 information including ITAAC (ITAAC is but one part of Tier 1 information)
- Quoting from the AP 1000 SER: To be certified, the Tier 1 information must verify the complete scope of the AP1000 design and that the regulations applicable to the AP1000 scope of design are met.
- The applicant provides a Tier 1 entry (subsection) for every system in its design. The amount of information in a given subsection is proportional to the safety significance of the particular system. The ITAAC portion of the Tier 1 information is used to verify that the as-built facility conforms to the applicable regulations.
- The staff reviews this information in detail as shown in the AP 1000 SER.
- Several areas that were investigated by the staff with Westinghouse for the AP 1000 are good examples for this course that demonstrate the use of this SRP to support this review.

SRP 14.3 (cont'd)

- These are several issues taken from the AP 1000 Tier 1 review by the NRC staff:
  - Containment wall thickness
  - CVCS instruments and controls
  - CRD penetration access and inspectability
  - Accessibility of components for inspection
  - Cable pulling procedures, standards, and ITAAC

Additional References

SRP 14.2	**IMC-2504**
SRP 14.2.1	**SECY-01-0124, GE Topical Reports, NRR LIC-100 / 101 / 500, WCAP Report, IM Part 9900, Info Notice 2002-26**
SRP 14.3	**IMC-2503**
SRP 14.3.2	**IMC-2503**
SRP 14.3.3	**IMC-2503, SECY-92-196**
SRP 14.3.4	**IMC-2503, SECY-90-016, SECY-92-053, SECY-92-137, SECY-93-087, SECY-95-132, SECY-97-044, SECY-02-059**
SRP 14.3.5	**IMC-2503**
SRP 14.3.6	**IMC-2503, SECY-00-77**
SRP 14.3.7	**IMC-2503**
SRP 14.3.8	**IMC-2503**
SRP 14.3.9	**IMC-2503, SECY-93-087, SECY-92-196**
SRP 14.3.10	**IMC-2503, SECY-91-041, SECY-05-197, SECY-06-098, NRR RS-002, NRR LIC-200**
SRP 14.3.11	**IMC-2503, SECY-93-087**
SRP 14.3.12	**IMC-2503, IP-65001.17, RIS-2008-05, NEI Ltr 2009, NRC Ltr 2009**

Additional References

- **IMC-2503, *Construction Inspection Program – ITAAC Inspections***
- **IMC-2504, *Construction Inspection Program – Non-ITAAC Inspections***
- **IM Part 9900, *10 CFR Part 50.59 Changes, Tests, and Experiments***
- **Info Notice 2002-26, *Failure of Steam Dryer Cover Plate After a Recent Power Uprate***
- **IP-65001.17, *Inspection of ITAAC Related Security SSCs***
- **NRR LIC-100, *Control of Licensing Basis for Operating Reactors***
- **NRR LIC-101, *License Amendment Review Procedures***
- **NRR LIC-200, *Standard Review Plan (SRP) Process***
- **NRR LIC-500, *Processing Requests for Reviews of Topical Reports***
- **NRR RS-002, *Processing Applications for Early site Permits***
- **RIS-2008-05, *Lessons Learned to Improve ITAAC Submittal***
- **SECY-90-016, *Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements***
- **SECY-91-041, *Early Site Permit Readiness Review***
- **SECY-92-053, *Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews***

Additional References

- **SECY-92-137,** ***Reviews of ITAAC for GE ABWR***
- **SECY-92-196,** ***Development of Design Acceptance Criteria (DAC) for ABWR***
- **SECY-93-087,** ***Policy, Technical, and Licensing Issues Pertaining to Evolutionary and ALWR Designs***
- **SECY-95-132,** ***Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems (RTNSS) in Passive Plant Designs***
- **SECY-97-044,** ***Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design***
- **SECY-00-77,** ***Certification of Two Evolutionary Designs***
- **SECY-01-124,** ***Power Uprate Application Reviews***
- **SECY-02-059,** ***Use of Design Acceptance Criteria for the AP1000 Standard Design***
- **SECY-05-197,** ***Review of Operational Programs in a Combined License Application and Generic Emergency Planning ITAAC***

Learning Questions

- **What is considered an "Initial Test Program" in SRP Chapter 14?**
- **How this ITP relates to the Design Certification ITAACs?**
- **Identify where Codes & Standards are found in an ITAAC.**

Module 11
Standard Review Plan
Chapter 17

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Module 11
Standard Review Plan Chapter 17

Instructor: C. Wesley Rowley, PE

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Learning Objectives

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- **Conduct an overview of SRP Chapter 17**
- **Discuss Safety Management and the role of the Quality Assurance Program and Processes**
- **Discuss the review criteria in SRP Chapter 17 and 10 CFR 50 Appendix B and how to relate to the design, construction, startup, and operations phases of the NPP project.**
- **Learn the role of codes and standards in Ch 17 of the SRP.**

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Significant Sub-topics

- Principles of Safety Management
- Program Scope
- Safety Management
- Quality Assurance Program
- Quality Assurance Processes
- Other quality assurance considerations

Safety Management Principles

- **Organizational commitment to safety is a cornerstone of all activities**
- **All levels of the organization operate in a learning environment and conduct their work with a questioning attitude**
- **Management demonstrates a commitment and responsibility**
- **Applicants for a license must submit a QA Program Description that fulfills these principles and meets the regulations**

Safety Management Scope

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- The scope of Safety Management includes all activities that may affect quality in all safety-related activities

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Safety Management Implementation

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- Effective organization and management team
- Foster a Safety Conscious Work Environment
- Training and Qualification of personnel

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NUREG-0800

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

17.5 QUALITY ASSURANCE PROGRAM DESCRIPTION - DESIGN CERTIFICATION, EARLY SITE PERMIT AND NEW LICENSE APPLICANTS

REVIEW RESPONSIBILITIES

Primary - The organization responsible for quality assurance (QA)

Secondary - None

I. AREAS OF REVIEW

The QA staff reviews and evaluates QA program descriptions (QAPDs) submitted by applicants for a design certification (DC), combined license (COL), early site permit (ESP), construction permit (CP), and operating license (OL). QAPDs submitted by applicants for DC, COL, ESP CP, and OL are reviewed and evaluated in accordance with the applicable sections of this standard review plan (SRP).

A QAPD submitted by a DC applicant may be a QA topical report or part of a safety analysis report (SAR). A QAPD submitted by a DC applicant would only address design QA activities in support of a DC. The QAPD would not address construction and design QA activities that occur once construction begins. The QAPD submitted by the DC applicant would be reviewed and evaluated by the NRC prior to NRC approval of the DC.

March 2007

USNRC STANDARD REVIEW PLAN

This Standard Review Plan, NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC's regulations. The Standard Review Plan is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide an acceptable method of complying with the NRC regulations.

The standard review plan sections are numbered in accordance with corresponding sections in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of Regulatory Guide 1.70 have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) are based on Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

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Standard Review Plan Chapter 17 Quality Assurance CONTENTS

SRP Implementation Guidance

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- **The expectation for use of the SRP by the staff is included:**
  - The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52
  - Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

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SRP Ch 17 Areas of Review

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1. Organization
2. QA Program
3. Design Control
4. Procurement Document Control
5. Instructions, Procedures, and Drawings
6. Document Control
7. Control of Purchased Material, Equipment, and Services
8. Identification and Control of Materials, Parts, and Components
9. Control of Special Processes
10. Inspection
11. Test Control
12. Control of Measuring and Test Equipment
13. Handling, Storage, and Shipping
14. Inspection, Test, and Operating Status
15. Nonconforming Material, Parts, or Components
16. Corrective Action
17. Quality Assurance Records
18. Audits

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Additional Areas in SRP17.5

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- **These additional areas are found for new reactor reviews**
  - Training and Qualification Criteria-Quality Assurance
  - Training and Qualification-Inspection and Test
  - QA Program Commitments
  - Non Safety-Related SSC Quality Controls
  - Independent Review

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SRP Chapter 17

- **In addition to the review areas and procedures, the SRPprovides guidance to NRC staff on:**
  - Review responsibilities
  - Acceptance criteria for review
  - Evaluation guidance
  - Many additional Regulatory Guide references
- **Guidance for a pre-docketing inspection**

Pre-docketing Inspection

- **Prior to docketing a CP application, the NRC performs a review of the applicant's QA program description relative to ongoing design and procurement activities. This review and associated inspection are performed right after tendering (submission to the NRC) of a CP application to determine that a satisfactory QA program has been established and is being implemented.**
- **The pre-docketing review emphasizes the areas of organization, QA program, design control, procurement document control, and audit.**
- **The SRP states that the application is not docketed unless the established and implemented program in these areas has no substantive deviation from NRC QA guidance applicable to activities conducted prior to docketing.**

Safety Evaluation Guidance

- **The QAPD acceptably describes the authority and responsibility of management and supervisory personnel, performance/verification personnel, and audit personnel.**
- **The organizations and persons responsible for performing the verification and audit functions have the authority and independence to conduct their activities without undue influence from those directly responsible for costs and schedules.**
- **The QAPD describes a philosophy and controls that, when properly implemented, comply with the requirements of 10 CFR 50.34(f)(3)(ii) and (iii), Appendix B to 10 CFR Part 50 pursuant to 10 CFR 50.34(b)(6)(ii) and 10 CFR 50.34(h), and GDC 1 of Appendix A to 10 CFR Part 50.**
- **The QA program for items that are important to safety is acceptable.**
- **The program for the QA treatment of nonsafety-related SSCs is acceptable.**
- **For a COL review, the findings include a specific conclusion that the implementation of the operational phase of the QAP complies with 10 CFR 50.54(a)(1) (proposed). In addition, the program and implementation will be identified in Table 13.4X (Operational Programs) of the FSAR.**

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SRP Review Procedure

- **The reviewer will select material from the procedures described in the SRP, as may be appropriate**
- **These review procedures are based on the identified SRP acceptance criteria.**
- **Manual Chapters 2501, 2502, and 2504 specify inspections to be performed to assess the applicant's or holder's interpretation and translation of the QAPD commitments into its procedures, processes, and organizational staffing.**
- **These inspections will focus on the effectiveness of the QAPD implementation.**
- **Through review of the information provided by the applicant or holder and, as required, meetings with the applicant or holder; review of applicable NRC inspection reports; and discussion with involved NRC inspectors, a judgment is made of the applicant's or holder's capability to carry out its QA responsibilities. The reviewer's satisfaction with the QA program commitments, the description of how the commitments will be met, the organizational arrangements, and the capabilities to fulfill the QAPD should lead to the conclusion of acceptability as described in Subsection IV of this document.**
- **For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.**

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The QAPD (Quality Assurance Program Description)

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- **High-level document that sets forth the applicant's over-arching philosophy of safety management, policy and principles.**
- **Establishes the manner that quality is achieved and commits the applicant for a Construction Permit or Operating License to implement the QAPD and a more detailed Quality Assurance Program (QAP), derived from the QAPD, through all phases of the life of the facility.**
- **Quality standards used to design, fabricate, erect, and test structures, systems, and components important to safety must be commensurate with the importance of the safety functions to be performed.**

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Organization

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- **The QAPD needs to contain an organizational description that addresses the organizational structure, functional responsibilities, levels of authority, and interfaces.**
- **The organizational description is to include the onsite and offsite organizational elements that function under the cognizance of the QA Program.**
- **These organizations include the applicant's organization and the principal contractors including the architect-engineer, the nuclear steam system supplier and the constructor, if different than the architect-engineer.**

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Quality Assurance Program

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- **The functional responsibilities encompassed by the QAPD are to include activities that may affect the quality of SCCs**
- **The QAP is to contain a detailed list of all SSCs and activities important to safety that may affect quality. These SSCs and activities are to be under the control of the QAP.**
- **Activities performed under the QAP are to be controlled by documented policies and procedures and instructions developed under the QAP and performed in a suitably controlled environment by individuals and organizations that have received appropriate training**
- **In the event that major portions of the applicant's scope of work that may affect SSCs important to safety are delegated, the applicant is to identify the scope of work delegated, the qualifications of applicant's organization responsible for the work and the organizations to which the work was delegated.**
- **The program should be periodically assessed by an independent source with sufficient technical competence in the assessment area. The source should have sufficient authorityand access to management**
- **Cost and schedule considerations should not unduly influence any decision-making which affects plant safety**

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Design Control

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- **To assure that final design of SSC meets applicable regulatory requirements as well as the licensing basis specified in the application**
- **Verification extent should be commensurate with the importance of the safety function, complexity, and degree of conformance with engineering practice**
- **Measures include design reviews**

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Corrective Action Program

- To correct and prevent recurrence of conditions adverse to quality
- Entails prompt identification of adverse conditions, with subsequent cause determination, documentation, classification, and corrective action
- Analyses should be performed to identify trends or repetitive recurrences which may indicate a widespread underlying problem

Procurement Document Control Program

- To ensure that procurement documents have been verified, are consistent with regulatory requirements, and the licensing basis and are clear and concise
- Documents should specify vendor reporting requirements, documentation to be provided with supplied items, and other special instructions
- Documents should also provide for access to vendor facilities for surveillance, audits, and inspections

Other Important Considerations

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- **Safety Culture**
- **Safety-Conscious work environment**
  - Excellent interactions and communications between entities within the operating organization as well as between the operating organization and external organizations
  - Safety should never be compromised to meet cost, schedule, or power demands
  - An atmosphere of "openness" should be created where personnel are encouraged to bring forth safety concerns without fear of reprisal

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NUREG 1055

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- **This NRC reference tells the story and lessons learned from failed QA programs**
  - The Quality Assurance Program must be implemented as an integral part of a comprehensive management system
  - The Quality Assurance Program must support prompt detection, communication and correction of quality problems in design, effective management system oversight of the design process, and the ability to control plant configuration and manage change

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Applicable Standards

- American Society for Mechanical Engineers, NQA-1-2008, "Quality Assurance Requirements for Nuclear Facility Applications"
- International Atomic Energy Agency, Safety Requirements, No. GS-R-3, "The Management System for Facilities and Activities"
- International Atomic Energy Agency, Safety Guide, No. GS-G-3.1, "Application of the Management System for Facilities and Activities"

Learning Questions

- **For the Corrective Action portion of QA Program Plan:**
  - What are some examples of corrective action for "design" ?
  - What are some examples of corrective action for "construction" ?
  - What are some examples of corrective action for "startup" ?
  - What are some examples of corrective action for "operation" ?

Module 12

ASME BPV Code Section III General Requirements for Division 1 and 2

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Module 12

ASME BPV Code Section III General Requirements for Division 1 and 2

Instructor: Gene Imbro, P.E.

Learning Objectives

- Learn the meaning of the term "Construction"as used in Section III
- Learn the scope and organization of Section III
- Learn when an N-Stamp component is required

ASME Section III – Early History

- **With the construction of commercial nuclear power plants it was recognized that "high" standards for passive component construction needed to be used so that they could operate for their life without attention.**
- **ASME Section III was first published in 1963 originally containing 134 pages of text and covering only 3 classes of nuclear vessels.**
- **Section III was developed by ASME from the Naval Reactors Program, "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components".**
- **"Construction" as used in Section III, Division 1 encompasses materials, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of an item.**

Major Sub-Topics

- Scope of Section III
- Organization of Section III
- Introduction
- Definitions
- Code Editions Addenda and Cases
- Design Specifications
- Code Classification
- Operating and Test Conditions
- Certification and Types of Certificates
- Quality Assurance
- Authorized Inspection
- Stamping and Data Reports

Scope of Section III

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- **There are three published Divisions of Section III which are:**
  - Division 1 - Metallic vessels, heat exchangers, storage tanks, piping systems, pumps, valves, core support structures, and supports
  - Division 2 – Concrete containment vessels.
  - Division 3 –Metallic containment systems for storage or transportation of spent nuclear fuel and high level nuclear waste.
- **An additional two divisions (as yet not approved for publication) have been created to address advanced reactors. They are:**
  - Division 4 – Fusion reactors
  - Division 5 – High Temperature Gas Cooled Reactors

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Organization of Section III

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Subsection NCA – General Requirements for Division 1 and 2

- **Division 1 Subsections:**
  - NB – Class1 Components
  - NC – Class 2 Components
  - ND – Class 3 Components
  - NE - Class MC Components
  - NF – Supports
  - NG – Core Support Structures
  - NH – Class 1 Components in elevated Temperature Service
  - Appendices
- **Division 2 Subsections:**
  - CC – Concrete Containments and Division 2 Appendices
- **Division 3 Subsections**
  - WA – General Requirements for Division 3
  - WB – Class TC Transportation
  - WC –Class SC Storage Containments

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Introduction

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- **The rules of this Section provide requirements for new construction and include consideration of mechanical and thermal stresses due to cyclic operation. An important limitation is that the rules do not cover deterioration that may occur in service as a result of radiation effects, corrosion, erosion, or instability of the material. While it is stated that these effects shall be taken into account, no guidance is provided.**

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Definitions

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- **The following definitions are required for a basic understanding of the Code process:**
  - Code Editions: documents issued at 3-year intervals by the ASME that include all revisions and additions previously included in Addenda and corrections included in Errata published since the previous edition.
  - Code Addenda: additions and revisions to individual sections of the Code published annually.
  - Code Cases: documents issued by ASME to clarify the intent of existing Code requirements or to provide alternative rules for construction.
  - Design Specification: a document prepared by the Owner or his designee that provides a complete basis for construction in accordance with this Section.
  - Code Class: the classification specified by the Owner (or his designee) and included in the Design Specification that establishes the rules for design and construction of items
  - Interpretations: Replies to technical inquiries published as part of the update to Section III

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Code Editions, Addenda and Cases

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- **The Owner shall establish the Code Edition and Addenda to be included in the Design Specifications.**
- **In no case shall the Code Edition and Addenda dates be earlier than:**
  - Three years prior to the date that a Construction Permit (CP) application is docketed
  - The latest edition and addenda endorsed by the regulatory authority having jurisdiction at the plant site at the time the CP application is docketed, or
  - The edition and addenda endorsed for a design certified or licensed by the regulatory authority
- **Code Cases are permissible and may be used beginning with the date of approval**
- **by the ASME Council and the American Concrete Institute for Division 2 design and construction.**

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Design Specifications

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- **It is the Owner's responsibility to provide Design Specifications for components , supports and appurtenances. They must contain sufficient detail to provide a complete basis for construction or design and must include the following:**
  - The functions and boundaries of the items covered
  - The design requirements
  - The environmental conditions including radiation
  - Material requirements including impact test requirements
  - Additional fracture mechanics data sufficient to use Appendix G
  - When operability of a component is a requirement, reference shall be made to other documents that specify the operating requirements
  - The effective Code Edition, Addenda, and Code Cases to be used for construction.

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Code Classification

- Construction rules are specified for items designated as Code Classes 1, 2, 3, CS, MC, and CC. These Code classes are applied to items of a nuclear power system and containment system. Within these systems the Code recognizes the different levels of importance associated with the function of each item. The Code classes allow a choice of rules that provide assurance of structural integrity and quality appropriate to the importance assigned to the individual items of a nuclear power plant.

- The regulatory guidance provided for classification is in NRC Regulatory Guide 1.26 entitled, "Quality Group Classifications and Standards for Water, Steam and Radioactive Waste Containing Components of Nuclear Power Plants"

Operating and Test Conditions

- Components and supports of a nuclear power system may be subjected to plant and system operating and test conditions required to be considered in the design and overpressure protection of the components and the design of supports in order to satisfy systems safety criteria.
- The definition of plant and system operating and test conditions and their significance to the design and operability of components and supports are beyond the scope of Section III.
- Appropriate guidance for the selection of these conditions may be found in the requirements of regulatory and enforcement authorities having jurisdiction at the site.

Certification and Types of Certificates

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- **ASME issues numerous certificates (too numerous to be listed here). Table NCA-8100-1 lists the types of certificates issued by ASME and the responsibilities of each Certificate Holder.**
- **The Certificate will identify the shop or field facility covered and state the scope of activities for which authorization is granted. ASME may limit or extend the scope of an authorization to any types or classes of items.**
- **Some important certificates are:**
  - Certificate of Authorization – issued to an organization for use of a Code Symbol Stamp, certifying a Data Report Form or performing welding or certifying joining
  - Quality Assurance Program Certificate may be issued to an organization who has documented a QA Program and whose ability to staff, equip, or otherwise implement the described QA Program has been evaluated and accepted by ASME.

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Quality Assurance

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- **ASME NQA-1, "*Quality Assurance Requirements for Nuclear Facility Applications*" Part 1, is the key document for compliance by N- Type Certificate holders. An N Certificate holder is the organization responsible for Code compliance with respect to material, design, fabrication, installation, examination, testing, inspection, certification, and stamping of items requiring an N Symbol Stamp.**

- **The various Requirements of NQA-1 that apply to QA organizations, QA systems, design control, procurement document control, processes, inspections, corrective actions, records, audits, etc. are as specified in NCA-4000 .**

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Authorized Inspection

- Inspection of items constructed to Section III is required by the Authorized Inspection Agency
- An Authorized Inspection Agency is one designated by, or is acceptable to a U.S. state or Canadian Province Accreditation shall be In accordance with ASME QAI-1, "Qualification for Authorized Inspection.
- The Authorized Inspection Agency shall notify the ASME when it enters into an agreement with an Owner or a Certificate Holder, or if an existing agreement is terminated. Similarly, the enforcement authority must also be notified when an agreement is written or terminated.
- The Authorized Inspection Agency shall employ Authorized Nuclear Inspectors qualified in accordance with ASME QAI-1 in order to perform the required inspections.

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Stamping and Data Reports

- The Code Symbol Stamp shall be applied by the Certificate Holder only with the authorization of the Inspector. In any case, the Code Symbol Stamp shall not be applied until completion of the required examination and testing. The completed Code Data Report Form indicates that the Inspector has inspected the item and authorized application of the Code Symbol Stamp.
- The sequence for stamping and completion of the Code Data Report shall be determined by agreement between the Authorized Nuclear Inspector and the Certificate Holder.
- The N Symbol Stamp shall only be applied after the pressure test requirements have been met and all other examinations, tests, and inspections have been satisfactorily completed.
- Other types of data reports and Code Symbol Stamps are covered in NCA-8000.

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Module 13

ASME BPV Code Section III Subsection NB – Class 1 Components

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Module 13

ASME BPV Code Section III Subsection NB – Class 1 Components

Instructor: Gene Imbro, P.E.

Learning Objectives

- Learn what "design by analysis" means and how it is used in Subsection NB
- Learn what nuclear facility components are typically classified as Class 1 and the bases for their classification
- Learn what requirements are covered by the rules of Subsection NB and what limitations apply

Major Sub-Topics

- Introduction
- Objective and Scope
- **Each Section III Subsection is divided into Articles. The Subsection NB Articles required for Class 1 Construction are:**
  - Article NB-2000 Materials
  - Article NB-3000 Design
  - Article NB- 4000 Fabrication and Installation
  - Article NB-5000 Examination
  - Article NB-6000 Testing
  - Article NB-7000 Overpressure Protection
  - Article NB-8000 Nameplates, Stamping and Reports

Introduction

- **Subsection NB Contains rules for the material, design, fabrication, testing, overpressure relief, marking, stamping, and preparation of reports by the Certificate Holder of items intended to conform to the requirements for Class 1 Construction.**
- **Class 1 construction typically has been applied to items within the Reactor Coolant Pressure Boundary of LWRs (PWRs and BWRs) and some items of PHWRs. Examples of such items are: Reactor Pressure Vessels, Pressurizers, Reactor Coolant: Piping, Pumps and Valves, the primary side of Steam Generators, etc.**

Objective and Scope

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- **The purpose of Subsection NB is to provide rules for Class 1 construction**
- **The rules of NB cover the requirements for strength and pressure integrity of items, the failure of which would violate the pressure retaining boundary. The rules cover initial construction, but a significant limitation is that they do not cover deterioration which may occur in service due to radiation effects, corrosion, or instability of material.**

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NB – 2000 Materials

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- **Article NB -2000 is comprised of the following Subarticles:**
  - NB-2100 General Requirements
  - NB-2200 Material Test Coupons and Specimens for Ferritic Steel
  - NB-2300 Fracture Toughness requirements
  - NB-2400 Welding Material
  - NB-2500 Examination and Repair of Pressure Retaining Material
  - NB-2600Material Organizations' Quality System Programs
  - NB-2700 Dimensional Standards

The slides which follow provide more detail for certain requirements of NB-2000

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NB – 2000 Materials

- **NB 2100 General Requirements for Material**
  - Pressure retaining material applies to items such as vessel shells, heads, and nozzles; pipes, tubes and fittings; valve bodies, bonnets and disks; pump casing and bolting which joins pressure retaining items.
  - Pressure-retaining material shall conform to one of the specifications for material given in Section II Materials, Part D, Subpart 1, Tables 2A and 2B and to all the applicable requirements of this Article.
  - Material made to specifications other than those specified in Section II, Part D, Subpart 1, Tables 2A and 2B may be used for the following applications:
    - safety valve disks
    - Control valve disks
    - Line valve disks for inlet connections ≤ NPS 2
  - Carbon steels, low alloy steels and high alloy chromium (Series 4XX) may be heat treated by quenching and tempering to enhance their impact properties. Postweld heat treatment at a temperature ≥ 1100 F is considered the tempering phase of the heat treatment.

NB – 2000 Materials

- **NB-2300 Fracture Toughness Requirements for Material**
  - Pressure-retaining material and material welded thereto must be impact tested except that the material below is not to be impact tested:
    - material with a nominal section thickness of 5/8 in. and less.
    - austenitic stainless steels and non ferrous material
    - some further exceptions are also listed in NB-2311
  - Impact test requirements and acceptance standards are as specified in other paragraphs under NB-2300

NB – 3000 Design

- **This Article contains more pages (101) than any other article in Subsection NB. It contains the following Subarticles:**
  - NB-3100 General Design
  - NB-3200 Design by Analysis
  - NB-3300 Vessel Design
  - NB-3400 Pump Design
  - NB-3500 Valve Design
  - NB-3600 Piping Design
- **In the slides that follow emphasis will be placed on some basic concepts of design by analysis**

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Design By Analysis

- **Terms Relating to Design By Analysis:**
  - Stress Intensity is twice the maximum shear stress which is the difference between the algebraically largest principal stress and the algebraically smallest principal stress at a point.
  - Primary stress is a normal or shear stress developed by an imposed loading necessary to satisfy the laws of equilibrium of external forces and moments.
  - Secondary stress is a normal or shear stress developed by constraint of adjacent material or self-constraint of the structure.
  - Local primary membrane stress is a membrane stress produced by pressure or other mechanical loading and associated with a discontinuity would if not limited produce excessive distortion in load transfer to other parts of the structure.
  - Peak stress is an increment of stress additive to the primary plus secondary stresses due to local discontinuities, local thermal stresses or stress concentrations. This stress is a possible source of fatigue

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Design by Analysis

- **Some additional information relating to design by analysis:**
  - Design stress intensity values (S_m) for materials are listed in Section II. Part D, Subpart 1, Tables 2A, 2B, and 4. The material cannot be used at metal and design temperatures that exceed the temperature limit in the applicability column for which stress intensity values are listed.
  - Each service condition that components are subjected to must be classified in accordance with NCA-2142 and Service Limits [NCA-2142.4(b)] designated in the Design Specifications in such detail as to provide a complete basis for design, construction and inspection in accordance with this Article. Also applicable:
  - Level B Conditions –the estimated duration shall be included in the Design Specification.
  - Level C Conditions-the total number stress cycles must be < 25 having an S_a value > than 10^6 cycles from the applicable fatigue design curve.

Design by Analysis

- **Requirements for acceptability of a design by analysis:**
  - The design shall be such that stress intensities will not exceed the limits in this Subarticle and in NB-3100 and tabulated in Section II, Part D, Subpart 1, Tables 2A, 2B, and 4. S_m denotes the design stress intensity values listed in those tables.
  - The design details must conform to the rules of NB-3100 (General Design) and
    those given in the Subarticle for the specific component e.g., NB-3400 Pump Design.
  - For configurations where compressive stresses occur, the critical buckling stress shall be accounted for.
  - Protection against nonductile ("brittle") fracture, shall be provided by performing an evaluation of service and test conditions by methods similar to Appendix G.
    Options to the use of Appendix G are specified in NB-3211 (d) (2) and (3).

Design by Analysis

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TABLE NB-3217-1
CLASSIFICATION OF STRESS INTENSITY IN VESSELS FOR SOME TYPICAL CASES (CONT'D)

Vessel Part	Location	Origin of Stress	Type of Stress	Classification
Nozzle (NB-3227.5)	Outside the limits of reinforcement defined by NB-3334	Pressure and external axial, shear, and torsional loads other than those attributable to restrained free end displacements of attached piping	General membrane stresses	P_m
		Pressure and external loads and moments other than those attributable to restrained free end displacements of attached piping	Membrane Bending	P_L P_b
		Pressure and all external loads and moments	Membrane Bending Peak	P_L Q F
	Nozzle wall	Gross structural discontinuities	Local membrane Bending Peak	P_L Q F
		Differential expansion	Membrane Bending Peak	Q Q F
Cladding	Any	Differential expansion	Membrane Bending	F F
Any	Any	Radial temperature distribution [Note (3)]	Equivalent linear stress [Note (4)]	Q
			Nonlinear portion of stress distribution	F
Any	Any	Any	Stress concentration (notch effect)	F

GENERAL NOTE: Q and F classification of stresses refers to other than design condition (Fig. NB-3222-1).

NOTES:
(1) If the bending moment at the edge is required to maintain the bending stress in the middle to acceptable limits, the edge bending is classified as P_b. Otherwise, it is classified as Q.
(2) Consideration shall also be given to the possibility of wrinkling and excessive deformation in vessels with a large diameter–thickness ratio.
(3) Consider possibility of thermal stress ratchet.
(4) Equivalent linear stress if defined as the linear stress distribution which has the same net bending moment as the actual stress distribution.

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Design by Analysis

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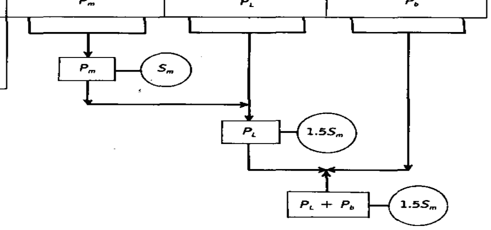
FIG. NB-3221-1 STRESS CATEGORIES AND LIMITS OF STRESS INTENSITY FOR DESIGN CONDITIONS

Stress Category	Primary		
	General Membrane	Local Membrane	Bending
Description (for examples see Table NB-3217-1)	Average primary stress across solid section. Excludes discontinuities and concentrations. Produced only by mechanical loads.	Average stress across any solid section. Considers discontinuities but not concentrations. Produced only by mechanical loads.	Component of primary stress proportional to distance from centroid of solid section. Excludes discontinuities and concentrations. Produced only by mechanical loads. [Note (1)]
Symbol [Note (2)]	P_m	P_L	P_b

Combination of stress components and allowable limits of stress intensities.



Legend
—— Use Design Loads

NOTES:
(1) Bending component of primary stress for piping shall be the stress proportional to the distance from centroid of pipe cross section.
(2) The symbols P_m, P_L, and P_b do not represent single quantities, but rather sets of six quantities representing the six stress components σ_t, σ_l, σ_r, τ_{lt}, τ_{lr}, and τ_{rt}.

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Design by Analysis

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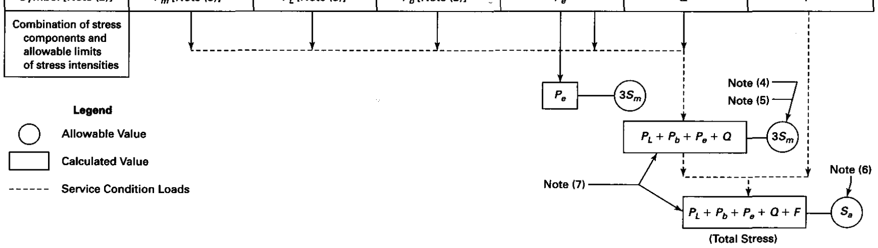
FIG. NB-3222-1 STRESS CATEGORIES AND LIMITS OF STRESS INTENSITY FOR LEVEL A AND LEVEL B SERVICE LIMITS

	Primary			Secondary		
Stress Category	General Membrane	Local Membrane	Bending	Expansion	Membrane plus Bending	Peak
Description (for examples see Table NB-3217-1)	Average primary stress across solid section. Excludes effects of discontinuities and concentrations. Produced by pressure and mechanical loads.	Average stress across any solid section. Considers effects of discontinuities but not concentrations. Produced by pressure and mechanical loads, including inertia earthquake effects.	Component of primary stress proportional to distance from centroid of solid section. Excludes effects of discontinuities and concentrations. Produced by pressure and mechanical loads, including inertia earthquake effects. [Note (1)]	Stresses which result from the constraint of free end displacement. Considers effects of discontinuities but not local stress concentration (not applicable to vessels).	Self-equilibrating stress necessary to satisfy continuity of structure. Occurs at structural discontinuities. Can be caused by pressure, mechanical loads, or differential thermal expansion. Excludes local stress concentrations.	(1) Increment added to primary or secondary stress by a concentration (notch). (2) Certain thermal stresses which may cause fatigue but not distortion.
Symbol [Note (2)]	P_m [Note (3)]	P_L [Note (3)]	P_b [Note (3)]	P_e	Q	F



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Design by Analysis

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FIG. NB-3222-1 STRESS CATEGORIES AND LIMITS OF STRESS INTENSITY FOR LEVEL A AND LEVEL B SERVICE LIMITS (CONT'D)

NOTES:

(1) Bending component of primary stress due to mechanical loads for piping shall be the stress proportional to the distance from centroid of pipe cross-section. For piping, the calculation of P_b stresses is not required for reversing dynamic loads (including inertia earthquake effects). See NB-3223(b)(2).

(2) The symbols P_m, P_L, P_B, P_e, Q, and F do not represent single quantities, but sets of six quantities representing the six stress components σ_t, σ_l, σ_r, τ_{lt}, τ_{lr}, and τ_{rt}.

(3) For Level B Service Limits for primary stress intensities generated by Level B Service Loadings, see NB-3223(a)(1).

(4) When the secondary stress is due to a temperature transient at the point at which the stresses are being analyzed or to restraint of free end deflection, the value of S_m shall be taken as the average of the tabulated S_m values for the highest and lowest temperatures of the metal during the transient. When part or all of the secondary stress is due to mechanical load, the value of S_m shall not exceed the value for the highest temperature during the transient.

(5) Special rules for exceeding $3S_m$ are provided in NB-3228.5.

(6) S_a is obtained from the fatigue curves, Fig. I-9.0. The allowable stress intensity for the full range of fluctuation is $2S_a$.

(7) The stresses in category Q are those parts of the total stress that are produced by thermal gradients, structural discontinuities, etc., and they do not include primary stresses that may also exist at the same point. However, it should be noted that a detailed stress analysis frequently gives the combination of primary and secondary stresses directly and, when appropriate, this calculated value represents the total of $P_m + P_b + Q$, and not Q alone. Similarly, if the stress in category F is produced by a stress concentration, the quantity F is the additional stress produced by the notch over and above the nominal stress. For example, if a point has a nominal stress intensity P_m and has a notch with a stress concentration factor K, then $P_m \le S_m$, $P_b = Q$, $Q = 0$, $F = P_m(K-1)$, and the peak stress intensity equals $P_m + P_m(K-1) = KP_m$. However, P_L is the total membrane stress that results from mechanical loads, including discontiuity effects, rather than a stress increment. Therefore, the P_L value always includes the P_m contribution.

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NB-4000 Fabrication and Installation

- **Article NB-4000 is comprised of the following Subarticles:**
  - NB- 4100 General Requirements
  - NB- 4200 Forming, Fitting, and Aligning
  - NB-4300 Welding Qualifications
  - NB-4400 Making, Examining and Repairing Welds
  - NB-4500 Brazing
  - NB-4600 Heat Treatment
  - NB-4700 Mechanical Joints
    - The slides which follow provide more detail for certain requirements of NB-4000

NB-4000 Fabrication and Installation

- **NB-4100 General Requirements**
  - The Certificate Holder shall certify (by application of the Code Symbol and completion of the Data Report) that the materials used comply with NB-2000 and that the fabrication/installation complies with this Article.
  - Material for pressure –retaining parts shall carry identification markings which must remain visible until the component is assembled or installed.
  - Material in which defects exceeding the limits of NB-2500, "EXAMINATION AND REPAIR OF PRESSURE-RETAINING MATERIAL", are known or discovered during fabrication or installation is unacceptable. However, the material may be used provided the condition is corrected in accordance with NB-2500 except that the depth of weld repair does not apply.

NB-4000 Fabrication and Installation

- **NB-4335 Impact Test Requirements**
  - Impact tests of the weld metal and heat affected zone (HAZ) is required for welding procedure qualification tests for production weld joints exceeding 5/8in. In thickness when the weld will be made on the surface or penetrate base metal that requires impact testing required by NB-2310. In addition, such testing of the weld metal is required for any weld repair to base material the requires impact testing in accordance with NB-2310.
  - Charpy V-notch tests of the heat HAZ of the welding procedure qualification test assembly are required whenever the thickness exceeds 5/8 in. and either of the base materials require impact testing in accordance with NB-2310

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NB-5000 Examination

- **Article NB-5000 is comprised of the following Subarticles:**
  - NB-5100 General Requirements for Examination
  - NB-5200 Required Examination of welds for Fabrication and Preservice Baseline
  - NB-5300 Acceptance Standards
  - NB-5400 Final Examination of Vessels
  - NB 5500Qualifications and Certification of NONdestructive Examination Personnel
    - The slide which follows provides more detail for certain requirements of NB-5000

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NB-5000 Examination

- **NB-5100 General Requirements for Examination**
  - Nondestructive examinations must be conducted in accordance with Section V except as modified by this Article
  - Fabrication -radiographic examination shall be performed in accordance with Section V, Article 2 .
  - Preservice- ultrasonic examinations required for weld preservice examination shall be in accordance with Section XI, Appendix I; eddy current examination shall be in accordance with Section V, Article 8 and surface examinations shall be in accordance with NB-5111(a).

NB-6000 Testing

- **Article NB-6000 is comprised of the following Subarticles:**
  - NB-6100 General Requirements
  - NB-6200 Hydrostatic Tests
  - NB-6300 Pneumatic Tests
  - NB-6400 Pressure Test Gages
  - NB-6600 Special Test Pressure Requirements
    - The slide which follows provides more detail for certain requirements of NB-6000

NB-6000 Testing

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- **NB-6100 General Requirements**
  - All pressure-retaining components, appurtenances, and completed systems must be pressure tested. The preferred method shall be a hydrostatic test using water as the test medium or an alternative liquid as permitted by the Design Specification.
  - The test should be made at a temperature to minimize the possibility of brittle fracture (Appendix G).
  - An installed system shall be hydrostatically at ≥ 1.25 times the lowest design pressure of any component within the overpressure protection boundary.
  - Maximum test pressure is limited by the NB-3226 stress limits.
  - The hydrostatic test pressure must be maintained a minimum of 10 min. prior to examination for leakage required by NB-6224.

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NB-7000 Overpressure Protection

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- **Article NB- 7000 is comprised of the following Subarticles:**
  - NB-7100 General Requirements
  - NB-7200 Overpressure Protection Report
  - NB-7300 Relieving Capacity
  - NB-7400 Set Pressures of Pressure Relief Devices
  - NB-7500 Operating and Design Requirements for Pressure Relief Valves
  - NB-7600 Nonreclosing Pressure Relief Devices
  - NB-7700 Certification
  - NB-7800 Marking, Stamping and Data Reports

The slides which follow provide more details for certain requirements of NB-7000

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NB-7000 Overpressure Protection

- **NB-7100 General Requirements**
  - A system must be protected from application of conditions of pressure and coincident temperature that would cause either the Design Pressure or Service Limits specified in the Design Specification to be exceeded.
  - Pressure relief devices are required when the operating conditions considered in the Overpressure Protection Report would cause the Service Limits in the Design Specification to be exceeded. Pressure relief devices include a pressure relief valve or a nonreclosing device such as a rupture disk.
  - For reclosing pressure relief devices, remote monitoring of valve position (fully open and fully closed) shall be provided.

NB-7000 Overpressure Protection

- **NB-7300 Relieving Capacity**
  - The total relieving capacity of the pressure relief devices shall prevent a rise in pressure of more than 10% above the Design Pressure of any component within the pressure-retaining boundary of the protected system under any expected system pressure transient conditions as summarized in the Overpressure Protection Report.
- **NB-7400 Set Pressures of Pressure Relief Devices**
  - The stamped set pressure of at least one of the pressure relief devices connected to the system shall not be greater than the Design Pressure of any component within the pressure-retaining boundary of the protected system. Additional relief devices may have higher stamped set pressures but those set pressures cannot result in total system pressures that exceed the system limitations of NB-7300.

NB-8000 Nameplates, Stamping and Reports

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- **NB-8100 General Requirements**
  - The requirements for nameplates, stamping and reports are as given in NCA-8000

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 14

ASME BPV Code Section III Subsections NC, ND, NE, NF, NG, NH and Appendices

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Module 14
ASME BPV Code Section III Subsections NC,ND,NE,NF,NG,NH and Appendices

Instructor: Gene Imbro

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Learning Objectives

- Learn the scope of the following: Subsections NC,ND,NE,NF,NG,NH and Appendices including what components and conditions that are covered
- Learn the purpose and content of the Appendices and the difference between "Mandatory" and "Nonmandatory"

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2 2

Major Sub-Topics

- Introduction
- Subsection NC, Class 2 Components
- Subsection ND, Class 3 Components
- Subsection NE, Class MC Components
- Subsection NF, Supports
- Subsection NG, Core Support Structures
- Subsection NH, Class1 Components in Elevated Temperature Service
- Appendices

Introduction

- This is the third (and final) of three training modules covering Section III. It covers the parts of Section III not covered by the previous modules, Subsections NCA and NB. Again, only requirements for new construction are addressed and only for items covered under Division 1 of Section III.

Organization of Section III

- **Subsection NCA –General Requirements for Division 1 and 2 (Module 12)**
- **Division 1 Subsections:**

Module 14

  - NB – Class1 Components (Module 13)
  - NC – Class 2 Components
  - ND – Class 3 Components
  - NE - Class MC Components
  - NF – Supports
  - NG – Core Support Structures
  - NH – Class 1 Components in elevated Temperature
  - Appendices
- Code does not provide guidance in the selection of specific classification of components. Such guidance is derived from systems safety criteria for specific types of nuclear power systems. These are found in engineering standards or requirements of regulatory and enforcement authorities having jurisdiction at the nuclear power plant site

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Subsection NC

- **Subsection NC contains rules for the material, design, fabrication, examination, testing, overpressure relief, marking stamping, and preparation of reports for items conforming to the requirements for Class 2 construction.**
- **These rules cover the strength and pressure integrity of items the failure of which would violate the pressure retaining boundary. The rules cover load stresses, but do not cover deterioration which may occur in service as a result of corrosion, radiation effects, or instability of materials.**

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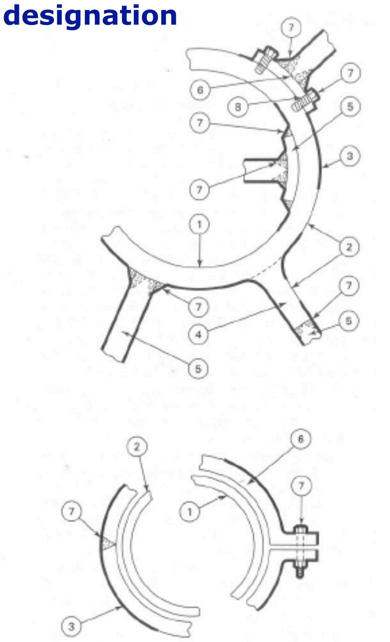
Major Sub-Topics

- Introduction
- Article NC-2000 Materials
- Article NC-3000 Design
- Article NC-4000 Fabrication and Installation
- Article NC-5000 Examination
- Article NC-6000 Testing
- Article NC-7000 Overpressure Protection
- Article NC-8000 Nameplates, Stamping and Reports

Example NC Class 2 pressure Boundary designation



1. Component shall conform to Subsection NC.
2. Pressure retaining portion of the component.
3. Jurisdictional boundary (heavy line).
4. Cast or forged attachment or weld buildup shall conform to Subsection NC.
5. Welded attachment shall conform to Subsection NC.
6. Bearing, clamped, or fastened attachment shall conform to Subsection NC.
7. Attachment connection shall conform to Subsection NC.
8. Drilled holes shall conform to Subsection NC.

Subsection ND

- Subsection ND contains rules for the materials, design, fabrication, examination, testing, overpressure relief, marking, stamping, and preparation of reports by the certificate holder for items conforming to the requirements for Class 3 construction.
- The rules of subsection ND cover the strength and pressure integrity of items the failure of which would violate the pressure retaining boundary. The rules cover load stresses, but do not cover deterioration which may occur in service as a result of corrosion, radiation effects, or instability of materials.

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Subsection ND

- The allowable stresses for design for materials listed in Tables 1A and 1B, section II, Part D, Subpart 1. The material shall not be used at metal and design temperatures that exceed the temperature limit in the applicability column for which stress values are given.
- Stress classification definitions defined in module 13, slide 10 apply

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Subsection ND

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- **The allowable stresses provided for various construction materials for operating temperature conditions (and temperature limits provided)**
  - Table 1A – ferrous materials (Carbon, Cr-Mo steels, MN-steels, silicon steels, nickel steels, etc.)
  - Table 1B – nonferrous materials (Chromium alloys, cobalt alloys, titanium alloys, zirconium alloys)
  - These are in Section II, Part D, Subpart 1

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Subsection NE

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- **Subsection NE contains rules for the material, design, fabrication, examination, inspection, testing, and preparation of reports for metal containment (MC) vessels.**
- **Only containment vessels and their appurtenances shall be classified as class MC. Piping, pumps, and valves which are part of the containment system shall be classified as Class 1 or Class 2.**
- **Subsection NE does not contain rules to cover all details of construction of Class MC containment vessels**

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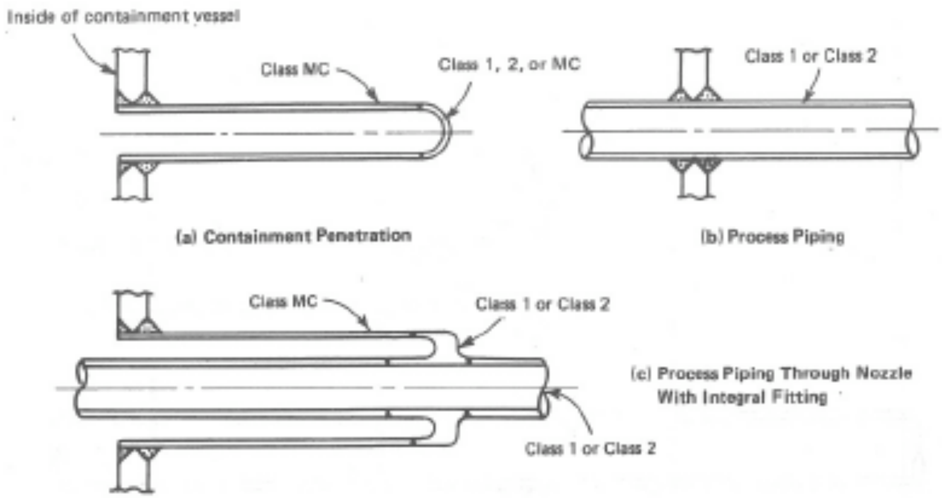
Subsection NE

FIG. NE-1120-1 TYPICAL CONTAINMENT PENETRATIONS



Subsection NF

- Subsection NF contains rules for the material, design, fabrication, examination, inspection, and preparation of certification documents (NS-1 Certificate of Conformance) for supports for components and piping which must conform the requirements for Class 1, 2, 3, and MC construction as set forth in Subsections NB (module 13) and NC, ND, and NE (discussed earlier), respectively.
- Deterioration (from corrosion, radiation, etc.) is not covered.
- NPP supports are those metal elements which transmit loads between components including piping systems, intervening element and building structure. Supports do not include structural elements designed to carry dynamic loads caused by postulated loss of pressure boundary integrity.

Subsection NF

- **Introduction**
- **Article NF-2000 Material**
- **Article NF-3000 Design**
- **Article NF- 4000 Fabrication and Installation**
- **Article NF-5000 Examination**
- **Article NF-8000 Certificates of Authorization and Certification Docs.**

- **For design by analysis Sm values listed in Tables 2A, 2B, and 4 or Section II, Part D, Subpart 1 shall be used (as for NC class 1 components).**

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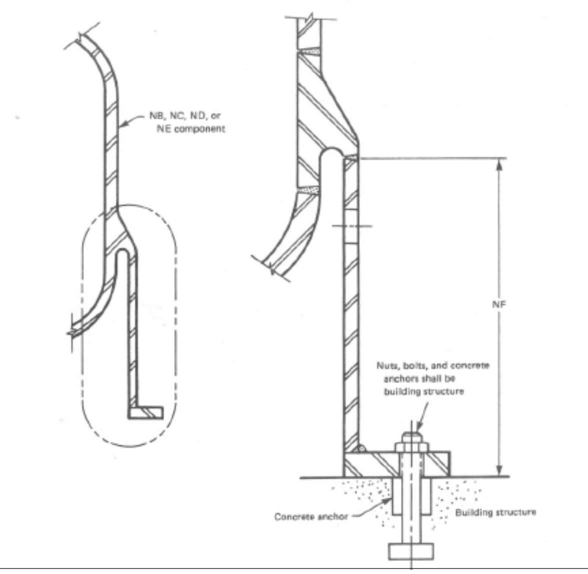
Subsection NF

FIG. NF-1132-2 TYPICAL EXAMPLE OF JURISDICTIONAL BOUNDARY BETWEEN COMPONENT SUPPORT AND THE BUILDING STRUCTURE



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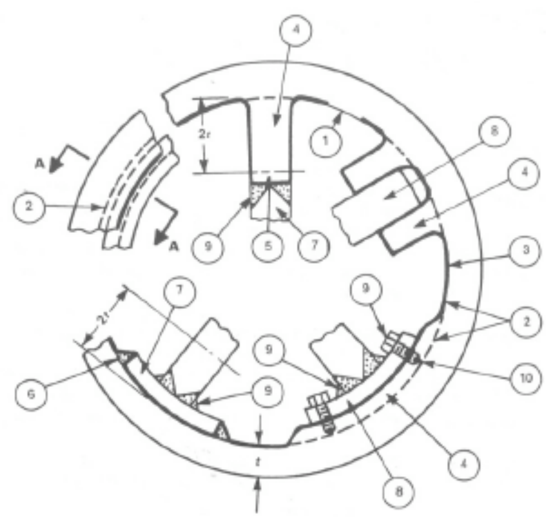
Subsection NG

- **Subsection NG establishes rules for materials, design, fabrication, examination, and reporting required for the manufacture and installation of core support structures**
- **NG Outline similar to NF (slide 15)**
- **Notes:**
  - NG rules are not directed to sealing against coolant leakage
  - The most severe loads usually result from abnormal conditions
  - Design functions are usually handled separately from fabrication functions of the Certificate Holder manufacturing core supports

Subsection NG



(1) Reactor pressure vessel conforms to Subsection NB.

(2) Pressure retaining portion of the reactor pressure vessel.

(3) Jurisdictional boundary (heavy line).

(4) Cast or forged attachment or weld buildup shall conform to Subsection NB.

(5) Beyond $2t$ from the pressure retaining portion of the reactor pressure vessel, the design rules of NC-3000 may be used as a substitute for the design rules of NB-3000.

(6) $2t$ or within $2t$ from the pressure retaining portion of the reactor pressure vessel, the first connecting weld shall conform to Subsection NB.

(7) Beyond $2t$ from the pressure retaining portion of the reactor pressure vessel or beyond the first connecting weld, the attachment shall conform to Subsection NG [see Note (1)].

(8) Attachment connection shall conform to Subsection NG [see Note (1)].

(9) At or within $2t$ from the pressure retaining portion of the component, the interaction effects of the attachment shall be considered in accordance with NB-3135.

(10) Drilled holes within the jurisdictional boundary shall conform to Subsection NB.

Subsection NH

- **Subsection NH contains rules for the materials, design, fabrication, examination, testing, and overpressure relief of Class 1 components that function when metal temperatures exceed those covered by subsection NB and Tables 2A, 2B, and 4 of Section II, Part D, subpart 1.**
- **The rules apply for Class 1 components regardless of the type of contained fluid.**
- **For conditions where creep effects are significant, the design shall consider the time-dependent material properties and structure behavior to guard against these failure modes:**
  - Ductile rupture from short term loadings
  - Creep rupture from long term loadings
  - Creep-fatigue failure
  - Gross distortion caused by creep

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Subsection NH

- **Subsection NH currently has no rule for accounting for crack like defects**
- **Crack like defects must be designed out.**
- **A new joint Section III and Section XI (flaw tolerance) task group has recently been formed which will deal with flaws for high temperature creep. This is mainly in anticipation of Generation IV reactors, currently under design, which will have components that operate in the creep regime.**

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Mandatory Appendices

- Appendix-I Design Fatigue Curves
- Appendix-II Experimental Stress Analysis
- Appendix-III Basis for Establishing Design Stress Values
- Appendix-IV Approval for New Material under BPV code
- Appendix-V Certificate Holders Report Forms, Instructions, etc.
- Appendix-VI Rounded Indications
- Appendix-XI Rules for Bolted Flange Connections for Class 2, 3, MC
- Appendix-XII Design Considerations for Bolted Connections
- Appendix-XIII Design Based on Stress Analysis
- Appendix-XIV Design Based on Fatigue Analysis

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Mandatory Appendices

- Appendix-XVIII Capacity Conversions for Pressure Relief Valves
- Appendix-XIX Integral Flat Head with a large Opening
- Appendix XX Submittal of Technical Inquires to BPV Committee
- Appendix-XXI Adhesive Attachment of Nameplates
- Appendix-XXII Rules for Reinforcement of Cone-to-Cylinder Junction Under External Pressure
- Appendix XXIII Qualifications and Duties of Specialized Professional Engineers

- Non-mandatory Appendices

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Module 15
Code Cases and Technical Inquiry Process

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Module 15
Code Cases and Technical Inquiry Process

Instructor: C. Wesley Rowley, P.E.

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Learning Objectives

U.S.NRC United States Nuclear Regulatory Commission Protecting People and the Environment

- Learn (what is the purpose of ASME Code Cases.
- Learn what is the purpose of ASME Technical Inquiries / Code Inquiries.
- Learn why the NRC uses ASME Code Cases.
- Understand that only the ASME approves code cases and code / technical inquires.

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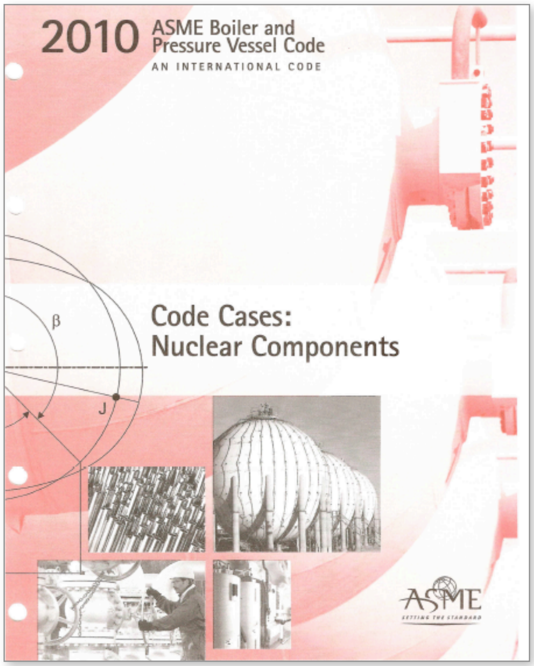
Significant Sub-topics

- **Code Cases as Alternative Requirements**
- **Approval Process for Code Cases**
- **Revised Code Cases**
- **Annulled Code Cases**
- **Summary of III / XI / OM Code Cases**
- **Regulatory Endorsement**
- **Format of Code / Technical Inquiry**
- **Approval Process of Inquiries**
- **Summary of Recent III / XI / OM Technical Inquiries**

ASME Code Cases

- **Used by most of the ASME codes – i.e.,**
  - BPV Code Section III
  - BPV Code Section XI
  - OM Code Section IST
- **Provides alternative requirements to the Code**
  - Specific paragraph requirement (i.e., N-307-3 NDE of Reactor Vessel Studs)
  - New program (i.e., OMN-3 Risk-Informed IST)
  - Use of new material (i.e., N-755 Polyethylene)



ASME BPV Code Sections III and XI

Code Cases Nuclear Components Contents

Numeric Index
Subject Index
Applicability Index
Code Cases in Numeric Order

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BPV Code Cases N-xxx

- **Large number of current code cases:**
  - Section III = 112
  - Section XI = 101
  - Total Current Nuclear BPV Code Cases = 213
- **Some have been current for decades:**
  - Code Case N-131-1 published 11 Dec 1981
  - Code Case N-155-2 published 21 Jan 1982
- **Some have been revised numerous times:**
  - Code Case N-4-13 last published 12 Feb 2008
  - Code Case N-71-18 last published 8 Dec 2000
- **Some have several revisions outstanding**
  - Code Case N-770 and N-770-1 published 26 Jan & 25 Dec 2009

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Code Case N-xxx Supplements

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INDEX TO NEW AND REVISED CASES

Section III, Divisions 1, 2, and 3, and Section XI

- **Nuclear Code Case Supplements are published approximately every six months:**
  - New Code Cases
  - Revised Code Cases
  - Annulled Code Cases

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Code Case N-xxx Supplements

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SUMMARY OF CHANGES

(10)

Revisions given below are identified on the pages by a margin note, (10).

NUMERIC INDEX
Affected Pages: ix – xiv

SUBJECT INDEX
Affected Pages: xv – xxi

APPLICABILITY INDEX FOR SECTION XI CASES
Affected Pages: xxiii – xxxvii

NEW AND REVISED CASES

Cases	Affected Pages
N-770-1	1 – 20
N-778	1

ANNULLED CASES

Cases	Affected Pages
N-122-2	1
N-318-5	1
N-391-2	1
N-392-3	1
N-626	1

ERRATA*

Cases	Affected Pages
N-517	1
N-759-2	5
N-779	1, 2

*Errata are identified on the above pages by a margin note, E, placed next to the affected area.

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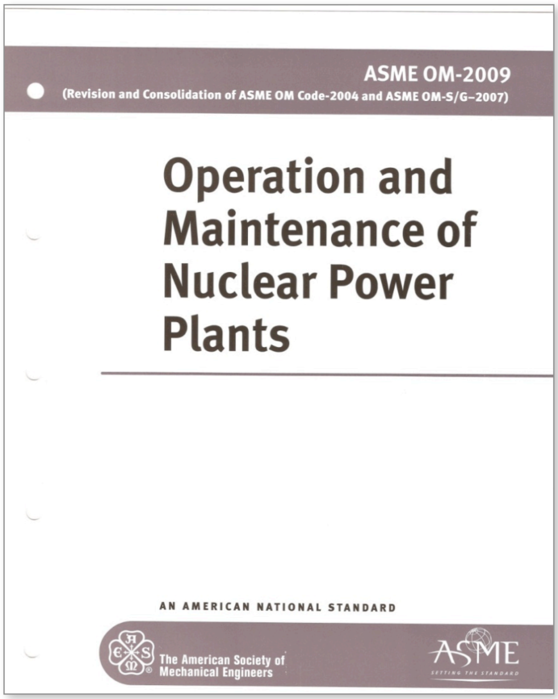

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ASME OM-2009 CONTENTS

Foreword
Preparation of Technical Inquiries
Preface
Summary of Changes

Division 1 OM Code

Division 2 OM Standards

Division 3 OM Guides

OM Interpretations
OM Code Cases

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OMN Code Cases

- **Small number of code cases:**
  - Subsection ISTA (general) = 1
  - Subsection ISTB (pumps) = 4
  - Subsection ISTC (valves) = 10
  - Subsection ISTD (snubbers) = 3
  - Subsection ISTE (RI-IST) = 1
  - Total Current OMN Code Cases = 19
- **Some have been revised:**
  - OMN-1-1 and OMN-13-1
- **Some have several revisions outstanding:**
  - Code Case OMN-1 and OMN-1-1

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Annulled Code Cases

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- It is rare for a code case to exist "forever".
- Sometimes the code case is revised.
- Sometime the technical content of the code case migrates into the code.
- Sometimes the code case becomes irrelevant and thus is annulled.

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NRC Use of Code Cases

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- Recall that code cases are an alternative to the code.
- Sometimes code cases are written as a "trial use" requirement.
- Sometimes code cases are only applicable to a handful of NPPs.
- It is more effective for the NRC to approve a code case (perhaps with conditions) than to respond to a plant specific licensing request.

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ASME Technical Inquiry Process

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- described in front of each ASME code or standard
- limited strictly to interpretations of the requirements, or to the consideration of revisions to the present requirements on the basis of new data or technology
- submitted to the Secretary of the germane ASME committee, Three Park Avenue, New York, NY 10016-5990

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Format of Technical Inquiry

U.S.NRC
United States Nuclear Regulatory Commission
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- Background
- Inquiry Structure
- Scope
- Proposed Reply

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Scope (Format of Technical Inquiry)

- The inquiry shall involve a single requirement or closely related requirements.
- An inquiry letter concerning unrelated subjects will be returned.

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Background (Format of Technical Inquiry)

- State purpose of inquiry, which would be either to obtain an interpretation of the Code requirement or to propose consideration of a revision to the present requirements.
- Provide concisely the information needed for the Committee's understanding of the inquiry (with sketches as necessary).
- Include reference to applicable code, edition, addenda, subsection, appendices, paragraphs, figures, and tables.

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Inquiry Structure (Format of Technical Inquiry)

U.S.NRC
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- **The inquiry should be stated in a condensed and precise question format, omitting superfluous background information, and, where appropriate, composed in such a way that "yes" or "no" (perhaps with provisos) would be an acceptable reply.**
- **The inquiry statement should be technically and editorially correct.**

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Proposed Reply (Format of Technical Inquiry)

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- **State what is believed that the Code requires.**
- **If, in the inquirer's opinion, a revision to the Code is needed, recommended wording shall be provided.**

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Approval of Technical / Code Inquiries

U.S.NRC United States Nuclear Regulatory Commission *Protecting People and the Environment*

- **Inquiry is sent to ASME Staff in New York**
- **Committee secretary assigns a tracking number**
- **Committee review and approval**
  - By a special committee
  - By the committee as a whole
- **Final product**
  - Posted of the ASME web site for early use
  - Published with the code edition

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Summary of Recent Technical / Code Inquiries

U.S.NRC United States Nuclear Regulatory Commission *Protecting People and the Environment*

- **ASME BPV Code Section III:**
  - Published in 2007 = 15
  - Published in 2008 = 21
  - Published in 2009 = 20
  - Published in 2010 = 20
- **ASME BPV Code Section XI:**
  - Published in 2007 = 30
  - Published in 2008 = 13
  - Published in 2009 = 17
  - Published in 2010 = 14
- **ASME OM Code Section IST:**
  - Published in 2004 = 2
  - Published in 2005 = 7
  - Published in 2006 = 3
  - Published in 2009 = 4

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Case Study

- **Problem: Case Study on Development of a Specific Technical Inquiry**
- **Situation – Plant ABC has buried check valves in the SWS Train A&B discharge piping between the auxiliary building and the ultimate heat sink (about a half mile of buried piping).**
- **Assume – components are safety related (Code Class 3).**
- **Code requirements:**
  - BPV Code Section XI requires, for buried components, a leak test or "delta" flow test (if components are unisolable).
  - ASME OM Code requires quarterly IST on all IST program CVs, unless the CV is in the condition monitoring program.

Case Study (cont'd)

- **Applicable code(s)?**
  - Is code edition / addenda germane?
  - Let's assume the plant is on the 1995 code for both ISI and IST (their current 10-year interval).
- **Background of the Technical Inquiry(s)**
- **Wording of the Technical Inquiry question**
- **Answer to the Technical Inquiry question**
  - Yes
  - No
  - Is additional explanative clause needed to the yes / no answer?
- **What is next in this Technical Inquiry process?**

Learning Questions

- What is the purpose of ASME Code Cases?
- What is the purpose of ASME Technical Inquiries / Code Inquiries?
- Why does the NRC use ASME Code Cases?
- Why does only the ASME approve code cases and code / technical inquires?

Module 16
50.55a and Related RGs

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 16
50.55a and Related RGs

Instructor: Gene Imbro, P.E.

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1

Learning Objectives

U.S.NRC United States Nuclear Regulatory Commission *Protecting People and the Environment*

- Learn how the NRC uses 10 CFR 50.55a to endorse certain ASME codes.
- Learn how the NRC selectively endorses certain ASME Code Cases.
- Learn how the NRC selectively rejects as "unacceptable" certain ASME Code Cases.
- Learn why IEEE Std 603 and IEEE Std 279 are the only IEEE standards referenced by 10 CFR 50.55a.

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Significant Sub-topics

- Endorsement of ASME BPV Code Section III
- Endorsement of ASME BPV Code Section XI
- Endorsement of ASME OM Code Section IST
- Use of Reg Guides 1.84 / 1.147 / 1.192 to endorse specific code cases for these three ASME codes
- Use of Reg Guide 1.193 to reject endorsement of certain ASME code cases
- Endorsement of IEEE Std 603 / IEEE Std 279

Major References

- **Title 10 CFR 50.55a, Codes and Standards**
- **NRC RG 1.84, Design, Fabrication, and Materials Code Case Acceptability, ASME Section III, Rev 35, July 2010**
- **NRC RG 1.147, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1**
- **NRC RG 1.192, Operations and Maintenance Code Case Acceptability, ASME OM Code**
- **NRC RG 1.193, ASME Code Cases Not Approved For Use, Rev 3, October 2010**

§ 50.55a Codes and standards. Each construction permit for a utilization facility is subject to the following conditions in addition to those specified in § 50.55. Each combined license for a utilization facility is subject to the following conditions in addition to those specified in § 50.55, except that each combined license for a boiling or pressurized water-cooled nuclear power facility is subject to the conditions in paragraphs (f) and (g) of this section, but only after the Commission makes the finding under § 52.103(g) of this chapter. Each operating license for a boiling or pressurized water-cooled nuclear power facility is subject to the conditions in paragraphs (f) and (g) of this section in addition to those specified in § 50.55.

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Title 10 CFR 50.55a CONTENTS

a. (1) Structures, systems, and components must be designed
b. The following standards have been approved for incorporation
c. *Reactor coolant pressure boundary*.
d. *Quality Group B components*.
e. *Quality Group C components*.
f. *Inservice testing requirements*.
g. *Inservice inspection requirements*.
h. *Protection and safety systems*.

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U.S.NRC United States Nuclear Regulatory Commission *Protecting People and the Environment*

10 CFR 50.55a

- **The regulation endorses certain codes & standards by reference; the requirements of those codes & standards become part of the regulations (legally binding), except as modified by the referencing statement.**
- **NRC rulemaking package updates this endorsement of the ASME codes for construction, ISI, and IST every few years.**
- **Currently endorses the 2004 code / addenda.**
- **Rulemaking package currently in progress endorses the 2007 code / addenda.**
- **Each rulemaking package typically adds some conditional endorsements regarding various technical issues.**

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10 CFR 50.55a (cont'd)

- **The currently published ASME codes are:**
  - BPV Code – 2010 edition (for nuclear construction and ISI)
  - OM – 2009 edition (for IST)
- **Wording in the regulation is hard to understand (legal jargon modified many times over decades).**
- **Three key "paragraphs" for ASME codes:**
  - (b)(1) for BPV Code Section III
  - (b)(2) for BPV Code Section XI
  - (b)(3) for OM Code Section IST

Code Case Endorsement

- **The ASME issues Code Cases as an approved alternative to the "code". Since the NRC endorses the "code", clearly they should decide if these code alternatives are acceptable.**
- **NRC uses four RGs to manage this Code Case endorsement process:**
  - RG 1.84 – NPP Construction (BPV Code Section III)
  - RG 1.147 – NPP ISI (BPV Code Section XI)
  - RG 1.192 – NPP IST (OM Code Section IST)
  - RG 1.193 – Unacceptable Code Cases

U.S. NUCLEAR REGULATORY COMMISSION

July 2010
Revision 35

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.84

(Draft was issued as DG-1191, dated June 2009)

DESIGN, FABRICATION, AND MATERIALS CODE CASE ACCEPTABILITY, ASME SECTION III

A. INTRODUCTION

General Design Criterion (GDC) 1, "Quality Standards and Records," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50), "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), requires, in part, that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Where generally recognized codes and standards are used, Criterion 1 requires that they be identified and evaluated to determine their applicability, adequacy, and sufficiency and be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

Criterion 30, "Quality of Reactor Coolant Pressure Boundary," of Appendix A to 10 CFR Part 50 requires, in part, that components that are part of the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest practical quality standards.

Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants," to 10 CFR Part 50 requires, in part, that measures be established for the control of special processing of materials and that proper testing be performed.

Provisions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code have been used since 1971 as one part of the framework to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety. Among other things, ASME standards committees develop improved methods for the construction and inservice inspection (ISI) of ASME Class 1, 2, 3, MC (metal containment), and CC

The NRC issues regulatory guides to describe and make available to the public methods that the NRC staff considers acceptable for use in implementing specific parts of the agency's regulations, techniques that the staff uses in evaluating specific problems or postulated accidents, and data that the staff needs in reviewing applications for permits and licenses. Regulatory guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions that differ from those set forth in regulatory guides will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

This guide was issued after consideration of comments received from the public.

Regulatory guides are issued in 10 broad divisions—1, Power Reactors; 2, Research and Test Reactors; 3, Fuels and Materials Facilities; 4, Environmental and Siting; 5, Materials and Plant Protection; 6, Products; 7, Transportation; 8, Occupational Health; 9, Antitrust and Financial Review; and 10, General.

Electronic copies of this guide and other recently issued guides are available through the NRC's public Web site under the Regulatory Guides document collection of the NRC's Electronic Reading Room at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML101800532.

United States Nuclear Regulatory Commission
Protecting People and the Environment

Regulatory Guide 1.84
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A. Introduction
B. Discussion
C. Regulatory Position
   1. Acceptable Code Cases
   2. Conditionally Acceptable Section III Code Cases
   3. Annulled Unconditionally Approved Section III Code Cases
   4. Annulled Conditionally Acceptable Section III Code Cases
   5. Code Cases Superceded by Revised Code Cases
D. Implementation Regulatory Analysis References

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Reg Guide 1.84

United States Nuclear Regulatory Commission
Protecting People and the Environment

- **Summary of Code Case Endorsement**
  - Cat C.1. – 74 code cases
  - Cat C.2. – 8 code cases
  - Cat C.3. – 290 code cases
  - Cat C.4. – 43 code cases
  - Cat C.5. – 137 code cases
- **There are a number of Code Cases that are not yet endorsed (i.e., Code Case N755 for Use of Polyethylene Pipe).**

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U.S. NUCLEAR REGULATORY COMMISSION

July 2010
Revision 16

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.147

(Draft was issued as DG-1192, dated June 2009)

INSERVICE INSPECTION CODE CASE ACCEPTABILITY, ASME SECTION XI, DIVISION 1

A. INTRODUCTION

General Design Criterion (GDC) 1, "Quality Standards and Records," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50), "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), requires, in part, that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, Criterion 1 requires that they be identified and evaluated to determine their applicability, adequacy, and sufficiency and be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

Provisions of the ASME International Boiler and Pressure Vessel (B&PV) Code have been used since 1971 as one part of the framework to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety. Among other things, ASME standards committees develop improved methods for the construction and inservice inspection (ISI) of ASME Class 1, 2, 3, MC (metal containment), and CC (concrete containment) nuclear power plant components. A broad spectrum of stakeholders participates in the ASME process, which helps to ensure that the various interests are considered.

The regulation in 10 CFR 50.55a(g), "Inservice Inspection Requirements," requires, in part, that Classes 1, 2, 3, MC, and CC components and their supports meet the requirements of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME BPV Code (Ref. 2) or

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The NRC issues regulatory guides to describe and make available to the public methods that the NRC staff considers acceptable for use in implementing specific parts of the agency's regulations, techniques that the staff uses in evaluating specific problems or postulated accidents, and data that the staff needs in reviewing applications for permits and licenses. Regulatory guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions that differ from those set forth in regulatory guides will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

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Electronic copies of this guide and other recently issued guides are available through the NRC's public Web site under the Regulatory Guides document collection of the NRC's Electronic Reading Room at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML101800536.

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   1. Acceptable Section XI Code Cases
   2. Conditionally Acceptable Section XI Code Cases
   3. Annulled Unconditionally Approved Section XI Code Cases
   4. Annulled Conditionally Acceptable Section XI Code Cases
   5. Code Cases That Have Been Superseded
D. Implementation

Regulatory Analysis

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Appendix A – Supplements Addressed in Proposed Rev 16 of Code Case 1.147

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United States Nuclear Regulatory Commission
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- **Summary of Code Case Endorsement**
  - Cat C.1. – 50 code cases
  - Cat C.2. – 26 code cases
  - Cat C.3. – 85 code cases
  - Cat C.4. – 10 code cases
  - Cat C.5. – 64 code cases
- **There are a number of Code Cases that are not yet endorsed (i.e., Code Case N769, N-770, N-773).**

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U.S. NUCLEAR REGULATORY COMMISSION — June 2003

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.192
(Draft was issued as DG-1089)

OPERATION AND MAINTENANCE CODE CASE ACCEPTABILITY, ASME OM CODE

A. INTRODUCTION

General Design Criterion (GDC) 1, "Quality Standards and Records," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, in part, that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, Criterion 1 requires that they be identified and evaluated to determine their applicability, adequacy, and sufficiency and be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

Provisions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code have been used since 1971 as one part of the framework to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety. ASME standards committees develop, among other things, improved methods for the construction, inservice inspection (ISI), and inservice testing (IST) of ASME Class 1, 2, 3, MC (metal containment) and CC (concrete containment) nuclear power plant components. A broad spectrum of stakeholders participates in the ASME process, which helps to ensure that the various interests are considered.

The NRC has committed through its Strategic Plan to use consensus standards to increase public involvement in the NRC's regulatory development process, consistent with the provisions of Public Law 104-113, the National Technology and Transfer Act of 1995, and Office of Management and Budget (OMB) Circular A-119, "Federal Participation in the Development and Use of Voluntary Consensus Standards and Conformity Assessment." To further the NRC's commitment in the Strategic Plan and because ASME Code provisions

This guide was issued after consideration of comments received from the public. Comments and suggestions for improvements in these guides are encouraged at all times, and guides will be revised, as appropriate, to accommodate comments and to reflect new information or experience. Written comments may be submitted to the Rules and Directives Branch, ADM, U.S. Nuclear Regulatory Commission, DC 20555-0001.

Regulatory guides are issued in ten broad divisions: 1, Power Reactors; 2, Research and Test Reactors; 3, Fuels and Materials Facilities; 4, Environmental and Siting; 5, Materials and Plant Protection; 6, Products; 7, Transportation; 8, Occupational Health; 9, Antitrust and Financial Review; and 10, General.

Single copies of regulatory guides (which may be reproduced) may be obtained free of charge by writing the Distribution Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by fax to (301)415-2289; or by email to DISTRIBUTION@NRC.GOV. Electronic copies of this guide are available on the Internet at NRC's web site at <WWW.NRC.GOV> in the Reference Library under Regulatory Guides. This guide is also available in the Electronic Reading Room at NRC's home page, along with other recently issued guides, Accession Number ML030730430.

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  C.1 Acceptable Code Cases
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Appendix A – OMN Code Cases Publication Information

Appendix B – Numerical Listing of OMN Code Cases

Reg Guide 1.192

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- **Summary of Code Case Endorsement**
  - Cat C.1. – 6 code cases
  - Cat C.2. – 6 code cases
  - Cat C.3. – 0 code cases (likely future category)
  - Cat C.4. – 0 code cases (likely future category)
  - Cat C.5. – 0 code cases (likely future category)

- **There are 5 new code cases (i.e., Code Cases OMN-14, 15, 16, 17, and 18) and 2 code case revisions (i.e., Code Case OMN-1-1 and OMN-13-1) that are not yet endorsed.**

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U.S. NUCLEAR REGULATORY COMMISSION

October 2010
Revision 3

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.193
(Draft was issued as DG-1193, dated June 2009)

ASME CODE CASES NOT APPROVED FOR USE

A. INTRODUCTION

In Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50), "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), Section 50.55a(c), "Reactor Coolant Pressure Boundary," requires, in part, that components of the reactor coolant pressure boundary be designed, fabricated, erected, and tested in accordance with the requirements for Class 1 components of Section III, "Rules for Construction of Nuclear Power Plant Components," of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code (Ref. 2) or equivalent quality standards. The regulations at 10 CFR 50.55a(f), "Inservice Testing Requirements," require, in part, that Class 1, 2, and 3 components and their supports meet the requirements of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) (Ref. 3) or equivalent quality standards. Finally, 10 CFR 50.55a(g), "Inservice Inspection Requirements," requires, in part, that Class 1, 2, 3, MC (metal containment), and CC (concrete containment) components and their supports meet the requirements of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME BPV Code or equivalent quality standards.

The ASME publishes a new edition of the BPV and OM Codes every 3 years and new addenda every year. The latest editions and addenda of Section III, Section XI, and the OM Code that the U.S. Nuclear Regulatory Commission (NRC) has approved for use by licensees are referenced in 10 CFR 50.55a(b). The ASME also publishes Code Cases for Section III and Section XI quarterly and Code Cases for the OM Code yearly. Code Cases provide alternatives developed and approved by the ASME.

The NRC issues regulatory guides to describe and make available to the public methods that the NRC staff considers acceptable for use in implementing specific parts of the agency's regulations, techniques that the staff uses in evaluating specific problems or postulated accidents, and data that the staff needs in reviewing applications for permits and licenses. Regulatory guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions that differ from those set forth in regulatory guides will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

This guide was issued after consideration of comments received from the public.

Regulatory guides are issued in 10 broad divisions—1, Power Reactors; 2, Research and Test Reactors; 3, Fuels and Materials Facilities; 4, Environmental and Siting; 5, Materials and Plant Protection; 6, Products; 7, Transportation; 8, Occupational Health; 9, Antitrust and Financial Review; and 10, General.

Electronic copies of this guide and other recently issued guides are available through the NRC's public Web site under the Regulatory Guides document collection of the NRC's Electronic Reading Room at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML101800540. The regulatory analysis may be found in ADAMS under Accession No. ML100560131.

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Regulatory Guide 1.193
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A. Introduction
B. Discussion
C. Regulatory Position

C.1 Unacceptable Section III Code Cases
C.2 Unacceptable Section XI Code Cases
C.3 Unacceptable OMN Code Cases

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Reg Guide 1.193

U.S.NRC
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- **Summary of Unacceptable Code Cases**
  - C.1 BPV Code Section III = 15 code cases
  - C.2 BPV Code Section XI = 32 code cases
  - C.3 OM Code Section IST = 1 code case

- **Above data is per Rev 3 dated October 2010**

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Rulemaking for 10 CFR 50.55a

- **Part 50.55a was revised on 4 November 2010 to endorse the following Regulatory Guides:**
  - RG 1.84 Rev 35, July 2010 (ASME BPV III Code Cases)
  - RG 1.147 Rev 16, July 2010 (ASME BPV XI Code Cases)
  - RG 1.192 Rev 0, June 2003 (ASME OMN Code Cases)
  - RG 1.193 Rev 3, October 2010 (Unacceptable Code Cases)
- **With the ASME code cases listed in these RGs, the NRC has authorized these alternative requirements to those requirements in the latest endorsed ASME BPV and OM codes.**

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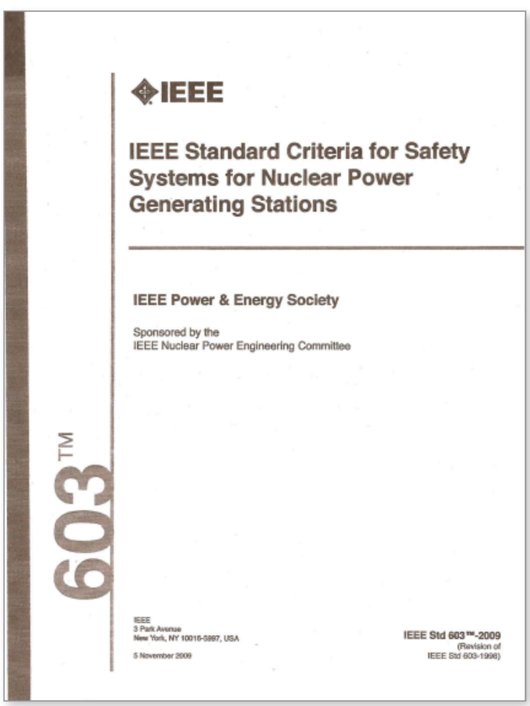
IEEE

IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations

IEEE Power & Energy Society

Sponsored by the
IEEE Nuclear Power Engineering Committee

603™

IEEE
3 Park Avenue
New York, NY 10016-5997, USA

5 November 2009

IEEE Std 603™-2009
(Revision of
IEEE Std 603-1998)

IEEE Std 603-2009 CONTENTS

1. Overview
2. Normative references
3. Definitions
4. Safety system design basis
5. Safety system criteria
6. Sense and command features – functional and design requirements
7. Execute features – functional and design requirements
8. Power source requirements

Annex A – Illustrations
Annex B – Electromagnetic Compatibility
Annex C – Bibliography

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IEEE Std. 603

- **Title: *IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations***
  - The edition endorsed in Part 50.55a is dated 1991.
  - The currently published edition by the IEEE is dated 2009 (which is a revision of IEEE Std 603 – 1998).
  - These safety criteria are important for power, control, and instrumentation systems, so they are incorporated by reference in 10 CFR 50.55a(h).

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IEEE Std. 279

- **IEEE-603 is a relatively new standard being endorsed by reference in 10 CFR 50.55a.**
- **The regulation endorses both IEEE-279 and IEEE-603.**
- **IEEE-279 is applicable to the ABWR Design Certification which was done before IEEE-603 was incorporated by reference into 50.55a.**
- **10 CFR 50.55a states that reference to IEEE-279 is allowed only prior to 1999.**

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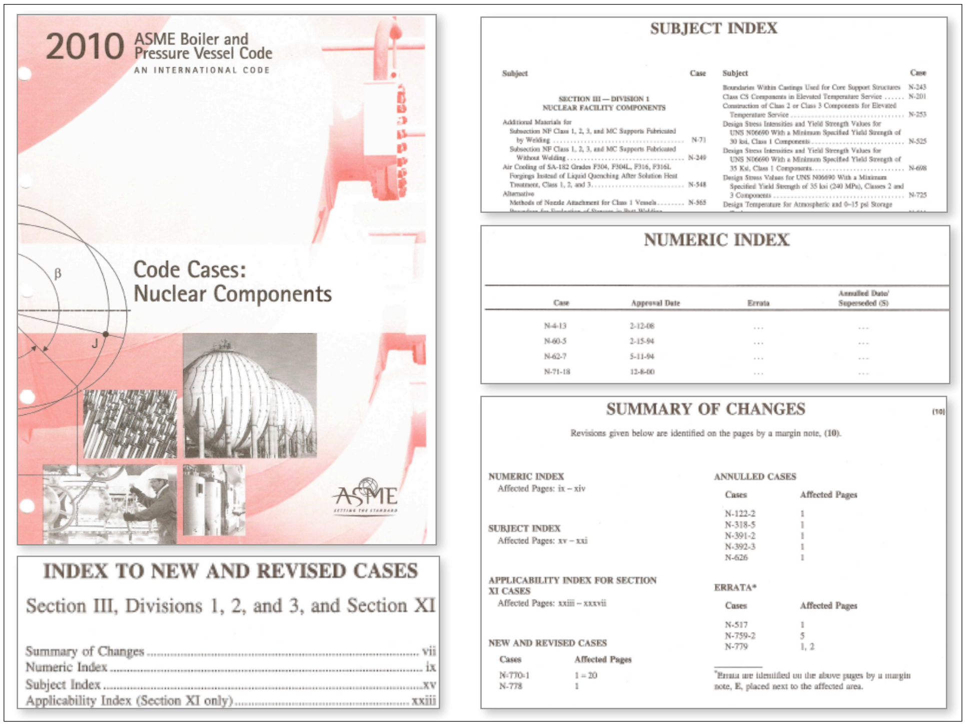

Case Study A

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- **Case Study on the NRC process of revising 10 CFR 50.55a to endorse a particular ASME code edition (specifically BPV and OM codes for 2010 edition).**
  - Step #1 is ???
  - Step #2 is ???
  - Step #3 is ???
  - Step #4 is ???
  - Step #5 is ???
  - Step #6 is ???

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Code Case N-655-1

CASES OF ASME BOILER AND PRESSURE VESSEL CODE

CASE
N-655-1

Approval Date: September 21, 2007

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

Case N-655-1
Use of SA-738, Grade B, for Metal Containment Vessels, Class MC
Section III, Division 1

Inquiry: May steel plate conforming to SA-738, Grade B, be used for the construction of metal containment conforming to the rules of Section III, Subsection NE?

Reply: It is the opinion of the Committee that SA-738, Grade B, may be used for the construction of metal containments conforming to the rules of Section III, Subsection NE, provided the following requirements are met:

(a) The maximum temperature shall not exceed 650°F (343°C).

(b) The allowable stress S, and the allowable stress intensity S_{mc} shall be 1.1 times the S values given for SA-738, Grade B in Section II, Part D, Subpart 1, Table 1A. The yield stress values S_y shall be given in Section II, Part D, Subpart 1, Table Y-1. For external pressure design, use the chart referenced for SA-738, Grade B, in Table 1A of Section II, Part D, Subpart 1.

(c) Exception to postweld heat treatment is permitted up through a nominal thickness of 1.75 in. (44 mm) under the conditions given in Table NE-4622.7(b)-1 including Note (2).

(d) SA-20/SA-20M Supplementary Requirement S1, "Vacuum Treatment" shall be mandatory for this material.

(e) SA-20/SA-20M Supplementary Requirement S20, "Maximum Carbon Equivalent for Weldability" shall be mandatory for this material.

(f) The exception from heat treatment of material test coupons below 2 in. (50 mm) given in NE-2211 shall not apply to this material.

(g) The Case number shall be shown on the Certificate Holder's Data Report.

Code Case N-757-1

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CASES OF ASME BOILER AND PRESSURE VESSEL CODE

CASE
N-757-1

Approval Date: September 21, 2007

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

Case N-757-1
Alternative Rules for Acceptability for Class 2 and 3 Valves, NPS 1 (DN 25) and Smaller With Welded and Nonwelded End Connections Other Than Flanges
Section III, Division 1

Inquiry: Under what rules may instrument, control and sampling line valves, NPS 1 (DN 25) and smaller, with welded and nonwelded end connections other than flanges, meet the design requirements of Section III, Division 1, Class 2 and 3 rules of NC-3512 and ND-3512, when the valve minimum wall thickness does not meet the requirements of ASME B16.34?

Reply: It is the opinion of the Committee that instrument, control, and sampling line valves, NPS 1 (DN 25) and smaller, having valve minimum wall thickness not in accordance with ASME B16.34, with welded and nonwelded end connections other than flanges, may meet the design requirements of Section III, Division 1, Class 2 and 3 rules of NC-3512 and ND-3512, provided the following additional requirements are met:

(a) Valves not meeting the wall thickness requirements of ASME B16.34, shall meet the pressure design rules of NC-3324 and ND-3324; an experimental stress analysis (Section III, Division 1, Appendix II); or Design Based on Stress Analysis (Section III, Division 1, Appendix XIII); and the design shall be qualified in accordance with the requirements of MSS-SP-105-2005, Section 5.

(b) The end connections shall meet the requirements of NC-3661 and ND-3661, NC-3671.3 or ND-3671.4, for welded, threaded, and flared, flareless and compression type fittings tube ends.

(c) Valve loadings including, but not limited to operation, closure, and assembly, shall be accounted for by one of the following methods: experimental stress analysis (Appendix II) or Design Based on Stress Analysis (Appendix XIII).

(d) All valves shall meet the requirements of NC-3521, ND-3521.

(e) Valve bonnets threaded directly into valve bodies shall have a lock weld or locking device to assure that the assembly does not disengage either through stem operation or vibration.

(f) This Case number shall be identified on the NPV-1 Data Report Form.

Code Case N-782

U.S.NRC
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CASES OF ASME BOILER AND PRESSURE VESSEL CODE

CASE
N-782

Approval Date: January 30, 2009

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

Case N-782
Use of Code Editions, Addenda, and Cases
Section III, Division 1

Inquiry: What Code Editions, Addenda, and Cases may be used as an alternative to NCA-1140(a)(2)(a) and NCA-1140(a)(2)(b)?

Reply: It is the opinion of the Committee that as an alternative to NCA-1140(a)(2)(a) and NCA-1140(a)(2)(b), the following requirements may be used:

(*a*) The Edition and Addenda endorsed for a design certified or licensed by the regulatory authority.

(*b*) This Case number shall be recorded on the documentation for the item.

Learning Questions

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- **How/why does the NRC use 10 CFR 50.55a to endorse certain ASME codes?**
- **How does the NRC selectively endorse certain ASME Code Cases?**
- **How does the NRC selectively reject as "unacceptable" certain ASME Code Cases?**
- **Why is IEEE Std 603 & IEEE Std 279 the only IEEE standards referenced by 10 CFR 50.55a?**

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Module 17

Pre-Service / In-Service Testing

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 17

Pre-Service / In-Service Testing

Instructor: C. Wesley Rowley, P.E.

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Learning Objectives

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- Learn the basics of an PST / IST Program
- Learn how a PST / IST program is developed for a new NPP
- Understand the strengths (and weaknesses) of an PST / IST Program
- Understand the current / future regulatory changes affecting the IST program

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Significant Sub-topics

- **IST Fundamentals**
- ASME OM Code / Standard / Guide
- Scope of Applicable Components
- General Requirements
- PST / IST Program for Pumps
- PST / IST Program for Power-Operated Valves
- PST / IST Program for Non-Power-Operated Valves
- PST / IST Program for Snubbers
- Risk-Informed IST
- Surveillance Procedures
- Interface with Plant Startup

IST Fundamentals

- **the "incredible accident"**
- **important system functions**
- **operational readiness**

Protecting the Public Health and Safety

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- conservative design for the RPV
- maintain pressure integrity
- functionally deliver lots of cooling water to core to remove heat
- defense-in-depth
- safety margin

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5

Important System Functions

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- in shutting down the reactor to the safe shutdown condition
- in maintaining the safe shutdown condition
- in mitigating the consequences of an accident

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Operational Readiness
PST / IST

- **certain pumps have to start or remain running**
- **certain valves have to operate or remain as is**
- **certain snubbers have to operate to maintain the integrity of the pressure boundary system during DBAs, as well as normal plant operation**
- **ensure operational readiness by periodically testing the functionality of these pumps / valves / snubbers**

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Significant Sub-topics

- IST Fundamentals
- **ASME OM Code / Standard / Guide**
- Scope of Applicable Components
- General requirements
- PST / IST Program for Pumps
- PST / IST Program for Power-Operated Valves
- PST / IST Program for Non-Power-Operated Valves
- PST / IST Program for Snubbers
- Risk-Informed IST
- Surveillance Procedures
- Interface with Plant Startup

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ASME OM Code / Standard / Guide

U.S.NRC United States Nuclear Regulatory Commission Protecting People and the Environment

- **Contents of ASME OM-2009**
  - Forward
  - Preparation of Technical Inquiries
  - Committee Roster
  - Preface
  - Summary of Changes
  - Division 1 Code
  - Division 2 Standards
  - Division 3 Guides

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U.S.NRC United States Nuclear Regulatory Commission Protecting People and the Environment

- **SECTION IST – RULES FOR INSERVICE TESTING**
  - Subsection ISTA – General Requirements
  - Subsection ISTB – Pump Requirements
  - Subsection ISTC – Valve Requirements
  - Subsection ISTD – Snubber Requirements
  - Subsection ISTE – Risk-Informed IST Requirements
  - Mandatory Appendices
  - Nonmandatory Appendices

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Appendices

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- **Mandatory Appendices**
  - Appendix I - Safety Relief Valves
  - Appendix II - Check Valves
  - Appendix III – MOVs
  - Appendix IV – AOVs (in the course of preparation)
- **Non-mandatory Appendices**
  - Appendix A - Preparation of Test Plans
  - Appendix J - CV Testing Following Valve Disassembly
  - Snubbers - Appendices B, C, D, E, F, G, H

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Division 2 OM Standards

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- **Part 2 Performance Testing of Closed Cooling Water Systems**
- **Part 3 Vibration Testing 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 25 Performance Testing of Emergency Core Cooling Systems**
- **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**

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Division 3 OM Guides

- **Part 5 Inservice Monitoring of Core Support Barrel Preload in PWR Power Plants**
- **Part 7 Requirements for Thermal Expansion Testing of NPP Piping Systems**
- **Part 11 Vibration Testing and Assessment of Heat Exchangers**
- **Part 14 Vibration Monitoring of Rotating Equipment**
- **Part 17 Performance Testing of Instrument Air Systems**
- **Part 19 Performance Testing of AOVs and HOVs**
- **Part 23 Inservice Monitoring of Reactor Internals Vibration in PWR Power Plants**

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Significant Sub-topics

- IST Fundamentals
- ASME OM Code / Standard / Guide
- **Scope of Applicable Components**
- General Requirements
- PST / IST Program for Pumps
- PST / IST Program for Power-Operated Valves
- PST / IST Program for Non-Power-Operated Valves
- PST / IST Program for Snubbers
- Risk-Informed IST
- Surveillance Procedures
- Interface with Plant Startup

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Scope of Applicable Components

- **10 CFR 50.2 Definitions**
  - Safety-related
    - Safe shutdown
    - Accident mitigation
- **10 CFR 50.55a Codes and Standards**
  - ASME Code Class 1
  - ASME Code Class 2
  - ASME Code Class 3
- **ASME OM Code (OM Division 1)**
  - Safe shutdown
  - Accident mitigation

IST Program Scope Determination – Sources

- **Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-waste Containing Components of Nuclear Power Plants"**
- **NUREG-0800, Standard Review Plan, Section 3.2.2**
- **NUREG-1482, Guidelines for Inservice Testing at Nuclear Power Plants**

Significant Sub-topics

- IST Fundamentals
- ASME OM Code / Standard / Guide
- Scope of Applicable Components
- **General Requirements**
- PST / IST Program for Pumps
- PST / IST Program for Power-Operated Valves
- PST / IST Program for Non-Power-Operated Valves
- PST / IST Program for Snubbers
- Risk-Informed IST
- Surveillance Procedures
- Interface with Plant Startup

Outline of Subsection ISTA
(from 2009 Edition)

- **ISTA - 1000 INTRODUCTION**
  - ISTA-1100 Scope
  - ISTA-1200 Jurisdiction
  - ISTA-1300 Application
  - ISTA-1400 Referenced Standards and Specifications
  - ISTA-1500 Owner's Responsibilities
  - ISTA-1600 Accessibility
- **ISTA - 2000 DEFINITIONS**
- **ISTA - 3000 GENERAL REQUIREMENTS**
  - ISTA-3100 Test and Examination Program
  - ISTA-3200 Administrative Requirements
  - ISTA-3300 Corrective Actions

Outline of ISTA – (cont'd)
(from 2009 Edition)

- **ISTA - 4000 INTRUMENTATION AND TEST EQUIPMENT**
  - ISTA-4100 Range and Accuracy
  - ISTA-4200 Calibration
- **ISTA - 5000 LATER**
- **ISTA - 6000 LATER**
- **ISTA - 7000 LATER**
- **ISTA - 8000 LATER**
- **ISTA - 9000 RECORDS AND REPORTS**
  - ISTA-9100 Scope
  - ISTA-9200 Requirements
  - ISTA-9300 Retention

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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.**

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ISTA-1100 Scope (cont'd)

- **It 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.**

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ISTA-1100 Scope (cont'd)

- **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;

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ISTA-1100 Scope (cont'd)

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b) pressure relief devices that protect systems or portions of systems 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.

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ISTA-1200 Jurisdiction

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- **The jurisdiction of Section IST covers individual components that have met all the requirements of the construction code commencing at the time when the construction code requirements have been met, irrespective of the physical location. When portions of systems or plants are completed at different times, the jurisdiction of this Section shall cover only those components on which all construction related to the components has been completed.**

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ISTA-1300 Application

- **ISTA-1310 Components Subject to Testing and Examination.**
  - Components identified in Section IST for testing or examination shall be included in the test plan (ISTA-3110). These components include nuclear power plant items such as pumps, valves, and dynamic restraints (snubbers).

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ISTA-1300 Application

- **ISTA-1320 Classifications.**
  - Optional construction of a component in a system boundary to a classification higher than the minimum class established in the component Design Specification (either upgraded from Class 2 to Class 1 or Class 3 to Class 2) shall not affect the overall system classification by which the applicable requirements of Section IST are determined.

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ISTA-1400 Referenced Standard and Specifications

- **When standards and specifications are referenced in Section IST, their revision date or indicator shall be as shown in Table ISTA-1400-1.**

Standard or Specification	Revision Date or Indicator
PTC 25	1994
API RP-527	3rd edition, 1991

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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.

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ISTA-1500 Owner's Responsibilities (cont'd)

b) design and arrangement of system components to include allowance for adequate access and clearance for conduct of tests and examinations;

c) preparation of plans and schedules;

d) preparation of written test and examination instructions and procedures;

ISTA-1500 Owner's Responsibilities (cont'd)

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'd)

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;

j) retention of all test and examination records for the service lifetime of the component or system; and

ISTA-1500 Owner's Responsibilities (cont'd)

k) documentation of a quality assurance program in accordance with the following:

1. Title 10, Code of Federal Regulations, Part 50; or
2. ASME NQA-1, Parts II and III.

ISTA-1600 Accessibility

- **Provisions for examination shall include access for the examination personnel and equipment necessary to conduct the test or examination.**

ISTA-2000 Definitions

- **Subsection ISTA has 32 definitions applicable to Subsections ISTA, ISTB, ISTC, ISTD, and / or ISTE.**
- **Some are general definitions, such as "Owner" and "operational readiness".**
- **Some are specific definitions such as "obturator", "instrument loop", or "system resistance".**

Excerpt from ISTA-1100 Definitions

- **Preservice Test – "test performed after completion of construction activities related to the component and before first electrical generation by nuclear heat, or in an operating plant, before the component is initially placed in service.**
- **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"**

Other Definitions

- **Performance testing – "a test to determine whether a system or component meets specified acceptance criteria"**
- **Reference Point – "a point of operation at which the reference values are established and inservice test parameters are measured for comparison with applicable acceptance criteria"**
- **Reference Value – "one or more values of parameters as measured or determined when the equipment is known to be operating acceptably"**

ISTA-3100 Test and Examination Program

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- **ISTA-3110 Test and Examination Plans**
- **ISTA-3120 Inservice Test Interval**
- **ISTA-3130 Application of Code Cases**
- **ISTA-3140 Application of Revised Code Cases**
- **ISTA-3150 Application of Annulled Code Cases**
- **ISTA-3160 Test and Examination Procedures**

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ISTA-3200 Administrative Requirements

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- **IST Plans**
- **Selection of components**
- **Application of requirements**
- **Use of Code Cases**
- **Revisions to Code Cases**
- **Tests and examinations**
  - Preservice test period
  - Initial inservice test interval
  - Successive inservice test intervals

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ISTA-3300 Corrective Action

- **Corrective actions requiring repair / replacement activities shall be performed in accordance with ASME BPV Code Section XI, as applicable.**
- **Other Corrective actions shall be performed in accordance with the Owner's quality assurance program.**

ISTA-4000 Instrumentation and Test Equipment

- **ISTA-4100 Range and Accuracy**
  - Instrumentation and test equipment used in performing the examination and testing program shall have the range and accuracy necessary to demonstrate conformance to specific examination or test requirements.
- **ISTA-4200 Calibration**
  - All instruments and test equipment used in performing the examination and testing program shall be calibrated and controlled in accordance with the Owner's administrative procedures or a quality assurance program approved by the Owner.

Subsection ISTA - (cont'd)

- **ISTA-5000 Reserved**
- **ISTA-6000 Reserved**
- **ISTA-7000 Reserved**
- **ISTA-8000 Reserved**
- **ISTA-9000 Records and Reports**

ISTA-9000 Records and Reports

- **ISTA-9100 Scope**
- **ISTA-9200 Requirements**
  - ISTA-9210 Owner's Responsibility
  - ISTA-9220 Preparation
  - ISTA-9230 Inservice Test and Examination Results
  - ISTA-9240 Record of Corrective Action
- **ISTA-9300 Retention**
  - ISTA-9310 Maintenance of Records
  - ISTA-9320 Reproduction
  - ISTA-9330 Test and Examination Records

Significant Sub-topics

- IST Fundamentals
- ASME OM Code / Standard / Guide
- Scope of Applicable Components
- General Requirements
- **PST / IST Program for Pumps**
- PST / IST Program for Power-Operated Valves
- PST / IST Program for Non-Power-Operated Valves
- PST / IST Program for Snubbers
- Risk-Informed IST
- Surveillance Procedures
- Interface with Plant Startup

Outline of ISTB (from 2009 Edition)

- **ISTB - 1000 INTRODUCTION**
  - ISTB-1100 Applicability
  - ISTB-1200 Exclusions
  - ISTB-1300 Pump Categories
  - ISTB-1400 Owner's Responsibility
- **ISTB - 2000 SUPPLEMENTAL DEFINITIONS**
- **ISTB - 3000 GENERAL TESTING REQUIREMENTS**
  - ISTB-3100 Preservice Testing
  - ISTB-3200 Inservice Testing
  - ISTB-3300 Reference Values
  - ISTB-3400 Frequency of Inservice Tests
  - ISTB-3500 Data Collection
- **ISTB - 4000 LATER**

Outline of ISTB (from 2009 Edition)

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- **ISTB - 5000 SPECIFIC TESTING REQUIREMENTS**
  - ISTB-5100 Centrifugal Pumps (Except Vertical Line Shaft Centrifugal Pumps)
  - ISTB-5200 Vertical Line Shaft Centrifugal Pumps
  - ISTB-5300 Positive Displacement Pumps
- **ISTB - 6000 MONITORING, ANALYSIS, AND EVALUATION**
  - ISTB-6100 Trending
  - ISTB-6200 Corrective Action
  - ISTB-6300 Systematic Error
  - ISTB-6400 Analysis of Related Conditions
- **ISTB - 7000 LATER**
- **ISTB - 8000 LATER**

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Outline of ISTB (from 2009 Edition)

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- **ISTB - 9000 RECORDS AND REPORTS**
  - ISTB-9100 Pump Records
  - ISTB-9200 Test Plans
  - ISTB-9300 Record of Tests
  - ISTB-9400 Record of Corrective Action

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PST / IST for Pumps

- **Design Basis Verification in the Pre-op Test**
- **Baseline Development in the PST**
- **Degradation Monitoring in the IST**
- **Periodic Maintenance & Repairs**
- **New Baseline Development in the IST**
- **New Degradation Monitoring in the IST**

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Significant Sub-topics

- IST Fundamentals
- ASME OM Code / Standard / Guide
- Scope of Applicable Components
- General Requirements
- PST / IST Program for Pumps
- **PST / IST Program for Power-Operated Valves**
- PST / IST Program for Non-Power-Operated Valves
- PST / IST Program for Snubbers
- Risk-Informed IST
- Surveillance Procedures
- Interface with Plant Startup

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Outline of ISTC (from 2009 Edition)

- **ISTC - 1000 INTRODUCTION**
  - ISTC-1100 Applicability
  - ISTC-1200 Exemptions
  - ISTC-1300 Valve Categories
  - ISTC-1400 Owner's Responsibility
- **ISTC - 2000 SUPPLEMENTAL DEFINITIONS**
- **ISTC - 3000 GENERAL TESTING REQUIREMENTS**
  - ISTC-3100 Preservice Testing
  - ISTC-3200 Inservice Testing
  - ISTC-3300 Reference Values
  - ISTC-3400 Later
  - ISTC-3500 Valve Testing Requirements
  - ISTC-3600 Leak Testing Requirements
  - ISTC-3700 Position Verification Testing
  - ISTC-3800 Instrumentation

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Outline of ISTC (from 2009 Edition)

- **ISTC-4000 later**
- **ISTC-5000 Specific Testing Requirements**
  - ISTC-5100 Power-Operated Valves (POVs)
  - ISTC-5200 Other Valves
- **ISTC-6000 Monitoring, Analysis, and Evaluation**
- **ISTC-7000 later**
- **ISTC-8000 later**
- **ISTC-9000 Records and Reports**

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PST / IST Program for Power-Operated Valves

- **ISTC-5100 Power-Operated Valves (POVs)**
  - ISTC-5110 Power-Operated Relief Valves
  - ISTC-5120 Motor-Operated Valves
  - ISTC-5130 Pneumatically Operated Valves
  - ISTC-5140 Hydraulically Operated Valves
  - ISTC-5150 Solenoid-Operated Valves
- **Appendix III Active Electric Motor Operated Valve Assemblies**
- **Appendix IV AOVs (in the course of preparation)**

Significant Sub-topics

- IST Fundamentals
- ASME OM Code / Standard / Guide
- Scope of Applicable Components
- General Requirements
- PST / IST Program for Pumps
- PST / IST Program for Power-Operated Valves
- **PST / IST Program for Non-Power-Operated Valves**
- PST / IST Program for Snubbers
- Risk-Informed IST
- Surveillance Procedures
- Interface with Plant Startup

PST / IST Program for Non-Power-Operated Valves

- **ISTC-5200 Other Valves**
  - ISTC-5210 Manually Operated Valves
  - ISTC-5220 Check Valves
  - ISTC-5230 Vacuum Breaker Valves
  - ISTC-5240 Safety and Relief Valves
  - ISTC-5250 Rupture Disks
  - ISTC-5260 Explosively Actuated Valves

- **Appendix I Pressure Relief Devices**
- **Appendix II Check Valve Condition Monitoring Program**

Significant Sub-topics

- IST Fundamentals
- ASME OM Code / Standard / Guide
- Scope of Applicable Components
- General Requirements
- PST / IST Program for Pumps
- PST / IST Program for Power-Operated Valves
- PST / IST Program for Non-Power-Operated Valves
- **PST / IST Program for Snubbers**
- Risk-Informed IST
- Surveillance Procedures
- Interface with Plant Startup

Outline of ISTD (from 2009 Edition)

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- **ISTD-1000 Introduction**
  - ISTD-1100 Application
  - ISTD-1200 Design and Operating Information
  - ISTD-1300 Examination and Test Results
  - ISTD-1400 later
  - ISTD-1500 Snubber Maintenance or Repair
  - ISTD-1600 Snubber Modification and Replacement
  - ISTD-1700 Deletions of Unacceptable Snubbers
  - ISTD-1800 Supported Component(s) or System Evaluation
- **ISTD-2000 Definitions**

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Outline of ISTD (from 2009 Edition)

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- **ISTD-3000 General Requirements**
  - ISTD-3100 General Examination Requirements
  - ISTD-3200 General Testing Requirements
- **ISTD-4000 Specific Examination Requirements**
  - ISTD-4100 Preservice Examination
  - ISTD-4200 Inservice Examination
- **ISTD-5000 Specific Testing Requirements**
  - ISTD-5100 Preservice Operational Readiness Testing
  - ISTD-5200 Inservice Operational Readiness Testing
  - ISTD-5300 The 10% Testing Sample
  - ISTD-5400 The 37 Testing Sampling Plan
  - ISTD-5500 Retests of Previously Unacceptable Snubbers

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Outline of ISTD (from 2009 Edition)

- **ISTD-6000 Service Life Monitoring**
  - ISTD-6100 Predicted Service Life
  - ISTD-6200 Service Life Evaluation
  - ISTD-6300 Cause Determination
  - ISTD-6400 Additional Monitoring Requirements for Snubbers That Are Tested Without Applying a Load to the Snubber Piston Rod
  - ISTD-6500 Testing for Service Life Monitoring Purposes

Significant Sub-topics

- IST Fundamentals
- ASME OM Code / Standard / Guide
- Scope of Applicable Components
- General Requirements
- PST / IST Program for Pumps
- PST / IST Program for Power-Operated Valves
- PST / IST Program for Non-Power-Operated Valves
- PST / IST Program for Snubbers
- **Risk-Informed IST**
- Surveillance Procedures
- Interface with Plant Startup

Outline of ISTE (from 2009 Edition)

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- **ISTE-1000 Introduction**
  - ISTE-1100 Applicability
  - ISTE-1200 Alternative
  - ISTE-1300 General
- **ISTE-2000 Supplemental Definitions**

Note A – there are 25 definitions in ISTE-2000.
Note B – there is a footnote reference to ASME RA-S-2002 (the PRA standard).

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Outline of ISTE (from 2009 Edition)

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- **ISTE-3000 General Requirements**
  - ISTE-3100 Implementation
  - ISTE-3200 Probabilistic Risk Assessment
  - ISTE-3300 Integrated Decision Making
  - ISTE-3400 Evaluation of Aggregate Risk
  - ISTE-3500 Feedback and Corrective Action

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Outline of ISTE (from 2009 Edition)

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- **ISTE-4000 Specific Component Categorization Requirements**
  - ISTE-4100 Component Risk Categorization
  - ISTE-4200 Component Safety Categorization
  - ISTE-4300 Testing Strategy Formulation
  - ISTE-4400 Evaluation of Aggregate Risk
  - ISTE-4500 Inservice Testing Program

Note A – The PRA provides the quantitative information.
Note B – The Expert Panel provides the qualitative information and make the categorization decision.

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Outline of ISTE (from 2009 Edition)

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- **ISTE-5000 Specific Testing Requirements**
  - ISTE-5100 Pumps
  - ISTE-5200 Check Valves
  - ISTE-5300 Motor Operated Valve Assemblies
  - ISTE-5400 Pneumatically and Hydraulically Operated Valves
  - ISTE-5500 later
- **ISTE-6000 Monitoring, Analysis, and Evaluation**
- **ISTE-7000 later**
- **ISTE-8000 later**
- **ISTE-9000 Records and Reports**

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Significant Sub-topics

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- IST Fundamentals
- ASME OM Code / Standard / Guide
- Scope of Applicable Components
- General Requirements
- PST / IST Program for Pumps
- PST / IST Program for Power-Operated Valves
- PST / IST Program for Non-Power-Operated Valves
- PST / IST Program for Snubbers
- Risk-Informed IST
- **Surveillance Procedures**
- Interface with Plant Startup

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Surveillance Procedures

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- **The plant Tech Specs identify that IST must be accomplished in accordance with the ASME OM Code.**
- **The ASME OM Code identifies the PST & IST for pumps, valves, and snubbers.**
- **The Surveillance Procedures describe:**
  - how the IST requirements are accomplished.
  - what the acceptance criteria are.
  - place to record the measured data.

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Significant Sub-topics

- IST Fundamentals
- ASME OM Code / Standard / Guide
- Scope of Applicable Components
- General Requirements
- PST / IST Program for Pumps
- PST / IST Program for Power-Operated Valves
- PST / IST Program for Non-Power-Operated Valves
- PST / IST Program for Snubbers
- Risk-Informed IST
- Surveillance Procedures
- **Interface with Plant Startup**

Plant Startup Interface

- **SSCs are checked for proper operation.**
- **Pre-op tests check that the system operates as designed.**
- **PST is developed as a baseline for the component performance, based on the Pre-op test.**
- **PST results are used to implement the IST program, including acceptance criteria.**

ASME Code Interpretations

- **The ASME O&M Committee issues code interpretations to technical inquiries.**
- **This interpretations are usually published concurrent with the ASME OM Code edition.**
- **Between 20 June 2007 and 25 July 2008 there were four technical inquiries answered.**
- **Between 22 November 2005 and 12 December 2005 there were three technical inquiries answered.**
- **Between 9 January 2004 and 20 June 2005 there were seven technical inquiries answered.**

ASME OMN Code Cases

- **Code Cases are alternatives to the Code.**
- **When the Code Case is approved, it has an indefinite life (needs to be annulled when no longer valid).**
- **When the ASME OM-2009 edition was published, there were 18 code cases (OMN-1 through OMN-18).**
- **Two of the 18 codes cases have been revised (OMN-1 and OMN-13).**

Case Study on Preservice Testing

- **Problem: The Watts Bar NPP will be performing PST for the Service Water pumps soon. What are some of the considerations for preparing that PST?**
  - Consideration #1 ?
  - Consideration #2 ?
  - Consideration #3 ?
  - Consideration #4 ?

Learning Questions

- **What are the basics of an PST / IST Program?**
- **How is a PST / IST program developed for a new NPP?**
- **What are the strengths of an PST / IST Program? And the weaknesses?**
- **What are the current regulatory changes affecting the IST program? And the future regulatory changes…?**

Module 18

Relation of 10 CFR 50 to Part 52

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Module 18
Relation of 10 CFR 50 to Part 52

Instructor: Gene Imbro, P.E.

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Learning Objectives

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- **Learn how Part 52 differs from Part 52**
- **Learn how 10 CFR Part 50 and Part 52 are inter-related.**
- **Learn which Sections of Part 50 and Part 52 reference each other.**

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Major References

- **10 CFR Part 50**
- **10 CFR Part 52**

Part 52

- **Part 52 Covers the issuance of:**
  - Early site permits - Subpart A
  - Standard Design Certifications - Subpart B
  - Combined Licenses - Subpart C
  - Standard Design Approvals - Subpart E
  - Manufacturing Licenses - Subpart F

Subpart D was reserved

Part 52

- **Subpart B - Standard Certified Designs**
  - Rulemaking
  - Included as an Appendix to 10 CFR 52.
    - Appendix A - US Advanced BWR
    - Appendix B - CE System 80+
    - Appendix C - Westinghouse AP-600
    - Appendix D - Westinghouse AP-1000
  - Change to a Certified Design requires a rulemaking

Part 52

- **Subpart C- Combined "*Operating*" License**
  - Application does not need to reference a Standard Certified Design or an Early Site Permit
  - Construction can begin when the COL is issued
  - Plant cannot operate until the ITAAC are completed and the Commission makes a finding in accordance with 10 CFR 52.103(g)
  - Combined license issued for 40 years commencing with the date of the 52.103(g) decision.
  - Standard Design Certification is valid for 15 years from the date of issuance.

Part 50

- **Two step process used to license the current fleet**
  - PSAR → ~2 years → Construction Permit
  - FSAR → ~?? years → Operating License
- **Prone to delays**
  - Lack of regulatory stability/predictability
    - Cost overruns
  - Intervention
    - Clam Shell Alliance - Seabrook
    - Abalone Alliance - Diablo Canyon

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Part 52

- **Part 52 was a process change rather a change in technical requirements.**
  - Substantial reliance on Part 50 requirements.
  - There are references in Part 52 to Part 50 requirements
  - Part 50 contains references to Part 52 applicability.

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Part 52 ↔ Part 50

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- **10 CFR 52.0 (b) Scope; applicability of 10 CFR I provisions**
  - Unless otherwise specifically provided for in this part, the regulations in 10 CFR Chapter I apply to a holder of or applicant for an approval, certification, permit, or license.
  - A holder of or applicant for an approval, certification, permit, or license issued under this part shall comply with all requirements in 10 CFR Chapter I that are applicable.

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Part 52 ↔ Part 50

U.S.NRC
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- **10 CFR 52.0 (b) Scope; applicability of 10 CFR I provisions (cont'd)**
  - A license, approval, certification, or permit issued under this part is subject to all requirements in 10 CFR Chapter I which, by their terms, are applicable to early site permits, design certifications, combined licenses, design approvals, or manufacturing licenses.

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Part 52 ↔ Part 50

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- **Early Site Permits**
  - **52.16 Contents of applications; general information.**
    - The application must contain all of the information required by 10 CFR 50.33(a) through (d) and (j) of this chapter.
  - **52.17 Contents of applications; technical information.**
    - The description of the quality assurance program for a nuclear power plant site shall include a discussion of how the applicable requirements of appendix B to part 50 of this chapter will be satisfied.

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Part 52 ↔ Part 50

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- **Early Site Permits (cont'd)**
  - **52.18 Standards for review of applications.**
    - Applications filed under this subpart will be reviewed according to the applicable standards set out in 10 CFR part 50 and its appendices and 10 CFR part 100.

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Part 52 ↔ Part 50

- **Standard Design Certifications**
  - **52.46 Contents of applications; general information.**
    - The application must contain all of the information required by 10 CFR 50.33(a) through (c) and (j)
  - **52.47 Contents of applications; technical information.**
    - **52.47(a)(2)(iv)(B)(*i*)** The principal design criteria for the facility. Appendix A to 10 CFR part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants…

Part 52 ↔ Part 50

- **Standard Design Certifications (cont'd)**
  - **52.47(a)(12)** The application must contain an analysis and description of the equipment and systems for combustible gas control as required by 10 CFR 50.44;

Part 52 ↔ Part 50

U.S.NRC
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- **In 52.47 reference is made to the following Part 50 Regs:**
  - 50.36, 50.36a, Technical Specifications
  - 50.48, Fire Protection
  - 50.49, Environmental Qualification
  - 50.60, 50.61, Pressurized Thermal Shock, etc
  - 50.62, Anticipated Transient Without Scram
  - 50.63, Station Blackout
  - 50.150, Aircraft Impact

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Part 50 ↔ Part 52

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- **50.34 Contents of applications; technical information.**
  - 50.34(f) Additional TMI-related items
    - Each applicant for a design certification, design approval, combined license, or manufacturing license under part 52 of this chapter shall demonstrate compliance with the technically relevant portions of the requirements in paragraphs (f)(1) through (3) of this section……

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Part 50 ↔ Part 52

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- **50.34a Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors.**
  - (d) Each application for a combined license under part 52 of this chapter shall include: (1) A description of the equipment and procedures for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems, …

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Part 50 ↔ Part 52

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- **50.34a Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors.**
  - (e) Each application for a design approval, a design certification, or a manufacturing license under part 52 of this chapter shall include:
    - (1) A description of the equipment for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems, under paragraph (a) of this section and…

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Part 50 ↔ Part 52

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- **50.36 Technical specifications.**
  - 50.36(a)(2)
  - Each applicant for a design certification or manufacturing license under part 52 of this chapter shall include in its application proposed generic technical specifications in accordance with the requirements of this section for the portion of the plant that is within the scope of the design certification or manufacturing license application.

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Part 50 ↔ Part 52

U.S.NRC
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- **50.36a Technical specifications on effluents from nuclear power reactors.**
  - 50.36a(2) Each holder of an operating license, and each holder of a combined license after the Commission has made the finding under §52.103(g) of this chapter, shall submit a report to the Commission annually that specifies the quantity of each of the principal radionuclides released to unrestricted areas in liquid and in gaseous effluents during the previous 12 months,

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Part 50 ↔ Part 52

U.S.NRC
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- **50.44 Combustible gas control for nuclear power reactors.**
  - (c) Requirements for future water-cooled reactor applicants and licensees. The requirements in this paragraph apply to all water-cooled reactor construction permits or operating licenses under this part, and to all water-cooled reactor design approvals, design certifications, combined licenses or manufacturing licenses under part 52 of this chapter, any of which are issued after October 16, 2003.

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Part 50 ↔ Part 52

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- **10 CFR 50, Appendix A, General Design Criteria**
  - Introduction - Under the provisions of 10 CFR 52.47, 52.79, 52.137, and 52.157, an application for a design certification, combined license, design approval, or manufacturing license, respectively, must include the principal design criteria for a proposed facility.

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Part 50 ↔ Part 52

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- **Appendix B to Part 50--Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants**
  - Every applicant for a combined license under part 52 of this chapter is required by the provisions of § 52.79 of this chapter to include in its final safety analysis report a description of the quality assurance applied to the design, and to be applied to the fabrication, construction, and testing of the structures, systems, and components of the facility and to the managerial and administrative controls to be used to assure safe operation.

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Part 50 ↔ Part 52

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- **10 CFR 50.46, ECCS Rule**
- **10 CFR 50.48, Fire Protection**
- **10 CFR 50.49, Environmental Qualification of Electric Equipment…**
- **10 CFR 50.61, Pressurized Thermal Shock**
- **10 CFR 50.62, Requirements for (ATWS)**
- **10 CFR 50.63, Station Blackout**

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Module 19

Inspection Tests Analysis and Acceptance Criteria (ITAAC)

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Module 19
Inspection Tests Analysis and Acceptance Criteria (ITAAC)

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Learning Objectives

- Review the definition and purpose of ITAAC: Inspection, Test, Analysis and Acceptance Criteria
- Understand the relationship of ITAAC to Mechanical Codes and Standards

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Topics in this Module

- 10CFR Part 52 and the role of ITAAC
- Differences with 10CFR Part 50
- Incorporation of Codes and Standards into ITAAC
- ITAAC Closure Process and 10CFR Part 52 requirements for fuel load
- Inspection input to ITAAC Closure
- AP 1000 ITAAC Example

10 CFR Part 52 and Role of ITAAC

- ITAAC are proposed by a vendor (e.g. Westinghouse) in their Design Certification Document (DCD) in accordance with the requirements of 52.47 (b) (1)-the content of the application to the NRC
- The application also includes the proposed design information, including applicable Codes and Standards for the design of the facility.
- The most important Codes and Standards and the ITAAC are included in the Tier 1 part of the application. Tier 1 contains the most important information associated with design and operation of the facility.

10 CFR Part 52 and Role of ITAAC

- The purpose of ITAAC are to ensure that the facility will be constructed and operated in accordance with the design certification.
- The ITAAC are reviewed and approved by the NRC staff along with the design information in the application.
- Most of the information in the application is contained in the Tier 2 Section, including information that describes the ITAAC and the Codes in the ITAAC
- The results of the staff review can be found in the staff Safety Evaluation Report. (For AP 1000, this is NUREG-1793)

Difference with Part 50

- ITAAC are only found in 10 CFR Part 52
- ITAAC tie together what must be inspected by the licensee to the design commitments and the safety analysis of the design-- all in one application.
- Part 50 inspections are conducted in accordance with separate procedures (e.g.Watts Bar)

ITAAC

- **There are typically hundreds of ITAAC in one Design Certification Document. They are found in the Tier 1 Section of the DCD along with all other Tier 1 information.**
- **ITAAC are primarily focused on the facility rather than operational programs**
- **Each ITAAC takes the tabular form of a detailed listing of:**
  - the specific design commitment
  - the associated inspection (I), test(T) or analysis(A), that needs to be performed to ensure the commitment has been fulfilled
  - the acceptance criterion(AC) applicable to the inspection or test.

ITAAC

- **It is important to remember that all ITAAC are performed by the licensee.**

- **ITAAC are inspected by the NRC in accordance with the guidance in the Construction Inspection Program**

ITAAC Including Codes and Standards

- **There are many ITAAC which include Codes and Standards.**
- **These ITAAC are "System Based" and include Codes for:**
  - design and construction of components and piping
  - non-destructive examination requirements for the weld integrity in piping and components
  - pressure boundary integrity
  - material fracture toughness
  - seismic qualification
  - environmental qualification
- **Additional information on a particular ITAAC can be found in related Tier 2 information in the DCD or in the applicable NRC staff safety evaluation of the system and/or ITAAC**

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Non-System Based ITAAC

- **Emergency Response Facilities**
- **Human Factors Engineering**
- **Buildings**
- **Initial Test Program**
- **Radiation Monitoring**
- **Reactor Coolant Pressure Boundary Leak Detection**
- **Design Reliability Assurance Program**

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AP 1000 Example: Reactor Coolant System

- **Design Commitment:**
  - Pressure boundary welds in components identified in Table 2.2.4-1 as ASME Code Section III meet ASME Code Section III requirements.
- **Inspection:**
  - Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.
- **Acceptance Criteria**
  - A report exists and concludes that the ASME Code Section III requirments are met for non-destructive examination of pressure boundary welds.

Taken from AP 1000 DCD

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Completing ITAAC

- **After receiving the license, the COL licensee must submit the schedule for completing ITAAC to the NRC (52.99(a))**
- **The licensee must inform NRC that ITAAC have been completed along with information to demonstrate acceptance criteria have been met (52.99(c))**
- **NRC must ensure that the ITAAC are performed (52.99(e))**
- **NRC must notify the public of ITAAC completion (52.99(e))**
- **The Commission makes the finding that all ITAAC have been completed before loading fuel (52.103 (g))**

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ITAAC Closure Process

- As stated previously, the licensee must inform the staff when an ITAAC has been completed
- The NRC staff, working with industry, has issued guidance for the scope and content of information to be submitted by a licensee to demonstrate that an ITAAC has been completed and closed (RG 1.215)
- An important part of the closure process are the ITAAC inspections done in the field by the NRC
- These inspections will be conducted in accordance with guidance provided to the inspectors as part of the Construction Inspection Program

NRC Construction Inspection Program

- The primary element of the Construction Inspection Program applicable to this course is the inspection of ITAAC associated with Mechanical Codes and Standards
- Inspection Strategies are Developed by the NRC staff to focus inspections on the most important areas of facility construction. These areas are determined by an integrated process that considers such attributes as safety significance (informed by risk assessments), potential for error, and opportunity to check the ITAAC later in the process through other inspection means.
- These inspections of "Targeted" ITAAC will ensure that design commitments, including those with Codes and Standards, have been carried out during the construction of the plant
- The "AC", or acceptance criteria in the ITAAC, will be tied to the Code information in Tier 1 and amplified in Tier 2 of the DCD.

Sampling Methodology

- The attributes listed on the previous slide that were used to inform the sampling methodology were based on expert judgment with the objective that inspections would be in areas that had the most value
- The most important inspections (highest value) were those that would be most likely to discover significant problems that could otherwise have gone undetected.
- Another necessary element of the sampling approach was to group all of the ITAAC activities into common groups. These groups are called families.
- Sampling can be used for common activities (families) but may not be appropriate for different activities.
- All ITAAC families are inspected to some degree.

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ITAAC Inspection

- ITAAC are grouped into approximately 70 families that are defined by a matrix that includes all plant equipment and all activities
- The ITAAC that have the most value in each family and that must be inspected are referred to as "Targeted" ITAAC. These are the ITAAC that are sampled.
- Judgment was used by the NRC to establish how many Targeted ITAAC were needed in each family. This is a balance between value and available resources.
- SECY 07 0047 goes through this in detail. Overall about 30-40% of ITAAC are inspected.

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ITAAC MATRIX

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...excerpt slide from an NRC / Region II presentation in June 2009 by NRC staff (at the ASME NTS-1 in Atlanta)

	A) As-Built Inspectiion	B) Welding	C) Construction Testing	D) Operations Testing	E) Qualification Criteria	F) Design/ Fabrication Requirements
01) Foundations & Buildings	A01	B01	C01	D01	E01	F01
02) Structural Concrete	A02	B02	C02	D02	E02	F02
03) Piping	A03	B03	C03	D03	E03	F03
04) Pipe Support & Restraints	A04	B04	C04	D04	E04	F04
05) Reactor Pressure Vessel & Internals	A05	B05	C05	D05	E05	F05
06) Mechanical Components	A06	B06	C06	D06	E06	F06
07) Valves	A07	B07	C07	D07	E07	F07
08) Electrical Components & Systems	A08	B08	C08	D08	E08	F08
09) Electrical Cable	A09	B09	C09	D09	E09	F09
10) I&C Components & Systems	A10	B10	C10	D10	E10	F10
11) Containment Integrity & Pen's	A11	B11	C11	D11	E11	F11
12) HVAC	A12	B12	C12	D12	E12	F12
13) Equipment Handle & Fuel Racks	A13	B13	C13	D13	E13	F13
14) Complex Sys w/ Multi-Comp	A14	B14	C14	D14	E14	F14
15) Fire Protection	A15	B15	C15	D15	E15	F15
16) Engineering	A16	B16	C16	D16	E16	F16
17) Security	A17	B17	C17	D17	E17	F17
18) EP	A18	B18	C18	D18	E18	F18
19) Radiation Protection	A19	B19	C19	D19	E19	F19

ITAAC Example

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… excerpt slide from an NRC / Region II presentation in June 2009 by NRC Staff (at the ASME NTS-1 in Atlanta)

TABLE 3.3-6 (BUILDINGS)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.a) Exterior walls and the basemat of the nuclear island have a water barrier up to site grade.	An inspection of the as-built exterior walls and the basemat of the nuclear island <u>up to floor elevation 100'-0",</u> for application of water barrier will be performed during construction before the walls are poured	A report exists that confirms that a water barrier exists on the nuclear island exterior walls up to site grade.

- **ITAAC= ITA + AC**
- **Standard 3 column format**
  - Design commitment (from design documents)
  - **Inspections, Tests, Analyses** (method)
  - **Acceptance criteria** (if met shows meeting the design commitment)

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NRC Inspection of ITAAC

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- **NRC is developing guidance for inspectors**
- **Guidance is expected to include:**
  - Which ITAAC to inspect
  - Applicable inspection procedures
  - What type of observations need to be made (e.g. measurement, walk-down, review)
  - What structures, components to inspect
  - Duration of the inspection
  - Particular planning considerations for the inspection
  - Need for interface with headquarters experts

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Summary

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- **Codes and Standards are explicitly identified in the Design Certification Document Tier 1 information and in the ITAAC for each of the certified designs**
- **The licensee is responsible for implementing all ITAAC**
- **This licensing and design bases for reactor designs, including particular Codes and Standards, may differ due to the age of the design certification**
- **NRC inspections will be conducted on the most important construction activities, including application of Codes and Standards**
- **Guidance will be provided for these inspections.**
- **All New Reactor Licensing and Design Certification Information is available under "New Reactors" on NRC's public website nrc.gov**

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Module 20

Nuclear Air and Gas Treatment

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 20
Nuclear Air and Gas Treatment

Instructor: C. Wesley Rowley, P.E.

Learning Objectives

U.S.NRC United States Nuclear Regulatory Commission *Protecting People and the Environment*

- **Learn how the ASME AG-1 Code is applied to Nuclear HVAC Systems.**
- **Learn correct application of ASME N509, N510, and N511 Standards to nuclear air & gas treatment equipment.**
- **Learn the important attributes for nuclear air & gas treatment preservice acceptance testing, including ALARA.**
- **Learn the variety of testing techniques available for the acceptance test program.**

Typical Containment Structure

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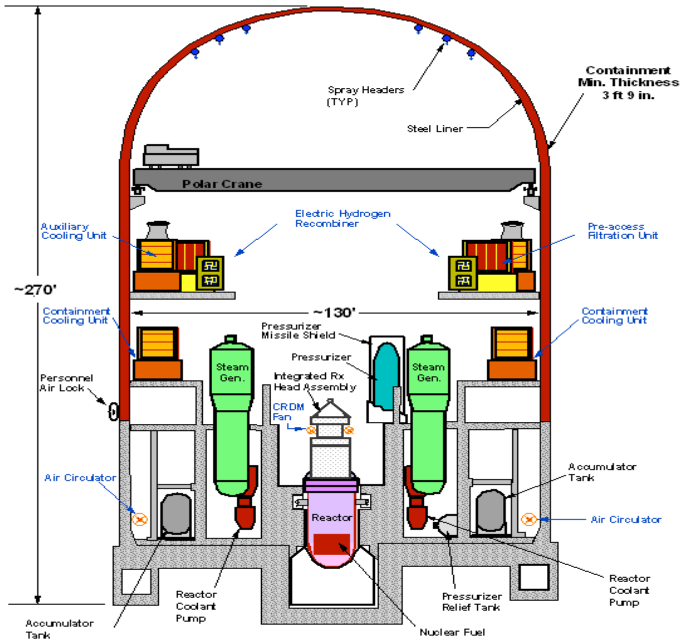

3

Lesson References

U.S.NRC
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- **ASME Standards**
  - ASME AG-1 Code
  - ASME N509 Standard
  - ASME N510 Standard
  - ASME N511 Standard
- **ASTM D3803 Standard**
- **NRC Regulatory Guides**
  - NRC RG 1.52
  - NRC RG 1.140

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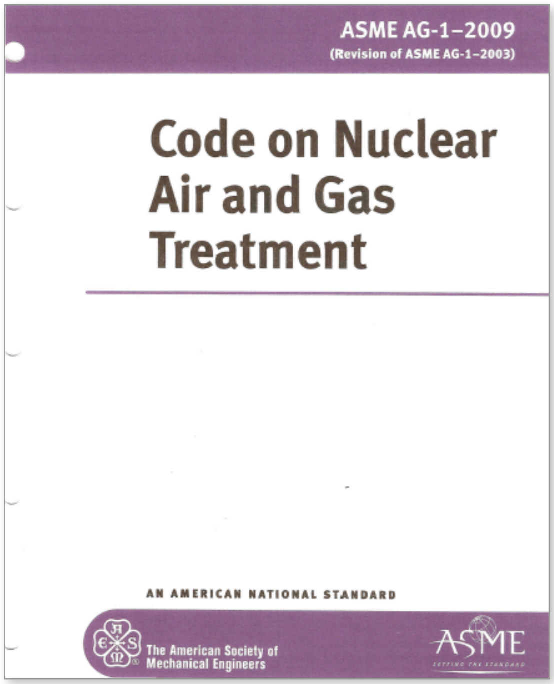

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ASME AG-1-2009 CONTENTS

Division I
General Requirements

Division II
Ventilation Air Cleaning and Ventilation Air Conditioning

Division III
Process Gas Treatment

Division IV
Testing Procedures

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Significant Sub-Topics

U.S.NRC
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- Variety of Nuclear HVAC and Process Gas Equipment
- ASME AG-1 Code Content
- Application of IEEE Class IE Standards
- Other Nuclear Air & Gas Treatment Standards
- Quality Assurance
- Nuclear HVAC Components
- Process Gas Treatment
- Installation Process
- Testing Process
- Testing Techniques
- Typical Problems
- NRC Guidance

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Variety of Nuclear HVAC Equipment

- **Fans & Blowers**
- **Ductwork**
- **Housings**
- **HEPA Filters**
- **Dampers & Louvers**
- **Refrigeration Equipment**
- **Conditioning Equipment**
- **Moisture Separators**
- **Mounting Frames**
- **Filters**
  - Low Efficiency
  - Medium Efficiency
  - HEPA
  - Type II Adsorber Cells
  - Type III Adsorber Cells
  - Metal Media
  - Sand
  - High Strength HEPA
- **Air Cleaning Equipment**
- **Instrumentation & Control**

7

Variety of Nuclear Process Gas Treatment Equipment

- **Pressure Vessels**
- **Piping**
- **Heat Exchangers**
- **Valves**
- **Compressors**
- **Noble Gas Holdup Equipment**
- **Other Radionuclide Equipment**
- **Hydrogen Recombiners**
- **Gas Sampling Equipment**

8

Nuclear Air Treatment Systems And Components

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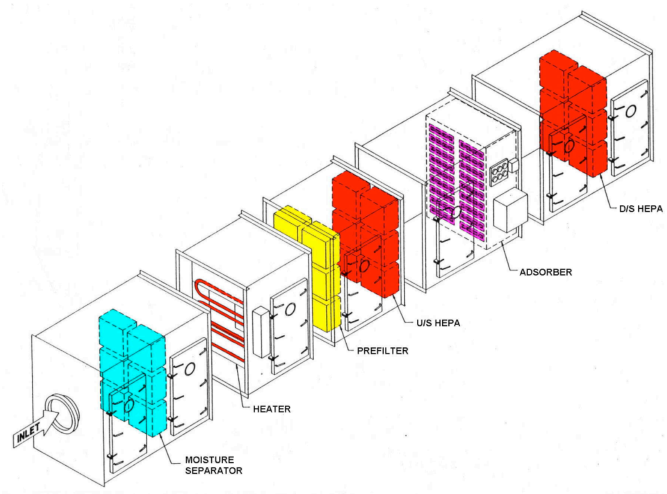

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AG-1 Code Division I General Requirements

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- **Section AA – Common Articles**
  - AA-1000 Introduction
  - AA-2000 Referenced Documents
  - AA-3000 Materials
  - AA-4000 Structural Design
  - AA-5000 Inspection and Testing
  - AA-6000 Fabrication, Joining, Welding, Brazing, Protective Coating, and Installation
  - AA-7000 Packaging, Shipping, Receiving, Storage, and Handling
  - AA-8000 Quality Assurance
  - AA-9000 Nameplates and Stamping
  - AA-10000 Repair and Replacement of Components
- **Non-mandatory Appendices**

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Excerpt From Division I (ASME AG-1 Code)

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- **AA-1100 Scope**
  - Provides minimum 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.
- **AA-1110 Purpose**
  - To assure that equipment used in nuclear facilities for nuclear safety-related air and gas treatment systems is acceptable in all aspects of performance, design, construction, and testing.

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Observations on the ASME AG-1 Code

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- **Code applies only to individual components in a system**
- **Nuclear air and gas equipment requirements differ from commercial equipment requirements**
- **Nuclear HVAC systems handles dangerous radionuclides**
- **Nuclear HVAC systems should maintain their integrity during accidents**

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Nuclear HVAC Construction Process

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- System Design
- Referenced Documents
- Materials
- Structural Design
- Fabrication, Joining, Welding, Brazing, Protective Coating, and Installation
- Packaging, Shipping, Receiving, Storage, and Handling
- Inspection and Testing
- Quality Assurance
- Nameplates and Stamping

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QA Hierarchy of Responsibility

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Owner

↓

Fabricator

↓

Material Supplier

↓

Material Manufacturer

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Source: Based on ASME AG-1 Code Division I

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Quality Assurance

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- **Design Verification Report**
  - Design Verification Stress Report
  - Design Verification Test Report
  - Design Verification by Comparative Evaluation
- **Certificate of Design Verification**
- **Certification of Test Results**
- **Qualifying the Suppliers of Sub-contracted Services and Material**
- **Traceability of Material**
  - Certified Material Test Reports
  - Marking, Nameplates, and Stamping
- **Compliance with ASME NQA-1**

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ASME AG-1 Code Division II
Ventilation Air Cleaning & Conditioning

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- **Section BA – Fans and Blowers**
- **Section DA – Dampers and Louvers**
- **Section SA – Ductwork**
- **Section HA - Housings**
- **Section RA – Refrigeration Equipment**
- **Section CA – Conditioning Equipment**
- **Section FA – Moisture Separators**
- **Section FB – Medium Efficiency Filters**
- **Section FC – HEPA Filters**
- **Section FD – Type II Adsorber Cells**
- **Section FE – Type III Adsorber Cells**
- **Section FF – Adsorber Media**

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ASME AG-1 Code Division II

Ventilation Air Cleaning & Conditioning (cont'd)

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- **Section FG – Mounting Frames, CONAGT Air-Cleaning Equipment, Nuclear Safety-Related Equipment**
- **Section FH – Other Adsorbers**
- **Section FI – Metal Media Filters (in the course of preparation)**
- **Section FJ – Low Efficiency Filters**
- **Section FK – Special Round and Duct-Connected HEPA Filters**
- **Section FL – Sand Filters (in the course of preparation)**
- **Section FM – High Strength HEPA Filters (in the course of preparation)**
- **Section IA – Instrumentation and Control**

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Nuclear HVAC Filtration

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Adsorbent Media – see Article FF-1000

- **Related Terms:**
  - activated carbon
  - batch
  - batch test
  - coimpregnants
  - grade or type
  - impregnated activated charcoal
    - iodides
    - amines
  - lot
  - qualification test
  - virgin activated carbon

Note that ASTM D3803 entitled "Standard Test Methods for Nuclear-Grade Activated Carbon" is the testing standard for this adsorbent media.

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Charcoal Adsorption

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- **ASTM D3803-1989, "Standard Test Methods for Nuclear-Grade Activated Carbon"**

- **Contents**
  1. Scope
  2. References

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ASME AG-1 Code Division III
Process Gas Treatment

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- **Section GA – Pressure Vessels, Piping, Heat Exchangers, and Valves**
- **Section GB – Noble Gas Hold Up Equipment**
- **Section GC – Compressors**
- **Section GD – Other Radionuclide Equipment**
- **Section GE – Hydrogen Recombiners**
- **Section GF – Gas Sampling**

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Process Gas Treatment

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Photo from "nucleartourist.com"

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ASME AG-1 Code Division IV

Testing Procedures

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- Section TA – Field Testing of Air Treatment Systems
- Section TB – Field Testing of Gas Processing Systems
- Section TC – Personnel Qualification
- Section TD – Laboratory Qualification

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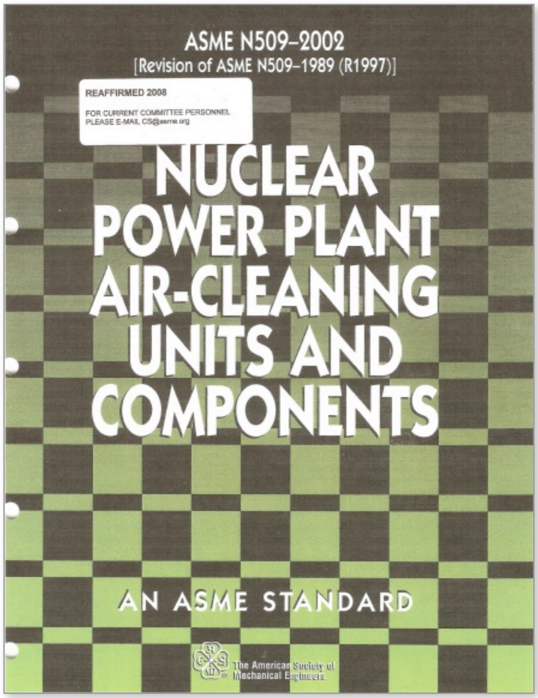
Testing Techniques

- **Visual Inspection**
  - Direct
  - Remote
  - Welded connections
    - Examination
    - Inspection
    - Testing
  - Bolted connections
    - Flange seating surfaces
    - Flange facing
    - Gaskets
    - Bolt torque
  - Fabrication tolerances
- **Pressure and Leak Testing**
- **Performance and Functional Testing**
- **Seismic Testing**
  - Proof testing
  - Fragility testing

Specialized Standards

- **ASME N509 – 2002 Nuclear Power Plant Air-Cleaning Units and Components**
- **ASME N510 – 2007 Testing of Nuclear Air Treatment Systems**
- **ASME N511 – 2007 In-Service Testing of Nuclear Air Treatment, Heating, Ventilating, and Air-Conditioning Systems**
- **ASTM D3803-1989 Standard Test Methods for Nuclear-Grade Activated Carbon**



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ASME N509-2002 CONTENTS

1 Scope
2 Applicable Documents
3 Terms and Definitions
4 Functional Design
5 Components
6 Packaging, Shipping, Receiving, Storage, and Handling of Components
7 Installation and Erection
8 Quality Assurance
9 Acceptance Testing
Mandatory Appendix

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ASME N509-2002 Standard

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4 Functional Design

- 4.1 General
- 4.2 Design Parameters
- 4.3 Size of Air-Cleaning Units
- 4.4 Environmental Design Condition
- 4.5 Structural Load Requirements
- 4.6 Air-Cleaning Units and Components That Must Withstand Fan Peak Pressure
- 4.7 Nuclear Air-Treatment System Configuration and Location
- 4.8 Maintainability Criteria

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ASME N509-2002 Standard (cont'd)

4 Functional Design (cont'd)

4.9 Monitoring of Operational Variables
4.10 Adsorbent Cooling
4.11 Fire Protection
4.12 Insulation
4.13 Testability
4.14 Pressure Boundary Leakage

ASME N509-2002 Standard (cont'd)

5 Components

5.1 HEPA Filters
5.2 Tray-Type Bed and Deep Bed Adsorber Cells
5.3 Prefilters and Postfilters
5.4 Moisture Separators
5.5 Air Heaters
5.6 Filter Housing
5.7 Fans
5.8 Fan Drives
5.9 Dampers
5.10 Ducts

Note – the requirements for many of these system components refer to ASME AG-1 Code … specific Sections.

ASME N509-2002 Standard (cont'd)

9 Acceptance Testing

- Acceptance testing shall be in accordance with ASME AG-1, Section TA and operational testing shall be in accordance with ASME N510.
- It is recommended that prefilters be installed before fan…
- Prefilters may have to be replaced after this evolution.
- For personnel protection, personnel should not enter housing…
- After installing the HEPA filters and adsorbers, the system heaters…
- All damper, valves, and controls shall be exercised…
- After completion of acceptance testing, the system shall be sealed and the fan controls locked out to protect components during the remainder of construction operations at the site.

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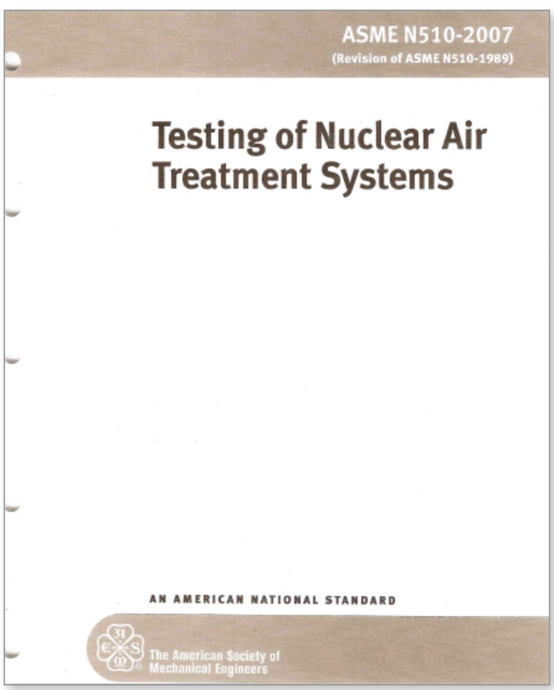

ASME N510-2007 CONTENTS

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6 Duct and Housing Leak Test
7 Airflow Capacity Test
8 Air-Aerosol Mixing Uniformity Test
9 HEPA Filter Bank In-Service Leak Test
10 Adsorber Bank In-Service Leak Test
11 System Bypass Test
12 Air Heater Performance Test
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ASME N510-2007 Standard

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1 SCOPE

- This Standard covers in-service testing of ASME N509 high-efficiency air treatment systems for nuclear power plants.

1.1 Use of This Standard

- This Standard provides a basis for the development of test programs and does not include acceptance criteria…
- Acceptance criteria shall be developed based on the design / function in accordance with ASME N509.
- This Standard is arranged so that the user may select…

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ASME N510-2007 Standard (cont'd)

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1.2 Limitations of This Standard

- This Standard covers the in-service (operational) testing of installed air treatment systems.
- This Standard shall be applied in its entirety to systems designed and built to ASME N509 specifications.
- Sections of this Standard may be used for technical guidance for testing air treatment systems designed to other criteria.
- ASME AG-1, Section TA covers the acceptance-testing program.

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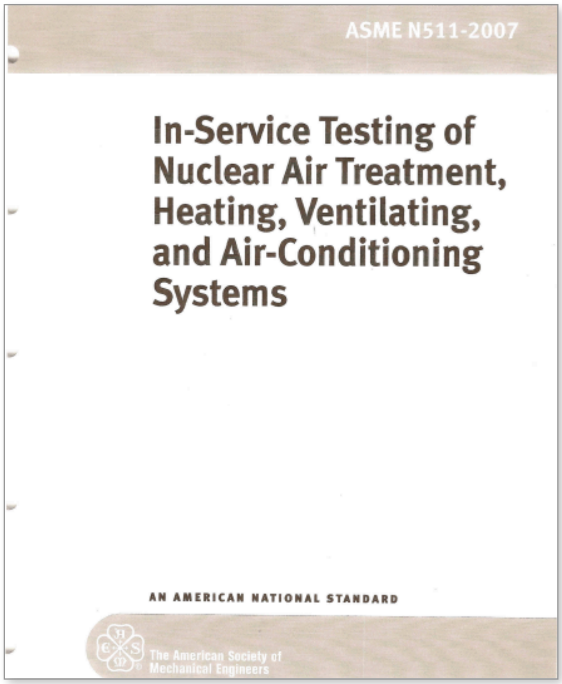
ASME N511-2007

In-Service Testing of Nuclear Air Treatment, Heating, Ventilating, and Air-Conditioning Systems

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ASME N511-2007
CONTENTS

ASME N511-2007 Standard

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1 INTRODUCTION

1.1 Scope

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

1.2 Purpose

- The purpose of this Standard is to provide requirements for in-service testing, the results of which are used to verify that the nuclear air treatment, heating, ventilating, and air-conditioning systems perform their intended function.

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ASME N511-2007 Standard (cont'd)

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1.3 Applicability

- This Standard applies to the in-service testing of nuclear air treatment, heating, ventilating, and air conditioning systems that have been designed, built, and acceptance tested in accordance with ASME AG-1.

- Sections of this Standard may be used for technical guidance when testing air treatment, heating, ventilating, and air-conditioning systems designed and built to other standards.

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NRC Guidance

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- **RG 1.52 Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants**
  - Redundant trains recommended
  - Physical separation recommended
  - Component protection devices (i.e., SRVs) recommended
  - Seismic Category I recommended
  - Design for harsh environment recommended
  - Limit air flow in each clean-up unit to 30,000 cfm
  - Use IEEE Std 603 for all instrumentation and controls
  - Design for ALARA
  - Minimize intake of on-site containments (i.e., D-G exhaust)

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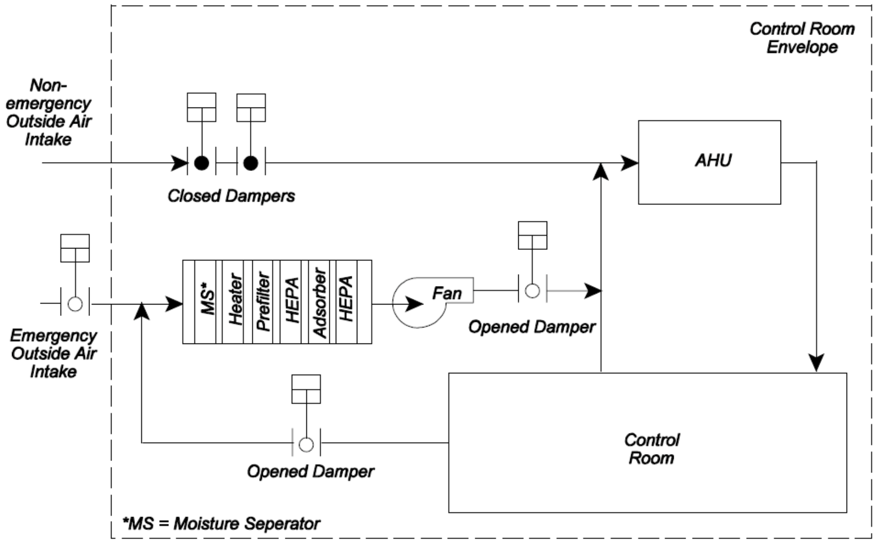
Example of a Control Room ESF Atmosphere Cleanup Train



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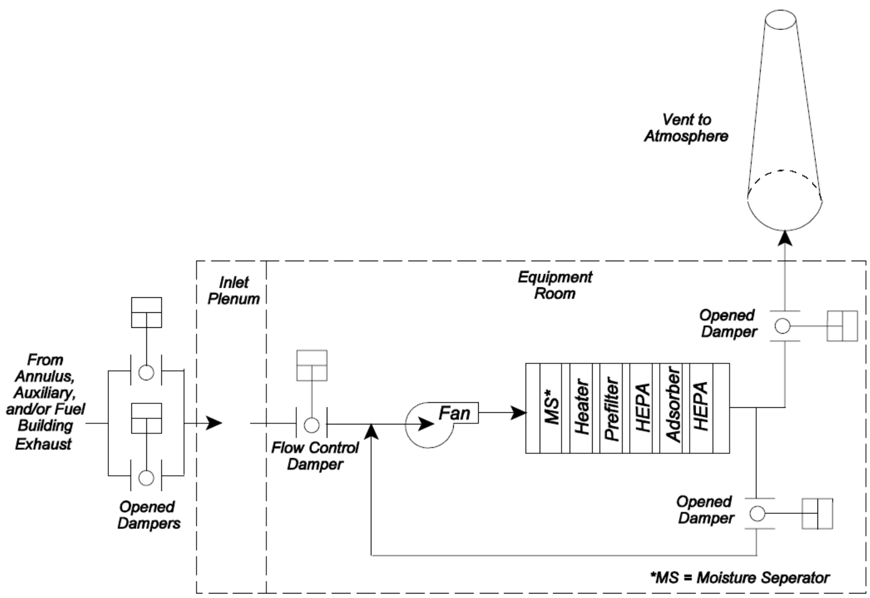
Example of a Shield, Annulus, and/or Fuel Building ESF Atmosphere Cleanup Train



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NRC Guidance (continued)

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- **RG 1.140 Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmospheric Cleanup Systems in Light-Water-Cooled Nuclear Power Plants**
  - For systems designed for normal plant operations
  - Does not apply to post-accident ESF systems
  - Particulate filtration and radioiodine adsorption units are included in the design for these systems

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Typical Problems

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- **Industrial contaminates and pollutants**
- **High temperature**
- **High relative humidity**
- **Condensation of moisture**
- **Poor layout for accessibility and working space**
- **Auto ignition of the adsorbent due to radiation induced heat**
- **Use of adsorbent not "qualified" per ASTM Std 3803**

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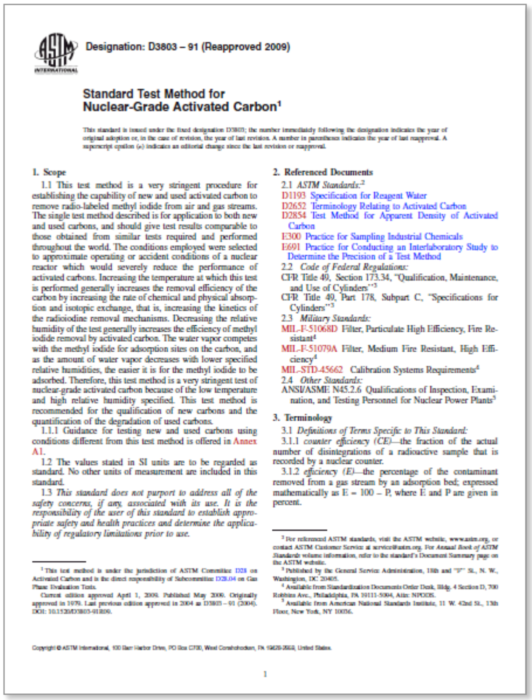

Designation: D3803 – 91 (Reapproved 2009)

Standard Test Method for Nuclear-Grade Activated Carbon[1]

This standard is issued under the fixed designation D3803; the number immediately following the designation indicates the year of original adoption or, in the case of revision, the year of last revision. A number in parentheses indicates the year of last reapproval. A superscript epsilon (ε) indicates an editorial change since the last revision or reapproval.

1. Scope

1.1 This test method is a very stringent procedure for establishing the capability of new and used activated carbon to remove radio-labeled methyl iodide from air and gas streams. The single test method described is for application to both new and used carbons, and should give test results comparable to those obtained from similar tests required and performed throughout the world. The conditions employed were selected to approximate operating or accident conditions of a nuclear reactor which would severely reduce the performance of activated carbons. Increasing the temperature at which this test is performed generally increases the removal efficiency of the carbon by increasing the rate of chemical and physical absorption and isotopic exchange, that is, increasing the kinetics of the radioiodine removal mechanisms. Decreasing the relative humidity of the test generally increases the efficiency of methyl iodide removal by activated carbon. The water vapor competes with the methyl iodide for adsorption sites on the carbon, and as the amount of water vapor decreases with lower specified relative humidities, the easier it is for the methyl iodide to be adsorbed. Therefore, this test method is a very stringent test of nuclear-grade activated carbon because of the low temperature and high relative humidity specified. This test method is recommended for the qualification of new carbons and the quantification of the degradation of used carbons.

1.1.1 Guidance for testing new and used carbons using conditions different from this test method is offered in Annex A1.

1.2 The values stated in SI units are to be regarded as standard. No other units of measurement are included in this standard.

1.3 *This standard does not purport to address all of the safety concerns, if any, associated with its use. It is the responsibility of the user of this standard to establish appropriate safety and health practices and determine the applicability of regulatory limitations prior to use.*

2. Referenced Documents

2.1 *ASTM Standards:*[2]

D1193 Specification for Reagent Water

D2652 Terminology Relating to Activated Carbon

D2854 Test Method for Apparent Density of Activated Carbon

E300 Practice for Sampling Industrial Chemicals

E691 Practice for Conducting an Interlaboratory Study to Determine the Precision of a Test Method

2.2 *Code of Federal Regulations:*

CFR Title 49, Section 173.34, "Qualification, Maintenance, and Use of Cylinders"[3]

CFR Title 49, Part 178, Subpart C, "Specifications for Cylinders"[3]

2.3 *Military Standards:*

MIL-F-51068D Filter, Particulate High Efficiency, Fire Resistant[4]

MIL-F-51079A Filter, Medium Fire Resistant, High Efficiency[4]

MIL-STD-45662 Calibration Systems Requirements[4]

2.4 *Other Standards:*

ANSI/ASME N45.2.6 Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Power Plants[5]

3. Terminology

3.1 *Definitions of Terms Specific to This Standard:*

3.1.1 *counter efficiency (CE)*—the fraction of the actual number of disintegrations of a radioactive sample that is recorded by a nuclear counter.

3.1.2 *efficiency (E)*—the percentage of the contaminant removed from a gas stream by an adsorption bed; expressed mathematically as E = 100 – P, where E and P are given in percent.

[1] This test method is under the jurisdiction of ASTM Committee D28 on Activated Carbon and is the direct responsibility of Subcommittee D28.04 on Gas Phase Evaluation Tests.

Current edition approved April 1, 2009. Published May 2009. Originally approved in 1979. Last previous edition approved in 2004 as D3803 – 91 (2004). DOI: 10.1520/D3803-91R09.

[2] For referenced ASTM standards, visit the ASTM website, www.astm.org, or contact ASTM Customer Service at service@astm.org. For *Annual Book of ASTM Standards* volume information, refer to the standard's Document Summary page on the ASTM website.

[3] Published by the General Service Administration, 18th and "F" St., N. W., Washington, DC 20405.

[4] Available from Standardization Documents Order Desk, Bldg. 4 Section D, 700 Robbins Ave., Philadelphia, PA 19111-5094, Attn: NPODS.

[5] Available from American National Standards Institute, 11 W. 42nd St., 13th Floor, New York, NY 10036.

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ASTM D3803-1991 (2009)
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1. Scope
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3. Terminology
4. Summary of Test Method
5. Significance and Use
6. Apparatus
7. Materials
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10. Preparation of Apparatus
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Case Study on Nuclear Air & Gas Treatment

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- **Problem: What makes Nuclear Air & Gas Treatment SSCs different for a NPP?**
  - What kind of air / gas is contained in these systems?
  - What kind of radioisotopes are processed in "nuclear air" versus "nuclear gas"?
  - What are some of the issues that need to be dealt with in these SSCs?

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Learning Questions

- How is the ASME AG-1 Code applied to Nuclear HVAC Systems?
- What is the correct application of the ASME N509, N510, and N511 Standards to nuclear air & gas treatment equipment?
- What are the important attributes for nuclear air & gas treatment preservice acceptance testing, including ALARA?
- Name the variety of testing techniques available for the acceptance test program.

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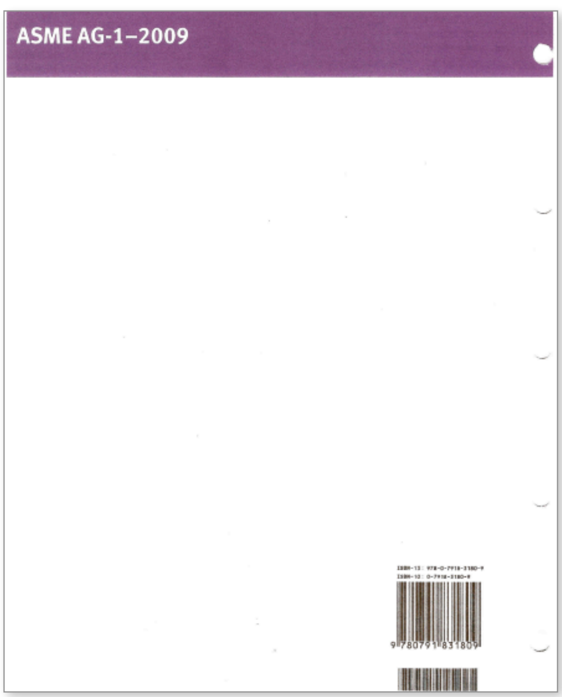

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Note - the ASME AG-1 Code is the primary reference for Nuclear Air & Gas Treatment Components.

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Module 21
Nuclear Cranes

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Module 21
Nuclear Cranes

Instructor: C. Wesley Rowley, P.E.

Learning Objectives

- **Learn the difference between "overhead" and "underhung" cranes.**
- **Learn what constitutes a "nuclear crane" from an "industrial" crane.**
- **Learn how to use a plant nuclear crane during construction.**

Significant Sub-topics

- Overview of Crane Standards
- Cranes Used in NPPs
- Types of Nuclear Cranes
- Organization of NOG-1 and NUM-1 Codes
- Common Crane Accidents
- Single Failure Proof Concept
- Nuclear Crane Commissioning
- Shop Testing
- Quality Assurance
- Construction Process
- Acceptance Testing
- Construction Use / Recertification

3

Overview of Crane Standards

- **ASME Nuclear Crane Standards**
  - ASME NOG-1 Standard, *"Rules for Construction of Overhead Gantry Cranes (Top Running Bridge, Multiple Girder)"*
  - ASME NUM-1 Standard, *"Rules for Construction of Cranes, Monorails and Hoists (with Bridge, or Trolley, or Hoist of the Underhung Type)"*
- ***ASME Nuclear Crane* Guide**
  - ASME STP-NU-015-2008, *"A Guide to American Crane Standards"*
- **Other ASME Crane Standards**

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Other Crane Standards

- ASME B30.2, Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)
- ASME B30.10, Hooks
- ASME B30.11, Monorails and Underhung Cranes
- ASME B30.16, Overhead Hoists (Underhung)
- ASME B30.17, Overhead and gantry Cranes (Top Running Bridge, Single Girder, Underhung Hoist)

Additional References

- NUREG-0554, Single-Failure-Proof Cranes for Nuclear Power Plants
- Whiting Crane Handbook, 4th Edition, ©1979
- OSHA Standard 29CFR1910.179

Cranes Used in NPPs

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- **Construction Cranes**
- **Plant Cranes**
  - Fabrication of Plant Cranes
  - Installation of Plant Cranes
  - Testing of Plant Cranes
  - Use of Plant Cranes During Construction
  - Use of Plant Cranes During Startup
  - Normal Use of Plant Cranes During Operation and Maintenance Throughout Life of the Plant

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Construction Cranes (non-nuclear)

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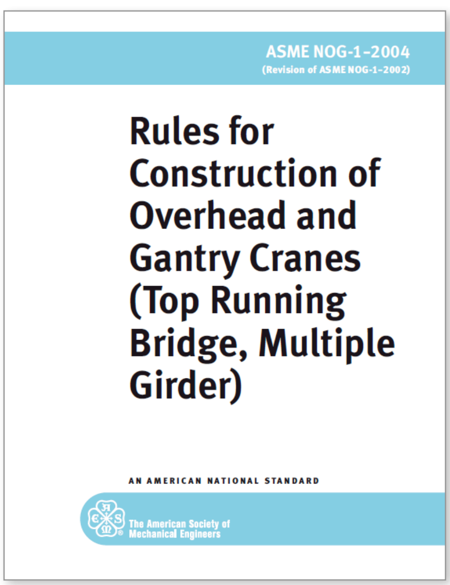
Types of Nuclear Cranes

- **Type I Nuclear Cranes – equipment that is used to handle critical loads and is required to withstand a seismic event**
  - Type IA – single failure proof features
  - Type IB – enhanced safety features
- **Type II Nuclear Cranes – equipment that is not used to handle critical loads and is required to withstand a seismic event**
- **Type III Nuclear Cranes – equipment that is not used to handle critical loads and is not required to withstand a seismic event**

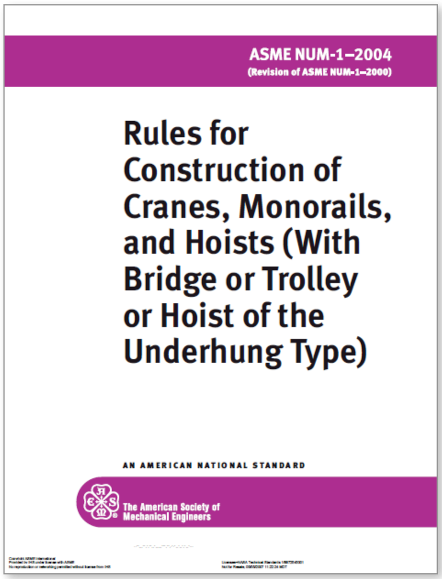

ASME NOG-1-2004
(Revision of ASME NOG-1-2002)

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

AN AMERICAN NATIONAL STANDARD

The American Society of Mechanical Engineers

ASME NOG-1-2004
CONTENTS


ASME NUM-1–2004
(Revision of ASME NUM-1–2000)

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

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Common Crane Accidents

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- **Where is the crane vulnerable to failure?**
- **How does a crane fail?**
- **How do most crane accidents occur?**
- **What safety systems can be added to the crane to help protect against these failures?**

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Common Crane Accidents (cont'd)

- **Most crane accidents are the result of the following:**
  - Over Capacity Lift
  - Two Blocking
  - Load Hang-up
  - Misreeving
  - Component Failure
- **For additional information on common crane accidents refer to the technical paper <u>ICONE-8077, X-Sam, The Single Failure Proof Crane System,</u> by Jim Nelson.**

Common Crane Accidents (cont'd)

- **Two-Blocking**
  - Two-blocking is the result of hoisting beyond the intended safe upper limit of hook travel to the point of solid contact between the load block and the upper block or hoist/trolley structure.
  - The usual result is immediate failure of the wire rope, due to overstress from rotational inertia or from the ropes being cut by the grooves of the drum or sheaves.

Common Crane Accidents (cont'd)

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- **Load Hang-up**
  - Load hang-up is entanglement or snagging of the load or other abrupt prevention of further motion of the hoist, trolley or bridge after the load has begun moving.
  - The closer the motor is to full speed, the more severe the effects of the hang-up will be.

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Common Crane Accidents (cont'd)

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- **Mis-spooling**
  - Mis-spooling means the wire rope leaves its machined groove on the rope drum, crossing over the groove "ridges" and piling up over other wraps of wire rope.
  - This can result in severe damage or rope failure, especially if the rope is crushed between the drum and its supporting structure.

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Single-Failure-Proof Concept

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- **The Standard Electric Hoist**
  - The hook is supported by a reeving system using a single wire rope.
  - The wire rope is wrapped around a wire rope drum.
  - The drum is driven by a low speed output shaft from a gear box
  - The high speed shaft of the gear reducer is coupled to both the electric motor and holding brake.
- **The single-failure-proof concept:**
  - What happens when a single component fails?
  - Will this single component failure cause the load to fall?
- **Only "credible" component failures are addressed.**

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Nuclear Crane Commissioning

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- **Crane Quality Assurance**
- **Shop Testing prior to delivery**
- **Shipment / Installation**
- **Site Preparation**
- **Acceptance Testing**
- **Construction Use**
- **Planned Engineered Lifts**

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Shop Testing

- **Shop testing requirements for the crane prior to shipment:**
  - ASME NOG-1, Section 7250
  - ASME NUM-I-8000 and NUM-III-8000
- **The Contractor is required to complete this testing in his shop using an approved procedure.**
- **The crane should be partially assembled and checked for proper dimensions, alignment, squareness and clearance.**

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Quality Assurance

- **The fabrication of critical components must comply with a Quality Assurance program described in:**
  - ASME NOG-1, Section 2000
  - ASME NUM-G-5000
- **The Code also requires the crane manufacturer to meet basic and supplemental requirements of ASME NQA-1 Quality Assurance Program.**

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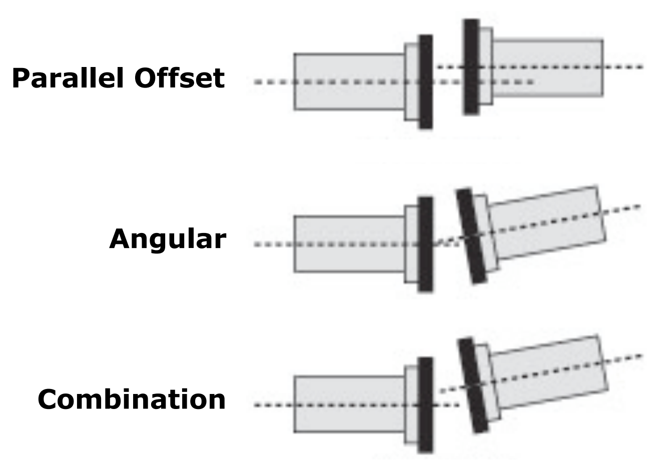
Shipment / Installation

- ASME NOG-1, Sections 7000 and 8000 cover requirements for shipment and installation.
- QA requirements are also invoked.

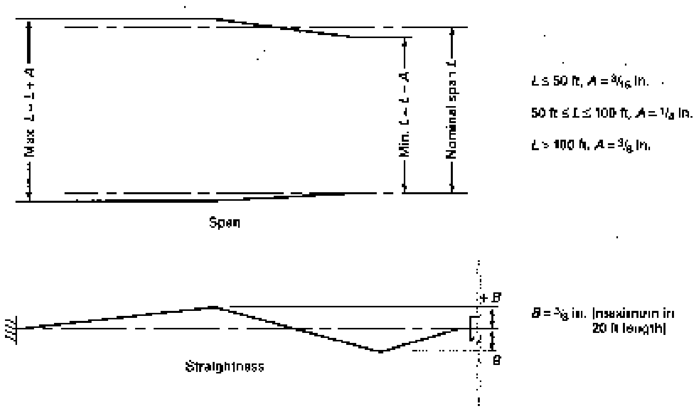
Coupling Alignment



Site Preparation

- Runway alignment must not be overlooked. This is usually done by the general contractor rather than the crane manufacturer.
- Runway needs to be correctly shimmed to assure flat runway splices.
- Alignment tolerances are provided in NOG-1 Fig. 4160-1 and Table NUM-III-8214.1. This includes runway Span, Straightness, Elevation, and Rail-to-Rail Elevation.
- The crane runway should be independently surveyed to assure alignment tolerances are provided.

Runway Alignment Tolerances



Runway Alignment Tolerance

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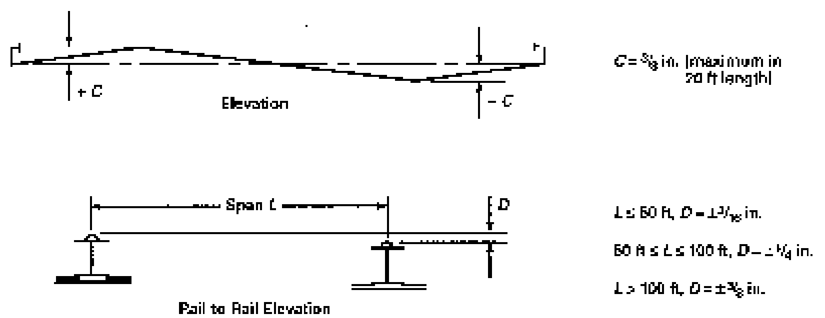

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Site Preparation (cont’d)

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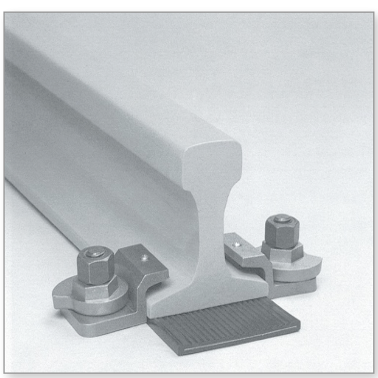

- **Forged Steel Rail Clips**
  - Lateral adjustment
  - Positive lateral rail restraint
  - Reduces crane skewing effects
  - Self-locking

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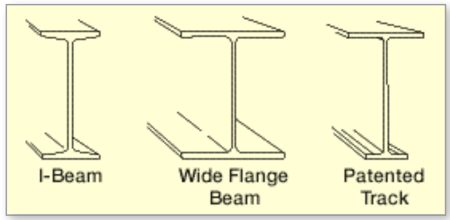
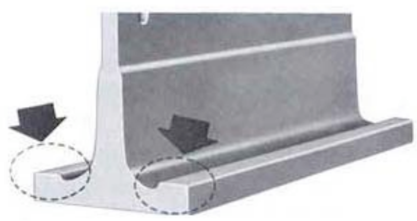
Site Preparation (cont'd)

- **Patented Track**
  - Special hardened steel bottom flange with raised tread to provide a perfect rolling surface.
  - The hardened steel T-section is welded to the web to complete the patented track runway beam.
  - A standard I-beam and wide-flange beam is rolled in one piece. This soft metal material provides a poor rolling surface.
  - Patented track is designed for high-repetition operations in harsh environments, and will provide a clean precise rolling surface. Especially if contamination is a concern!





Acceptance Testing

- **ASME NOG-1, Section 7000 covers acceptance testing.**
  - specifies using an approved procedure.
  - includes no-load testing, 100% load testing and 125% load testing.
  - provides complete functional testing of the crane at no load and full load.
  - provides for a rated load test at 125% of the capacity of the crane (to assure safety margins are present).
- **ASME NUM-1**

Commercial Motor Drive

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- **Flux Vector drive Safety Considerations**
  - Motor torque proving at start
  - Roll back detection
  - Brake check at start
  - Speed deviation detection
  - Over speed protection
  - Brake check at stop
  - Encoder check at load float
- **Exceptional motor performance while using a lower cost AC squirrel cage motor.**
- **In addition this system provides a lot of internal enhanced safety features.**

Flux Vector Drive

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Construction Use

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- **Temporary use of the plant crane during construction after installation is discussed in ASME NOG-1 Section 7430.**
- **Conditions for use, inspection and recertification after use during construction are addressed.**
- **Temporary use of the plant crane during construction after installation is not addressed in ASME NUM-1.**

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Planned Engineered Lifts

- **NOG-1, Section 9000 Planned Engineered Lifts covers "special engineered lifts" in excess of the crane rated capacity.**
- **Capacity limitations (Ref NOG Section 9200) and frequency limitations (Ref NOG Section 9300) are provided.**

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Planned Engineered Lifts (cont'd)

- Never make a lift in excess of the crane rated capacity unless the crane design is checked to assure this lift can be made safety without damaging the crane.
  - Brakes will be marginalized.
  - Machinery alignment may be compromised.
  - Crane or building structure issues may be present that could require restricted movement.
  - Additional inspection and load testing is required.

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Planned Engineered Lifts (cont'd)

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Crane design margins are not provided to allow an overcapacity lift.

The crane manufacturer or a qualified engineer must review this first.

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Planned Engineered Lifts (cont'd)

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- **Excerpt from the Whiting Crane Handbook:**

  Section VI, Part A, Design Factor

  The "design factor" is often incorrectly called "factor of safety" in crane specifications. The term "factor of safety" is misleading in that it implies a level of protection greater than actually exists. It should not be used.

  The "design factor" is a broader term in that it includes consideration of life expectancy and material characteristics as well as stress levels. The use of a design factor provides a margin to allow for variations in the properties of materials, manufacturing tolerances, operating conditions and design assumptions. Under no condition does it imply authorization or protection for users to load the crane beyond the rated load. Such practice is in violation of OSHA Standard 29CFR1910.179 and represents hazardous operation.

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Defined Load Path

- **In most industrial applications, the load path is obvious.**
- **In many NPP applications the load path is not obvious, thus it needs to be planned:**
  - Proximity to the nearby SSCs
  - Proximity to other crane activities
  - Assess the risk of a load drop on critical component (how minimize the consequences)

Case Study on ASME B30 Crane

- **What do we have to do to a “B30 Crane” to make it a “nuclear crane”?**
  - Design
  - Fabrication
  - Installation
  - Testing
  - Operations & Maintenance

Learning Questions

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- **What is the difference between "overhead" and "underhung" cranes?**
- **What constitutes a "nuclear crane" from an "industrial" crane?**
- **How is a plant nuclear crane used during construction?**

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Module 22
ASME B31.1 Power Piping

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Module 22
ASME B31.1 Power Piping

Instructor: C. Wesley Rowley, P.E.

Learning Objectives

- **Learn how the ASME B31.1 Code is used in NPPs.**
- **Learn similarities and differences between the ASME B31.1 Code and ASME BPV Code Section III.**
- **Learn typical systems in a NPP that are designed and constructed using the requirements of ASME B31.1 Code.**

Significant Sub-topics

- Scope and Definitions
- Design
- Materials
- Dimensional Requirements
- Fabrication, Assembly, and Erection
- Inspection, Examination, and Testing
- Mandatory Appendices
- Non-mandatory Appendices
- Technical / Code Inquiries
- Code Cases

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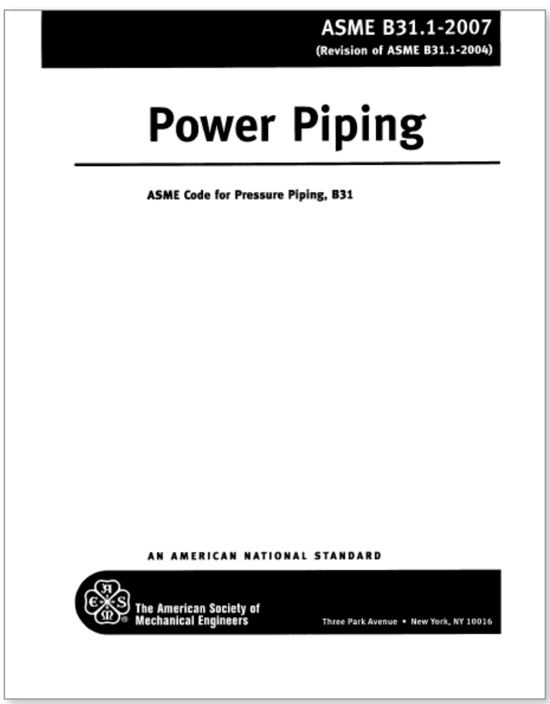
ASME B31.1-2007
(Revision of ASME B31.1-2004)

Power Piping

ASME Code for Pressure Piping, B31

AN AMERICAN NATIONAL STANDARD

The American Society of Mechanical Engineers

Three Park Avenue • New York, NY 10016

ASME B31.1-2007
CONTENTS

1. Scope and Definitions
2. Design
3. Materials
4. Dimensional Requirements
5. Fabrication, Assembly, and Erection
6. Inspection, Examination, and Testing

Mandatory Appendices
Nonmandatory Appendices
Index

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Chapter I Scope and Definitions

- **100 General**
  - 100.1 Scope
    - This Code prescribes requirements for the design, materials, fabrication, erection, test, and inspection of piping systems.
    - Piping system as used in this Code includes pipe, flanges, bolting, gaskets, valves, relief devices, fittings, and the pressure containing portion of other piping components …

Chapter I (cont'd)

- **100 General (cont'd)**
  - 100.2 Definitions
    - Some commonly used terms relating to piping are defined in this Paragraph.
    - Terms related to welding generally agree with AWS A3.0 Standard.
    - Some welding terms are defined with specified reference to piping.
    - For welding terms used in this Code, but not shown here, definitions of AWS A3.0 apply.
    - There are 88 terms defined in this Paragraph.

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Partial Listing of Defined Terms

- Anchor
- Assembly
- Branch connection
- Component
- Equipment connection
- Erection
- Fabrication
- Inspection
- Maximum allowable stress
- Maximum allowable working pressure
- Mechanical joint
- Nominal thickness
- Pipe and tube
- Pipe supporting elements
- Restraint
- Shall / should / may
- Steel
- Supplementary steel

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Chapter II Design

- Part 1 Conditions and Criteria
- Part 2 Pressure Design of Piping Components
- Part 3 Selection and Limitations of Piping Components
- Part 4 Selection and Limitations of Piping Joints
- Part 5 Expansion, Flexibility, and Pipe Supporting Element
- Part 6 Systems

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Part 1 Conditions and Criteria

101 Design Conditions

101.1 General
101.2 Pressure
101.3 Temperature
101.4 Ambient Influences
101.5 Dynamic Effects
101.6 Weight Effects
101.7 Thermal Expansion and Contraction Loads

Part 1 Conditions and Criteria

- **102 Design Criteria**

102.1 General
102.2 P-T Ratings for Piping Components
102.3 Allowable Stress Values and Other Stress Limits
102.4 Allowances
- 102.4.1 Corrosion or Erosion
- 102.4.2 Threading and Grooving
- 102.4.3 Weld Joint Efficiency Factors
- 102.4.4 Mechanical Strength
- 102.4.5 Bending
- 102.4.6 Casting Quality Factors

Excerpt from ASME B31.1

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Fig. 102.4.5 Nomenclature for Pipe Bends

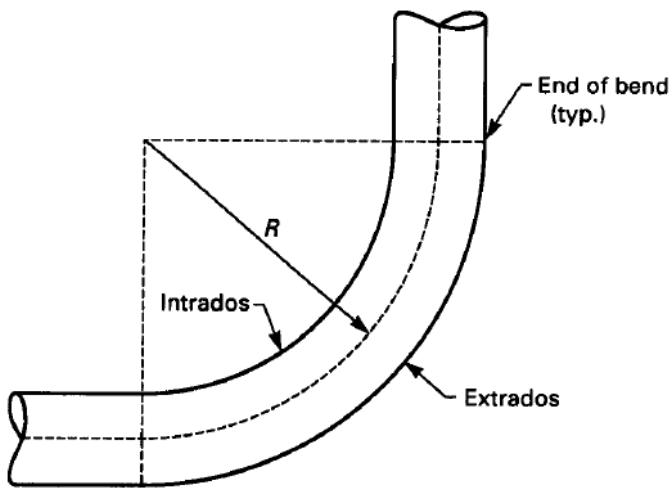

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Table 102.4.5

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Table 102.4.5 Bend Thinning Allowance

Radius of Bends	Min. Thickness Recommended Prior to Bending
6 pipe diameters or greater	$1.06t_m$
5 pipe diameters	$1.08t_m$
4 pipe diameters	$1.14t_m$
3 pipe diameters	$1.25t_m$

GENERAL NOTES:

(a) Interpolation is permissible for bending to intermediate radii.

(b) t_m is determined by eq. (3) or (3A) of para. 104.1.2(A).

(c) Pipe diameter is the nominal diameter as tabulated in ASME B36.10M, Tables 1, and ASME B36.19M, Table 1. For piping with a diameter not listed in these Tables, and also for tubing, the nominal diameter corresponds with the outside diameter.

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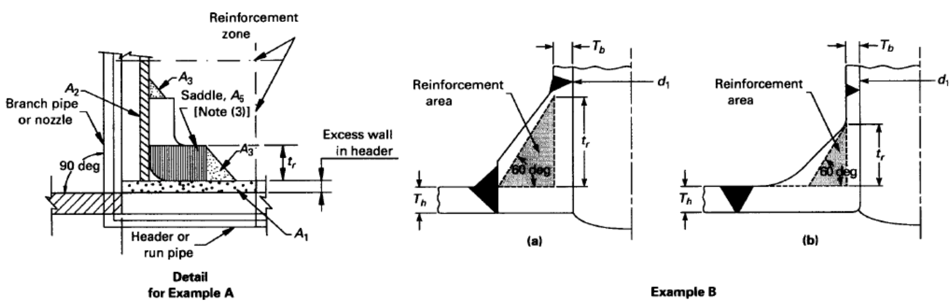
Design and Fabrication Details (from ASME B31.1)

Fig. 104.3.1(D) Reinforcement of Branch Connections (Cont'd)



Another Design Portion

- **Part 5 Expansion, Flexibility, and Pipe Supporting Element**
  - 119 Expansion and Flexibility
    - 119.1 General
    - 119.2 Stress Range
    - 119.3 Local Overstrain
    - 119.5 Flexibility
    - 119.6 Properties
    - 119.7 Analysis
    - 119.8 Movements
    - 119.9 Cold Spring
    - 119.10 Reactions

Another Design Portion (cont'd)

- **Part 5 Expansion, Flexibility, and Pipe Supporting Element**
  - 120 Loads on Pipe Supporting Elements
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    - 120.2 Supports, Anchors, and Guides
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      - 120.2.3 Anchors or Guides
      - 120.2.4 Supplementary Steel
        - » Designed in accordance with AISC (American Institute of Steel Construction) specifications

Another Design Portion (cont'd)

- **Part 5 Expansion, Flexibility, and Pipe Supporting Element**
  - 121 Design of Pipe Supporting Elements
    - 121.1 General
    - 121.2 Allowable Stress Values
    - 121.3 Temperature Limitations
    - 121.4 Hanger Adjustments
    - 121.5 Hanger Spacing
    - 121.6 Springs
    - 121.7 Fixtures
    - 121.8 Structural Attachments
    - 121.9 Loads and Supporting Structures
    - 121.10 Fabricating Pipe Supports

121.5 Hanger Spacing

121.5 Hanger Spacing

Supports for piping with the longitudinal axis in approximately a horizontal position shall be spaced to prevent excessive sag, bending and shear stresses in the piping, with special consideration given where components, such as flanges and valves, impose concentrated loads. Where calculations are not made, suggested maximum spacing of supports for standard and heavier pipe are given in Table 121.5. Vertical supports shall be spaced to prevent the pipe from being overstressed from the combination of all loading effects.

Table 121.5 Suggested Pipe Support Spacing

Nominal Pipe Size, NPS	Suggested Maximum Span: Water Service, ft	Water Service, m	Steam, Gas, or Air Service, ft	Steam, Gas, or Air Service, m
1	7	2.1	9	2.7
2	10	3.0	13	4.0
3	12	3.7	15	4.6
4	14	4.3	17	5.2
6	17	5.2	21	6.4
8	19	5.8	24	7.3
12	23	7.0	30	9.1
16	27	8.2	35	10.7
20	30	9.1	39	11.9
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124 Limitations on Materials

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- 124.4 Cast Gray Iron
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- 124.7 Nonferrous Metals
- 124.8 Cladding and Lining Materials
- 124.9 Nonmetallic Pipe
- 124.10 Deterioration of Materials in Service

125 Materials Applied to Miscellaneous Parts

- 125.1 Gaskets
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Chapter IV Dimensional Requirements

126 Material Specifications and Standards for Standard and Nonstandard Piping Components

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- See Table 126.1 Specifications and Standards
- ASTM Ferrous Material Specifications
- ASTM Nonferrous Material Specifications
- API Specification
- ANSI / NFPA Code
- MSS Standard Practices
- ASME Codes and Standards
- AWS Specifications
- AWWA Standards
- National Fire Codes
- PFI Standards
- FCI Standard

126.2 Nonstandard Piping Components

126.3 Referenced Documents

Chapter V Fabrication, Assembly, and Erection

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127 Welding
128 Brazing and Soldering
129 Bending and Forming
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131 Welding Preheat
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133 Stamping
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Chapter VI Inspection, Examination, and Testing

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136 Inspection and Examination
- 136.1 Inspection
- 136.2 Inspection and Qualification of Authorized Inspector for Boiler External Piping
- 136.3 Examination
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Mandatory Appendix A

Table A-1 Carbon Steel
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Quality Control Requirements for Boiler External Piping (BEP)

Nonmandatory Appendices

- **Appendix II Design of Safety Valve Installations**
- **Appendix III Nonmetallic Piping and Piping Lined With Nonmetals**
- **Appendix IV Corrosion Control**
- **Appendix V Recommended Practice for Operation, Maintenance, and Modification**
- **Appendix VI Approval of New Materials**
- **Appendix VII Design of Restrained Underground Piping**

ASME B31.1 Interpretations

- **There were two interpretations published with the 2004 Edition of ASME B31.1 Code.**
  - One dealt with Para 131.6 Interruption of Welding.
  - One dealt with Para 132.4.3 discussing the term "nominal material thickness" as used in Table 132 Post Weld Heat Treatment.
- **There were seven interpretations published with the 2005 Addenda of ASME B31.1 Code.**
- **There were six interpretations published with the 2006 Addenda of ASME B31.1 Code.**

B31.1 Code Cases

- **There were published 13 code cases with the 2004 Edition of ASME B31.1 Code.**
- **There was published one code case with the 2005 Addenda of ASME B31.1 Code.**
- **There was published one code case with the 2006 Addenda of ASME B31.1 Code.**

Case Study on Service Water Piping in NPP

- **Problem: How to determine the proper wall thickness of service water piping?**
  - Design Specification details
  - Joining details
  - Support details

Note – we will use conceptual details vice calculating details.

Case Study (cont'd)

104.1.2 Straight Pipe Under Internal Pressure

(A) Minimum Wall Thickness. The minimum thickness of pipe wall required for design pressures and for temperatures not exceeding those for the various materials listed in the Allowable Stress Tables, including allowances for mechanical strength, shall not be less than that determined by eq. (3) or (3A), as follows:

$$t_m = \frac{PD_o}{2(SE + Py)} + A \qquad (3)^3$$

$$t_m = \frac{Pd + 2SEA + 2yPA}{2(SE + Py - P)} \qquad (3A)^3$$

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Learning Questions

- **How is the ASME B31.1 Code used in NPPs?**
- **What are the similarities between the ASME B31.1 Code and ASME BPV Code Section III? …and the differences?**
- **What are the typical systems in a NPP that are designed and constructed using the requirements of ASME B31.1 Code?**

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 23
Process Piping

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Module 23
Process Piping

Instructor: C. Wesley Rowley, P.E.

Learning Objectives

- **Learn how the ASME B31.3 Code is used in NPPs / nuclear facilities.**
- **Learn similarities and differences between the ASME B31.3 Code and ASME B31.1 Code / ASME BPV Code Section III.**
- **Learn typical systems in a NPP / nuclear facility that are designed and constructed using the requirements of ASME B31.3 Code.**

Significant Sub-Topics

- Scope and Definitions
- Design
- Materials
- Standards for Piping Components
- Fabrication, Assembly, and Erection
- Inspection, Examination, and Testing
- Nonmetallic Piping and Piping Lined With Nonmetals
- Piping for Category M Service
- High Pressure Piping
- Mandatory Appendices
- Non-mandatory Appendices
- Technical / Code Inquiries
- Code Cases

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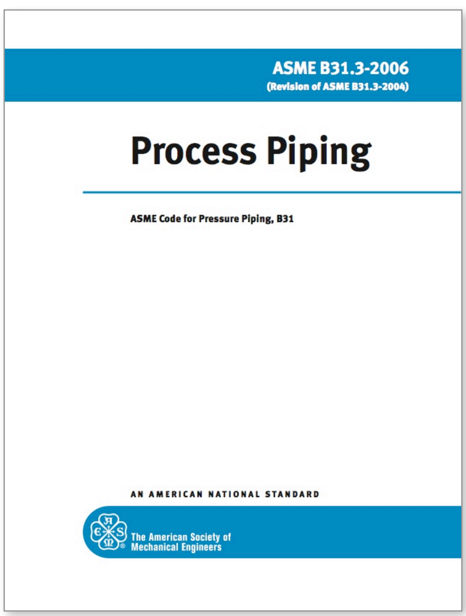

ASME B31.3 -2006, Process Piping CONTENTS

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  § 300.2 Definitions
  § 300.3 Nomenclature
  § 300.4 Status of Appendices

§ 300 GENERAL STATEMENTS

a) Identification

b) Responsibilities

c) Intent of the Code

d) Determining the Code Requirements

e) High Pressure Piping

f) Appendices

§300.1 Scope

- **Rules for the Process Piping Code Section B31.3 have been developed considering piping typically found in petroleum refineries; chemical, paper, pharmaceutical, textile, semiconductor, and cryogenic plants; and related processing plants and terminals.**

  300.1.1 Content and Coverage
  300.1.2 Packaged Equipment Piping
  300.1.3 Exclusions

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§300.2 Definitions

- **There are 125 definitions in §300.2**
- **An important definition is for Fluid Service:**
  a) Category D Fluid Service
  b) Category M Fluid Service
  c) High Pressure Fluid Service
  d) Normal Fluid Service

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Chapter II Design

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Part 1 Conditions and Criteria

301 Design Conditions
302 Design Criteria

Part 2 Pressure Design of Piping Components

303 General
304 Pressure Design of Components

Part 3 Fluid Service Requirements for Piping Components

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306 Fittings, Bends, Miters, Laps, and Branch Connections
307 Valves and Specialty Components
308 Flanges, Blanks, Flange Facings, and Gaskets
309 Bolting

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Chapter II Design (cont'd)

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Part 4 Fluid Service Requirements for Piping Joints

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311 Welded Joints
312 Flanged Joints
313 Expanded Joints
314 Threaded Joints
315 Tubing Joints
316 Caulked Joints
317 Soldered and Brazed Joints
318 Special Joints

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Chapter II Design (cont'd)

Part 5 Flexibility and Support

319 Piping Flexibility

321 Piping Support

Part 6 Systems

322 Specific Piping Systems

- Instrument Piping
- Pressure Relieving Systems – reference to BPV Code Section VIII

Process Design Problem

- **Assignment – manufacturer Chlorine gas using the gas extraction method**
- **Process – Chlorine can be manufactured by electrolysis of a sodium chloride solution (brine). The production of chlorine results in the co-products caustic soda (sodium hydroxide, NaOH) and hydrogen gas (H2).**
- **Challenge – These two co-products, as well as chlorine itself, are highly reactive.**

Chapter III Materials

323 General Requirements

- Materials and Specifications
- Temperature Limitations
- Impact Testing Methods and Acceptance Criteria
- Fluid Service Requirements for Materials

325 Materials – Miscellaneous

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Chapter IV
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- **Ratings of Components**
- **Referenced Documents**
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  - ASTM standards
  - API standards
  - AWWA standards
  - MSS standards
  - SAE standards
  - BS standards
  - NFPA standards

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Chapter V
Fabrication, Assembly, and Erection

Chapter VI
Inspection, Examination, and Testing

Chapter VII
Nonmetallic Piping and Piping Lined With Nonmetals

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Part 1	**Conditions and Criteria**
Part 2	**Pressure Design of Piping Components**
Part 3	**Fluid Service Requirements for Piping Components**
Part 4	**Fluid Service for Piping Joints**
Part 5	**Flexibility and Support**
Part 6	**Systems**
Part 7	**Materials**
Part 8	**Standards for Piping Components**
Part 9	**Fabrication, Assembly, and Erection**
Part 10	**Inspection, Examination, and Testing**

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Chapter VIII
Piping for Category M Fluid Service

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Part 1	**Conditions and Criteria**
Part 2	**Pressure Design of Metallic Piping Components**
Part 3	**Fluid Service Requirements for Metallic Piping Components**
Part 4	**Fluid Service Requirements for Metallic Piping Joints**
Part 5	**Flexibility and Support of Metallic Piping**
Part 6	**Systems**
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Part 9	**Fabrication, Assembly, and Erection of Metallic Components**
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Parts 11	**Through 20 Corresponding to Chapter VII**

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Chapter IX
High Pressure Piping

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Appendices

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Appendices (cont'd)

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Appendix L	**Aluminum Alloy Pipe Flanges**
Appendix M	**Guide to Classifying Fluid Services**
Appendix P	**Alternative Rules for Evaluating Stress Range**
Appendix Q	**Quality System Program**
Appendix S	**Piping System Stress Analysis Examples**
Appendix V	**Allowable Variation in Elevated Temperature Service**
Appendix X	**Metallic Bellows Expansion Joints**
Appendix Z	**Preparation of Technical Inquiries**

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User Technical Support

- **Codes Cases**
  - The B31 Committee publishes the code cases.
  - With the B31.3-2006 edition one code case was published (B31 Case 178, *Providing an Equation for Longitudinal Stress for Sustained Loads in ASME B31.3 Construction*)
- **Technical Inquiries**
  - Volume 19 (1 Apr 01 to 31 Oct 03) = 49
  - Volume 20 (1 Nov 03 to 31 Oct 05) = 52

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Committee Membership

- **B31 Standards Committee – about 35 people, who are representatives from the various piping committees**
- **B31.3 Piping Committee – about 45 people, who are representative from the various segments of the process industry.**

Note A – there is no one from the nuclear industry on the B31.3 piping committee.

Note B – there are 4 people from B31.3 that are also members of B31.

Case Study

- **Problem: What are the application differences between ASME B31.3 and ASME B31.1 / BPV Section III?**
  - Pressure & Temperature?
  - Fluid Environment?
  - External Loads?
  - Risk?

Learning Questions

- **How is the ASME B31.3 Code used in NPPs / nuclear facilities?**
- **What are the similarities and differences between the ASME B31.3 Code, ASME B31.1 Code, and the ASME BPV Code Section III?**
- **What are the typical systems in a NPP / nuclear facility that are designed and constructed using the requirements of ASME B31.3 Code?**

Module 24
ASME Welding Overview

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Module 24
ASME Welding Overview

Instructor: Gene Imbro, P.E.

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Learning Objectives

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- **Learn the difference between "welding" and "nuclear welding".**
- **Learn the three major aspects of welding.**
- **Learn correct application of "essential variables" and "non-essential variables" concepts to welding and brazing processes.**

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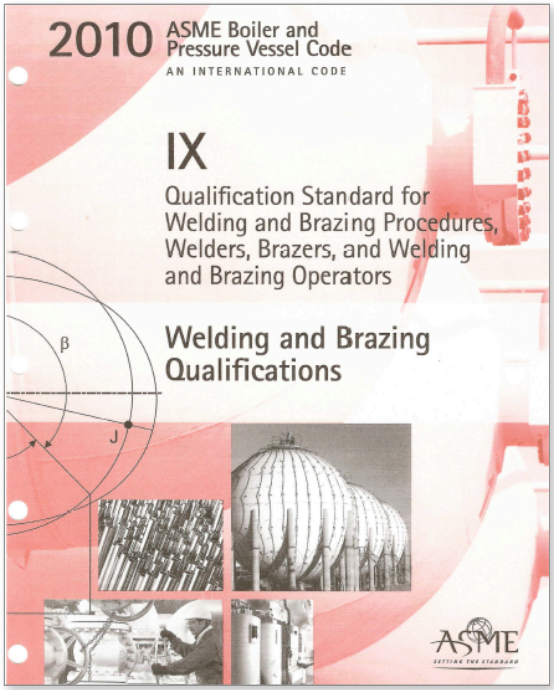

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CONTENTS

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Types of ASME BPV Codes

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- **Construction Codes**
  - BPV Code Section I – Power Boilers
  - BPV Code Section III – Nuclear Facilities
  - BPV Code Section IV – Heating Boilers
  - BPV Code Section VIII – Pressure Vessels
  - BPV Code Section X – FRP Pressure Vessels
- **Service Codes**
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  - BPV Code Section V – NDE
  - BPV Code Section IX – Welding & Brazing
- **Other Codes**
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Outline of Sub-topics

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- Qualification for Welding Prior to Work
- Welding Equipment
- Temper Bead Welding Features
- Welding Discontinuities
- Soldering and Brazing
- Welding Procedure Specification (WPS)
- WPS Qualification
- Welder Performance Qualification (WPQ)
- Surface NDE Methods for Welds

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Contents of Section IX

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- **Part QW Welding**
  - Article I Welding General Requirements
  - Article II Welding Procedure Qualifications
  - Article III Welding Performance Qualifications
  - Article IV Welding Data
  - Article V Standard Welding Procedure Specifications
- **Part QB Brazing**
- **Appendices**
- **Index**

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Part QW Article I

QW-100 General

- QW-110 Weld Orientation
- QW-120 Test Positions for Groove Welds
- QW-130 Test Positions for Fillet Welds
- QW-140 Types and Purposes for Tests and Examinations
- QW-150 Tension Tests
- QW-160 Guided-Bend Tests
- QW-170 Notch Toughness Tests
- QW-180 Fillet Weld Tests
- QW-190 Other Tests and Examinations

QW-101 Scope

- **The rules in this Section apply to the preparation of Welding Procedure Specification and the qualification of welding procedures, welders, and welding operators for all types of manual and machine welding processes permitted in this Section.**
- **These rules may also be applied, insofar as they are applicable, to other manual or machine welding processes permitted in other Sections.**

… excerpt from ASME BPV Code Section IX.

Part QW Articles II and III

QW-200 General

- QW-210 Preparation of Coupons
- QW-250 Welding Variables
- QW-290 Temper Bead Welding

QW-300 General

- QW-310 Qualification Test Coupons
- QW-320 Retest and Renewal of Qualification
- QW-350 Welding Variables for Welders
- QW-360 Welding Variables for Welding Operators
- QW-380 Special Processes

Part QW Article IV

QW-400 Variables

- QW-410 Technique
- QW-420 Base Metal Groupings
- QW-430 F-Numbers
- QW-440 Weld Metal Chemical Composition
- QW-450 Specimens
- QW-460 Graphics
- QW-470 Etching – Processes and Reagents
- QW-490 Definitions

Base Metal Groupings

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- **Steel and Steel Alloys**
- **Aluminum and Aluminum-base Alloys**
- **Copper and Copper-base Alloys**
- **Nickel and Nickel-base Alloys**
- **Titanium and Titanium-base Alloys**
- **Zirconium and Zirconium-base Alloys**

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Part QW Article V

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- **QW-500 General**
  - QW-510 Adaption of SWPSs
  - QW-520 Use of SWPSs Without Discrete Demonstration
  - QW-530 Forms
  - QW-540 Production Use of SWPSs

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Welding Procedure Specification (WPS)

- **A written document**
- **Required for all welding**
- **Provides directions to the welder for making welds in accordance with Code requirements**
- **Specifies the conditions under which welding must be performed**
  - Base metals that are permitted
  - Filler metal that must be used
  - Preheat and post weld heat treatments etc.
  - Such conditions are referred to in the code as welding variables

Welding Procedures Specification (WPS)

- **All WPS must address:**
  - Welding variables required by Section IX, Article II
  - Article II specifies welding variables for each welding process
  - Essential and nonessential variables are specified
  - Additional supplemental essential variables required when notch toughness qualification is required by the construction code

Purpose of WPS Qualification

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- **Determine weldment proposed for construction is capable of providing required properties for its intended application**
- **Welding procedure qualification establishes the properties of the weldment**
- **Welding procedure qualification does not establish the skill of the welder**
- **Any WPS used by a manufacture or contractor will be qualified by that organization**

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Procedure Qualification Record (PQR)

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- **What occurred during welding the test coupon**
- **Results of coupon testing**
- **Essential variables used**
- **Supplementary essential variables for each process used**
- **Results of the required testing and examination**

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Welder Performance Qualification

- **Required prior to welding:**
  - Determines welder’s ability to deposit sound weld metal
  - Test weld coupon using qualified welding procedure
  - Pipe test weld coupon used for pipe weld qualifications
  - Plate test weld coupon used for plate welding qualifications
  - Physical testing of test welds and or nondestructive examination for weld soundness

Welder Performance Qualification (WPQ) Record

- **Record of essential variables**
- **Type of test**
- **Test results**
- **The ranges qualified**

Welder Identification

- **Each qualified welder is assigned an identifying number, letter or symbol by the manufacture or contractor, which shall be used to identify the work of the welder**

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Welder Retests

- **A welder who fails one or more of the tests prescribed may be retested**
- **The code prescribes conditions for retesting**

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Expiration and Renewal of Qualifications

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- **When he / she has not welded with a process during a 6 month period**
- **When there is a specific reason to question his / her ability to make welds that meet the specification**

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Definitions

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welding, machine: welding with equipment that has controls that are manually adjusted by the welding operator or adjusted under the welding operator's direction in response to visual observation of the welding, with the torch, gun, or electrode holder held by a mechanical device. See also *welding, automatic*.

welding, manual: welding wherein the entire welding operation is performed and controlled by hand.

welding operator: one who operates machine or automatic welding equipment.

weldment: an assembly whose constituent parts are joined by welding, or parts which contain weld metal overlay.

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… excerpt from ASME BPV Code Section IX.

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Basic Welding Processes

- **Stick Welding**
- **Gas Tungsten Arc Welding (GTAW) – frequently referred to as TIG welding**
- **Gas Metal Arc Welding (GMAW) – frequently referred to as MIG welding**

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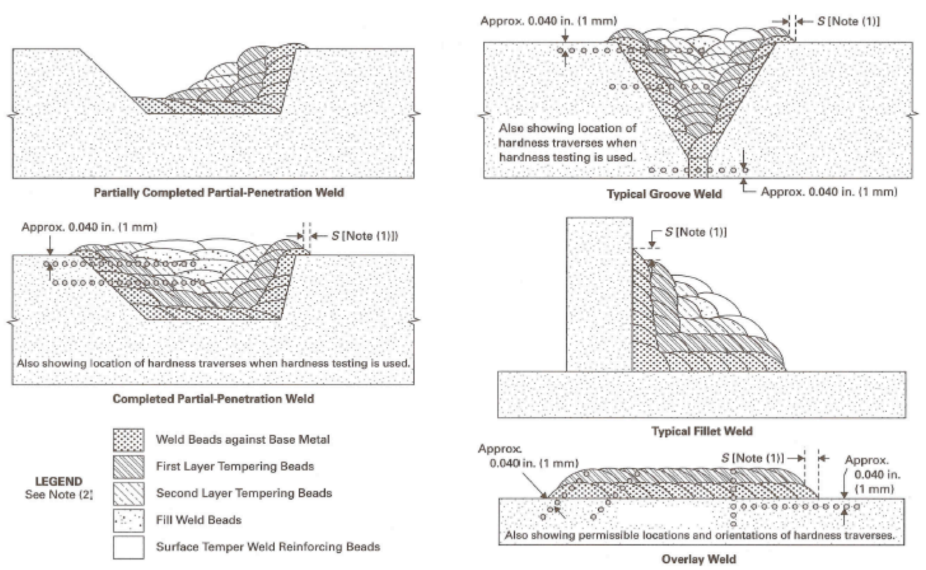
QW-462.12 NOMENCLATURE FOR TEMPER BEAD WELDING



GENERAL NOTES:

(a) Weld beads shown above may be deposited in any sequence that will result in placement of the beads as shown.

(b) Surface temper reinforcing beads may cover the entire weld surface, or may only be placed at the toe of the weld; they may or may not be mechanically removed.

NOTES:

(1) The distance, S, is measured from the toe of the weld to the edge of the temper beads. Measurements shall be made parallel to the base metal surface.

(2) Beads near the finished surface may be both tempering beads and surface temper reinforcing beads.

Weld Bead / Passes

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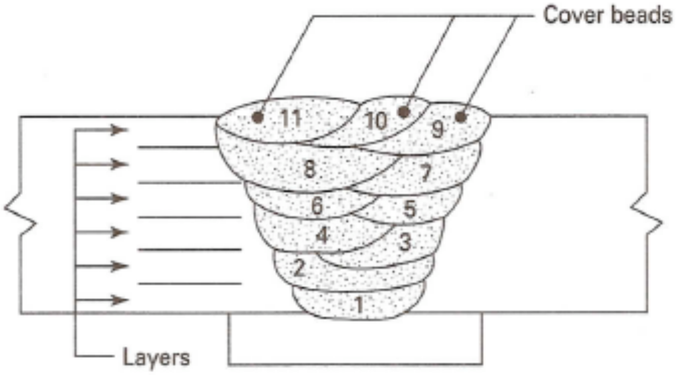
QW/QB-492.1 TYPICAL SINGLE AND MULTIBEAD LAYERS



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… excerpt from ASME BPV Code Section IX.

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Weld Bead / Passes

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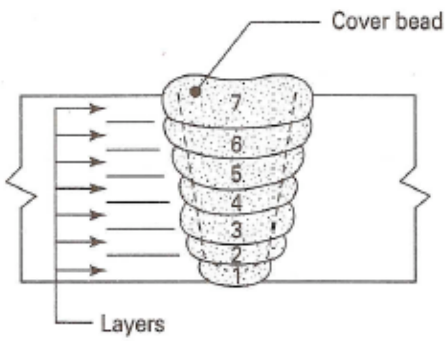
QW/QB-492.2 TYPICAL SINGLE BEAD LAYERS



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… excerpt from ASME BPV Code Section IX.

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NDE Methods for Welds

- **Surface**
  - Visual Examination
  - Liquid Penetrant
  - Magnetic Particle
- **Volumetric**
  - Radiographic – X ray or isotope
  - Ultrasonic

Nondestructive Examination (NDE) Procedures

- **Detailed written procedures are required by all code sections**
- **Procedures must comply with ASME Boiler and Pressure Vessel Code, Section V articles for method used**
- **Proven by actual demonstration to satisfaction of the Inspector**
- **Examination requirements specified in construction Code**
- **Acceptance criteria specified in construction Code**

Acceptance Criteria for Weld Examination

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- **Specified by construction Code**
  - ASME BPV Code Section III
  - ASME B31.7 Code
  - ASME B31.1 Code
  - ASME BPV Code Section VIII

- **Based on weld type and service requirements**

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Weld Profiles (Source: ASME AG-1 Code)

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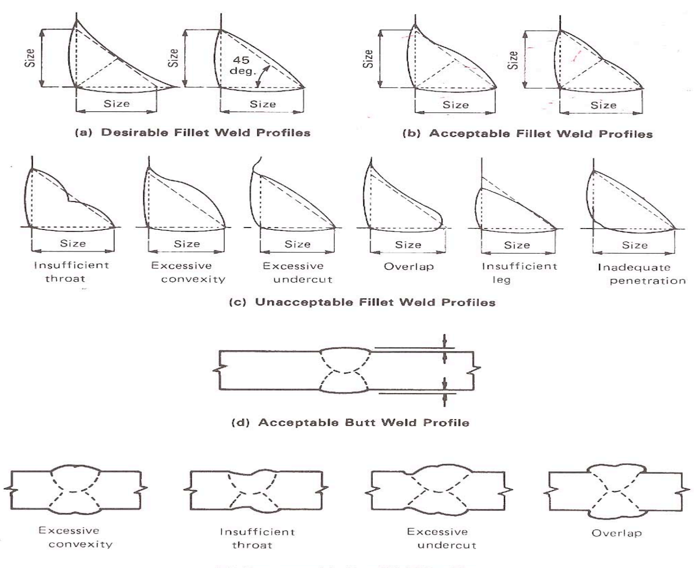

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FIG. AA-6300-1 ACCEPTABLE AND UNACCEPTABLE WELD PROFILES
(Courtesy of the American Welding Society)

Welding Discontinuities

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- **Undercut**
- **Incomplete fusion**
- **Porosity**
- **Slag Inclusions**
- **Cracks**

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Welding Incomplete Fusion

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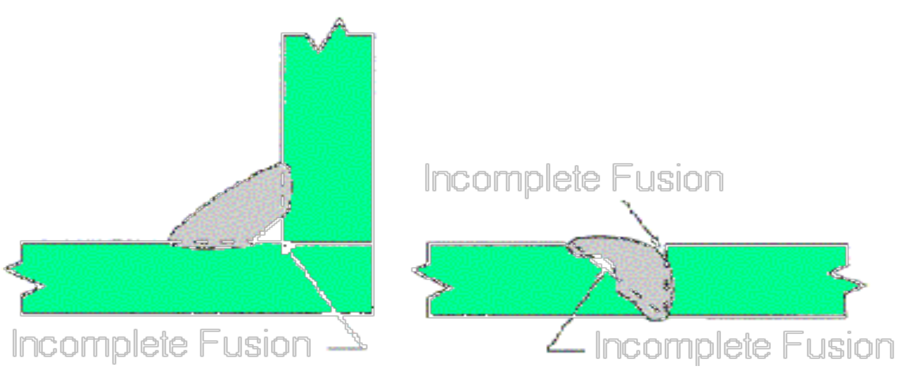

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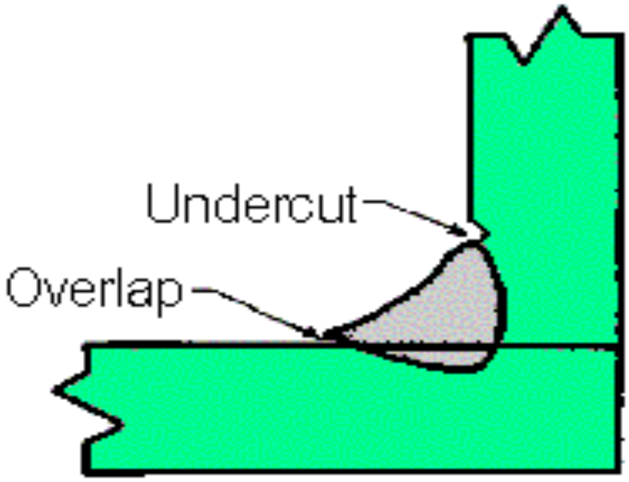
Welding Undercut



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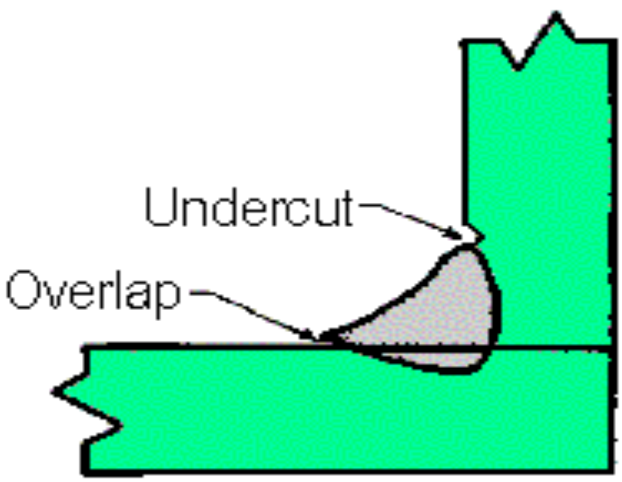
Welding Overlap



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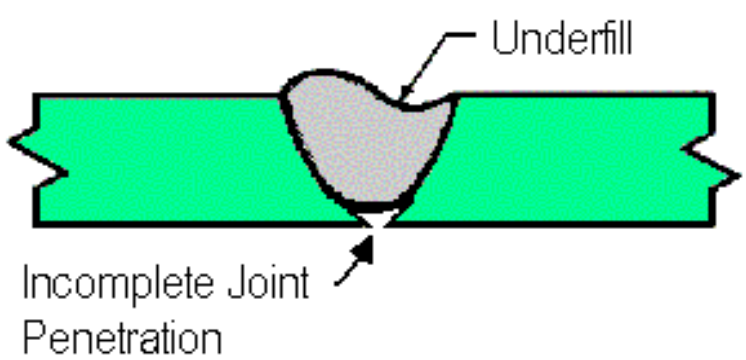
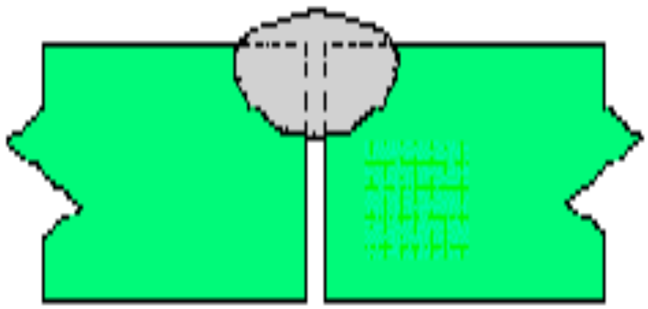
Welding Joint Penetration

- **Under fill - A condition in which the weld face or root surface extends below the adjacent surface of the base metal.**
- **Incomplete Joint Penetration - A joint root condition in a groove weld in which weld metal does not extend through the joint thickness**

Welding Joint Penetration



Welding Partial Penetration



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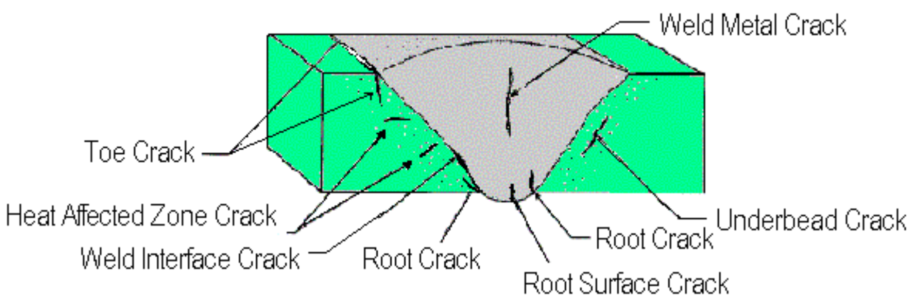
Weld Cracking



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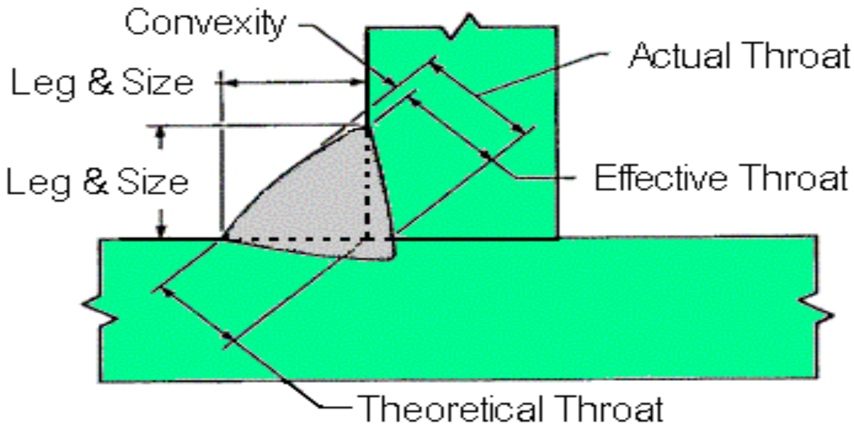
Convex Fillet Weld Without Discontinuities



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Code Interpretation

Interpretation: IX-10-02

Subject: QW-300.2(b)

Date Issued: August 18, 2009

File: 09-747

Question: Is the manufacturer or contractor required to provide full supervision during the performance qualification testing, so that issues such as the essential variables and inspections during the test can be verified and satisfied for each welder or welding operator qualified?

Reply: Yes.

…. excerpt from ASME BPV Code Section IX (2010 Edition)

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ASME BPV IX
Committee on Welding & Brazing

- **ASME Organization**
- **BPV IX Main Committee**
  - Subgroup on General Requirements
  - Subgroup on Materials
  - Subgroup on Performance Qualification
  - Subgroup on Procedure Qualification
  - Subgroup on Brazing
- **Other BPV Main Committees**
- **Other Pressure Technology C&S Main Committees**

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Related Welding Standards

- **ASME BPV Code Section IX Welding and Brazing Qualifications**
- **Other Welding Standards:**
  - American Welding Society (AWS)
    - AWS D1.1 Structural Welding Code - Steel
  - American Petroleum Institute (API)
    - API 1104 Welding of Pipelines and Related Facilities

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Nuclear Welding

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- **Documentation**
  - Use the right WPS
  - Approved procedure
  - Qualified welder / welding operator
  - Right material
- **Traceability for each weld**

NPPs (both operating plants and plants under construction) plan and document their welding processes.

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Heated Fusion of Non-Metallic Materials

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- **High Density Polyethylene (HDPE) piping products.**
- **Currently not addressed in Section IX**
- **Generic fusing qualification requirements are being addressed in Section III Code case N-755.**
- **Section IX is working to incorporate HDPE fusing qualification requirements.**

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Case Study on Nuclear Welding

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- **Problem: Weld two modules of 24-inch service water piping together in the Auxiliary Building.**
  - As the area construction superintendent, what events need to occur to support this welding?
  - As the field welding engineer, what information do I need to generate?
  - As the field craft welding supervisor, what information do we need?

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Case Study (cont'd)

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- **Area Construction Superintendent (in the Auxiliary Building)**
  - Task #1 – ??
  - Task #2 – ??
  - Task #3 – ??

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Case Study (cont'd)

- **Field Welding Engineering (for the Auxiliary Building)**
  - Task #1 – ??
  - Task #2 – ??
  - Task #3 – ??

Case Study (cont'd)

- **Field Craft Welding Supervisor (in the Auxiliary Building)**
  - Task #1 – ??
  - Task #2 – ??
  - Task #3 – ??

Case Study (cont'd)

- **What are the possible and likely field implementation problems with this welding task?**
  - For the Area Construction Superintendent
  - For the Field Welding Engineer
  - For the Field Craft Welding Supervisor

Learning Questions

- **What is the difference between "welding" and "nuclear welding"?**
- **What are the three major aspects of welding?**
- **What is the correct application of "essential variables" and "non-essential variables" concepts to the welding and brazing processes?**

UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 25
ASME B16 Standards

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Module 25
ASME B16 Standards

Instructor: Gene Imbro, P.E.

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Learning Objectives

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- **Learn what the term "class" means for the B16 Standards.**
- **Learn what "pressure rating designation" means.**
- **Learn what "NPS" means. Learn how it is used for flanges ... and for valves.**
- **Learn whether the B16 Standards are performance-based, prescriptive-based, or risk-based.**

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Significant Sub-topics

- Scope
- Pressure – Temperature Ratings
- Component Size and Type
- Marking
- Material
- Dimensions & Tolerances
- Pressure Testing
- Specific Requirements of ASME B16.5
- Specific Requirements of ASME B16.11
- Specific Requirements of ASME B16.34
- Specific Requirements of ASME B16.47

Background on ASME B16 Committee

- **Originally formed in 1920 under the American Standards Association (ASA).**
- **This is how it got its "B16" designation.**
- **The B16 Committee consists of 19 members from a wide varieties of organizations (including the NRC).**
- **A B16 subcommittee is responsible for each of B16.x standards.**
- **Each committee and subcommittee has a chairman, vice-chairman, ASME Staff secretary, and a wide variety of members from multiple segments of industry.**

ASME B16 Standards

- **Title – B16 American National Standards for Piping, Pipe Flanges, Fittings, and Valves**
  - B16.1-2005 *Gray Iron Pipe Flanges and Flanged Fittings (Classes 25, 125, and 250)*
  - B16.3-2006 *Malleable Iron Threaded Fittings: Classes 150 and 300*
  - B16.4-2006 *Gray Iron Threaded Fittings: Classes 125 and 250*
  - B16.5-2003 *Pipe Flanges and Flanged Fittings NPS ½ Through NPS 24 Metric / Inch Standard*
  - B16.9-2007 *Factory-Made Wrought Buttwelding Fittings*
  - B16.10-2000 (R2003) *Face-to-face and End-to-end Dimensions of Valves*
  - B16.11-2009 *Forged Fittings, Socket-Welding, and Threaded*
  - B16.12-1998 (R2006) *Cast Iron Threaded Drainage Fittings*

ASME B16 Standards (cont'd)

- B16.14-1991 *Ferrous Pipe Plugs, Bushings, and Locknuts with Threads*
- B16.15-2006 *Cast Copper Alloy Threaded Fittings*
- B16.18-2001 (R2005) *Cast Copper Alloy Solder Joint Pressure Fittings*
- B16.20-2007 *Metallic Gaskets for Pipe Flanges: Ring-joint, Spiral-Wound, and Jacketed*
- B16.21-2005 *Nonmetallic Flat Gaskets for Pipe Flanges*
- B16.22-2001 (R2005) *Wrought Copper and Copper Alloy Solder Joint Pressure Fittings*
- B16.23-2002 (R2006) *Cast Copper Alloy Solder Joint Drainage Fittings – DWV*
- B16.24-2006 *Cast Copper Alloy Pipe Flanges and Flanged Fittings: Classes 150, 300, 600, 900, 1500, and 2500*
- B16.25-2007 *Buttwelding Ends*

ASME B16 Standards (cont'd)

- B16.26-2006 *Cast Copper Alloy Fittings for Flared Copper Tubes*
- B16.28-1994 *Wrought Steel Buttwelding Short Radius Elbows and Returns*
- B16.29-2007 *Wrought Copper and Copper Alloy Solder Joint Drainage Fittings – DWV*
- B16.33-2002 (R2007) *Manually Operated Metallic Gas Valves for Use in Gas Piping Systems up to 125 psi (Sizes NPS ½ Through NPS 2)*
- B16.34-2004 *Valves – Flanged, Threaded, and Welding Ends*
- B16.36-2006 *Orfice Flanges*
- B16.38-2007 *Large Metallic Valves for Gas Distribution Manually Operated, NPS 2½ (DN 65) to NPS 12 (DN 300), 125 psig (8.6 bar) Maximum*
- B16.39-1998 (R2006) *Malleable Iron Threaded Pipe Unions: Classes 150, 250, and 300*

ASME B16 Standards (cont'd)

- B16.40-2008 *Manually Operated Thermoplastic Gas Shutoffs and Valves in Gas Distribution Systems*
- B16.41-1983 (R1989) *Functional Qualification Requirements for Power Operated Active Valve Assemblies in Nuclear Power Plants*
- B16.42-1998 (R2006) *Ductile Iron Pipe Flanges and Flanged Fittings: Classes 150 and 300*
- B16.44-2002 (R2007) *Manually Operated Metallic Gas Valves for Use in Aboveground Piping Systems Up to 5 psi*
- B16.45-1998 (R2006) *Cast Iron Fittings for Solvent® Drainage Systems*
- B16.47-2006 *Large Diameter Steel Flanges NPS 26 Through NPS 60 Metric / Inch Standard*
- B16.48-2005 *Steel Line Blanks*
- B16.49-2007 *Factory-Made Wrought Steel Buttwelding Induction Bends for Transportation and Distribution Systems*
- B16.50-2001 (R2008) *Wrought Copper and Copper Alloy Braze-joint Pressure Fittings*

Typical B16 Standard Format

- Scope and General
- Pressure Ratings
- Size and Type
- Marking
- Material
- Dimensions
- Additional Tolerances
- Proof Testing
- Figures
- Tables
- Mandatory Appendices (e.g., I, II, III)
- Nonmandatory Appendices (e.g., A, B, C)

Some Common B16 Terms / Phrases

- Ceiling Pressure – effectively sets pressure limits based on selected materials stress
- Component – a pipe, fitting, or valve
- Convention – uses ASTM Practice E29 for fixing significant digits (where limits, maximum, and minimum values are designated)
- Denotation – applies to pressure rating designation and size
- Fitting – consists of flange, pipe, and weld
- Flange – series A and series B (for large diameters)
- NPS – nominal pipe size
- Tolerances
- Valve – consists of valve body, valve internals, and flange / weld end

Selected ASME B16 Standards

- **B16.5** ***Pipe Flanges and Flanged Fittings - NPS ½ Through NPS 24 - Metric / Inch Standard***
- **B16.11** ***Forged Fittings, Socket-Welding and Threaded***
- **B16.34** ***Valves – Flanged, Threaded, and Welded End***
- **B16.47** ***Large Diameter Steel Flanges - NPS 26 Through NPS 60 - Metric / Inch Standard***

ASME B16.5 Standard

Title: Pipe Flanges and Flanged Fittings NPS ½ Through NPS 24 Metric / Inch Standard

1 SCOPE

1.1 General
1.2 References
1.3 Time of Purchase, Manufacturer, or Installation
1.4 User Accountability
1.5 Quality Systems
1.6 Relevant Units
1.7 Selection of Materials
1.8 Convention
1.9 Denotation

ASME B16.5 Standard (cont'd)

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2 Pressure – Temperature Ratings

2.1 General
2.2 Flanged joints
2.3 Ratings of Flanged joints
2.4 Rating Temperature
2.5 Temperature Considerations
2.6 System Hydrostatic Testing
2.7 Welding Neck Flanges
2.8 Straight Hub Welding Flanges
2.9 Multiple Material Grades

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ASME B16.5 Standard (cont'd)

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3 Component Size

3.1 Nominal Pipe Size
3.2 Reducing Fittings
3.3 Reducing Flanges

4 Marking

4.1 General
4.2 Identification Markings

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ASME B16.5 Standard (cont'd)

4.2 Identification Markings

4.2.1 Name
4.2.2 Material
4.2.3 Rating Designation
4.2.4 Conformance
4.2.5 Temperature
4.2.6 Size
4.2.7 Ring joint Flanges
4.2.8 Multiple Material Marking

ASME B16.5 Standard (cont'd)

5 Materials

5.1 General
5.2 Mechanical Properties
5.3 Bolting
5.4 Gaskets

Note: Materials required for flanges and flanged fittings are listed in Table 1A. Recommended bolting materials are listed in Table 1B. (information excerpted from Para 5.1)

Table 1A List of Material Specifications (Cont'd)

Material Group	Nominal Designation	Pressure–Temperature Rating Table	Applicable ASTM Specifications [Note (1)]		
			Forgings	Castings	Plates
3.11	44Fe–25Ni–21Cr–Mo	2-3.11	A 479 Gr. N08904	...	A 240 Gr. N08904
3.12	26Ni–43Fe–22Cr–5Mo	2-3.12	...	...	B 620 Gr. N08320
	47Ni–22Cr–20Fe–7Mo	2-3.12	...	...	B 582 Gr. N06985
	46Fe–24Ni–21Cr–6Mo–Cu–N	2-3.12	B 462 Gr. N08367	A 351 Gr. CN3MN	B 688 Gr. N08367
3.13	49Ni–25Cr–18Fe–6Mo	2-3.13	...	...	B 582 Gr. N06975
	Ni–Fe–Cr–Mo–Cu–Low C	2-3.13	B 564 Gr. N08031	...	B 625 Gr. N08031
3.14	47Ni–22Cr–19Fe–6Mo	2-3.14	...	...	B 582 Gr. N06007
	40Ni–29Cr–15Fe–5Mo	2-3.14	B 462 Gr. N06030	...	B 582 Gr. N06030
	58Ni–33Cr–8Mo	2-3.14	B 462 Gr. N06035	...	B 575 Gr. N06035
3.15	42Ni–42Fe–21Cr	2-3.15	B 564 Gr. N08810	...	B 409 Gr. N08810
3.16	35Ni–19Cr–1¼Si	2-3.16	B 511 Gr. N08330	...	B 536 Gr. N08330
3.17	29Ni–20.5Cr–3.5Cu–2.5Mo	2-3.17	...	A 351 Gr. CN7M	...
3.19	57Ni–22CR–14W–2Mo–La	2-3.19	B 564 Gr. N06230	...	B 435 Gr. N06230

GENERAL NOTES:
(a) For temperature limitations, see notes in Tables II-2-1.1 through II-2-3.17 of Mandatory Appendix II.
(b) Plate materials are listed only for use as blind flanges and reducing flanges without hubs (see para. 5.1). Additional plate materials listed in ASME B16.34 may also be used with corresponding B16.34, Standard Class ratings.

NOTE:
(1) ASME Boiler and Pressure Vessel Code, Section II materials may also be used, provided the requirements of the ASME specification are identical to or more stringent than the corresponding ASTM specification for the Grade, Class, or Type listed.

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ASME B16.5 Standard (cont'd)

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6 Dimensions

- 6.1 Flanged Fittings Wall Thickness
- 6.2 Fitting Center-to-Contact Surface and Center-to-End
- 6.3 Flat Face Flanges
- 6.4 Flange Facings
- 6.5 Flange Bolt Holes
- 6.6 Bolting Bearing Surfaces
- 6.7 Welding End Preparation for Welding Neck Flanges
- 6.8 Reducing Flanges
- 6.9 Threaded Flanges
- 6.10 Flange Bolting Dimensions
- 6.11 Gaskets for Line Flanges
- 6.12 Auxiliary Connections

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ASME B16.5 Standard (cont'd)

7 Tolerances

- 7.1 General
- 7.2 Center-to-Contact Surfaces and Center-to-End Tolerances
- 7.3 Facings
- 7.4 Flange Thickness
- 7.5 Welding End Flanges and Hubs
- 7.6 Length Through Hub on Welding Neck Flanges
- 7.7 Flange Bore Diameter
- 7.8 Drilling and Facing

ASME B16.5 Standard (cont'd)

8 Pressure Testing

- 8.1 Flange Test
- 8.2 Flange Fitting Test
  - 8.2.1 Shell Pressure Test
  - 8.2.2 Test Conditions
  - 8.2.3 Test Fluid
  - 8.2.4 Test Duration
  - 8.2.5 Acceptance

ASME B16.5 Standard (cont'd)

- **Some observations on the B16.5 Standard:**
  - There are 226 pages, so these presentation slides are just an inkling of the detail in this standard.
  - There are 22 tables in the text of the standard and another 22 tables in Appendix II / 5 tables in Appendix E.

ASME B16.11 Standard

Title: Forged Fittings, Socket-Welding and Threaded

1 Scope and General

1.1 Scope

1.1.1 Fitting Types / Configurations

1.1.2 Special Fittings

1.1.3 Welding

1.2 General

1.2.1 Referenced Standards

1.2.2 Codes and Regulations

1.2.3 Service Conditions

1.2.4 Quality Systems

1.2.5 Standard Units

ASME B16.11 Standard (cont'd)

Table 6 Types of Fittings by Class Designation and NPS Size Range

Description	Socket-Welding Class Designation 3000	6000	9000	Threaded Class Designation 2000	3000	6000
45-deg, 90-deg elbows,	1/8–4	1/8–2	1/2–2	1/8–4	1/8–4	1/8–4
tees, crosses,	1/8–4	1/8–2	1/2–2	1/8–4	1/8–4	1/8–4
couplings, half-couplings,	1/8–4	1/8–2	1/2–2	...	1/8–4	1/8–4
and caps	1/8–4	1/8–2	1/2–2	...	1/8–4	1/4–4
Street elbows	...	...	...	...	1/8–2	1/8–2
Square, hex, round plug,	...	...	...	1/8–4 [Note (1)]	1/8–4 [Note (1)]	1/8–4 [Note (1)]
hex, and flush bushing	...	...	...	1/8–4 [Note (1)]	1/8–4 [Note (1)]	1/8–4 [Note (1)]

NOTE:
(1) Plugs and bushings are not identified by class designation. They may be used for ratings up through Class 6000 designation.

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ASME B16.11 Standard (cont'd)

Table 7 Correlation of Fittings Class With Schedule Number or Wall Designation of Pipe for Calculation of Ratings

Class Designation of Fitting	Type of Fitting	Pipe Used for Rating Basis [Note (1)] Schedule No.	Wall Designation
2000	Threaded	80	XS
3000	Threaded	160	...
6000	Threaded	...	XXS
3000	Socket-welding	80	XS
6000	Socket-welding	160	...
9000	Socket-welding	...	XXS

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ASME B16.11 Standard (cont'd)

2 Pressure Ratings

2.1 General

2.1.1 Basis of Rating

2.1.2 Nonstandard Pipe Wall Thickness

2.1.3 Combination End Fittings

2.2 Pressure Test Capability

"…the fitting shall be capable of withstanding a hydrostatic test pressure required by the applicable piping code for seamless pipe of material equivalent to the fitting forging and of the schedule or wall thickness correlated with the fitting Class and end connection of Table 7."

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ASME B16.11 Standard (cont'd)

3 Size and Type

3.1 General

The relationship is typically as follows:

NPS	DN
1/8	6
1/4	8
3/8	10
1/2	15
3/4	20
1	25
1 1/4	32
1 1/2	40
2	50
2 1/2	65
3	80
4	100

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ASME B16.11 Standard (cont'd)

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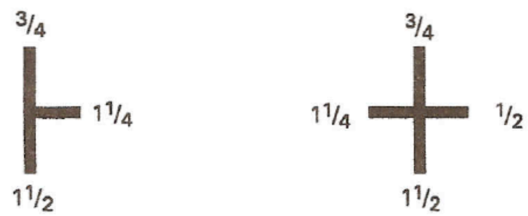
3.2 Reducing Fitting Size

Fig. 1 Method of Designating Outlets of Reducing Tees and Crosses



$1\frac{1}{2} \times \frac{3}{4} \times 1\frac{1}{4}$ $1\frac{1}{2} \times \frac{3}{4} \times 1\frac{1}{4} \times \frac{1}{2}$

Tee Cross

GENERAL NOTE: See para. 3.2.

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ASME B16.11 Standard (cont'd)

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Table 8 Nominal Wall Thickness of Schedule 160 and Double Extra Strong Pipe

	Schedule 160		XXS	
NPS	mm	in.	mm	in.
1/8	3.15	0.124	4.83	0.190
1/4	3.68	0.145	6.05	0.238
3/8	4.01	0.158	6.40	0.252

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ASME B16.11 Standard (cont'd)

4 Marking

- 4.1 General
  - 4.1.1 Specific Marking
    - (a) Manufacturer's Name or Trademark
    - (b) Material Identification
    - (c) Product Conformance
    - (d) Class Designation
    - (e) Size
  - 4.1.2 Omission of Markings

5 Material

- 5.1 Standard Materials

ASME B16.11 Standard (cont'd)

6 Dimensions

- 6.1 General
- 6.2 Socket Fittings
- 6.3 Threaded Fittings
- 6.4 Collars
- 6.5 Reducing Fittings

7 Additional Tolerances

- 7.1 Concentricity of Bores
- 7.2 Coincidence of Axis

8 Proof Testing

ASME B16.11 Standard (cont'd)

- **Mandatory Appendix I Inch Tables**
  - Table I-1 Socket-Welding Fittings
  - Table I-2 Forged Threaded Fittings
  - Table I-3 Forged Threaded Fittings – Street Elbows
  - Table I-4 Threaded Fittings
  - Table I-5 Plugs and Bushings
- **Mandatory Appendix II References**
- **Nonmandatory Appendix A Quality System Program**

ASME B16.34 Standard

- **Title: Valves – Flanged, Threaded, and Welding End**

1 Scope

1.1 General
1.2 Applicability
1.3 Selection of Valve Types and Material Service Conditions
1.4 Convention
1.5 Denotation
1.6 References

ASME B16.34 Standard (cont'd)

2 Pressure – Temperature Ratings

- 2.1 General
- 2.2 Rating Temperature
- 2.3 Temperature Effects
- 2.4 Guidance for the Use of Flanged Valve Ratings
- 2.5 Variances
- 2.6 Multiple Material Grades
- 2.7 Local Operating Conditions

3 Nominal Pipe Size

ASME B16.34 Standard (cont'd)

4 Marking

- 4.1 General
- 4.2 Identification Markings
- 4.3 Identification Plate
  - 4.3.1 Attachment
  - 4.3.2 Pressure Markings
  - 4.3.3 Special Markings
- 4.4 Conformity
  - 4.4.1 Designation
  - 4.4.2 Compliance

ASME B16.34 Standard (cont'd)

5 Materials
- 5.1 General
- 5.2 Material Selection
- 5.3 Electrical Continuity
- 5.4 Flange Removal

6 Dimensions
- 6.1 Body Dimensions
- 6.2 End Dimensions
- 6.3 Auxiliary Connections
- 6.4 Valve Joints
- 6.5 Stems
- 6.6 Installation Limitations
- 6.7 Wafer or Flangeless Valves

ASME B16.34 Standard (cont'd)

7 Pressure Testing
- 7.1 Shell Test
- 7.2 Valve Closure Tests
- 7.3 Leakage Detection Devices
- 7.4 Surface Protection

8 Requirements for Special Valves
- 8.1 Scope
- 8.2 General
- 8.3 Required Examination
- 8.4 Defect Removal and Repair

ASME B16.34 Standard (cont'd)

- Mandatory Appendix I **Radiography Examination: Procedure and Acceptance Standards**
- Mandatory Appendix II **Magnetic Particle Examination: Procedure and Acceptance Standards**
- Mandatory Appendix III **Liquid Penetrant Examination: Procedure and Acceptance Standards**
- Mandatory Appendix IV **Ultrasonic Examination: Procedure and Acceptance Standards**
- mandatory Appendix V **Requirements for Limited Class Valves**
- Mandatory Appendix VI **Basis Equations for Minimum Wall Thickness**
- Mandatory Appendix VII **Pressure-Temperature Ratings: U.S. Customery Units**
- Mandatory Appendix Viii **Reference Standards and Specifications**

ASME B16.34 Standard (cont'd)

- Nonmandatory Appendix A **Relationship Between Nominal Pipe Size And Inside Diameter**
  - Table A-1 Inside Diameter, d (versus class ratings)
- Nonmandatory Appendix B **Method Used For Establishing Pressure-Temperature Ratings**
  - B-1 General Considerations
  - B-2 Standard Class Rating Method
  - B-3 Special Class Rating Method
  - B-4 Intermediate Rating Class Method
  - B-5 Maximum Ratings
- Nonmandatory Appendix C **Quality System Program**

ASME B16.34 Standard (cont'd)

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- **Some observations on the B16.34 Standard:**
  - Total pages in standard is 192
  - Total number of figures in standard is 17 (each for a different kind of valve)
  - Total number of tables in standard is 53 (some extend for multiple pages)

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ASME B16.47 Standard

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Title: Large Diameter Steel Flanges NPS 26 Through NPS 60 Metric / Inch Standard

1 Scope

1.1 General
1.2 Flange Series
1.3 References
1.4 Time of Purchase, Manufacture, or Installation
1.5 User Accountability
1.6 Quality Systems
1.7 Relevant Units
1.8 Service Conditions
1.9 Convention
1.10 Denotation

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ASME B16.47 Standard (cont'd)

2 Pressure-Temperature Ratings

- 2.1 General
- 2.2 Flanged Joints
- 2.3 Ratings of Flanged Joints
- 2.4 Rating Temperature
- 2.5 Temperature Considerations
- 2.6 System Pressure Testing
- 2.7 Welding Neck Flanges
- 2.8 Multiple Material Grades

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ASME B16.47 Standard (cont'd)

3 Component Size: Nominal Pipe Size

4 Marking

- 4.1 General
- 4.2 Identification Marking
  - 4.2.1 Name
  - 4.2.2 Materials
  - 4.2.3 Rating Designation
  - 4.2.4 Conformance
  - 4.2.5 Temperature
  - 4.2.6 Size
  - 4.2.7 Ring-joint Flange
  - 4.2.8 Multiple Material Marking

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ASME B16.47 Standard (cont'd)

5 Materials
- 5.1 General
- 5.2 Mechanical Properties
- 5.3 Bolting
- 5.4 Gaskets

6 Dimensions
- 6.1 Flange Facings
- 6.2 Flange Bolt Holes
- 6.3 Bolting Bearing Surfaces
- 6.4 Welding End Preparation for Welding Neck Flanges
- 6.5 Flange Bolting Dimensions
- 6.6 Gaskets
- 6.7 Hub Dimensions

ASME B16.47 Standard (cont'd)

7 Tolerances
- 7.1 Facings
- 7.2 Flange Thickness
- 7.3 Welding End Flange Ends and Hubs
- 7.4 Hub Length for Welding Neck Flanges
- 7.5 Drilling and Facing

8 Pressure Testing

Figures (3)

Tables (41)

ASME B16.47 Standard (cont'd)

- Mandatory Appendix I **Pressure-Temperature Ratings and Dimensional Data for Classes 75, 150, 300, 400, 600, and 900 flanges in u.S. Customery units (3 figures and 38 tables)**
- Mandatory Appendix II **Quality System Program**
- Mandatory Appendix III **References**

- Nonmandatory Appendix A **Methods Used for Establishing Pressure-temperature Ratings**
- Nonmandatory Appendix B **Gaskets (Other Than Ring-joint)**

Case Study

- **Case Study on determining the requirements for special class valves (perhaps a LPSI isolation valve).**
  - What is a "special class valve" ?
  - What does this "valve" include?
  - What are some example special requirements for a "special class valve" ?
  - Provide some examples of these processes for the special requirements.
  - Are repairs to the "special class valve" allowed?
  - What are two reasons why a NPP would want to use a "Special Class Valve" for LPSI isolation?

Learning Questions

- **What does the term "class" mean for the ASME B16 Standards?**
- **What does the phrase "pressure rating designation" mean?**
- **What does "NPS" mean? How is it used for flanges? …for fittings? …for valves?**
- **Are the ASME B16 Standards performance-based, prescriptive-based, or risk-based?**

UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 26
ASME BPV Code Section II, Materials

AdSTM
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Module 26

ASME BPV Code Section II, Materials

Instructor: C. Wesley Rowley, P.E.

AdSTM Advanced Systems Technology and Management, Inc.

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Learning Objectives

- **Learn how the ASME develops / modifies a material specification.**
- **Understand the backup information use and storage process for ASME material specifications.**
- **Understand how the ASME manages proprietary material data.**

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Significant Sub-topics

- Control allowable structural materials
- Use of ASTM material specifications in Section II
- Part A to Section II
- Part B to Section II
- Part C to Section II
- Part D to Section II
- Part E (proposed) to Section II

Control of Allowable Structural Materials

- **Construction Codes**
  - BPV Code Section I
  - BPV Code Section III
  - BPV Code Section VIII
  - Other Construction Codes
- **Section II Materials**

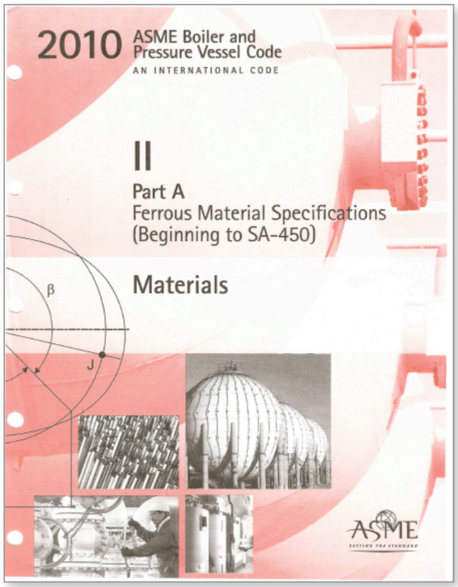
Service Sections to BPV Code

- **Three Service Sections**
  - Materials (Section II)
  - Nondestructive Testing (Section V)
  - Welding and Brazing (Section IX)
- **Section II Materials**
  - Ferrous Material Specifications (Part A)
  - Nonferrous Material Specifications (Part B)
  - Specifications for Welding Rods, Electrodes, and Filler Metals (Part C)
  - Properties (Part D)
  - Nonmetallic Materials (Part E) … in preparation

Typical Material Proposal

1. **Scope**
2. **Referenced Document**
3. **General Requirements**
4. **Ordering Information**
5. **Materials and Manufacture**
6. **Chemical Composition**
7. **Mechanical Properties**
8. **Number of Tests**
9. **Reports of Testing**
10. **Repair by Welding**
11. **Marking of Forgings**
12. **Certificate of Compliance**
13. **Key Words**

Table 1 – Chemical Requirements
Table 2 – Tensile Requirements



ASME BPV Code
Section II, Materials
CONTENTS

- List of Sections
- Foreword
- Statements of Policy
- Personnel
- ASTM Personnel
- Preface
- Specifications Listed by Material
- Specification Removal
- Guidelines on Submittal of Technical Inquiries to BPV Committee
- Guideline on the Approval of New Materials Under the BPV Code
- Guideline on Acceptable ASTM Editions
- Guidelines on Acceptable Non-ASTM Editions
- Summary of Changes
- List of Changes in Record Number Order

Specifications

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Section II Materials

- **Part A Ferrous Material Specifications**
  - two volumes
- **Part B Nonferrous Material Specification**
- **Part C Specifications for Welding Rods, Electrodes, and Filler Material**
- **Part D Properties**
  - customary
  - metric
- **Part E Nonmetallic Materials**
  - in the course of preparation

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Part A
Ferrous Material Specifications
CONTENTS

- Specifications
- Mandatory Appendix
- Nonmandatory Appendix

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Part A Specifications

- **Ferrous Material Specifications**
  - Published in two 2-inch thick paper volumes
    - Beginning to SA-450
    - SA-451 to End
  - Total of 184 specifications
  - Most are joint ASME / ASTM specifications
  - Some are ASME specifications

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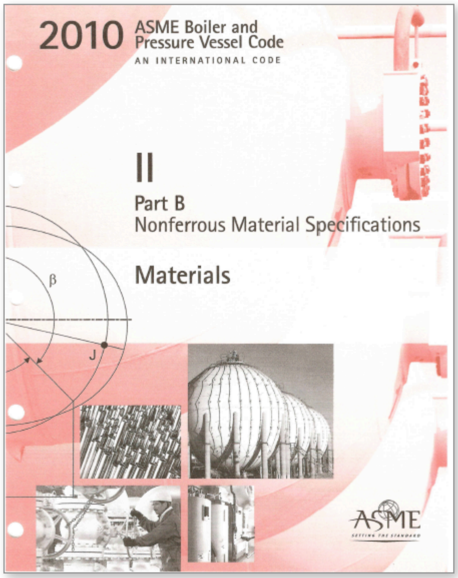

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Part B
Nonferrous Material Specifications
CONTENTS

- Specifications
- Mandatory Appendix
- Nonmandatory Appendix

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Part B Specifications

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- **Nonferrous Materials**
  - Published in one 2-inch paper volume
  - Total of 139 specifications
  - Most are ASME specifications
  - Some are joint ASME / ASTM specifications

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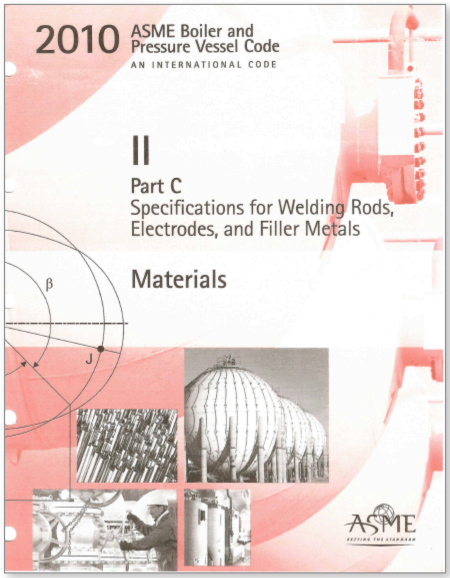

Part C Contents

- List of Sections
- Foreword
- Statement of Policy
- Personnel
- AWS Personnel
- Submittal of Technical Inquiries
- Guideline on Approval of New Material
- Preface
- Summary of Changes List of Changes in BC Order
- Specifications
- Mandatory Appendix
- I – Standard Units for Use in Equations

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Part C Specifications

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- **Specifications for Welding Rods, Electrodes, and Filler Metal**
  - Published in one 2-inch paper volume
  - Total of 33 specifications
  - All are joint ASME / AWS specifications

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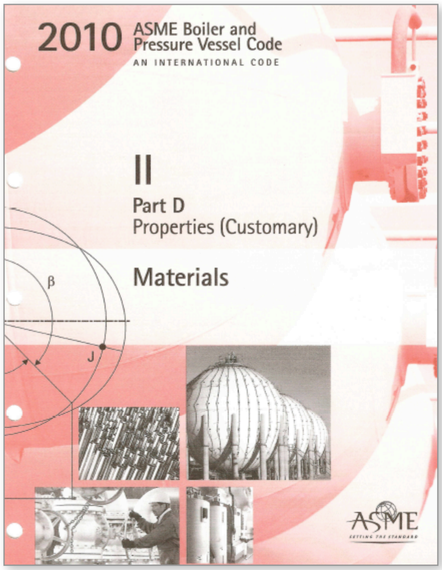

Part D Properties
CONTENTS

- List of Sections
- Statements of Policy
- Personnel
- Summary of Changes
- List of Changes in Record Number Order

- Subpart 1 Stress Tables
- Subpart 2 Physical Properties Tables
- Subpart 3 Charts and Tables for Determining Shell Thickness of Components Under External Pressure

- Tables
- Mandatory Appendices
- Nonmandatory Appendices

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Part D Specifications

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- **Properties**
  - Subpart 1 Stress Tables
    - Section I
    - Section III
    - Section VIII
  - Subpart 2 Physical Properties Tables
    - Thermal Expansion
    - Moduli of Elasticity
    - Poisson’s Ration and Density of Materials
  - Subpart 3 Charts and Tables for Determining Shell Thickness of Components Under External Pressure

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ASME BPV SCII / SWG-NMM
BPV Code Part E, Rev 0, Draft #13 (Record 07-1947)

200X ASME BOILER & PRESSURE VESSEL CODE

II

Part E -

Non-Metallic Material Specifications

AN INTERNATIONAL CODE

Materials

3 May 2010 Page i

ASME BPV Code
Section II Materials
Part E
Nonmetallic Materials

- Foreword
- Statements of Policy
- Personnel
- ASTM Personnel
- Preface
- Introduction
- Acceptable ASTM Editions
- Specifications Listed by Materials
- Specification Removal
- Preparation of Technical Inquiries to the Boiler and Pressure Vessel Committee
- Summary of Changes

- **Specifications**

Part E Specifications
(Proposed – page 1)

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Nonmetallic Materials Specifications

- **SE-A Thermoplastics**
- **SE-A1 Polyethylene**
  - SE-A1-1 PE-material
  - SE-A1-2 PE pipe
  - SE-A1-3 PE molded fittings
  - SE-A1-4 PE fabricated fittings
  - SE-A1-5 PE electro-fusion fittings
  - SE-A1-6 PE with reinforced fibers
- **SE-A2 Polystyrene-Modified-Polyphenylene**
  - SE-A2-1 PPE-material
    - Grade PPE – 2210G30A40452G1125F11
    - Grade PPE – 410G30A09325
    - Grade PPE – 210G30A50553

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Part E Specifications
(Proposed – page 2)

- **SE-B Thermoset Plastics**
  - SE-B1 Epoxy
  - SE-B2 Polyester
  - SE-B3 Vinyl Ester
- **SE-D Graphite**
  - SE-D1 Impregnated Graphite
  - SE-D2 Reactor Core Graphite
- **SE-E Fiber Reinforced Plastics**
  - SE-E1 FRP Pipe
  - SE-E2 FRP with Fiberglass
    - Grade 1 Epoxy xxx and E-glass
    - Grade 2 Epoxy yyy and S-glass
  - SE-E3 FRP with Carbon
  - SE-E4 FRP with Kevlar

Regulatory Use of Section II

- **Materials which appear in Section II are approved for use for ASME Class components unless specifically conditioned by NRC**
- **Materials which do not appear in Section II are not acceptable for use unless a specific request is made to the NRC and the NRC approves it.**
- **Verification of material through comparison of CMTR data with Section II data**

Case Study on Adoption of Foreign Materials into Section II

- **Is it possible for foreign material to be approved for inclusion into Section II?**
- **If yes, how would it be accomplished?**
- **If no, why not?**

Learning Questions

- **How does the ASME develops / modifies a material specification?**
- **What is the backup information use and storage process for ASME material specifications?**
- **How does the ASME manage proprietary material data?**

Module 27
ASTM Standards Overview

AdSTM
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Module 27
ASTM Standards Overview

Instructor: C. Wesley Rowley, PE

Learning Objectives

- **Learn how to read an ASTM material specification to find allowable engineering properties and physical properties.**
- **Learn the difference between an engineering property and a physical property.**
- **Learn the most common ASTM material specifications in use in a NPP.**

Significant Sub-topics

U.S.NRC
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- Format of ASTM material specifications
- Grades of materials
- Engineering properties (i.e., stress tables)
- Physical properties
- Relation to ASME BPV Code Section II
- ASTM committee structure for approving material specification
- Ferrous materials
- Non-ferrous materials
- Nonmetallic materials

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Type of ASTM Standards

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- **Standard Guide**
- **Standard Practice**
- **Standard Specification**
- **Standard Terminology**
- **Standard Test Method**

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Format of ASTM Standards

1. Scope
2. Referenced Documents
3. Terminology
4. XXX
5. XXX
6. XXX
7. Report
8. Keywords

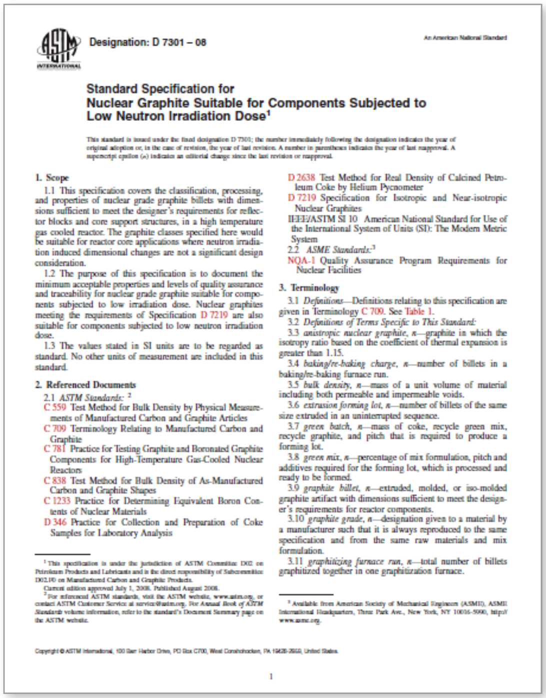
Annexes (mandatory)
Appendices (nonmandatory)
Supplementary Requirements

5

Publishing Formats

- ASTM web site (www.astm.org)
- ASTM Licensing Agreements for Sales
- ASME BPV Code Section II for Material
  - Standard Specifications
- ASME BPV Code Section V for NDE
  - Standard Practices
  - Standard Test Method

6



Designation: D 7301 – 08

An American National Standard

Standard Specification for
Nuclear Graphite Suitable for Components Subjected to Low Neutron Irradiation Dose[1]

This standard is issued under the fixed designation D 7301; the number immediately following the designation indicates the year of original adoption or, in the case of revision, the year of last revision. A number in parentheses indicates the year of last reapproval. A superscript epsilon (ε) indicates an editorial change since the last revision or reapproval.

1. Scope

1.1 This specification covers the classification, processing, and properties of nuclear grade graphite billets with dimensions sufficient to meet the designer's requirements for reflector blocks and core support structures, in a high temperature gas cooled reactor. The graphite classes specified here would be suitable for reactor core applications where neutron irradiation induced dimensional changes are not a significant design consideration.

1.2 The purpose of this specification is to document the minimum acceptable properties and levels of quality assurance and traceability for nuclear grade graphite suitable for components subjected to low irradiation dose. Nuclear graphites meeting the requirements of Specification D 7219 are also suitable for components subjected to low neutron irradiation dose.

1.3 The values stated in SI units are to be regarded as standard. No other units of measurement are included in this standard.

2. Referenced Documents

2.1 *ASTM Standards:* [2]

C 559 Test Method for Bulk Density by Physical Measurements of Manufactured Carbon and Graphite Articles
C 709 Terminology Relating to Manufactured Carbon and Graphite
C 781 Practice for Testing Graphite and Boronated Graphite Components for High-Temperature Gas-Cooled Nuclear Reactors
C 838 Test Method for Bulk Density of As-Manufactured Carbon and Graphite Shapes
C 1233 Practice for Determining Equivalent Boron Contents of Nuclear Materials
D 346 Practice for Collection and Preparation of Coke Samples for Laboratory Analysis
D 2638 Test Method for Real Density of Calcined Petroleum Coke by Helium Pycnometer
D 7219 Specification for Isotropic and Near-isotropic Nuclear Graphites
IEEE/ASTM SI 10 American National Standard for Use of the International System of Units (SI): The Modern Metric System

2.2 *ASME Standards:*[3]

NQA-1 Quality Assurance Program Requirements for Nuclear Facilities

3. Terminology

3.1 *Definitions*—Definitions relating to this specification are given in Terminology C 709. See Table 1.

3.2 *Definitions of Terms Specific to This Standard:*

3.3 *anistropic nuclear graphite, n*—graphite in which the isotropy ratio based on the coefficient of thermal expansion is greater than 1.15.

3.4 *baking/re-baking charge, n*—number of billets in a baking/re-baking furnace run.

3.5 *bulk density, n*—mass of a unit volume of material including both permeable and impermeable voids.

3.6 *extrusion forming lot, n*—number of billets of the same size extruded in an uninterrupted sequence.

3.7 *green batch, n*—mass of coke, recycle green mix, recycle graphite, and pitch that is required to produce a forming lot.

3.8 *green mix, n*—percentage of mix formulation, pitch and additives required for the forming lot, which is processed and ready to be formed.

3.9 *graphite billet, n*—extruded, molded, or iso-molded graphite artifact with dimensions sufficient to meet the designer's requirements for reactor components.

3.10 *graphite grade, n*—designation given to a material by a manufacturer such that it is always reproduced to the same specification and from the same raw materials and mix formulation.

3.11 *graphitizing furnace run, n*—total number of billets graphitized together in one graphitization furnace.

[1] This specification is under the jurisdiction of ASTM Committee D02 on Petroleum Products and Lubricants and is the direct responsibility of Subcommittee D02.F0 on Manufactured Carbon and Graphite Products.
Current edition approved July 1, 2008. Published August 2008.

[2] For referenced ASTM standards, visit the ASTM website, www.astm.org, or contact ASTM Customer Service at service@astm.org. For *Annual Book of ASTM Standards* volume information, refer to the standard's Document Summary page on the ASTM website.

[3] Available from American Society of Mechanical Engineers (ASME), ASME International Headquarters, Three Park Ave., New York, NY 10016-5990, http://www.asme.org.

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ASTM D7301 – 08
Standard Specification
CONTENTS

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ASTM Committee Structure

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- **130 Technical Committees**
- **Some example industrial committees:**
  - Alloys (Steel, Stainless Steel, and Related) A01
  - Coating (Protective) and Lining Work for Power Generation Facilities D33
  - Fasteners F16
  - Nuclear Energy (US Tag ISO/TC85) 085
  - Nuclear Fuel Cycle C26
  - Nuclear Technology and Applications E10
  - Testing (Mechanical) E28
  - Testing (Nondestructive) E07
  - Underground Utilities (Technology) F36

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ASTM Committee E10

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- **Committee Scope**
  - To promote the advancement of nuclear science and technology and the safe application of energy by:
  - Standardizing measurement techniques and specifications for radiation effects and dosimetry including materials response, instrument response, and fuel burnup.
  - Standardizing the nomenclature and definitions used in or relating to testing methods or instruments in support of nuclear industry.
  - Maintaining a broad expertise in application of nuclear science and technology especially the measurement of radiation effects from environments of nuclear reactor, particle accelerators, indigenous space, spacecraft, and radionuclides.
  - Maintaining a broad expertise in the applications of radionuclides.
  - Sponsoring scientific and technical symposia and publications in the Committee's fields of specialization.
  - Performing liaison with related ASTM Committees and other technical societies and organizations, both national and international.
  - Advising other technical committees of the Society in our field of expertise.

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ASTM Committee E10

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- **Nuclear Technology and Applications**
  - Formed in 1951
  - Has 225 members from 20 countries
  - Has seven technical subcommittees
  - Has jurisdiction over 105 standards
  - Addresses radiation dosimetry, nuclear structural materials, and decontamination & decommissioning
  - Publishes in the Annual Book of ASTM Standards, Vol. 12.02

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Types of Materials

- Ferrous materials
- Graphite materials
- Non-ferrous materials
- Reinforcing fiber material
- Thermoplastic materials
- Thermoset plastic materials

Engineering Properties

- Tensile Strength
- Tensile Modulus
- Compressive Strength
- Compressive Modulus
- Flexural Strength
- Flexural Modulus

Physical Properties

- Thermal Expansion
- Thermal Conductivity
- Thermal Diffusivity
- Modulus of Elasticity
- Poisson's Ratio
- Density of Materials

Relation to Section II (ASME BPV Code)

- Use many ASTM material specifications
- Section II defines the stress limitations (Part D Stress Tables) for the construction sections of the BPV Code.
  - Maximum allowable stress values
  - Tensile strength values
  - Yield strength values
  - Factors for limiting permanent strain

Ferrous Materials

- **Example Material Specifications**
  - ASTM A6 / A6M -07, Standard Specification for General Requirements for Rolled Structural Steel Bars. Plates, Shapes, and Sheet Piling
  - ASTM A36 / A36M, Standard Specification for Carbon Structural Steel
  - ASTM A283 / A283M, Standard Specification for Low and Intermediate Tensile Strength Carbon Steel Plates
  - ASTM A786 / A786M, Standard Specification for Rolled Steel Floor Plates

Non-ferrous Materials

- **Example Material Specifications**
  - ASTM B151 / B151M – 05, Standard Specification for Copper-Zinc Alloy (Nickel Silver) and Copper-Nickel Rod and Bar
  - ASTM B169 / B169M – 05, Standard Specification for Aluminum Bronze Sheet, Strip, and Rolled Bar
  - ASTM B248 / B248M – 96, Standard Specification for General Requirements for Wrought Copper and Copper-Alloy Plate, Sheet, Strip, and Rolled Bar
  - ASTM B265 – 09, Standard Specification for Titanium and Titanium Alloy Strip, Sheet, and Plate

Non-metallic Materials

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- **Thermoplastics**
  - Polyethylene
- **Thermoset Plastics**
  - Epoxy
  - Polyester
  - Vinyl Ester
- **Graphite**
- **Reinforcing Fibers**
  - Glass
  - Carbon
  - Kevlar

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Polyethylene Materials

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- **Example ASTM Standards**
  - ASTM D638-08, Standard Test Method for Tensile Properties of Plastics
  - ASTM D3035-06, Standard Specification for PE Plastic Pipe (DR-PR) Based on Controlled Outside Diameter
  - ASTM D3350-06, Standard Specification for PE Plastics Pipe and Fittings PE compounds
  - ASTM D4218-08, Standard Test Method for Determination of Carbon Black Content in PE Compounds By the Muffle-Furnace Technique
  - ASTM F412-06, Standard Terminology Relating to Plastic Piping Systems
  - ASTM F714-06a, Standard Specification for PE Plastic Pipe (SDR-PR) Based on Outside Diameter
  - ASTM F2620-06 Standard Practice for Heat Fusion Joining of PE Pipe and Fittings

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Case Study

- **Problem: Case Study on how to get a new material defined by an ASTM material specification.**
  - The responsible Committee needs to sponsor the activity. Who typically requests the Committee to be the sponsor?
  - A proposed material specification needs to be developed. Who typically develops the proposal?
  - What kind of technical information is within this material specification?
  - What is the ASTM approval process for the material specification?

Learning Questions

- **How do you find allowable engineering properties and physical properties from an ASTM material specification?**
- **What is the difference between an engineering property and a physical property?**
- **Identify some of the most common ASTM material specifications in use in a NPP.**

UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 28
NDT Personnel Qualification

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Module 28
NDT Personnel Qualification

Instructor: C. Wesley Rowley, P.E.

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Learning Objectives

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- **Learn the personnel qualification requirements of ASNT CP-189.**
- **Learn the important differences between ASNT CP-189 and ASNT TC-1A (and ANSI N45.2.6).**
- **Learn how the various "Levels" of NDE qualification are used in NPPs.**

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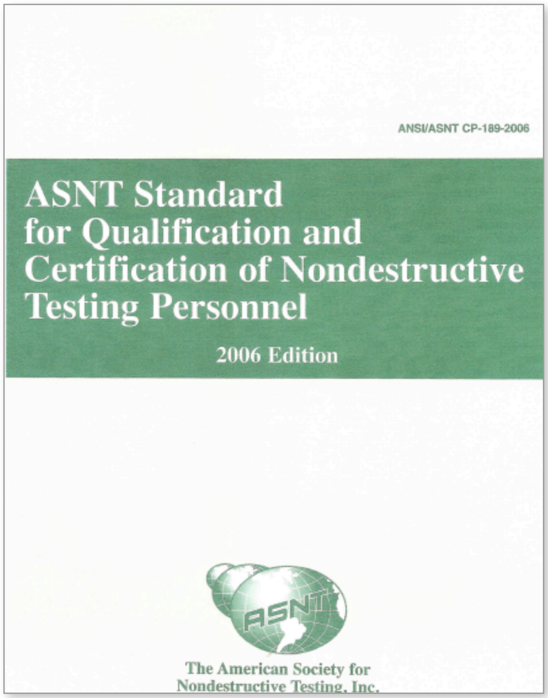
Significant Sub-topics

- Major References
- Terminology
- Major Categories of NDT
- NDT Methods
- Training & experience
- NDT Qualification
- NDT Certification & recertification
- Levels I / II / III
- Role of the NDT Employer
- NDT Records
- ASNT Central Certification Program

Major References

- **ASNT Standards**
  - CP-189-2006 ASNT Standard for Qualification and Certification of Nondestructive Testing Personnel
  - Recommended Practice No. SNT-TC-1A 2006
  - CP-105-2006 ASNT Standard Topical Outlines for Qualification of Nondestructive Testing Personnel
- **ASME Standards**
  - BPV Code, Section III (Nuclear Construction)
  - BPV Code, Section V (NDE)
  - BPV Code, Section XI (Inservice Inspection)
- **ASTM Standards**
  - ASTM E1316 Standard Terminology for Nondestructive Examination (Section A – Common NDT Terms)



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ASNT CP-189-2006
CONTENTS

1.0 Scope
2.0 Definitions
3.0 Levels of Qualification
4.0 Qualification Requirements
5.0 Qualification and Certification
6.0 Examinations
7.0 Expiration, Suspension, Revocation, and Reinstatement of Employer Certification
8.0 Employer Recertification
9.0 Records
10.0 Referenced Publications

Table 1 – Minimum Number of Examination Questions

Appendices

CP-189 Inquiry Form

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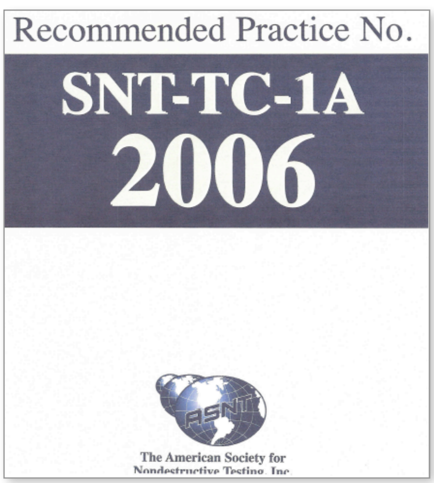

SNT-TC-1A
CONTENTS

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Foreword

Personnel Qualification and Certification in Nondestructive Testing
- Scope
- Definitions
- Nondestructive Testing Methods
- Levels of Qualification
- Written Practice
- Education, Training, and Experience Requirements for Initial Qualification
- Training Programs
- Examinations
- Certification
- Technical Performance Evaluation
- Interrupted Service
- Recertification
- Termination
- Reinstatement

Basic Examination

Appendix
- Example Questions for Level I and Level II
- Answers to Example Questions

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Scope of SNT-TC-1A

- Effectiveness of NDT depends on the capabilities of NDT personnel.
- This Recommended Practice provides guidelines for qualification and certification of NDT personnel.
- This guidelines are to aid the employers of NDT personnel in recognizing the essential factors for qualifying NDT personnel.
- It is recognized that these guidelines may not be appropriate for certain employer's circumstances and / or applications.
- In developing a written practice the employer should review the detailed recommendations and modify them, as necessary, to meet particular needs.

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Terminology

- ASNT – American Society of Nondestructive Testing
- NDT – non-destructive testing
- NDE – non-destructive examination
- PdM – predictive maintenance

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Major Categories of NDT

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- **Surface NDT**
  - Liquid Penetrant
  - Eddy Current
  - Magnetic Particle
  - Visual
- **Volumetric NDT**
  - Acoustic Emission
  - Radiographic
  - Ultrasonic

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NDE Methods

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- **Used in BPV Section III**
  - Radiographic
  - Ultrasonic
  - Magnetic Particle
  - Liquid Penetrant
  - Leak Testing
  - Visual
  - Leak Testing
- ***Note – Reference is ASNT Recommended Practice TC-1A***
- **Used in BPV Section XI**
  - Visual Examination
  - Surface Examination
    - Magnetic Particle
    - Liquid Penetrant
    - Eddy Current
    - Ultrasonic
  - Volumetric Examination
    - Radiographic
    - Ultrasonic
    - Eddy Current
    - Acoustic Emission
- ***Note – Reference is ASNT CP-189 Standard***

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Training and Experience

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- **Training** (excerpt from ASNT CP-189, para 4.1)

  Candidates for certification as NDT Level I or Level II shall complete sufficient organized training to become familiar with the principles of the method and practices of the applicable test technique.This training shall be conducted in accordance with a course outline approved by an NDT Level III.

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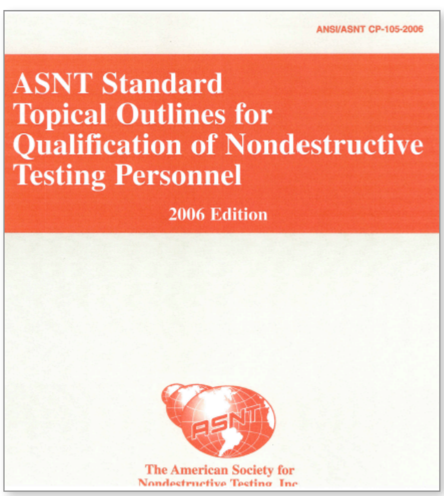

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ASNT CP-105-2006
CONTENTS

- Scope
- Acoustic Emission
- Alternating Current Field Measurement Testing
- Eddy Current Testing
- Flux Leakage Testing
- Remote Field Testing
- Electromagnetic Testing
- Laser Testing
- Holography / Shearography Testing
- Liquid Penetrant Testing
- Magnetic Particle Testing
- Neutron Radiographic Testing
- Radiographic Testing
- Thermal / Infrared Testing
- Ultrasonic Testing
- Vibration Analysis Testing
- Visual Testing
- Basic Examination Level III

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Training and Experience (cont'd)

- **Experience** (excerpt from ASNT CP-189, para 4.2)
  - Candidates for certification shall have acquired the practical experience to assure they are capable of performing the duties of the level in which certification is being sought.
    - NDT Level I and Level II – see Appendix A
    - NDT Level II Limited – see Appendix B

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Training and Experience (cont'd)

- **Previous Training and Experience** (excerpt from ASNT CP-189, para 4.3)
  - NDT Level I and Level II
    - A candidate's previous training and experience may be accepted by the employer's NDT Level III if documented and verified.
    - Any claimed training or experience which is not documented and cannot be verified shall be considered invalid.
  - NDT Level III
    - The employer shall verify and document the current validity of a candidate's ASNT Level III certificate.

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Training and Experience (cont'd)

- **NDT Instructor** (excerpt from CP-189, para 4.4)
  - Criteria – An NDT instructor shall meet at least one of the following criteria:
    - Possess a current ASNT Level III certificate in NDT method…
    - Have academic credentials at least equivalent to a B.S. … and possess adequate knowledge of NDT method to be taught.
    - Be graduate of two-year school … And gave five or more years of experience as an NDT Level II in the NDT method to be taught.
    - Have ten or more years of NDT experience as an NDT Level II or equivalent in the NDT method to be taught.
  - Designation
    - The NDT instructor shall be designated by an NDT Level III individual.
    - The designation shall become part of the individual's qualification records.

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Training and Experience (cont'd)

- **Outside Services** (excerpt from CP-189, para 4.5)
  - At the option of the employer, an outside organization may be engaged to perform the duties of an NDT Level III.
  - For organizations other than ASNT, the employer shall be responsible for evaluating the organization to assure the services are in accordance with the employer's certification procedure and this standard, and so documented.
  - An NDT Level III of the engaged outside organization shall be responsible for the services provided.

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Levels of Qualification

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- **NDT Level III**
- **NDT Level II**
- **NDT Level II Limited**
- **NDT Level I**
- **Trainee**
- **NDT Instructor**

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NDT Certification

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- **Procedure** (excerpt from ASNT CP-189, para 5.1)
  - The employer shall develop and maintain a procedure detailing the program that will be used for qualification and certification of NDT personnel in accordance with this standard.
- **Procedure Requirements** (excerpt from ASNT CP-189, para 5.2)
  - The procedure shall describe the minimum requirements for certifying personnel in each NDT method and the levels of qualification desired. The procedure shall satisfy the requirements of this standard.

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NDT Certification (cont'd)

- **The procedure shall include, as a minimum, the following:**
  - Personnel duties and responsibilities
  - Training requirements
  - Experience requirements
  - Examination requirements
  - Records and documentation requirements (including control, responsibility, and retention period)
  - Recertification requirements
- **Approval** (excerpt from ASNT CP-189, para 5.3)
  - The employer's certification procedure shall be approved by an NDT Level III designated by the employer.

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NDT Recertification

- **Expiration, Suspension, Revocation, and Reinstatement of Employer's Certification** (excerpt from ASNT CP-189, para 7.0)
  - Expiration
  - Suspension
  - Revocation
  - Reinstatement
    - Suspended Certification
    - Expired or Revoked Certification
    - Expired for Termination of Employment
      - The employer has maintained the personnel certification records required in Section 9 (para 9.0 Records).
      - The employee certification did not expire during termination.
      - The employee is being reinstated within 12 months of termination.

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NDT Recertification

- **Employer Recertification** (excerpted from ASNT CP-189, para 8.0)
  - NDT Level I and NDT Level II
    - Every five years … documented 350 hours of experience and successfully passed examination.
    - Every ten years … repeat examination.
  - NDT Level III – shall be recertified by the employer every five years by verifying the individual's ASNT Level III certificate is current in each method for which recertification is being sought.

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Role of the NDT Employer

- **Shall Be Responsible for NDT Employee**
  - Training
  - Qualification
  - Certification
- **May Subcontract Portions to**
  - NDT Specialty Organizations
  - ASNT Central Certification Program
- **Shall Be Responsible for**
  - Records Generation
    - NDT Employee
    - NDT Procedure
    - NDT Results
  - Records Retention

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NDT Records

- Responsibility for documentation is the normally the employer.
- The ASNT Central Certification Program may responsible for some senior NDT personnel.
- Certification documentation must include training record, certification record, experience record, record of previous experience, current examinations, and vision examination record.

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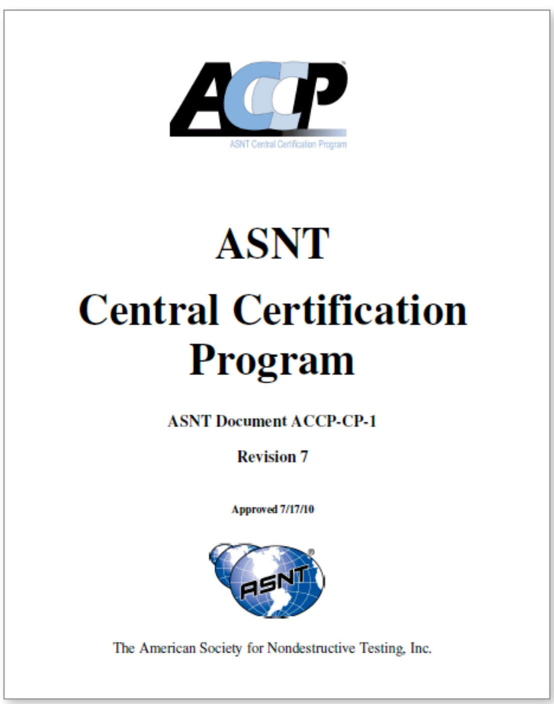
ACCP
ASNT Central Certification Program

ASNT

Central Certification Program

ASNT Document ACCP-CP-1

Revision 7

Approved 7/17/10

ASNT

The American Society for Nondestructive Testing, Inc.

ASNT Document ACCP-1
CONTENTS

1 Scope
2 Definitions
3 Categories of Qualification
4 Responsibilities
5 Industrial Sectors
6 Eligibility for Examination
7 Qualification Examinations
8 Examination Results
9 Eligibility for Certification
10 Certification
11 Certification Validity
12 Renewal by Points
13 Renewal by Examination
14 Documentation
15 Applicant Rights
16 Program Change Notification
17 Accommodation for Disabilities
Appendices

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ASNT Central Certification Program

- **Reference – ACCP CP-1, Revision 7 (17 July 2010)**
  - Standardized requirements
  - Administered by an accredited certification body
  - Administered and maintained by ASNT (other organizations may be licensees of ASNT – i.e., AWS)
  - Independent, transportable certifications
  - National and international acceptance
  - ACCP Level II and ACCP Professional Level III
  - Pressure Equipment (PE) is one of the industrial sectors

Case Study on Qualification of LP Technicians

- **Problem: A small business has hired three technicians and needs to get them "certified" to perform LP NDT on fabricated castings.**
  - What are the germane experience factors?
    - Level I
    - Level II
  - What are the germane training factors?
    - Level I
    - Level II
  - What are the germane certification factors?

Learning Questions

- **What are the major personnel qualification requirements of ASNT CP-189?**
- **What are the important differences between ASNT CP-189 and ASNT TC-1A?**
- **How are the various "Levels" of NDE qualification used in NPP construction projects? In operating NPPs?**

Lesson 29

ASME BPV Code Section XI Pre-Service Testing

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Lesson 29
ASME BPV Code Section XI Pre-Service Testing

Instructor: Gene Imbro

Learning Objectives

- **Learn the meaning of the Section XI terms**
  - Inservice Inspection,
  - Pre-service Testing,
  - Flaw Evaluation, and
  - Flaw Acceptance

Major Subtopics

- **Three divisions**
  1. Light-water cooled reactors (483 pages)
     - Subsections for light-water cooled reactors
       - IWA – General requirements
       - IWB – Requirements for Class 1
       - IWC – Requirements for Class 2
       - IWD – Requirements for Class 3
       - IDE – Requirements for Class MC
       - IWF – Requirements for Supports
       - IWL – Requirements for Concrete Components
       - Mandatory Appendices
       - Non-Mandatory Appendices

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Major Subtopics (cont'd)

- **Three divisions (cont'd)**

  2. Gas-cooled reactors (2 pages)

  3. Liquid metal cooled reactors (1 page)

4

Introduction - Early History

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- **Early power plant designers used "high" standards so that passive components of reactors could operate for their life without attention.**
- **In 1966, the AEC (NRC) recognized an inspection program will be necessary for pressure-containing components.**

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Objective & Scope

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- **A committee was developed and accepted as a subgroup of the ASME Section III Boiler and Pressure Vessel Committee.**
- **ASME Section XI code was published in 1970, originally containing 24 pages of text.**
  - Today, contains over 500 pages and covers Class 1, 2 and 3 systems primarily for light-water reactors

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IWA – General Requirements

- **Points user to other Subsections (i.e., IWB-, IWC-, etc.)**
- **IWA-1000 – Scope and Responsibility**
- **IWA-2000 – Examination and Inspection**
  - Duties, qualification, access for inspectors (Including NRC inspectors)
  - Examination methods (visual-VT, surface-MP/EC, volumetric-UT/R, alternative-AE; more details later)

IWA – General Requirements

- **IWA-2000 – Examination and Inspection (cont'd)**
  - Qualifications of Nondestructive Examination Personnel – ASNT qualified Level 1 < Level 2 < Level 3
  - Inspection program – (details later)
  - Extent of examination (excludes welds for repairs in base metals)
  - Weld reference system (i.e., 0-degrees is top of pipe)

IWA – General Requirements

- **IWA-3000 – Standards for Examination Evaluation**
  - Significant digits for limiting values
  - Flaw characterization (discussed later)
  - Linear flaws detected by surface or volumetric examination (more detail later)
- **IWA-4000 – Repair/Replacement Activities**
  - General Requirements
  - Items for Repair/Replacement Activities
  - Design

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IWA – General Requirements

- **IWA-4000 – Repair/Replacement Activities**
  - Welding, Brazing, Metal Removal, Fabrication and Installation
  - Examination and Testing
  - Alternative Welding Methods, i.e., temper bead welding for repair of ferritic materials to avoid post-weld heat treatment
  - Heat Exchanger Tubing – plugging, explosive welding, friction welds, etc.

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IWA – General Requirements

- **IWA-5000 – System Pressure Tests**
  - General
  - System Test Requirements
  - Test Records
- **IWA-6000 – Records and Reports**
  - Scope
  - Requirements
  - Retention
- **IWA-9000 – Glossary**

IWB – Class 1 Components

- **IWB-1000 – Scope and Responsibility**

- **IWB-2000 – Examination and Inspection**
  - Pre-service
  - Inspection Schedule
  - Examination and Pressure Test Requirements

IWB – Class 1 Components

- **IWB-3000 – Acceptance Standards**
  - Evaluation of examination results
  - Supplemental examinations
  - Standards
  - Acceptance Standards- Workmanship flaw tables
  - Analytical Evaluation of Flaws
    - 3610 – 4” and thicker ferritic steel components
    - 3620 – Less then 4” thick ferritic steel components
    - 3630 – Steam generator tubing

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IWB – Class 1 Components

- **IWB-3000 – Acceptance Standards**
  - Analytical Evaluation of Flaws (cont'd)
    - 3630 – Steam generator tubing
    - 3640 – Flaws in austenitic and ferritic piping
    - 3660 – RPV head penetration nozzle flaws
    - 3700 – Analytical evaluation of operating plant events
  - Non-Mandatory Appendix A, C, G, H, K, L, O, Q
- **IWB-5000 – System Pressure Tests**

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IWC & IWD – Class 2 & 3 Components

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- **IWC-XXXX for Class 2 piping (39 pages)**
- **IWD-XXXX for Class 3 piping (10 pages)**
- **Generally IWC and IWD have much less detail than IWB, and IWC and IWD will frequently refer user to IWB.**
  - Exception might be some criteria specific to Class 2/3 piping, i.e., flow-accelerated corrosion (FAC) also called erosion-corrosion.

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Other Subsections

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- **IWE-XXXX for Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants (10 pages)**
  - i.e., drywell containment vessel for BWRs (Oyster Creek corrosion)
- **IWF-XXXX for Requirements for Class 1, 2, 3 and MC component supports of Light-Water Cooled Plants (6 pages)**
- **IWL-XXXX for Requirements for Class CC**

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Other Subsections

- **IWL-XXXX for Requirements for Class CC Concrete Components of Light-Water Cooled Plants (14 pages)**
- **Appendices**
  - Mandatory (I-X)
  - Nonmandatory (A-R)

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Inspections

TABLE IWB-2411-1
INSPECTION PROGRAM A

Inspection Interval	Inspection Period, Calendar Years of Plant Service	Minimum Examinations Completed, %	Maximum Examinations Credited, %
1st	3	100	100
2nd	7	33	67
	10	100	100
3rd	13	16	34
	17	40	50
	20	66	75
	23	100	100
4th	27	8	16
	30	25	34
	33	50	67
	37	75	100
	40	100	. . .

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Inspections

- **The Code allows option for inspection programs, but a 10-year interval was chosen based on historical failure rate data**

- **Risk-based inspection currently being used to determine inspection frequencies of different components**
  - Non-mandatory Appendix R

Inspection Methods

- **Originally concerned with fatigue cracking.**

  - Note, the design sections of the ASME Code are made for preclusion of overload failures of unflawed components and fatigue failures, not any other degradation modes.

  - SCC is much more common in nuclear plants – Code does good job in designing to avoid fatigue failures

Inspection Methods (cont'd)

- **UT was chosen over RT for superiority in locating and sizing fatigue cracks**
  - UT can be performed from one surface
  - Appendices were developed for techniques for improving UT reliability
    - Appendix I for Vessels in 1973
    - Appendix III for piping in 1975
    - Currently eight appendices for UT

Inspection of Class 1 Systems

- **Systems subject to examination include;**
  - Reactor coolant system (RCS)
  - Portions of the auxiliary systems connected to RCS
  - Portions of the Emergency Core Coolant System (ECCS)
- **ISI requirements were developed during and after the design/order of most US power plants**

Inspection of Class 1 Systems (cont'd)

- **Prior to 1977**
  - Only 5% of each circumferential and 10% of each longitudinal vessel weld was required
    - Except vessel-to-flange and head-to-flange welds
- **After 1977**
  - 100% of the length of 25% of piping circumferential welds, and all circumferential dissimilar welds are required to be inspected

Flaw Characterization

- **If a flaw is found, it must first be characterized**
- **The code gives guidance in the form of figures for determining flaw size, etc. for analyses**
- **The figures are for all flaw types**
  - Surface, subsurface, multiple, planar, non-planar, etc.

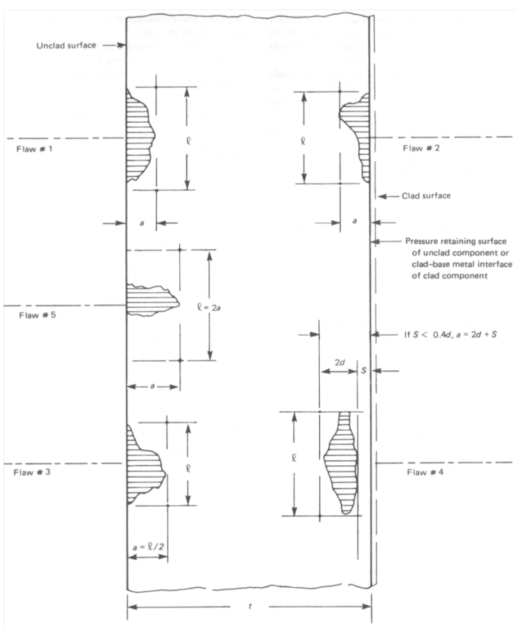

FIG. IWA-3310-1 SURFACE PLANAR FLAWS ORIENTED IN PLANE NORMAL TO PRESSURE RETAINING SURFACE
Illustrative Flaw Configurations and Determination of Dimensions a and ℓ
($\frac{1}{2}$ in. = 13 mm)

Flaw Characterization

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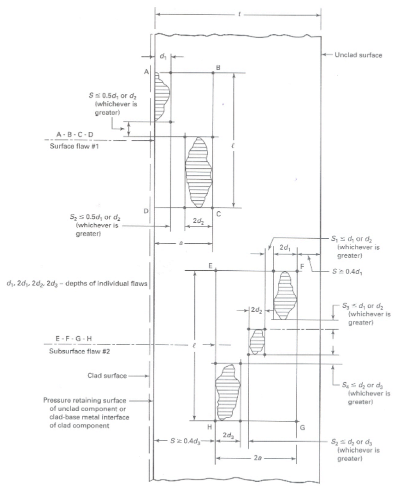
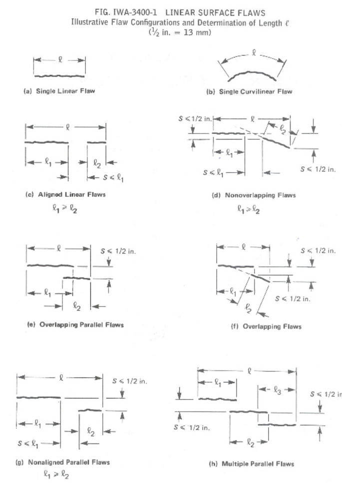
- **Spacing criteria (S) being updated**
  - Flaw interaction difference for subcritical crack growth like SCC or fatigue cracks than for failure criteria





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Flaw Acceptance Standards

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- **After the flaw is characterized, its size is compared with the Acceptance Standards**
- **These "Acceptance Standard" flaw sizes are also known as "Workmanship flaws"**

TABLE IWB-3410-1
ACCEPTANCE STANDARDS

Examination Category	Component and Part Examined	Acceptance Standard
B-A, B-B	Vessel welds	IWB-3510
B-D	Full penetration welded nozzles in vessels	IWB-3512
B-F, B-J	Dissimilar and similar metal welds in piping and vessel nozzles	IWB-3514
B-G-1	Bolting greater than 2 in. (50 mm) in diameter	IWB-3515 IWB-3517
B-G-2	Bolting 2 in. (50 mm) in diameter and less	IWB-3517
B-K	Welded attachments for vessels, piping, pumps, and valves	IWB-3516
B-L-1, B-M-1	Welds in pumps and valves	IWB-3518
B-L-2, B-M-2	Pump casings and valve bodies	IWB-3519
B-N-1, B-N-2, B-N-3	Interior surfaces and internal components of reactor vessels	IWB-3520
B-O	Control rod drive and instrument nozzle housing welds	IWB-3523
B-P	Pressure retaining boundary	IWB-3522
B-Q	Steam generator tubing	IWB-3521

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Flaw Acceptance

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- **Flaws that are smaller than these flaw sizes are acceptable for continued service without any evaluation**
  - Tables being updated recently
- **Flaws that are larger than the acceptable flaw size can either be repaired, or replaced, or found acceptable by analytical evaluation**

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Flaw Acceptance (cont'd)

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TABLE IWB-3510-1
ALLOWABLE PLANAR FLAWS
Material: Ferritic steels that meet the requirements of NB-2331 and G-2110(b) of Section III

	Volumetric Examination Method, Nominal Wall Thickness,[1,2] t, in. (mm)					
	$2^1/_2$ (65) and less		4 (100) to 12 (300)		16 (400) and greater	
Aspect Ratio,[1] a/ℓ	Surface Flaw,[5] a/t, %	Subsurface Flaw,[3,4] a/t, %	Surface Flaw,[5] a/t, %	Subsurface Flaw,[3,4] a/t, %	Surface Flaw,[5] a/t, %	Subsurface Flaw,[3,4] a/t, %
0.0	3.1	3.4 Y	1.9	2.0 Y	1.4	1.5 Y
0.05	3.3	3.8 Y	2.0	2.2 Y	1.5	1.7 Y
0.10	3.6	4.3 Y	2.2	2.5 Y	1.7	1.9 Y
0.15	4.1	4.9 Y	2.5	2.9 Y	1.9	2.1 Y
0.20	4.7	5.7 Y	2.8	3.3 Y	2.1	2.5 Y
0.25	5.5	6.6 Y	3.3	3.8 Y	2.5	2.8 Y
0.30	6.4	7.8 Y	3.8	4.4 Y	2.9	3.3 Y
0.35	7.4	9.0 Y	4.4	5.1 Y	3.3	3.8 Y
0.40	8.3	10.5 Y	5.0	5.8 Y	3.8	4.3 Y
0.45	8.5	12.3 Y	5.1	6.7 Y	3.9	4.9 Y
0.50	8.7	14.3 Y	5.2	7.6 Y	4.0	5.6 Y

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Analytical Evaluation of Flaws

- **The Code separates the evaluation of flaws into five categories**
  - Ferritic components where t > 4 inches (102mm)
  - Ferritic components where t < 4 inches
  - Steam generator tubing
  - Ferritic and austenitic piping
  - PWR head penetration nozzles

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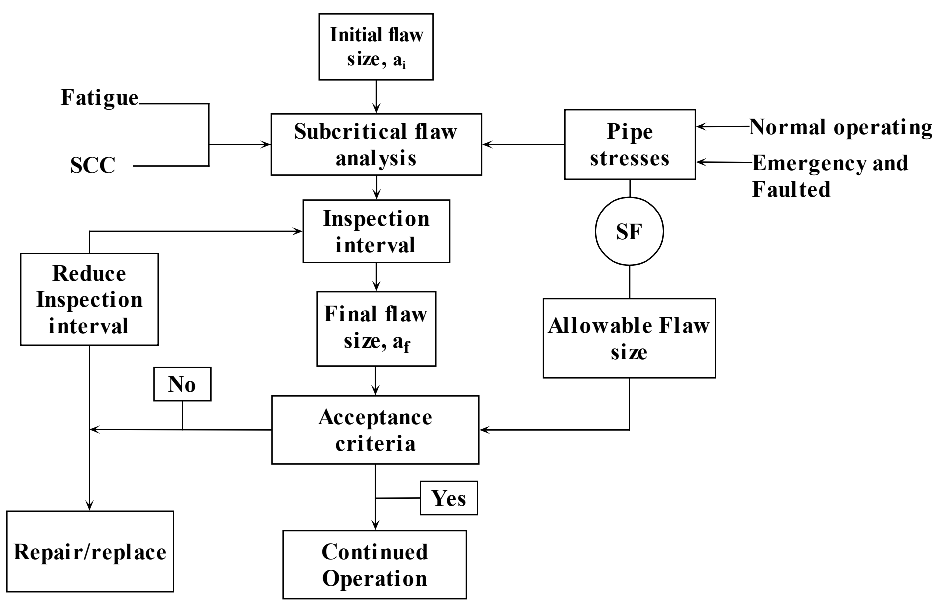
Flaw Evaluation Flow Chart



Flaw Evaluation for Class 1 Piping

- **The Code gives the user choices in evaluating flaws in piping**
- **If the flaws exceed the workmanship size flaws, they can be analyzed by;**
  - Following the procedures in Nonmandatory Appendix C
  - Following the procedures in Nonmandatory Appendix H

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Flaw Evaluation for Class 1 Piping (cont'd)

- **If the flaws exceed the workmanship size flaws, they can be analyzed by (cont'd):**
  - Performing an alternate procedure, i.e., finite element analyses and demonstrating that the allowable loads have the following safety factors
    - Service level A – 2.7 (Normal)
    - Service level B – 2.4 (Upset – occasional deviation)
    - Service level C – 1.8 (Emergency – infrequent requires shut down for inspection & repair prior to re-start)
    - Service level D – 1.4 (Faulted – extremely low probability eg safe shutdown earthquake)

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Appendix C – Alternative Pipe Flaw Evaluation Criteria

- **An Appendix C analyses has the following steps:**
  - Determine flaw size
  - Resolve size into circumferential and axial components
  - Determine stresses perpendicular to flaw for Service Level A-D
  - Perform a flaw growth analysis to determine flaw size at end-of-evaluation time period

Appendix C – Alternative Pipe Flaw Evaluation Criteria (cont'd)

- **An Appendix C analyses has the following steps (cont'd)**
  - Obtain material properties at operating conditions
  - Determine failure mode
  - Determine allowable flaw size or allowable stress (with appropriate safety factors)
    - Note the term "Safety Factor" is being changed to "Structural Factor"
  - Apply acceptance criteria
  - Recently updated for DMW Welds (In82/182 – SCC susceptible materials in PWRs.

Section XI Code Cases and Relief Request

- **The Boiler and Pressure Vessel Committee meets regularly to consider proposed additions and revisions to the Code and to formulate Cases to clarify the intent of existing requirements or provide, when the need is urgent, rules for materials or constructions not covered by existing Code rules**
  - >200 code cases in existence (N-4 to N-785 as of 10/09)
  - ½ of them are for Section XI – rest for all other divisions of the code

Section XI Code Cases and Relief Request (cont'd)

- **The BPV Revisions (cont'd)**
  - More recent ones deal with evaluation of PWSCC cracking inspection, evaluations, and repairs, i.e.,
    - N-735 – Successive inspections of Class 1 and 2 pipe welds
    - N-740 – Dissimilar weld metal overlay for repair of Class 1, 2, and 3 items
    - N-755 – Polyethylene Piping for Safety-Related Class 3 Service Water

Evolving Areas of Section XI

- **Plastic pipe being approved for service water lines (Code Case N-755)**
  - Next step is inspection and flaw evaluation – lack of fusion girth welds
- **Buried service water line flaw acceptance criteria**
  - Developing new procedures in Section XI for corrosion in steel buried pipes
    - Similar to natural gas/oil line issues, but level of tolerable leakage sensitive to contaminates in the line

Evolving Areas of Section XI

- **Gen IV reactors developing design criteria in Section III NH**
  - Flaw acceptance criteria for creep/fatigue design may be in Section XI

Evolving Areas of Section XI

- **Code is not based on guidance for avoidance of SCC (most common type of degradation mechanism in existing plant high energy systems) – needs some significant improvements!**
  - SCC requires combination of material, water environment, high stresses

Evolving Areas of Section XI

- **Code is not based on guidance for avoidance of SCC (cont'd)**
  - Usual cure is to change to a new material, try adjusting water chemistry, stress mitigation, repairs (overlays) or replacements
  - For new plant construction, need better guidance on how to fabricate welds with reduced or no tensile residual stresses on wetted surface of pressure boundary

Module 30
Valve Functional Specification

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Module 30
Valve Functional Specification

Instructor: Gene Imbro, P.E.

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Learning Objectives

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- **Learn how valves are functionally specified in an NPP design specification.**
- **Learn what are the specification differences for self-operated and power-operated valves.**

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Significant Sub-Topics

- Major References
- Scope of ANSI / ASME N278.1
- Valve Application Characteristics
- Structural Requirements
- Operational Requirements
- Seat Leakage Limits
- Pressure Relief Valve Characteristics
- Special Material Requirements
- Installation Requirements
- Maintenance Requirements

Major References

- **ANSI / ASME N278.1-1975 (1992), Self-Operated and Power Operated Valves Functional Specification Standard**
- **ASME BPV Code – 2010, Section III, Rules for Construction of Nuclear Power Plant Components**
- **Subsection NCA - 3250, Design Specification**
- **Subsection NB, Class 1 Components**
  - λ Paragraph NB-3500, Valve Design

AMERICAN NATIONAL STANDARD
REACTOR PLANTS AND THEIR MAINTENANCE

Self-Operated and Power-Operated
Safety-Related Valves
Functional Specification Standard

ANSI N278.1 - 1975

REAFFIRMED 1992
FOR CURRENT COMMITTEE PERSONNEL
PLEASE SEE ASME MANUAL AS-11

SECRETARIAT
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United Engineering Center 345 East 47th Street New York, N. Y. 10017

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CONTENTS

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Scope of the Standard

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- **This standard establishes requirements for functional specification for safety-related self-operated and power-operated valves for applications in nuclear power plant.**

- ***Question – how does this Functional Specification relate to the Design Specification required by ASME BPV Code Section III, Sub-section NCA?***

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Functional/Design Specification Relationship

- **The Functional specification:**
  - Supplements the piping and valve codes and standards for pressure integrity.
  - Can be provided as part of the Sec III Design Specification or as a separate document.
  - Is intended to assure that the operating conditions and safety related functions are addressed.
  - Permits valve and actuator manufacturers to provide the proper valve for the application (product design and materials).

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Functional Specification

- **Identifies the safety related function(s) of the valve.**
- **Also identifies the following:**
  - Structural requirements
  - Operational requirements
  - Seat leakage limits
  - Pressure relief valve characteristics
  - Special material requirements
  - Installation requirements
  - Maintenance requirements

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Application Characteristics

- **Power-operated versus self-operated**
  - Safety – Relief Valve?
  - Frequent versus infrequent use
  - Low leakage versus nominal leakage
  - Normally open versus normally closed

Note A – frequent use is operating over 500 times during service life.

Note B – low leakage is less than 2 cc/hr per inch of diameter.

Note C – nominal leakage is less than 10 cc/hr per inch of diameter.

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Structural Requirements

- **Design pressure and temperature**
  - Normal operating pressure and temperature
  - Flow capacity at stated differential pressure
  - Time-temperature data for significant thermal transients, with numbers of cycles (Class 1 valves)
  - Seismic accelerations and dynamic loadings without loss of valve functionality
  - Loadings from structural supports and restraints
  - Fundamental frequency of valve assembly

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Operational Requirements

- **Anticipated modes of valve operation**
  - Power versus operating conditions
  - Environmental conditions
  - Operating conditions during:
  - Installation testing
  - System hydrostatic testing
  - Preoperational testing
  - Startup testing
  - Normal and abnormal plant operation
  - Inservice testing and exercising

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Seat Leakage Limits

- **Acceptance leakage limits for SRVs**
- **Acceptance leakage limits for:**
  - Low leakage valves
  - Nominal leakage valves
  - Direction of leakage and pressure differential

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Pressure Relief Valve Characteristics

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- Fluid
- Set pressure
- Set pressure range
- Set pressure tolerance
- Discharge capacity
- Overpressure
- Blowdown
- Static and dynamic back pressure (minimum and maximum)
- Response time

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Special Material Requirements

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- Unique material requirements of the valve, actuator, and actuator controls
- Items shall include:
- Unacceptable part or trim materials
- Halogen limits for gaskets and packing
- Limitations on non-ferrous materials
- Special surface preparations or coatings
- Unusual process fluid chemistry

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Installation Requirements

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- **Valve orientation**
- **SRV piping arrangements**
- **Details of water seal arrangements**

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Maintenance Requirements

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- **Special provision for valve maintenance shall be specified.**

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Learning Questions

- **What are the specification differences for self-operated and power-operated valves?**
- **What is the difference between a safety valve and a relief valve.**

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 31
ANSI/ANS-56.3-1983

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Module 31
ANSI/ANS-56.3-1983

Instructor: Gene Imbro, P.E.

Learning Objectives

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- **Design provisions for overpressure protection of low pressure systems connected to high pressure portions of the RCBP to which attached**

ANSI/ANS 56.3

- **Title: Overpressure protection of low pressure systems connected to the reactor coolant pressure boundary**

Major Sub-topics

- Overpressure protection methods
- Instrumentation and control design requirements for overpressure protection systems
- Testing requirements for devices of the overpressure protection systems, including preoperational and inservice testing

Introduction

1. Purpose and Scope
2. Methods of Overpressure Protection
3. Instrumentation and Control
4. Testing of Overpressure Protection Devices

5

Purpose and Scope

- **Purpose**
  - To specify minimum requirements for the overpressure protection of low pressure systems connected to the reactor coolant pressure boundary of LWRs

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Purpose and Scope

- **Scope**
  - Covers the interaction of those low pressure systems with the high pressure portions of the RCPB, which, if overpressurized by the reactor coolant system, could result in the failure of the low pressure system and:
    - The prevention of an orderly reactor shutdown, or
    - The inability to main a safe shutdown condition assuming makeup is provided by normal makeup systems only, or

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Purpose and Scope

- **Scope (cont'd)**
  - The release of radioactive materials at the site boundary exceeding 0.5 rem to the whole body or its equivalent to any part of the body

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Methods of overpressure protection

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- **Design basis**
  - Single failure criterion
  - The components interfacing with the RCBP shall have the same safety classification as the RCBP component that it connected to
  - The pressure relief devices shall be of the same safety classification as the low pressure system to which attached

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Methods of overpressure protection

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- **Methods for integrated overpressure protection**
  - Involving pressure control (isolation and/or flow limitation) and protection against single failure of pressure control devices (isolation valves and pressure reducing devices)
  - Isolation
    - At least two valves in series shall be provided to isolate the low pressure system whenever the reactor pressure is above the design pressure of the low pressure system

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Methods of overpressure protection

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- Isolation (cont'd)
  - Using two power operated valves in series for isolation
  - Using a single check valve and one power operated valve for isolation
  - Two check valves in series for systems that perform emergency core cooling

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Methods of overpressure protection

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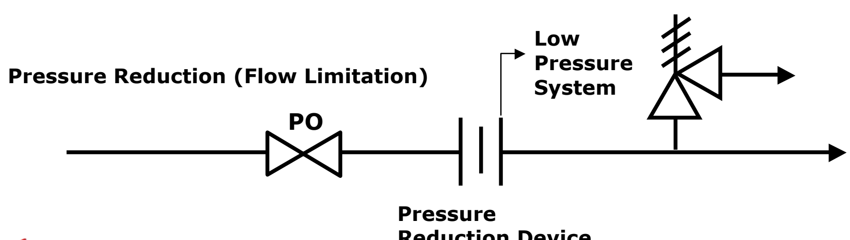
- Pressure Reduction
  - The low pressure system pressure design shall include pressure relief protection with sufficient capacity for the worst case pressure transient
  - The low pressure system design shall include pressure relief protection for the single active failure of a flow limiting valve or inservice degradation of flow control orifices



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Methods of overpressure protection

- Pressure reduction (cont'd)
  - A power operated isolation valve shall be provided in series with the flow limiting device. The power operated valve shall be operable from the control room
  - The discharge from the relief valve shall be contained

Instrumentation and Control

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- **Design basis**
  - Overpressure protection of low pressure systems connected to RCBP is a safety related function
- **Design criteria**
  - Design of the I and C for overpressure protection system shall meet the requirements of applicable IEEE Stds
  - Power operated valves shall be capable of either remote operation from the control room or local station, both subject to intervention or proper interlocks

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Instrumentation and Control

- **Design criteria (cont'd)**
  - Open/Close status shall be indicated in the control room
  - Control room indication shall be provided to indicate when isolation is necessary
  - Process variables to be sensed may include, but not limited to the following:
    - High pressure system pressure with associated set point to prevent opening of isolation valves

Instrumentation and Control

- Process variables (cont'd)
  - High pressure system pressure with associated set point to initiate automatic isolation, alarm, or both
  - Low pressure system pressure with associated set point to initiate automatic isolation, alarm, or both

Testing of Overpressure Protection Devices

- **Objectives for the testing programs**
  - Capability to reliably perform its intended function
  - Operability over the design service life
  - Detection of degrading conditions
- **Pre-operational testing**
  - Isolation valves shall be subjected to pre-operational testing as part of the system test

Testing of Overpressure Protection Devices

- **Pre-operational testing (cont'd)**
  - Safety and relief valves shall be bench tested and the set points verified. Retesting is required after each adjustment
  - Pressure reducing devices shall be subjected to pre-operational testing to verify their design parameters and function and shall be equipped with provisions for pressure drop determination

Testing of Overpressure Protection Devices

- **Operational testing**
  - Shall comply with the requirements in ASME Section XI
  - Valves and their actuating devices and signals shall be periodically functionally tested
  - Pressure reducing devices shall be periodically checked against specified design parameters
  - Check valves shall be periodically tested to ensure their isolation functions

Module 32
Nondestructive Examination

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 32
Nondestructive Examination

Instructor: C. Wesley Rowley, P.E.

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Learning Objectives

- **Learn which NDE methods are acceptable within ASME BPV Code Section V Subsection A and used in NPPs.**
- **Learn which ASTM standards are adopted by ASME BPV Code Section V Subsection B and used in NPPs.**
- **Learn which Section V code cases are applicable to NPPs.**

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  - Working Group on Acoustic Emissions
  - Working Group on Radiography
  - Working on Ultrasonics

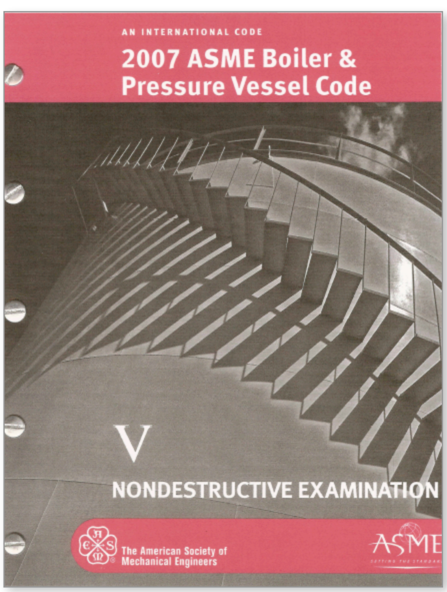
Significant Sub-topics

- Major References
- Subsection A Nondestructive Methods of Examination
- Subsection B Documents Adopted by Section V
- Mandatory and Non-mandatory Appendices
- Index (by subject)
- Code Cases

Major References

- **ASME BPV Code**
  - Section V Nondestructive Examination
    - Subsection A Nondestructive Methods of Examination
    - Subsection B Documents Adopted by Section V
- **ASNT Standards**
  - CP-189 Qualification and Certification of Nondestructive Testing Personnel
  - TC-1A Recommended Practice

5



CONTENTS

Subsection A Nondestructive Methods of Examination

Article 1
General Requirements

T-110 Scope
T-120 General
T-130 Equipment
T-150 Procedure
T-160 Calibration
T-170 Examinations and Inspections
T-180 Evaluation
T-190 Records / Documentation

T-110 SCOPE

(a) This Section of the Code contains requirements and methods for nondestructive examination (NDE), which are Code requirements to the extent they are specifically referenced and required by other Code Sections or referencing document.

These NDE methods are intended to detect surface and internal imperfections in materials, welds, fabricated parts, and components.

They include radiographic examination, ultrasonic examination, liquid penetrant examination, magnetic particle examination, eddy current examination, visual examination, leak testing, and acoustic emission examination.

T-110 Scope (cont'd)

See Nonmandatory Appendix A of this Article for a listing of common imperfections and damage mechanisms, and the NDE methods that are generally capable of detecting them.

(b) For general terms such as Inspection, Flaw, Discontinuity, Evaluation, etc., refer to Mandatory Appendix I.

MANDATORY vs. NON-MANDATORY APPENDICES

- Mandatory – must be followed when implementing Section V.
- Non-Mandatory – not required to be followed. They may be followed and used as guidance.

Article 1
Nonmandatory Appendix

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- **Appendix A – Imperfection vs Type of NDE Method**
  - Table A-110 lists common imperfections and the NDE methods that are general capable of detecting flaws.
  - Listed Imperfections
    - Service Induced Imperfections
    - Welding Imperfections
    - Product Form Imperfections

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ARTICLE 1 2010 SECTION V

TABLE A-110
IMPERFECTION VS TYPE OF NDE METHOD

	Surface [Note (1)]		Sub-surf. [Note (2)]		Volumetric [Note (3)]				
	VT	PT	MT	ET	RT	UTA	UTS	AE	UTT
Service-Induced Imperfections									
Abrasive Wear (Localized)	●	◑	◑	...	●	◑	◑	...	◑
Baffle Wear (Heat Exchangers)	●	...	...	◑	...	...	...	...	...
Corrosion-Assisted Fatigue Cracks	○	◑	●	...	○	●	...	●	...
Corrosion -Crevice	●	...	...	...	...	...	...	...	○
-General / Uniform	...	...	...	○	◑	...	◑	...	●
-Pitting	●	●	○	...	●	○	○	◑	○
-Selective	●	●	○	...	...	...	...	...	○
Creep (Primary) [Note (4)]	...	...	...	...	...	...	...	...	...
Erosion	●	...	...	...	●	○	◑	...	◑
Fatigue Cracks	○	●	●	◑	◑	●	...	●	...
Fretting (Heat Exchanger Tubing)	◑	...	...	◑	...	...	...	...	◑
Hot Cracking	...	◑	◑	...	◑	○	...	◑	
Hydrogen-Induced Cracking	...	◑	◑	...	○	◑	...	◑	...
Intergranular Stress-Corrosion Cracks	...	...	...	...	...	○	...	...	...
Stress-Corrosion Cracks (Transgranular)	○	◑	●	○	◑	◑	...	◑	...
Welding Imperfections									
Burn Through	●	...	...	...	●	◑	...	...	○
Cracks	○	●	●	◑	◑	●	○	●	...
Excessive/Inadequate Reinforcement	●	...	...	...	●	◑	○	...	○
Inclusions (Slag/Tungsten)	...	...	◑	◑	●	◑	○	○	...
Incomplete Fusion	◑	...	◑	◑	◑	●	◑	◑	...
Incomplete Penetration	◑	●	●	◑	●	●	◑	◑	...
Misalignment	●	...	...	...	●	◑	...	...	...
Overlap	◑	●	●	○	...	○	...	...	...
Porosity	●	●	○	...	●	◑	○	○	...
Root Concavity	●	...	...	...	●	◑	○	○	○
Undercut	●	◑	◑	○	●	◑	○	○	...
Product Form Imperfections									
Bursts (Forgings)	○	●	●	◑	◑	◑	◑	●	...
Cold Shuts (Castings)	○	●	●	○	●	◑	◑	○	...
Cracks (All Product Forms)	○	●	●	◑	◑	◑	○	●	
Hot Tear (Castings)	○	●	●	◑	◑	◑	○	○	...
Inclusions (All Product Forms)	...	...	◑	◑	●	◑	○	○	...
Lamination (Plate, Pipe)	○	◑	◑	...	...	○	●	○	●
Laps (Forgings)	○	●	●	○	◑	...	○	○	
Porosity (Castings)	●	●	○	...	●	○	○	○	...
Seams (Bar, Pipe)	○	●	●	◑	○	◑	◑	○	...

Legend:
AE – Acoustic Emission
ET – Electromagnetic (Eddy Current)
MT – Magnetic Particle
PT – Liquid Penetrant
RT – Radiography
UTA – Ultrasonic Angle Beam
UTS – Ultrasonic Straight Beam
UTT – Ultrasonic Thickness Measurement
VT – Visual
● – All or most standard techniques will detect this imperfection under all or most conditions.
◑ – One or more standard technique(s) will detect this imperfection under certain conditions.
○ – Special techniques, conditions, and/or personnel qualifications are required to detect this imperfection.

GENERAL NOTE: Table A-110 lists imperfections and NDE methods that are capable of detecting them. It must be kept in mind that this table is very general in nature. Many factors influence the detectability of imperfections. This table assumes that only qualified personnel are performing nondestructive examinations and good conditions exist to permit examination (good access, surface conditions, cleanliness, etc.).

NOTES:
(1) Methods capable of detecting imperfections that are open to the surface only.
(2) Methods capable of detecting imperfections that are either open to the surface or slightly subsurface.
(3) Methods capable of detecting imperfections that may be located anywhere within the examined volume.
(4) Various NDE methods are capable of detecting tertiary (3rd stage) creep and some, particularly using special techniques, are capable of detecting secondary (2nd stage) creep. There are various descriptions/definitions for the stages of creep and a particular description/definition will not be applicable to all materials and product forms.

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Table A-110 Imperfection vs Type of NDE Method

Three groupings of imperfections

- Service-Induced
- Welding
- Product Form

vs

Three groupings of NDE methods

- Surface
- Sub-surface
- Volumetric

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Article 5 Ultrasonic Examination Methods for Materials

T-510 Scope
T-520 General
T-530 Equipment
T-560 Calibration
T-570 Examination
T-580 Evaluation
T-590 Documentation
Figure
Table
Mandatory Appendices

Mandatory Appendices

I. Ultrasonic Examination of Pumps and Valves
II. Inservice Examination of Nozzle Inside Corner Radius and Inner Corner Regions
III. Glossary of Terms for Ultrasonic Examination
IV. Inservice Examination of Bolts

- IV-510 Scope
- IV-530 Equipment
- IV-531 Calibration Blocks
  - IV-531.1 Material
  - IV-531.2 Reflectors
- IV-560 Calibration
- IV-570 Examination

Section V UT and RT as Applied Under Section III

- **Except for very specific joint configurations, UT is not permitted.**
- **Consideration of using UT in lieu of RT where use of RT is specified.**
- **Section III Code Case address use of UT in lieu of RT**

Article 14

- **Performance demonstration approach to qualify NDE systems**
- **Three levels of rigor – low, intermediate, and high - each represents an increased level of "reliability" in detection and/or sizing.**
- **Methodology may be applied to all NDE methods**

NDE of Non-Metallic Products

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- **High Density Polyethylene (HDPE) piping products.**
- **Section V ultrasonic requirements may not be adequate for HDPE.**
- **Generic NDE requirements and criteria are currently addressed by Section III Code Case N-755.**

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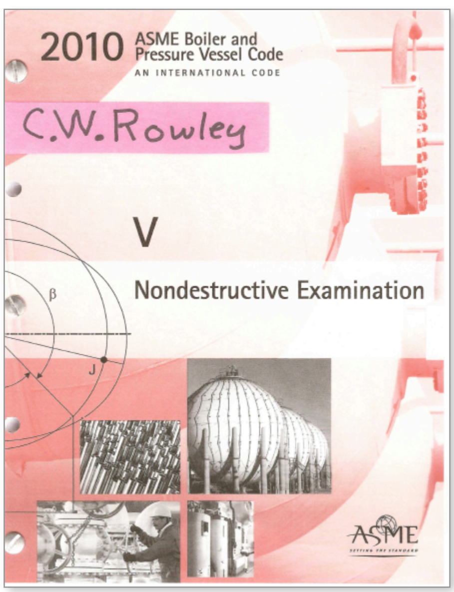

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CONTENTS (continued)

Subsection B Documents Adopted by Section V

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Applicable ASTM Standards

- Standard Guides
- Standard Practices
- Standard Specifications
- Standard Test Methods
- Standard Terminology

STANDARD TEST METHOD FOR RADIOGRAPHIC EXAMINATION OF METALLIC CASTINGS

SE-1030

(Identical with ASTM Specification E 1030-05)

1. Scope

1.1 This test method provides a uniform procedure for radiographic examination of metallic castings using radiographic film as the recording medium.

1.2 Due to the many complex geometries and part configurations inherent with cast products, it is necessary to recognize potential limitations associated with obtaining complete radiographic coverage on castings. Radiography of areas where geometry or part configuration does not allow achievement of complete coverage with practical radiographic methods shall be subject to mutual agreements between purchaser and supplier. The use of alternative nondestructive methods for areas that are not conducive to practical radiography shall also be specifically agreed upon between purchaser and supplier.

1.3 The radiographic method is highly sensitive to volumetric discontinuities that displace a detectable volume of cast material. Discontinuities that do not displace an appreciable volume of material, however, such as cracks or other planar-type indications, may not be detected with radiography unless the radiation beam is coincidentally aligned with the planar orientation of the discontinuity. In view of this limitation, it may be considered appropriate to use the radiographic method in conjunction with additional nondestructive methods that maintain reliable detection capabilities for these types of discontinuities. The use of additional methods shall be specifically agreed upon between the purchaser and supplier.

1.4 The radiographic techniques stated herein provide adequate assurance for defect detectability; however, it is recognized that, for special applications, specific techniques using more or less stringent requirements may be required than those specified. In these cases, the use of alternate radiographic techniques shall be as agreed upon between purchaser and supplier (also see Section 5).

1.5 The values stated in inch-pound units are to be regarded as standard.

1.6 *This standard does not purport to address all of the safety concerns, if any, associated with its use. It is the responsibility of the user of this standard to establish appropriate safety and health practices and determine the applicability of regulatory limitations prior to use.*

2. Referenced Documents

2.1 *ASTM Standards:*

E 94 Guide for Radiographic Examination

E 155 Reference Radiographs for Inspection of Aluminum and Magnesium Castings

E 186 Reference Radiographs for Heavy-Walled [2 to 4½-in. (51 to 114-mm)] Steel Castings

E 192 Reference Radiographs for Investment Steel Castings for Aerospace Applications

E 272 Reference Radiographs for High-Strength Copper-Base and Nickel-Copper Alloy Castings

E 280 Reference Radiographs for Heavy-Walled [4½ to 12-in. (114 to 305-mm)] Steel Castings

E 310 Reference Radiographs for Tin Bronze Castings

E 446 Reference Radiographs for Steel Castings Up to 2 in. (51 mm) in Thickness

E 505 Reference Radiographs for Inspection of Aluminum and Magnesium Die Castings

E 543 Practice for Agencies Performing Nondestructive Testing

E 689 Reference Radiographs for Ductile Iron Castings

SE-1030 CONTENTS

1. Scope
2. Referenced Documents
3. Terminology
4. Significance and Use
5. Basis of Application
6. Apparatus
7. Reagents and Materials
8. Requirements
9. Procedure
10. Radiograph Evaluation
11. Reference Radiographs
12. Report
13. Precision and Bias
14. Keywords

Appendixes

ARTICLE 23
ULTRASONIC STANDARDS

STANDARD PRACTICE FOR
ULTRASONIC EXAMINATION OF
HEAVY STEEL FORGINGS

SA-388/SA-388M

(Identical with ASTM Specification A 388/A 388M-07)

1. Scope

1.1 This practice covers the examination procedures for the contact, pulse-echo ultrasonic examination of heavy steel forgings by the straight- and angle-beam techniques. The straight-beam techniques include utilization of the DGS (Distance Gain Size) method. See Appendix X3.

1.2 This practice is to be used whenever the inquiry, contract, order, or specification states that forgings are to be subject to ultrasonic examination in accordance with Practice A 388/A 388M.

1.3 The values stated in either inch-pound or SI units are to be regarded as the standard. The values stated in each system are not exact equivalents; therefore, each system must be used independently of the other. Combining values from the two systems may result in nonconformance with the specification.

1.4 This specification and the applicable material specifications are expressed in both inch-pound units and SI units. However, unless the order specifies the applicable "M" specification designation (SI units), the material shall be furnished to inch-pound units.

1.5 *This standard does not purport to address all of the safety concerns, if any, associated with its use. It is the responsibility of the user of this standard to establish appropriate safety and health practices and determine the applicability of regulatory limitations prior to use.*

2. Referenced Documents

2.1 *ASTM Standards:*

A 469/A 469M Specification for Vacuum-Treated Steel Forgings for Generator Rotors

A 745/A 745M Practice for Ultrasonic Examination of Austenitic Steel Forgings

E 317 Practice for Evaluating Performance Characteristics of Ultrasonic Pulse-Echo Testing Instruments and Systems Without the Use of Electronic Measurement Instruments

E 428 Practice for Fabrication and Control of Metal, Other Than Aluminum Reference, Blocks Used in Ultrasonic Inspection

E 1065 Guide for Evaluating Characteristics of Ultrasonic Search Units

2.2 *ANSI Standard:*

B46.1 Surface Texture

2.3 *Other Document:*

Recommended Practice for Nondestructive Personnel Qualification and Certification SNT-TC-1A (1988 or later)

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SA-388 / SA-388M CONTENTS

1. Scope
2. Referenced Documents
3. Terminology
4. Ordering Information
5. Apparatus
6. Personnel Requirements
7. Preparation of Forging for Ultrasonic Examination
8. Procedure
9. Recording
10. Report
11. Quality Levels
12. Keywords

Supplementary Requirements
Appendixes

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ARTICLE 25, SE-709 2010 SECTION V

STANDARD GUIDE FOR MAGNETIC PARTICLE EXAMINATION

SE-709

(Identical with ASTM Specificaiton E 709-01)

1. Scope

1.1 This guide describes techniques for both dry and wet magnetic particle examination, a nondestructive method for detecting cracks and other discontinuities at or near the surface in ferromagnetic materials. Magnetic particle examination may be applied to raw material, semi-finished material (billets, blooms, castings, and forgings), finished material, and welds, regardless of heat treatment or lack thereof. It is useful for preventive maintenance examination.

1.1.1 This guide is intended as a reference to aid in the preparation of specifications/standards, procedures, and techniques.

1.2 This guide is also a reference that may be used as follows:

1.2.1 To establish a means by which magnetic particle examination, procedures recommended or required by individual organizations, can be reviewed to evaluate their applicability and completeness.

1.2.2 To aid in the organization of the facilities and personnel concerned in magnetic particle examination.

1.2.3 To aid in the preparation of procedures dealing with the examination of materials and parts. This guide describes magnetic particle examination techniques that are recommended for a great variety of sizes and shapes of ferromagnetic materials and widely varying examination requirements. Since there are many acceptable differences in both procedure and technique, the explicit requirements should be covered by a written procedure (see Section 21).

1.3 This guide does not indicate, suggest, or specify acceptance standards for parts/pieces examined by these techniques. It should be pointed out, however, that after indications have been produced, they must be interpreted or classified and then evaluated. For this purpose there should be a separate code, specification, or a specific agreement to define the type, size, location, degree of alignment and spacing, area concentration, and orientation of indications that are unacceptable in a specific part versus those which need not be removed before part acceptance. Conditions where rework or repair are not permitted should be specified.

1.4 This guide describes the use of the following magnetic particle method techniques.

1.4.1 Dry magnetic powder (see 8.4),

1.4.2 Wet magnetic particle (see 8.5),

1.4.3 Magnetic slurry/paint magnetic particle (see 8.5.8), and

1.4.4 Polymer magnetic particle (see 8.5.8).

1.5 *Personnel Qualification* — Personnel performing examinations in accordance with this guide shall be qualified and certified in accordance with ASNT Recommended Practice No. SNT-TC-1A, ANSI/ASNT Standard CP-189, NAS 410, or as specified in the contract or purchase order.

1.6 *Nondestructive Testing Agency* — If a nondestructive testing agency as described in Practice E 543 is used to perform the examination, the testing agency shall meet the requirements of Practice E 543.

1.7 *Table of Contents:*

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SE-709 CONTENTS

1. Scope
2. Referenced Documents
3. Terminology
4. Summary of Guide
5. Significance and Use
6. Equipment
7. Examination Area
8. Magnetic Particle Materials
9. Part Preparation
10. Sequence of Operations
11. Types of Magnetizing Currents
12. Part Magnetization Techniques
13. Direction of Magnetic Fields
14. Magnetic Field Strength
15. Application of Dry and Wet Magnetic Particles
16. Interpretation of Indications
17. Recording of Indications
18. Demagnetization
19. Post Examination Cleaning
20. Evaluation of System Performance / Sensitivity
21. Procedure
22. Acceptance Standards
23. Safety
24. Precision and Bias
25. Keywords

Annex and Appendices

2010 SECTION V ARTICLE 30, SE-1316

ARTICLE 30
TERMINOLOGY FOR NONDESTRUCTIVE EXAMINATIONS STANDARD

STANDARD TERMINOLOGY FOR NONDESTRUCTIVE EXAMINATIONS

SE-1316

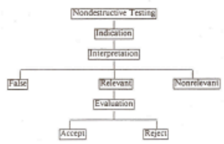
(Identical with ASTM E 1316-02a, except for editorial differences.)

1. Scope

1.1 This Standard defines the terminology used in the standards prepared by the E-7 Committee on Nondestructive Testing. These nondestructive testing (NDT) methods include: acoustic emission, electromagnetic testing, gamma- and X-radiology, leak testing, liquid penetrant examination, magnetic particle examination, neutron radiology and gaging, ultrasonic examination, and other technical methods.

1.2 Section 4 defines terms that are common to multiple NDT methods, whereas, the subsequent sections define terms pertaining to specific NDT methods. An alphabetical list of the terms defined in this Standard is given in Appendix XI, which also identifies the section in which each term is defined.

1.3 As shown on the chart below, when nondestructive testing produces an indication, the indication is subject to interpretation as false, nonrelevant, or relevant. If it has been interpreted as relevant, the necessary subsequent evaluation will result in the decision to accept or reject the material. With the exception of *accept* and *reject*, which retain the meaning found in most dictionaries, all the words used in the chart are defined in Section 4.



2. Referenced Documents

2.1 *ASTM Standards:*

E 94 Guide for Radiograhic Examination

E 127 Practice for Fabricating and Checking Aluminum Alloy Ultrasonic Standard Reference Blocks

E 215 Practice for Standardizing Equipment for Electromagnetic Examination of Seamless Aluminum-Alloy Tube

E 494 Practice for Measuring Ultrasonic Velocity in Materials

E 566 Practice for Electromagnetic (Eddy-Current) Sorting of Ferrous Metals

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1. Scope
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9. Liquid Penetrant Examination
10. Magnetic Particle Examination
11. Neutron Radiology
12. Ultrasonic Examination
13. Infrared Examination
14. Optical Holography
15. Visual and Optical Methods

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Article 30 Process Flowchart

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- **Nondestructive Testing**
- **Indication**
- **Interpretation**
- **Relevant / Nonrelevant / False**
- **Evaluation**
- **Accept / Reject**

Note that this is effectively the process flow chart on the previous slide.

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Use of Section V Editions / addenda and Code Cases

- **Which editions/addenda of Section V are permissible for use**
- **Are Section V Code cases permitted for use**
- **Does the use of Section V Code cases require prior NRC approval**

Alternatives to Section V Requirements

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- **When Section V is invoked, all provisions must be followed.**
- **Alternatives to Section V provisions may be proposed via the requirements of Section III or XI, and must receive NRC approval to use.**

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Case Study

ASME Code Case N-307-3 (Section XI Division 1)

CASES OF ASME BOILER AND PRESSURE VESSEL CODE

CASE N-307-3

Approval Date: March 28, 2001

The ASME Boiler and Pressure Vessel Standards Committee took action to eliminate Code Case expiration dates effective March 11, 2005. This means that all Code Cases listed in this Supplement and beyond will remain available for use until annulled by the ASME Boiler and Pressure Vessel Standards Committee.

Case N-307-3
Ultrasonic Examination of Class 1 Bolting, Table IWB-2500-1, Examination Category B-G-1
Section XI, Division 1

Inquiry: When ultrasonic examinations are conducted from the end of the bolt or stud or from the center-drilled hole of bolts or studs to satisfy the examination requirements of Table IWB-2500-1, Examination Category B-G-1, may the examination volume be limited to the cylindrical region defined by A-B-C-D-E-F-A in Fig. 1, and may the surface examination requirement of Table IWB-2500-1, Examination Category B-G-1, Item No. B6.30, Reactor Closure Studs when removed be eliminated?

Reply: It is the opinion of the Committee that, when conducting ultrasonic examinations from the end of the bolt or stud or from the center-drilled hole of bolts or studs to satisfy the examination requirements of Table IWB-2500-1, Examination Category B-G-1, the examination volume may be limited to the cylindrical region defined by A-B-C-D-E-F-A in Fig. 1. The surface examination requirement of Table IWB-2500-1, Examination Category B-G-1, Item No. B6.30, Reactor Vessel Closure Studs when removed, may be eliminated.

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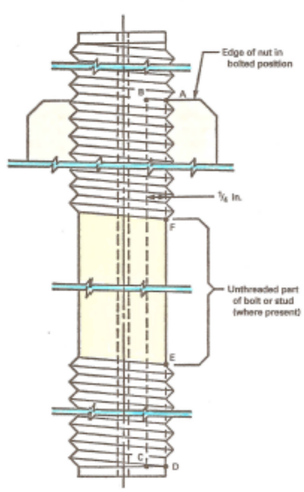
CASE (continued)
N-307-3

CASES OF ASME BOILER AND PRESSURE VESSEL CODE

FIG. 1 REVISED EXAMINATION VOLUME



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References Within the Case BPV Code Section XI

Table IWB-2500-1
Examination Categories

Examination Category B-G-1
Pressure Retaining Bolting, Greater Than 2 in. (50 mm) in Diameter

Reactor Vessel

Item No. B6.30
Reactor Vessel Closure Studs

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Table IWB-2500-1

1998 Edition

- **Reactor Vessel**
  - B6.10 Closure Head Nuts
  - B6.20 Closure Studs, in place
  - B6.30 Closure Studs, when removed
  - B6.40 Threads in Flange
  - B6.50 Closure washers, bushings

2007 Edition

- **Reactor Vessel**
  - B6.10 Closure Head Nuts
  - B6.20 Closure Studs
  - B6.40 Threads in Flange
  - B6.50 Closure washers, bushings

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Learning Questions

- **Which NDE methods are acceptable within ASME BPV Code Section V Subsection A and used in NPPs?**
- **Which ASTM standards are adopted by ASME BPV Code Section V Subsection B and used in NPPs?**
- **Which Section V code cases are applicable to NPPs?**

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Module 33
Pressure Vessel Construction

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Module 33
Pressure Vessel Construction

Instructor: C. Wesley Rowley, P.E.

Learning Objectives

- **Learn what typical components in a NPP are designed and constructed using the requirements of ASME BPV Code Section VIII Div 1.**
- **Learn what typical components in a NPP are designed and constructed using the requirements of ASME BPV Code Section VIII Div 2.**
- **Learn the typical application issues of ASME BPV Code Section VIII for NPPs.**

Significant Sub-topics

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- Major References
- General Requirements
- Responsibilities and Duties
- Materials Requirements
- Design By Rule Requirements
- Design By Analysis Requirements
- Fabrication Requirements
- Inspection and Examination Requirements
- Pressure Testing Requirements
- Pressure Vessel Overpressure Protection
- Mandatory and Non-mandatory Appendices
- Technical / Code Inquiries
- Code Cases

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Major References

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- **ASME BPV Code**
  - Section I Rules for Construction of Power Boilers
  - Section III Rules for Construction of Nuclear Facility Components
  - Section VIII Rules for Construction of Pressure Vessels
    - Division 1 Rules for Pressure Vessels
    - Division 2 Alternative Rules for Pressure Vessels
    - Division 3 Alternative Rules for High Pressure Vessels
  - Section X Rules for Construction of FRP Vessels

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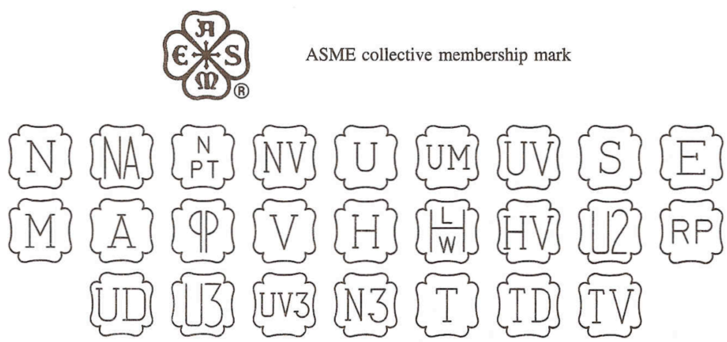
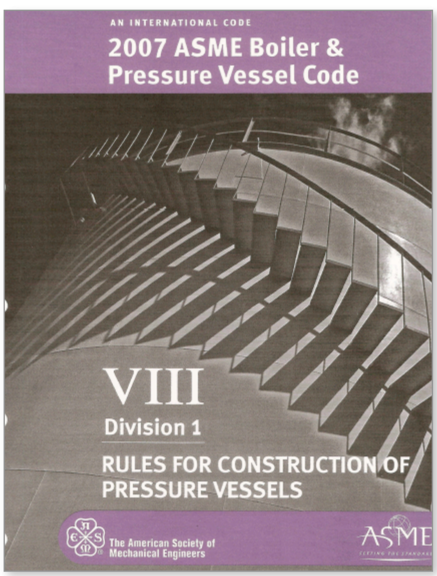
ASME Stamps / Marks





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- Subsection B Requirements Pertaining to Methods of Fabrication of Pressure Vessels
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- Part UF Fabricated by Forging
- Part UB Fabricated by Brazing
- Subsection C Requirements Pertaining to Classes of Materials
- Part UCS Carbon and Low Alloy Steels
- Part UHA High Alloy Steel
- Part UCI Cast Iron
- Part UCL Claddings and Linings
- Part UCD Cast Ductile Iron
- Part UHT Ferritic Steels with Heat Treatment
- Part ULW Layered Construction
- Part ULT Low Temperature Application
- Part UHX Shell-and-Tube Heat Exchangers
- Part UIG Impregnated Graphite
- Appendices

General Requirements
BPV Code Section VIII Div 1

- **Subsection A General Requirements**
  - Part UG General Requirements for All Methods of Construction and All Materials
  - Scope
    - UG-1 Scope
    - The requirements of Part UG are applicable to all pressure vessels and vessel parts and shall be used in conjunction with the specific requirements in Subsections B and C and the Mandatory Appendices that pertain to the method of fabrication and the material used.
  - **Materials**
    - UG-4 General
    - UG-5 Plate
    - UG-6 Forgings
    - UG-7 Castings
    - UG-8 Pipe and Tubes
    - UG-9 Welding Materials

General Requirements (cont'd)

- **Materials** (cont'd)
  - UG-10 Material Identified With or Produced to a Specification Not Permitted by This Division, and Material Not Fully Identified
  - UG-11 Prefabricated or Preformed Pressure Parts
  - UG-12 Bolts and Studs
  - UG-13 Nuts and Washers
  - UG-14 Rods and Bars
  - UG-15 Product Specification
- **Design**
  - UG-16 General
  - UG-17 Methods of Fabrication in Construction
  - UG-18 Materials in Combination
  - UG-19 Special Construction
  - UG-20 Design Temperature
  - UG-21 Design Pressure
  - UG-22 Loadings

General Requirements (cont'd)

- **Design** (cont'd)
  - UG-23 Maximum Allowable Stress Loadings
  - UG-24 Castings
  - UG-25 Corrosion
  - UG-26 Linings
  - UG-27 Thickness of Shells Under Pressure
  - UG-28 Thickness of Shells and Tubes
  - UG-29 Stiffening Rings for Cylindrical Shells Under External Pressure
  - UG-30 Attachment of Stiffening Rings
  - UG-31 Tubes, and Pipe When Used As Tubes or Shells
  - UG-32 Formed Heads, and Sections, Pressure on Concave Side
  - UG-33 Formed Heads, Pressure on Convex Side
  - UG-34 Unstayed Flat Heads and Covers
  - UG-35 Other Types of Closures

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General Requirements (cont'd)

- Openings and Reinforcements (UG-36 thru UG-46)
- Braced and Stayed Surfaces (UG-47 thru UG-50)
- Ligaments (UG-53 thru UG-55)
- Fabrication (UG-75 thru UG-103)
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  - UG-115 General
  - UG-116 Required Marking
  - UG-117 Certificates of Authorization and Code Symbol Stamps
  - UG-118 Methods of Marking
  - UG-119 Nameplates
  - UG-120 Data Reports

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General Requirements (cont'd)

- Overpressure Protection (UG-125 thru UG-140)
  - UG-125 General
  - UG-126 Pressure Relief Valves
  - UG-127 Nonreclosing Pressure Relief Devices
  - UG-128 Liquid Pressure Relief Valves
  - UG-129 Marking
  - UG-130 Code Symbol Stamp
  - UG-131 Certification of Capacity of Pressure Relief Valves
  - UG-132 Certification of Capacity of Pressure Relief Valves in Combination with Nonreclosing Pressure Relief Devices
  - UG-133 Determination of Pressure Relieving Devices
  - UG-134 Pressure Settings and Performance Requirements
  - UG-135 Installation
  - UG-136 Minimum Requirements for Pressure Relief Valves
  - UG-137 Minimum Requirements for Rupture Disk Devices
  - UG-138 Minimum Requirements for Pin Devices
  - UG-140 Overpressure Protection by System Design

Subsection B
BPV Code Section VIII Div 1

- **Requirements Pertaining to Methods of Fabrication of Pressure Vessels**
  - Part UW Requirements for Pressure Vessels Fabricated by Welding
    - General
    - Materials
    - Design
    - Fabrication
    - Marking and Reports
    - Pressure Relief Valves

Welded Vessels

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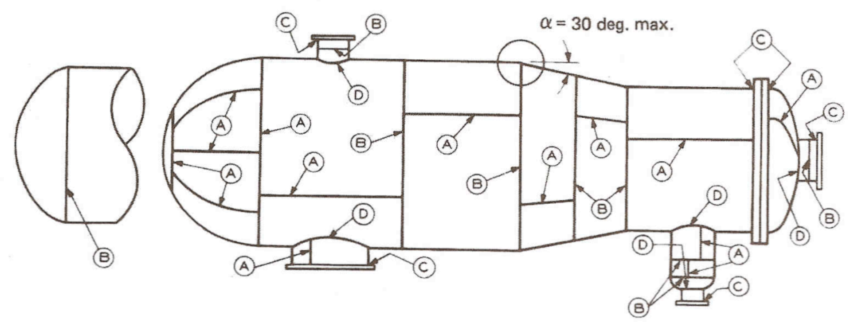
FIG. KE-321 ILLUSTRATION OF WELDED JOINT LOCATIONS TYPICAL OF CATEGORIES A, B, C, AND D



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Subsection B (cont'd)
BPV Code Section VIII Div 1

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- **Requirements Pertaining to Methods of Fabrication of Pressure Vessels (cont'd)**
  - Part UF Requirements for Pressure Vessels Fabricated by Forging
  - Part UB Requirements for Pressure Vessels Fabricated by Brazing

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Subsection C
BPV Code Section VIII Div 1

- **Requirements Pertaining to Classes of Materials**
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    - General
    - Materials
    - Design
    - Low Temperature Operation
    - Fabrication
    - Inspection and Tests
    - Marking and Reports
    - Pressure Relief Devices
  - Part UNF Requirements for Pressure Vessels Constructed of Nonferrous Materials

Subsection C (cont'd)
BPV Code Section VIII Div 1

- **Requirements Pertaining to Classes of Materials (cont)**
  - Part UHA Requirements for Pressure Vessels Constructed of High Alloy Steel
  - Part UCI Requirements for Pressure Vessels Constructed of Cast Iron
  - Part UCL Requirements for Pressure Vessels Constructed of Material with Corrosion Resistant Integral Cladding, Weld Metal Overlay Cladding, or with Applied Linings
  - Part UCD Requirements for Pressure Vessels Constructed of Cast Ductile Iron
  - Part UHT Requirements for Pressure Vessels Constructed of Ferritic Steels with Tensile Properties Enhanced by Heat Treatment
  - Part ULW Requirements for Pressure Vessels Fabricated by Layered Construction

Subsection C (cont'd)
BPV Code Section VIII Div 1

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  - Part UIG Requirements for Pressure Vessels Constructed of Impregnated Graphite
- **Mandatory Appendices**
- **Nonmandatory Appendices**

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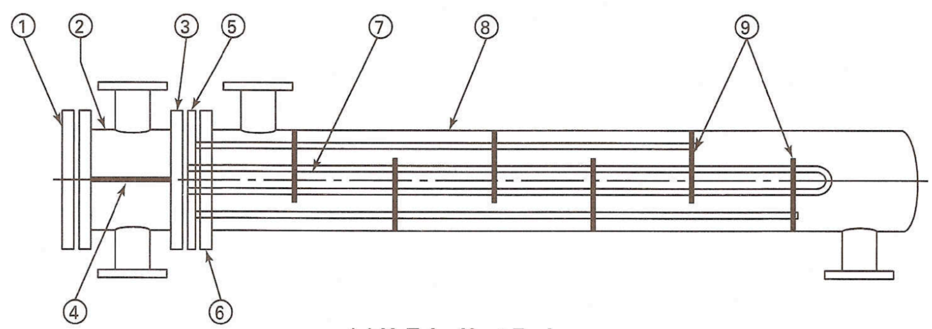
FIG. UHX-3 TERMINOLOGY OF HEAT EXCHANGER COMPONENTS

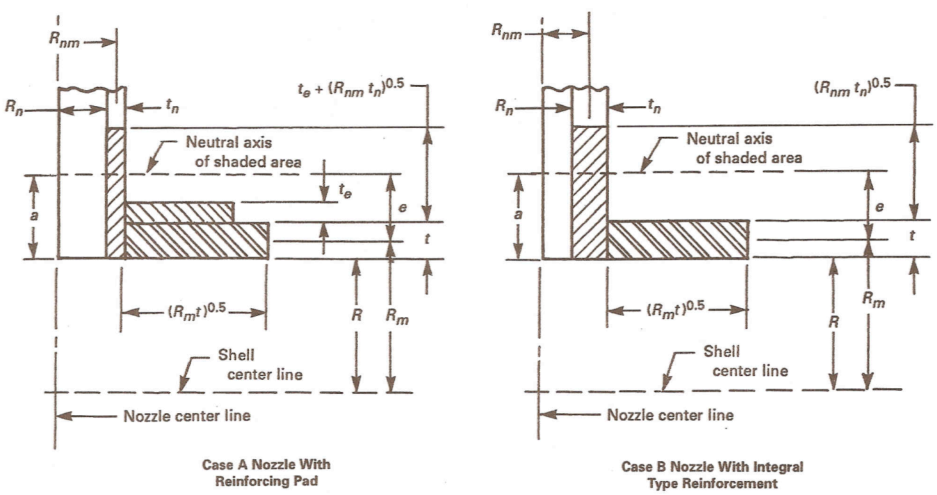


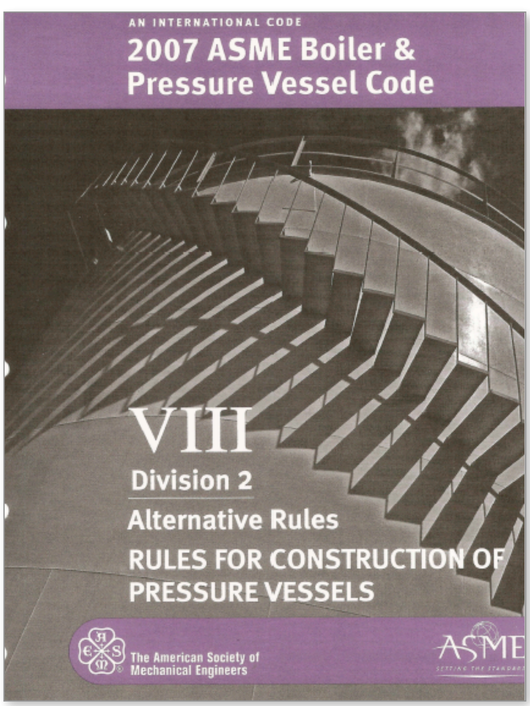
(a) U–Tube Heat Exchanger

1 Channel cover (bolted flat cover)
2 Channel
3 Channel flange
4 Pass partition
5 Stationary tubesheet
6 Shell flange
7 Tubes
8 Shell
9 Baffles or support plates
10 Floating head backing device
11 Floating tubesheet
12 Floating head
13 Floating head flange
14 Shell cover
15 Expansion bellows
16 Distribution or vapor belt

Figure from Div 1 Mandatory Appendix 1 (Large Openings in Shells)

FIG. 1-7-1





ASME BPV Code Section VIII Div 2 CONTENTS

Part 1 General Requirements
Part 2 Responsibilities and Duties
Part 3 Material Requirements
Part 4 Design By Rule Requirements
Part 5 Design By Analysis Requirements
Part 6 Fabrication Requirements
Part 7 Inspection and Examination Requirements
Part 8 Pressure Testing Requirements
Part 9 Pressure Vessel Overpressure Protection

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(Section VIII Division 2)

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  - Annex 2.C Report Forms and Maintenance of Records
  - Annex 2.D Guide for Preparing Manufacturer's Data Reports
  - Annex 2.E Quality Control System
  - Annex 2.F Contents and Methods of Stamping
  - Annex 2.G Obtaining and Using Code Stamps
  - Annex 2.H Guide to Information Appearing on the Certificate of Authorization

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Part 3 (cont'd)
(Section VIII Division 2)

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  - 3.11 Material Toughness Requirements
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  - 3.14 Physical Properties
  - 3.15 Design Fatigue Curves
  - 3.16 Nomenclature
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  - 3.19 Figures

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  - 4.11 Design Rules for Jacketed Vessels

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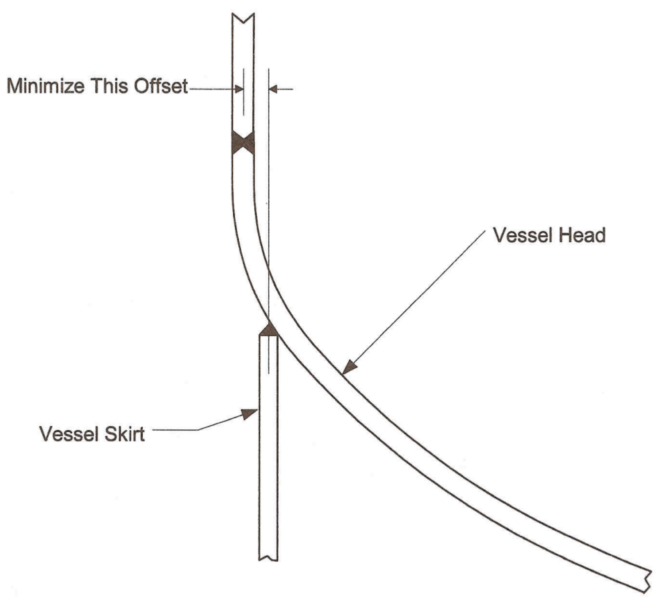

Figure 4.15.7 – Skirt Attachment Location on Vertical Vessels

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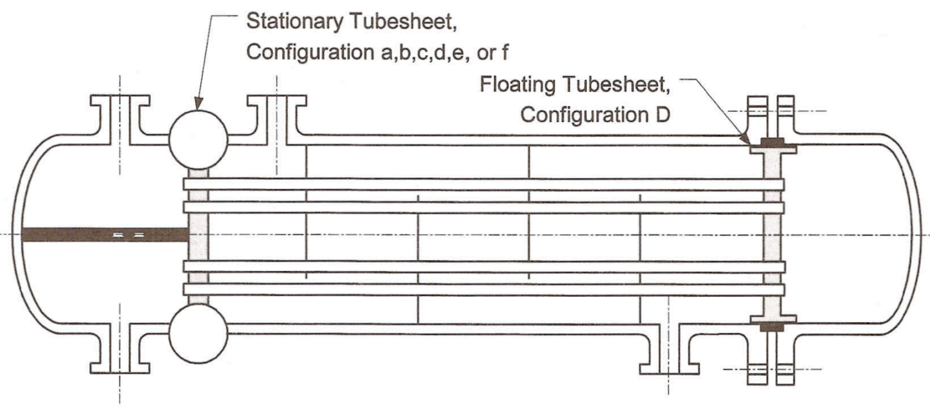

(c) Typical Floating Tubesheet Exchanger With An Internally Sealed Floating Tubesheet

Figure 4.18.10 – Floating Tubesheet Heat Exchangers

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(Section VIII Division 2)

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Part 5 (cont'd) (Section VIII Division 2)

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Part 6 (Section VIII Division 2)

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- **Fabrication Requirements**
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  - 6.4 Preheating and Heat Treatment of Weldments
  - 6.5 Special Requirements for Clad or Weld Overlay Linings and Lined Parts
  - 6.6 Special Requirements for Tensile Property Enhanced Q and T Ferritic Steels
  - 6.7 Special Requirements for Forged Fabrication
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  - 6.9 Nomenclature
  - 6.10 Tables
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  - 7.3 Verification and Examination Prior to Welding
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(Section VIII Division 2)

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  - 8.2 Hydrostatic Testing
  - 8.3 Pneumatic Testing
  - 8.4 Alternative Pressure Testing
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(Section VIII Division 2)

- **Pressure Vessel Overpressure Protection**
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  - 9.2 Pressure Relief Valves
  - 9.3 Nonreclosing Pressure Relief Devices
  - 9.4 Calculation of Rated Capacity for Different Relieving Pressures and / or Fluids
  - 9.5 Marking and Stamping
  - 9.6 Provisions for Installation of Pressure Relieving Devices
  - 9.7 Overpressure Protection by Design
  - Annex 9.A Best Practices for the Installation and Operation of Pressure Relief Devices

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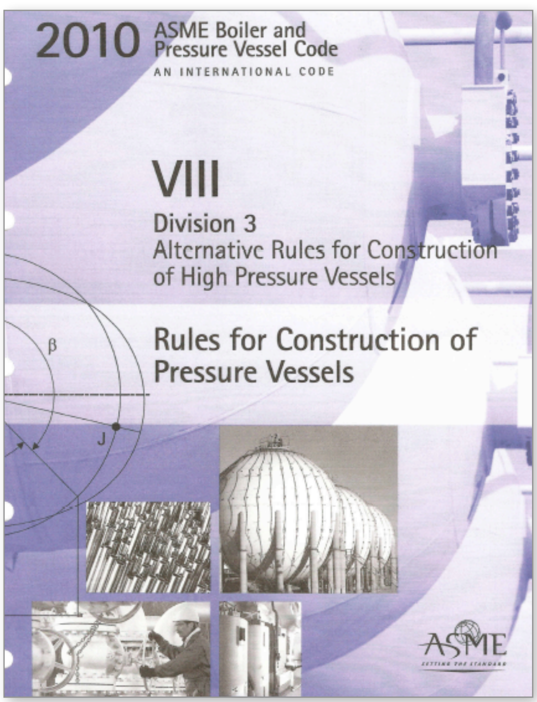

ASME BPV Code Section VIII Div 3 CONTENTS

Part KG General Requirements
Part KM Material Requirements
Part KD Design Requirements
Part KF Fabrication Requirements
Part KR Pressure Relief Devices
Part KE Examination Requirements
Part KT Testing Requirements
Part KS Marking, Stamping, Reports, and Records
Mandatory Appendices
Nonmandatory Appendices

Technical Inquiries

- **Technical Inquiries published in 2007 Edition**
  - Division 1 = 34
  - Division 2 = 1
  - Division 3 = 1
- **Technical Inquiries published in 2010 Edition**
  - Division 1 = 20
  - Division 2 = 4
  - Division 3 = 0
- **Numerical Index**
- **Subject Index**

Code Cases

- **ASME BPV Code Section VIII (2007 Edition)**
  - Thru Supplement 11 for Div 1 = 133
  - Thru Supplement 11 for Div 2 = 38
  - Thru Supplement 11 for Div 3 = 27
- **ASME BPV Code Section VIII (2010 Edition)**
  - Thru Supplement 1 for Div 1 = 136
  - Thru Supplement 1 for Div 2 = 39
  - Thru Supplement 1 for Div 3 = 26
- **Supplements are published every six months**

Case Study

- **Study on Code Interpretation VIII-1-10-04 related to Nonmandatory Appendix G, G-2**
  - Question: Nonmandatory Appendix G makes reference to several documents that may be used as guidance in the design of vessel supports and attachments. When a Manufacturer uses one of these referenced documents, such as the Manual for Steel Construction, do all the requirements of that document then become a mandatory Code requirement?
  - Answer: ??

Learning Questions

- **What typical components in a NPP are designed and constructed using the requirements of ASME BPV Code Section VIII Div 1?**

- **What typical components in a NPP are designed and constructed using the requirements of ASME BPV Code Section VIII Div 2?**

- **What are the typical application issues of ASME BPV Code Section VIII for NPPs?**

Module 34
PWR & BWR System Classification

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Module 34
PWR & BWR System Classification

Instructor: Gene Imbro, P.E.

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Learning Objectives

- **Learn about the reason systems are classified**
- **Learn the basis for system classification for BWRs and PWRs**
- **Learn how the classification scheme is applied to the design of mechanical systems, structures and components (SSCs)**
- **Learn how the classification selected relates to other Codes and Standards**
- **Learn the interface requirements for change of Safety Class within a system.**

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Significant Sub Topics

- Overview of the ANS Classification Standards
- Definitions that relate to system classification.
- Similarity and differences between classification criteria for BWRs and PWRs.
- Interface requirements between safety classes
- Applicability of related mechanical and electrical design standards based on safety classification.

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Major References

- **Title 10, Code of Federal Regulations, Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants."**
- **ANSI/ANS-51.1-1983, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants**
- **ANSI/ANS-52.1-1983, Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants**
- **NUREG-0800, Standard Review Plan**
  - Chapter 3.2.1, Seismic Classification
  - Chapter 3.2.2, System Quality Group Classification

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Major References

- Regulatory Guide 1.26, Rev 4, dated March 2007 Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants
- American National Standard Criteria for Protection Systems for Nuclear Power Generating Stations, ANSI/IEEE Std 279-1971.
- American National Standard Application of Single-Failure Criterion to Nuclear Power Generating Station Class 1E Systems. ANSI/IEEE-379-1977
- Regulatory Guide 1.53, Application of the Single-failure Criterion to Safety Systems, Rev 2, dated Nov. 2003
- SECY-77-439, "Single-Failure Criterion," August 1977 (ADAMS Accession No. (ML060260236).

Overview of Classification Standards

- **ANS 51.1-1983 and 52.1-1983 have been withdrawn**
- **Additional legacy standards, included**
  - ANSI/ANS - N18.2 for PWR classification
  - ANSI/ANS - N212 for BWR classification

Purpose of the Standards

- **The content of both 51.1 and 52.1 reflects an attempt to achieve the following objectives:**
  - To establish a consistent set of requirements for LWR power plants;
  - To establish a disciplined, systematic method for defining nuclear safety;
  - To establish and delineate the functional nuclear safety requirements for the design of nuclear power plants;
  - To provide a uniform basis for design safety requirements which may be reflected in regulatory documents.

Definitions

- **Active component**
  - A component in which mechanical movement must occur to accomplish the nuclear safety function of the component.
- **Active failure.**
  - A malfunction, excluding passive failures, of a component that relies on mechanical movement to complete its intended nuclear safety function upon demand.

Definitions (cont'd)

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- **Passive failure.**
  - The blockage of a process flow path or failure of a component to maintain its structural integrity or stability, such that it cannot provide its intended nuclear safety function upon demand.
- **Initiating occurrence.**
  - A single occurrence and its consequential effects that place the plant or portion of the plant in an off-normal condition. An initiating occurrence is not the single failure defined elsewhere herein. An initiating occurrence can be an equipment failure, a human error, a natural hazard, or a man-made hazard

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Definitions (cont'd)

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- **Single failure**
  - A random failure and its consequences in addition to an initiating occurrence, that result in the loss of capability of a component to perform its intended nuclear safety function.
- **Short term.**
  - In the context of the single failure criterion, that period of operation up to 24 hours following the initiating event, but for purposes of design of the emergency core cooling and containment spray systems, the short term shall be considered to terminate upon transfer of these systems to the long-term cooling mode.

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Application of the Single Failure Criterion

- **The single failure need only be assumed in the nuclear safety-related components needed to respond in the initiating occurrence.**
- **A single failure must also be assumed in the electrical nuclear safety-related equipment that is required to support the safety functions defined in 10 CFR 50.2**

Classification of SSCs

- **Systems are classified into 4 Safety Classes based on their importance to safety.**
  - Safety Class 1
    - Reactor coolant pressure boundary
  - Safety Class 2
    - Primary reactor containment and penetrations
    - Containment heat removal
    - Emergency core cooling
    - Refueling water storage tank

Classification of SSCs

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- **Safety Class 3**
  - Component cooling water
  - Safety-related service water
  - Auxiliary feedwater except for Class 2 containment penetration
  - Safety-related chillers and room coolers
  - Spent fuel cooling
  - Normal makeup system (CVCS)
  - Diesel fuel oil and lube oil cooling and jacket water cooling

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Classification of SSCs

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- **Non-nuclear safety (NNS)**
  - SSCs that are not relied on to perform a safety function
    - Main turbine, main condenser and generator
    - Potable water system
    - Main steam system (downstream of MSIVs)
    - Condensate and Feedwater system (upstream of main feed isolation valves)
    - Switchyard
    - Compressed air (typically)

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Reg Guide 1.26

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- **Reg Guide 1.26, Quality Group Classifications And Standards For Water-, Steam-, And Radioactive-waste-containing Components Of Nuclear Power Plants, Rev 4, dated March 2007**
- **Provides guidance for system classification**
  - Quality Group A ↔ Safety Class 1
  - Quality Group B ↔ Safety Class 2
  - Quality Group C ↔ Safety Class 3
  - Quality Group D ↔ Nonnuclear Safety (NNS)

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Reg Guide 1.26

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Table 1

Components	QUALITY STANDARDS		
	Quality Group B	**Quality Group C**	**Quality Group D**
Pressure Vessels	ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components,"[a,b] Class 2	ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components,"[a,b] Class 3	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, "Rules for Construction of Pressure Vessels" (Ref. 7)
Piping	As above	As above	ASME B31.1 (Ref. 8)
Pumps	As above	As above	Manufacturers' standards
Valves	As above	As above	ASME B31.1 (Ref. 8)
Atmospheric Storage Tanks	As above	As above	API-650 (Ref. 9), AWWA D-100 (Ref. 10), or ASME B96.1 (Ref. 11)
0–15 psig Storage Tanks	As above	As above	API-620 (Ref. 12)

[a] See 10 CFR 50.55a for guidance regarding the ASME Code and addenda to be applied.

[b] Other regulatory guides or Commission regulations cover the specific applicability of code cases, where appropriate. Applicants proposing the use of code cases not covered by guides or regulations should demonstrate that an acceptable level of quality and safety would be achieved.

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Correspondence of Industry Codes and Standards to Safety Classes

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- **Safety-related pressure boundary design**
  - Section III of the ASME Boiler and Pressure Vessel Code is required for the design of all safety-related pressure boundary components by 10 CFR 50.55a
  - ASME Section III also has design classes that apply to SC-1, 2 and 3 or the Reactor Coolant Pressure Boundary and Quality Groups B and C.
    - Class 1 components are required to be designed to Subsection NB rules
    - Class 2 components are required to be designed to Subsection NC rules and
    - Class 3 components are required to be design to Subsection ND rules

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Correspondence of Industry Codes and Standards to Safety Classes

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- **All electrical equipment performing a safety function is required to meet IEEE-603**
- **Other NRC requirements applicable to Class 1E safety-related electrical equipment include:**
  - 10CFR50.49, Environmental qualification and
  - 10 CFR 50, Appendix A, Criterion 2, Natural Phenomena…Seismic
  - 10 CFR 50, Appendix A, Criterion 4, Environmental Effects

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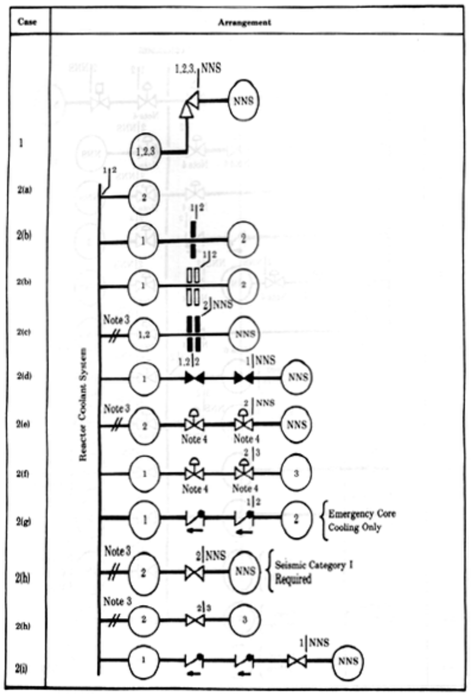
Correspondence of Industry Codes and Standards to Safety Classes

- **Additional NRC guidance for Class 1E systems and components is provided in the following Regulatory Guides:**
  - 1.53, Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems
  - 1.73, Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants
  - 1.89, Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants
  - 1.100, Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants

Standards for Class 1E Equipment

Standards for Safety Class 3 Electrical Equipment (Class 1E)

Standard	Modules	Sensors	Systems	Cables	Connectors	Switchgear	Transformers	Diesel-Generators	Batteries	Motors	Valve Actuators[33]	Penetrations[34]	Control Boards	Components	Motor Control Centers
IEEE-279[35]			X												
IEEE-308			X												
IEEE-334										X					
IEEE-336			X												
IEEE-338			X												
IEEE-344[36]	X	X		X	X	X	X	X	X	X	X	X	X	X	X
IEEE-352			X												
IEEE-379[37]			X												
IEEE-383				X											
IEEE-450									X						
IEEE-577			X												



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System Interface Requirements

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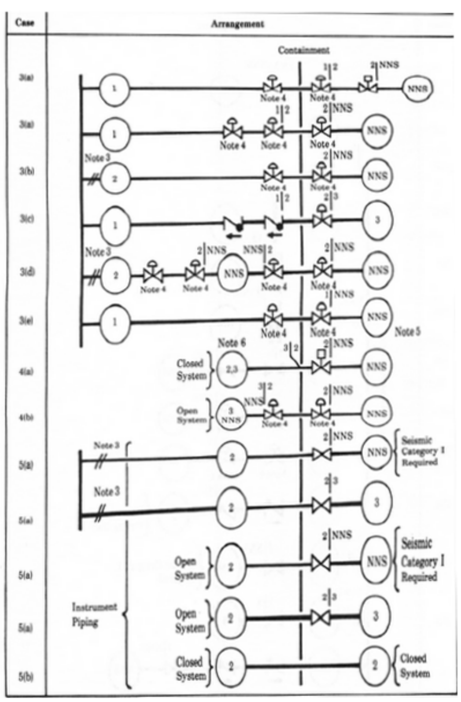

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System Interface Requirements

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NUREG-0800

- **Chapter 3.2.1, "Seismic Classification"**
  - Regulatory Guide 1.29, "Seismic Design Classification."
  - Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear PowerPlants."
  - 10 CFR 50, Appendix A, Criterion 61
- **Chapter 3.2.2, "System Quality Group Classification"**
  - Regulatory Guide 1.26
  - 10 CFR 50.55a Codes and Standards

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System or Component	Quality Group	References
1. Combustible Gas Control System	**B (1)**	**RG 1.7**
2. Compressed Air Systems required to perform a safety function	**C**	**SRP 9.3.1**
3. Containment Isolation System:	**A/B (2)**	**SRP 6.2.4**
a. Penetrations including associated piping and isolation valves	A/B (2)	**RG 1.141**
b. Instrument lines penetrating containment	B (3)	**RG 1.11**
c. Isolation barriers comprised of closed systems inside containment	B (2)	**SRP 6.2.4**
d. Isolation barriers comprised of closed systems outside containment	B(2)	**SRP 6.2.4**
e. Closed systems in secondary containment proposed as boundaries to produce bypass leakage	B (4)	**Branch Technical Position 6-3**
4. Emergency Diesel Engine		**RG 1.137**
a. Fuel oil storage and Transfer System	C (5)	
b. Cooling Water System	C	
c. Starting System	C	
d. Lubrication System	C	
e. Combustion Air Intake and Exhaust System	C	
5. Equipment and Floor Drainage System	C (6)	**SRP 9.33**
6. Gas Treatment Systems which are considered as engineered safeguards systems	**B**	

Reg Guide 1.29

- **Reg Guide 1.29 "Seismic Design Classification" Rev 4, dated March 2007**
  - Reactor coolant pressure boundary
  - Reactor core and reactor vessel internals
  - Systems or portions thereof that are required for (1) emergency core cooling,(2)post-accident containment heat removal, or (3) post-accident containment atmosphere cleanup (e.g., hydrogen removal system)
  - Systems or portions thereof that are required for (1) reactor shutdown, (2) residual heat removal, or (3) cooling the spent fuel storage pool

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Reg Guide 1.29

- **Those portions of SSCs of which continued function is not required but of which failure could reduce the functioning of any plant feature included in items 1.a through 1.q above to an unacceptable safety level or could result in incapacitating injury to occupants of the control room should be designed and constructed so that the SSE would not cause such failure.**
- **At the interface between Seismic Category I and non-Seismic Category I SSCs, the Seismic Category I dynamic analysis requirements should be extended to either the first anchor point in the non-seismic system or a sufficient distance into the non-Seismic Category I system so that the Seismic Category I analysis remains valid.**

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Module 35
Containment Isolation

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Module 35
Containment Isolation

Instructor: Gene Imbro, P.E.

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Learning Objectives

- Learn how the General Design Criteria relate to the design of containment isolation provisions
- Learn about the different schemes for containment isolation of fluid systes
- Learn about the ANS-56.2/ANSI N271-1976 Standard
- Learn about any limitations imposed by Reg Guide 1.141, Rev 1, July 2010

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Significant Sub Topics

- **Review of Applicable General Design Criteria**
- **Review ANS-56.2/ANSI-N271-1976 Standard, Containment Isolation Provisions For Fluid Systems**
- **Review Regulatory Guide 1.141, Rev 1, July 2010, Containment Isolation Provisions For Fluid Systems**

Major References

- **10 CFR 50, Appendix A, General Design Criteria**
- **ANS-56.2/ANSI-N271-1976 Standard, Containment Isolation Provisions For Fluid Systems**
- **Regulatory Guide 1.141, Rev 1, July 2010**

10 CFR 50, Appendix A

- **10 CFR 50, Appendix A**
  - Section V. Reactor Containment
    - Criterion 54, Systems Penetrating Containment
    - Criterion 55, Reactor Coolant Pressure Boundary Penetrating Containment
    - Criterion 56, Primary Containment Isolation
    - Criterion 57, Closed Systems Isolation Valves

10 CFR 50, Appendix A

- **Criterion 54, Systems Penetrating Containment**
  - Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

10 CFR 50, Appendix A

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- **Criterion 55–Reactor coolant pressure boundary penetrating containment**
  - Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

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10 CFR 50, Appendix A

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- **Criterion 55–Reactor coolant pressure boundary penetrating containment (cont'd)**
  1. One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
  2. One automatic isolation valve inside and one locked closed isolation valve outside containment; or
  3. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or…

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10 CFR 50, Appendix A

- **Criterion 55–Reactor coolant pressure boundary penetrating containment (cont'd)**
  4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

- **Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.**

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10 CFR 50, Appendix A

- **Criterion 56–Primary containment isolation**
  - Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

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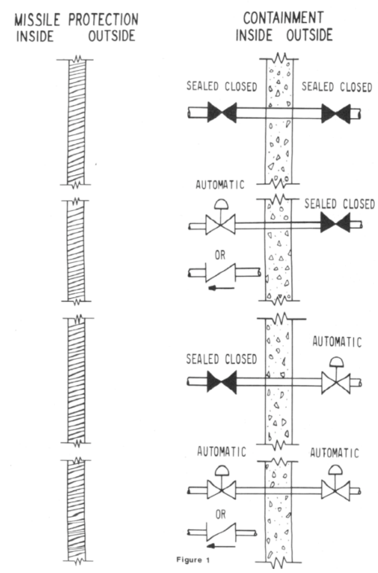
GDC 55 & 56 Configurations



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10 CFR 50, Appendix A

- **Criterion 57–Closed system isolation valves.**
  - Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

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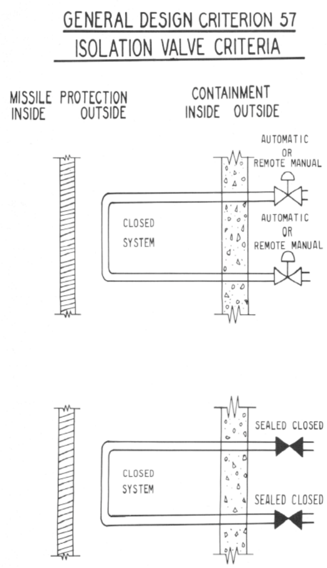
GDC 57 Configurations



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ANSI N-271-1976

- **Scope**
  - This standard specifies minimum design, actuation, testing, and maintenance requirements for the containment isolation of fluid systems after a LOCA. These fluid systems penetrate the primary containment of light water reactors and include piping systems (including instrumentation and control) for all fluids entering or leaving the containment.

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ANSI N-271-1976

- **Purpose**
  - Containment isolation is the closure of isolation provisions in lines penetrating the containment in the event of the loss-of-coolant accident within the containment or any other accident which calls for actuation of the same containment isolation provisions.

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ANSI N-271-1976

- **3.5 Criteria for closed systems inside containment.**
  - Closed system inside containment shall:
    1. Not communicate with either the primary coolant or the containment atmosphere,
    2. Be missile, pipe whip, and jet force protected,
    3. Meet Safety Class 2 design requirements,
    4. Withstand temperature equal to containment design temperature, (cont'd)

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ANSI N-271-1976

- Closed system inside containment shall:
  5. Withstand external pressure equal to containment structural integrity test pressure,
  6. Withstand loss-of-coolant accident transients and environment,
  7. Meet Seismic Category I design requirements,
  8. Be protected against overpressure from thermal expansion of contained fluid when isolated, and
  9. Be capable of being leak tested.

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ANSI N-271-1976

- **3.7 Criteria for Piping Outside Containment and Between the Containment Isolation Valves**
  1. Meet Safety Class 2 design requirements
  2. Withstand the containment design temperature
  3. Withstand internal pressure from containment structural integrity test
  4. Withstand loss·of·coolant accident transient and environment
  5. Meet Seismic Category I design requirements
  6. Be protected against a high energy line break outside of containment when needed for containment isolation.

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- **Section 3.6, "Other Defined Basis"**
  - Instrument lines with closed systems both inside and outside of containment, such as containment pressure instrumentation, which are designed to withstand the maximum containment structural integrity test pressure, the containment design temperature, and are protected from missiles and dynamic effects are acceptable without isolation valves.

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- **Section 3.6, "Other Defined Basis" (cont'd)**
  - The isolation function associated with the following systems maybe accomplished remote manually instead of automatically:
    1. Engineered safety feature systems,
    2. Systems which are not required to function following a loss-of-coolant accident but, if available, can be used to accomplish a function similar to an engineered safety feature system. Examples of such systems are BWR feedwater systems and PWR fluid systems required for reactor coolant pump operation.
  - Remote manual closure of valves is acceptable when provisions are made to detect possible failure of the fluid lines inside or outside containment, or both, and the capability is maintained to remote manually isolate these lines.

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ANSI 56.2 - 1984

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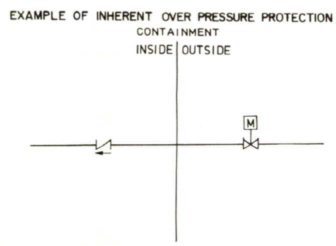

FIGURE 3

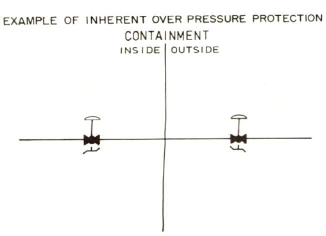

FIGURE 4

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ANSI 56.2 - 1984

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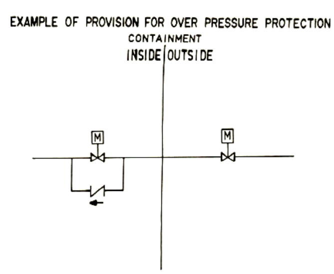
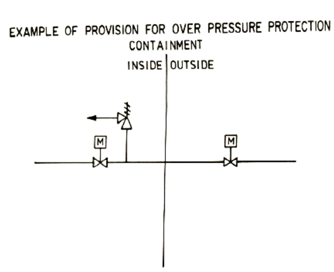


FIGURE 5



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ANSI N-271-1976

- **In summary, the main benefit of this standard is that in addition to indicate GDC requirements for isolation, it presents guidance on what are acceptable "Other Defined Bases" for Criterion 55 and 56.**
- **The standard also has two appendices:**
  - Appendix A, devoted to BWR containment isolation configurations for different systems,and
  - Appendix B devoted to PWR containment isolation configurations for various PWR systems

Regulatory Guide 1.141

- **ANSI-N-271-1976 endorsed as being generally acceptable subject to 10 conditions imposed by the staff. Several are relaxations…**
  - 3.6.4, System integrity inspections may be applied to closed systems inside the containment in lieu of leak testing.
  - 3.6.6, Under conditions specified in the Reg Guide, relief valves discharging outside the containment may be used as containment isolation devices in the forward flow direction.

Regulatory Guide 1.141

- The licensee should provide thermally induced overpressure protection for liquid-filled piping between containment isolation barriers inside containment.
- 4.2.3, states, "Sealed closed isolation valves are under administrative controls and do not require position indication in the control room for valve status." Because the containment isolation valves are components of the containment isolation system, which is an engineered safety feature system, all power-operated valves should have position indication in the control room.

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Regulatory Guide 1.141

- 4.2.4, states, "Isolation valve closure shall be completed when an isolation signal is received, and the valve shall not be opened until the signal is removed and deliberate operator action is taken (reset switch)."
  - The reactor operator should not be able to override a containment isolation signal in such a way that would return any isolation valve to its normal (pre-accident) condition by a single action.
  - The licensee should not consider the use of procedural controls to prevent the reopening of a valve upon reset/override as an acceptable design alternative.
  - The design of the reset/override capability should require a deliberate separate operator action, in addition to the reset/override of the signal, for the reopening of each isolation valve.

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- 4.2.5, states, "Diversity in means of actuation of automatic isolation valves in series should be considered to preclude common mode failure."
  - The NRC staff'sposition is that the licensee should provide diversity in the parameters sensed (i.e., types of isolation signals) for the initiation of containment isolation.
- 4.4.2 states, "For power-operated isolation valves, which do not receive a containment isolation signal, the primary mode shall be a remote manual initiation signal from the main control room."
  - A containment isolation signal should automatically isolate all nonessential systems, as required in 10 CFR 50.34(f)(2)(xiv).

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Regulatory Guide 1.141

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- 4.4.8 gives general requirements for closed systems
  - In addition, all branch lines and their isolation valves in closed systems both inside and outside the containment should meet the design criteria of Section 3.5 or Section 3.6.7 of ANSI N271-1976 if, the branch lines constitute one of the containment isolation barriers.
- Section 4.6.3 cites Regulatory Guide 1.7 for guidance in determining radiation exposures for a loss-of-coolant accident.
  - Regulatory Guide 1.89, gives more appropriate guidance to determine radiation exposures for a loss-of-coolant accident.

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Regulatory Guide 1.141

- 4.14 states, “The piping between isolation barriers or piping, which forms part of isolation barriers, shall meet the requirements of 3.7 and applicable requirements for isolation barriers.”
  - Piping between isolation barriers should meet the applicable requirements of Section 3.5 or Section 3.7

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Module 36
ASME PTC 32.1 Standard

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Module 36
ASME PTC 32.1 Standard

Instructor: C. Wesley Rowley, P.E.

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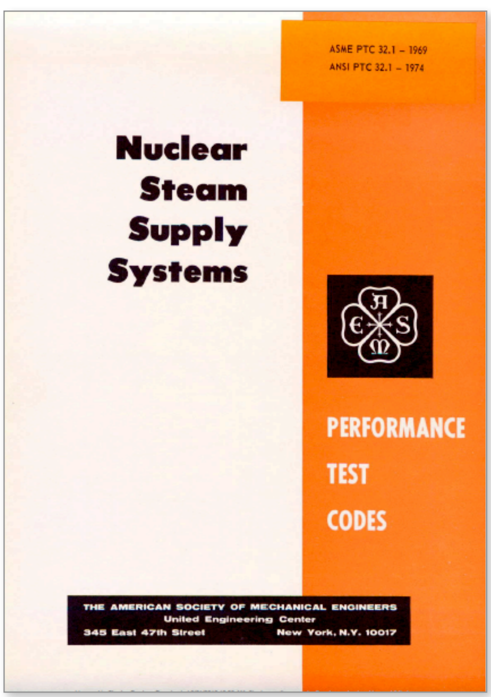

ASME PTC 32.1-1974
Test Code for Nuclear Steam Supply Systems
CONTENTS

Introduction
1. Object and Scope
2. Symbols and Their Descriptions
3. Guiding Principles
4. System Thermal Performance
5. Computations
6. Reporting Results

Learning Objectives

- **Learn the guiding principles for a NSSS thermal calculation of power.**
- **Learn how to apply the "energy loss method" to the "envelope boundary" for the thermal calculations.**
- **Learn how instrument calibration can affect the overall thermal calculation.**

ASME PTC 32.1 Standard

- **Title: Nuclear Steam Supply Systems**
  - It is one of the many ASME Performance Test Codes.
  - It provides a method for testing a Nuclear Steam Supply System as an entity.
  - It was first issued in 1969 and last revised in 1974.
  - It is still a widely used standard, hence this training module.
  - A somewhat related standard is PTC 32.2 Nuclear Reactor Fuel.

Major Sub-topics

- Introduction
- References
- Objective and Scope
- Energy Loss Method
- Symbols and Definitions
- "Envelope Boundary"
- Guiding Principles
- System Thermal Performance
- "Test" versus "Run"
- Computations
- Reporting Results

Introduction

- **NSSS performance testing instructions**
- **The systems are defined as nuclear reactors that have been designed for the generation of thermal energy and the equipment required to transfer that energy from the fuel to the working fluid of the power cycle.**
- **Application in ASME PTC 32.1 is limited to LWRs (specifically PWRs and BWRs).**

References

- **PTC 1 – Code on General Instructions**
- **PTC 2 – Code on Definitions and Values**
- **PTC 6 – Code on Steam Turbines**
- **PTC 19 – Supplements on Instruments and Apparatus**
  - PTC 19.1 – General Considerations
  - PTC 19.2 – Pressure Measurement
  - PTC 19.3 – Temperature Measurement
  - PTC 19.5 – Measurement of Quantity of Materials
  - PTC 19.6 – Electrical Measurements in Power Circuits
  - PTC 19.11 – Methods for Determination of Quality and Purity of Steam
  - PTC 19.21 – Leak Detection and Leakage Measurement

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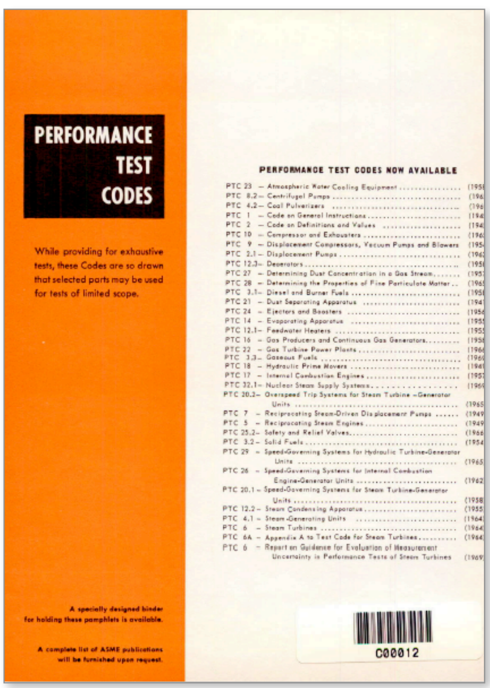

PERFORMANCE TEST CODES

While providing for exhaustive tests, these Codes are so drawn that selected parts may be used for tests of limited scope.

PERFORMANCE TEST CODES NOW AVAILABLE

PTC 23 – Atmospheric Water Cooling Equipment .... (195
PTC 8.2 – Centrifugal Pumps .... (196
PTC 4.2 – Coal Pulverizers .... (196
PTC 1 – Code on General Instructions .... (194
PTC 2 – Code on Definitions and Values .... (194
PTC 10 – Compressor and Exhausters .... (196
PTC 9 – Displacement Compressors, Vacuum Pumps and Blowers (195
PTC 7.1 – Displacement Pumps .... (196
PTC 12.3 – Deaerators .... (195
PTC 27 – Determining Dust Concentration in a Gas Stream .... (195
PTC 28 – Determining the Properties of Fine Particulate Matter .. (196
PTC 3.1 – Diesel and Burner Fuels .... (195
PTC 21 – Dust Separating Apparatus .... (194
PTC 24 – Ejectors and Boosters .... (195
PTC 14 – Evaporating Apparatus .... (195
PTC 12.1 – Feedwater Heaters .... (195
PTC 16 – Gas Producers and Continuous Gas Generators .... (195
PTC 22 – Gas Turbine Power Plants .... (196
PTC 3.3 – Gaseous Fuels .... (196
PTC 18 – Hydraulic Prime Movers .... (194
PTC 17 – Internal Combustion Engines .... (195
PTC 32.1 – Nuclear Steam Supply Systems .... (196
PTC 20.2 – Overspeed Trip Systems for Steam Turbine –Generator Units .... (1965
PTC 7 – Reciprocating Steam-Driven Displacement Pumps .... (1949
PTC 5 – Reciprocating Steam Engines .... (1949
PTC 25.2 – Safety and Relief Valves .... (1966
PTC 3.2 – Solid Fuels .... (1954
PTC 29 – Speed-Governing Systems for Hydraulic Turbine-Generator Units .... (1965)
PTC 26 – Speed-Governing Systems for Internal Combustion Engine-Generator Units .... (1962)
PTC 20.1 – Speed-Governing Systems for Steam Turbine-Generator Units .... (1958)
PTC 12.2 – Steam Condensing Apparatus .... (1955)
PTC 4.1 – Steam-Generating Units .... (1964)
PTC 6 – Steam Turbines .... (1964)
PTC 6A – Appendix A to Test Code for Steam Turbines .... (1964)
PTC 6 – Report on Guidance for Evaluation of Measurement Uncertainty in Performance Tests of Steam Turbines (1969)

A specially designed binder for holding these pamphlets is available.

A complete list of ASME publications will be furnished upon request.

C00012

A S M E

P T C 3 2 .1

B a c k

C o v e r

Objective and Scope

- Purpose of the standard is to establish procedures for conducting tests to determine the thermal performance of a NSSS as a unit.
- Tests include capacity, reactor power level, efficiency, and other related operating characteristics such as steam pressure, moisture content, and solids in the steam.
- The reactor power level is defined as the rate of energy release in the reactor core in units of Mw.
- The capacity of a NSSS is defined as the actual evaporation rate of steam in pounds per hour delivered at specified conditions of the working fluid.
- The efficiency of a NSSS is defined as the ratio of energy absorbed by the working fluid to energy

Energy Loss Method

- Since there is no accurate, direct method of measuring the energy released by the nuclear fuel, the only method used for calculating the energy balance is the direct measurement of the energy output, losses and credits.

- This is referred to as the “energy loss method”.

PWR NSSS Energy Balance

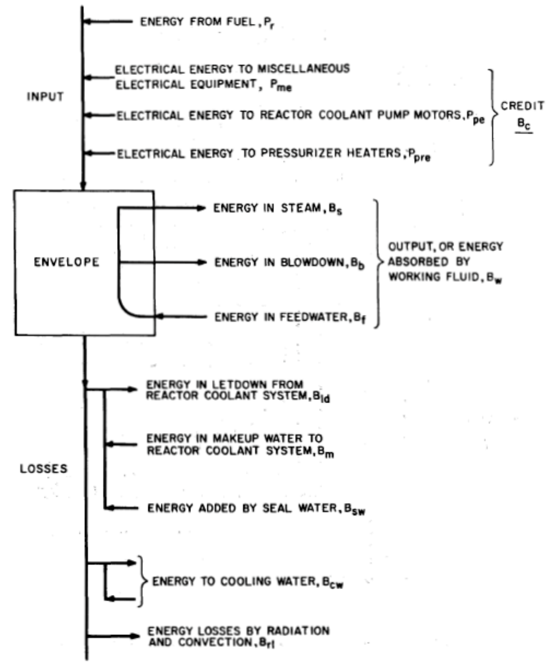

FIG. 2 ENERGY BALANCE OF NUCLEAR STEAM SUPPLY SYSTEM – PRESSURIZED WATER REACTOR

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"Envelope Boundary"

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- **The "envelope boundary" we are using is basically the Rankin Cycle.**
- **In concept we have:**
  - Energy from the nuclear fuel
  - Energy converted to electrical power in the main turbines
  - Energy rejected to ultimate heat sink to convert the low pressure steam to condensate
- **We have refined the "envelope boundary" to achieve more accuracy in our calculations.**

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PWR NSSS Envelope Boundary

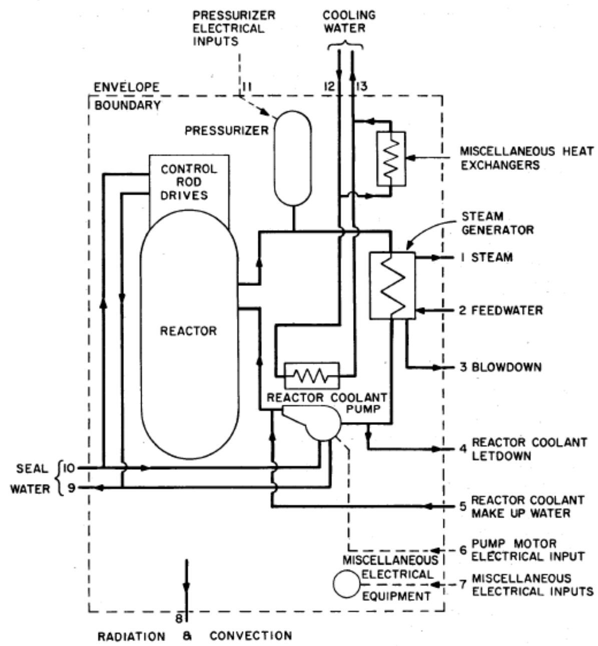

FIG. 1 NUCLEAR STEAM SUPPLY SYSTEM DIAGRAM – PRESSURIZED WATER REACTOR

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Symbols and Definitions

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- **There are 51 symbols used in this standard.**
- **The basic symbols are:**
  - A – surface area (sq ft)
  - B – energy transfer rate (Btu/hr)
  - C – system energy capacity (Mw-hr/F)
  - k – enthalpy (Btu/lb)
  - η – NSSS efficiency (%)
  - P – power (kw)
  - Pr – reactor power level (Mw)
  - p – pressure (psia)
  - T – temperature (F)
  - U – heat transfer coefficient (Btu/hr-ft2-F)
  - W – flow (lbs/hr)
  - x – steam quality (mass percent)

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Guiding Principles

- Definition of the "envelope boundary".
- Should include or exclude certain pieces of equipment.
- Energy credits may be assigned if not measured.
- Definition of permissible deviation from specified operating conditions between duplicate runs.
- Establishment of number and duration of runs, number of load points, observations and readings to be taken, procedures to be followed, and basis for rejection of runs.

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Guiding Principles (cont'd)

- Identification of instrumentation, calibration of instruments, methods of measurement, and equipment to be used in testing the unit.
- Corrections to be made for deviations from specified operating conditions.
- Allowances for errors of measurement and sampling are permissible.
- At least two runs shall be made to demonstrate the attainment of capacity or reactor power level.
- Prior to "run" the NSSS shall be operated for sufficient time to show that steady state operating conditions have been achieved.

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System Thermal Performance

- **The capacity of the NSSS is defined as the rate of steam generation in pounds per hour delivered at specified conditions of the working fluid.**
- **The power level of the reactor is equal to the rate of fuel energy input (expressed in Mw).**
- **The "loss method" shall be based on accurate and complete information which will make possible the determination of output, losses, and credits.**
- **The efficiency is equal to the output divided by the output and losses.**

System Thermal Performance (cont'd)

- **Data Required:**
  - Temperature, pressure, and flow rates of any medium representing energy quantities in the output or losses and power inputs to other items which represent credits.
  - Quality of the steam.
  - Pump power.
  - Certain items may be determined to be insignificant in comparison with the accurancy required.
  - Certain other items may be found to be small, yet significant enough to be included in the determinations.

"Test" versus "Run"

- **Word "test" is applied only to the entire investigation.**
- **Word "run" is applied to a subdivision of the investigation (or "test").**
- **A "run" consists of a complete set of observations made for a period of time with one or more of the independent variables maintained virtually constant.**

Computations

- **<u>Capacity</u>, W_s (steam flow = feedwater flow – blowdown flow) ($W_s = W_f - W_b$)**
- **<u>Reactor Power Level</u>, P_r (reactor power level = energy rate absorbed by the working fluid + energy rate losses from the steam supply system + energy rate credits to the steam supply system) $P_r = B_w + B_l - B_c \ / \ 3{,}412{,}141$**
- **<u>NSSS efficiency</u> (%)**

  $\eta = B_w \times 100 \ / \ B_w + B_l$

Computations (cont'd)

Energy (rate) loss, B_l

$$B_l = B_{ld} - B_m - B_{sw} + B_{cw} + B_{rl}$$

where

B_{ld} = energy (rate) in letdown from reactor coolant system (Btu/hr)

B_m = energy (rate) in makeup to reactor coolant system (Btu/hr)

B_{sw} = energy (rate) in seal water (Btu/hr)

B_{cw} = energy (rate) to cooling water (Btu/hr)

B_{rl} = energy (rate) loss from energy envelope by radiation or convection (Btu/hr)

Reporting Results

- **To report the test results, ASME PTC 32.1 provides sample test forms.**
- **For the PWR the sample test form identifies 37 items of data (parameters) including the units (i.e., pressure is psia, steam temperature in oF, feedwater flow in lbs/hr), cooling water inlet enthalpy in Btu/lb, pressurizer heater power in kw).**
- **For the BWR the sample test form identifies 38 items of data (parameters).**

Case Study

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- **Thermal Performance Calculation of PWR NSSS**
  - Envelope Simplifying Assumptions:
    - Input
      - Nuclear fuel
      - RCP motors
      - Pressurizer heaters
      - Miscellaneous electrical equipment (insignificant)
    - Losses (assume all are insignificant)
    - Feedwater flow = Steam flow (blowdown is zero)

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Case Study (cont'd)

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- **Measured Data:**
  - Item #1 - ??
  - Item #2 - ??
  - Item #3 - ??
  - Item #4 - ??
  - Item #5 - ??
  - Item #6 - ??

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Case Study (cont'd)

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- **Applicable Equations:**
  - Equation #1 - ??
  - Equation #2 - ??
  - Equation #3 - ??
  - Equation #4 - ??

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Case Study (cont'd)

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- **Calculations:**
  - Calculation #1 – ??
  - Calculation #2 – ??
  - Calculation #3 – ??
  - Calculation #4 – ??

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Case Study (cont'd)

- **Results:**

  Capacity ____________ lbs/hr

  Reactor Power Level ____________ Mwt

  Efficiency ___________ %

Quiz / Learning Questions

- **What are the guiding principles for a NSSS thermal calculation of power?**
- **How is the "energy loss method" applied to the "envelope boundary" for the thermal calculations?**
- **How can instrument calibration affect the overall thermal calculation?**

Module 37

Seismic Qualification of Equipment

Advanced Systems Technology and Management, Inc.

UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 37
Seismic Qualification of Equipment

Instructor: C. Wesley Rowley, P.E.

Learning Objectives

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- **Learn how equipment responds to seismic activity.**
- **Learn the general seismic qualifications methods for active equipment.**
- **Learn the common terms and definitions used in seismic qualification.**

Significant Sub-Topics

- Seismic Key Terms
- General Design Criteria
- Key References
- ASME QME-1 Standard
- Earthquake Environment and Equipment Response
- Qualification Principles
- Qualification Specification
- Qualification Program

3

Key References

- **ASME Standards**
  - ASME QME-1 Code
- **IEEE Standards**
  - IEEE-344
- **NRC Regulatory Guides**
  - RG 1.29
  - RG 1.60*
  - RG 1.61*
  - RG 1.92*
  - **RG 1.100**

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General Design Criterion (10 CFR 50 Appendix A)

- **Title: GDC 2 "Design Bases for Protection Against Natural Phenomena"**
  - Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect:
    1. appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated,
    2. appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and
    3. the importance of the safety functions to be performed.

Key Definitions

- *Operating basis earthquake ground motion (OBE)* is the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional. The operating basis earthquake ground motion is only associated with plant shutdown and inspection unless specifically selected by the applicant as a design input.

- *Safe-shutdown earthquake ground motion (SSE)* is the vibratory ground motion for which certain structures, systems, and components must be designed to remain functional.

… definitions excerpted from 10 CFR 50 Appendix S.

Earthquake Criteria (10 CFR 50 Appendix S)

U.S.NRC United States Nuclear Regulatory Commission *Protecting People and the Environment*

- **Title: "Earthquake Engineering Criteria for Nuclear Power Plants"**
  - I. **Introduction - implements GDC 2**
  - II. **Scope**
  - III. **Definitions**
  - IV. **Application to Engineering Design**
    - a) Vibratory Ground Motion.
    - b) Operating Basis Earthquake Ground Motion
    - c) Required Plant Shutdown
    - d) Required Seismic Instrumentation

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Earthquake Criteria (10 CFR 50 Appendix S)

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- **The OBE was changed from ½ the SSE to a value defined by licensee.**
- **If a value of greater than 1/3 the SSE is selected, an analysis must be performed to justify that all SSCs necessary for continued operation without undue risk to the health and safety of the public remain functional and within applicable stress, strain, and deformation limits.**

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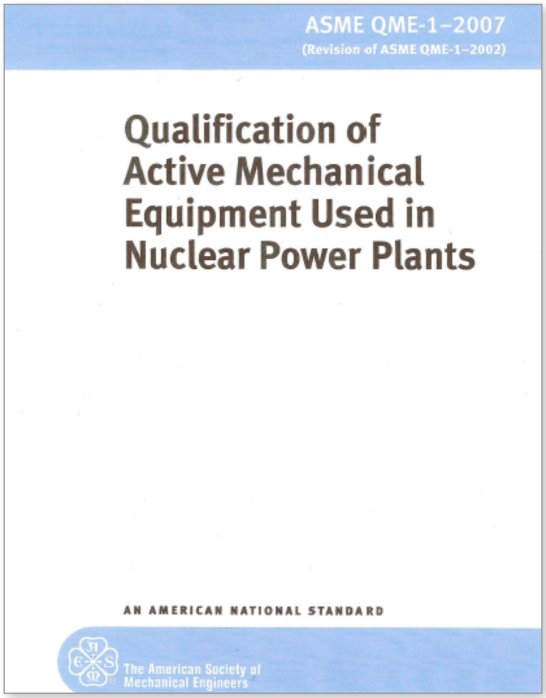

CONTENTS

ASME QME-1 Section QR General Requirements

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ASME QME-1 Section QR General Requirements

U.S.NRC
United States Nuclear Regulatory Commission
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- **Scope - Applicable to mechanical equipment**
  - Pumps
  - Dynamic restraints (Snubbers)
  - Valves
- **Qualification of mechanical equipment to meet functional requirements during and after postulated accidents**

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ASME QME-1 Section QR General Requirements

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- **Definitions**
  - Design life - the time during which satisfactory performance can be expected for a specific set of service conditions
  - Installed life - the interval from installation to removal during which the equipment or component thereof may be subject to design service condition and system demands

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ASME QME-1 Section QR General Requirements

U.S.NRC
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- **Definitions (cont'd)**
  - Qualification Life - the period of time, prior to the start of a design basis event, for which the active mechanical equipment was demonstrated to meet the design requirements for the specified service conditions.
  - Aging: the cumulative effects of operational, environmental, and system conditions on equipment during a period of time up to, but not including, design basis events or the process of simulating these effects.

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ASME QME-1 Section QR General Requirements

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- **Aging**
  - Determination of significant aging mechanisms
    - Thermal
    - Radiation
    - Erosion/Corrosion
    - Vibration
    - Wear
    - Chemical attack

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ASME QME-1 Section QR General Requirements

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- **An aging mechanism is considered significant if it satisfies any of the following criteria:**
  - a) in normal service environments, the aging mechanism promotes the same malfunction as that which may result from exposure to abnormal or design basis event service conditions
  - b) the aging mechanism adversely affects the ability of the active mechanical equipment to perform its function in accordance with its specification requirements

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ASME QME-1 Section QR General Requirements

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- c) the deterioration caused by the aging mechanism is not amenable to assessment by in-service test/inspection or surveillance
- d) in the normal service environment, the aging mechanism causes degradation during the design life of the active mechanical equipment that is significant compared with degradation caused by the design basis event

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Qualification Principles

- **Determine significant aging mechanisms (e.g., thermal, radiation, corrosion, vibration, wear, or aggressive chemical attack).**
- **Determine the qualified life (vs design life).**
- **Develop a schedule of item replacement to address the differences in qualified vs design life.**
- **Restraints are used to control dynamic system responses.**

ASME QME-1 Section QR General Requirements

- **Qualification Methods**
  - Test
  - Analysis
  - Earthquake Experience Data
  - Similarity

  or
  - Combination of the above

Nonmandatory Appendix QR-A Seismic Qualification of Active Mechanical Equipment

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QR-A1000 Scope
QR-A2000 Purpose
QR-A3000 References
QR-A4000 Definitions
QR-A5000 Earthquake Environment and Equipment Response
QR-A6000 Seismic Qualification Requirements
QR-A7000 Qualification Methods
QR-A8000 Documentation
Attachment A Guidelines for Qualification by Similarity (Indirect Method)
Attachment B Examples of Qualification of Pumps and Valves by Analysis
Attachment C Qualification of Pumps and Valves Using Natural Earthquake Experience Data

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Earthquake Environment and Equipment Response

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QR-A5000
QR-A5100 Earthquake Environment
QR-A5200 Active Mechanical Equipment on Foundations
QR-A5300 Active Mechanical Equipment on Structures
QR-A5400 Active Mechanical Equipment on Systems (in Line)
QR-A5500 Nonlinear Equipment Response
QR-A5600 Simulating the Earthquake
- QR-A5610 Required Input Motion
- QR-A5620 Response Spectrum
- QR-A5630 Time History
- QR-A5640 Acceleration or Equivalent Static Load Design Values

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Damping

- **Damping is the measure of the energy dissipation of a material or structural system as it responds to dynamic excitation.**
  - Structural (or global) Damping
  - Piping Damping
  - Electrical Distribution System Damping
  - HVAC Duct Damping
  - Mechanical and Electrical Component Damping
  - Local Damping

Qualification Specification

- **Dynamic Restraints (snubbers)**
  - Define the required performance characteristics.
  - Provide values and ranges of dynamic restraint.
  - Identify the required environmental conditions.
- **Pump Assembly**
  - Identify the pump assembly specified function.
  - Identify the pump assembly boundary.
  - Describe interface attachments and loads.
- **Valve Assembly**
  - Identify the functional requirements.
  - Identify the valve assembly boundary.
  - Identify the required environmental conditions.

Note: all components need to be reconciled with the Design Specification of the NPP (see ASME BPV Code Section III).

Qualification Program

- **Dynamic Restraints (snubbers)**
  - Define the level of the force – displacement relationship.
  - Predict the degradation of this relationship when subjected to operational and severe environmental conditions.
- **Pumps**
  - Identify potential malfunctions.
  - Select qualification method for pump, driver, shaft-seal system, power transmission device, and auxiliary equipment.
  - Qualify for aging and dynamic loading.
- **Valves**
  - Address environmental and aging, sealing capability, end loading, seismic, and functional requirements.
  - Account for dimensional variations of critical clearances of essential-to-function parts.
  - Diagnostic testing shall be performed during the qualification testing.

Nonmandatory Appendix QR-B Guide for Qualification of Nonmetallic Parts

QR-B1000 Scope

QR-B2000 Purpose

QR-B3000 References

QR-B4000 Definitions

QR-B5000 Requirements

QR-B5100 General

QR-B5200 Identification and Specification of Qualification Requirements

QR-B5300 Selection of Qualification Methods

QR-B5400 Preservation of Qualification

QR-B5500 Documentation

Nonmandatory Appendix QR-B (cont'd)

QR-B6000 Methods of Qualification

- QR-B6100 General
- QR-B6200 Arrhenius Model
- QR-B6300 Testing
  - QR-B6310 Thermal Aging
  - QR-B6320 Radiation Aging
  - QR-B6330 Mechanical Wear Aging
- QR-B6400 Use of Experience
- QR-B6500 Qualification by Analysis

QR-B7000 Documentation

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ASME QME-1 Section QDR Qualification of Dynamic Restraints

QDR-1000 Scope

QDR-2000 Purpose

QDR-3000 Definitions

QDR-4000 Qualification Principles and Philosophy

- QDR-4100 Hydraulic Snubbers
- QDR-4200 Mechanical Snubbers
- QDR-4300 Gap Restraints

QDR-5000 Functional Specification

QDR-6000 Qualification Program

- QDR-6100 General Requirements
- QDR-6200 Parent Restraint Qualification
- QDR-6300 Candidate Qualification
- QDR-6400 Extension of Qualification

QDR-7000 Documentation Requirements

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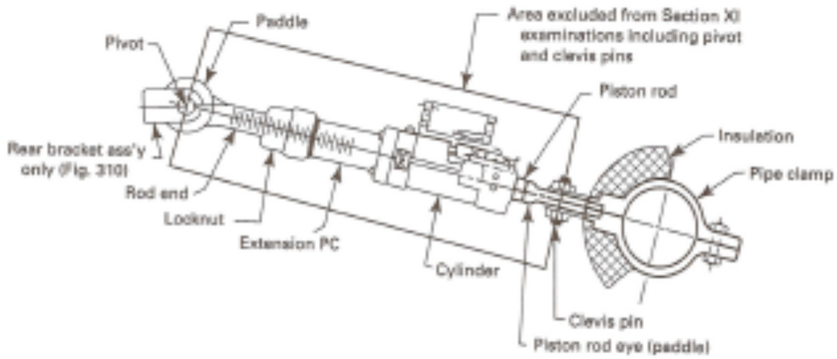
Key Aspects of Snubber Assembly



… figure from ASME BPV Code Section XI.

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ASME QME-1 Section QP Qualification of Pump Assemblies

QP-1000 Scope
QP-2000 Purpose
QP-3000 References
QP-4000 Definitions
QP-5000 Qualification Principles and Philosophy
QP-6000 Qualification Specification
QP-7000 Qualification Program
- QP-7100 General Requirements
- QP-7200 Review for Potential Malfunctions
- QP-7300 Selection of Qualification Methods
- QP-7400 Aging
- QP-7500 Dynamic Loading

QP-8000 Documentation

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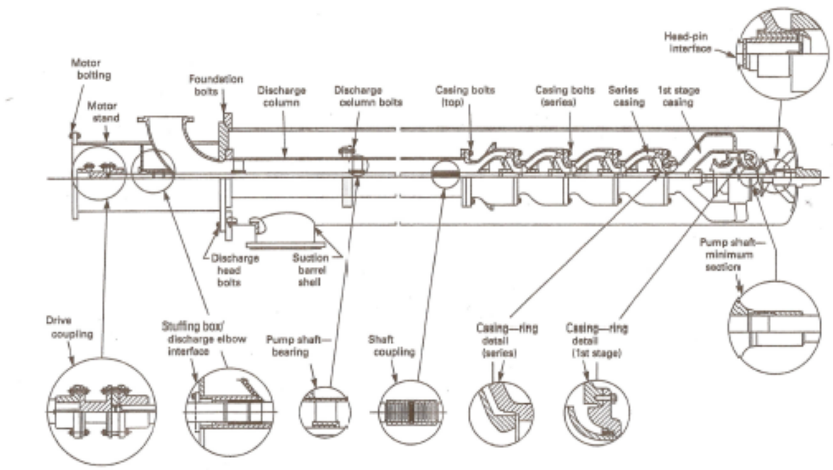
Key Aspects of Pump Assembly

Fig. B-1 Pump Assembly



…. figure from ASME QME-1-2007.

ASME QME-1 Section QV Qualification of Valve Assemblies

QV-1000 Scope
QV-2000 Purpose
QV-3000 References
QV-4000 Definitions
QV-5000 Qualification Principles and Philosophy
QV-6000 Qualification Specification
QV-7000 Qualification Program
QV-8000 Documentation

ASME QME-1 Section QV (cont'd)

QV-7000 Qualification Program

- QV-7100 General Requirements
- QV-7200 Analysis Guidelines
- QV-7300 Specific Qualification Requirements for Valve Assemblies
- QV-7400 Qualification Requirements for Power-Operated Valve Assemblies
- QV-7500 Qualification Requirements for Self-Actuated Check Valve Assemblies
- QV-7600 Qualification Requirements for Safety and Relief Valve Assemblies

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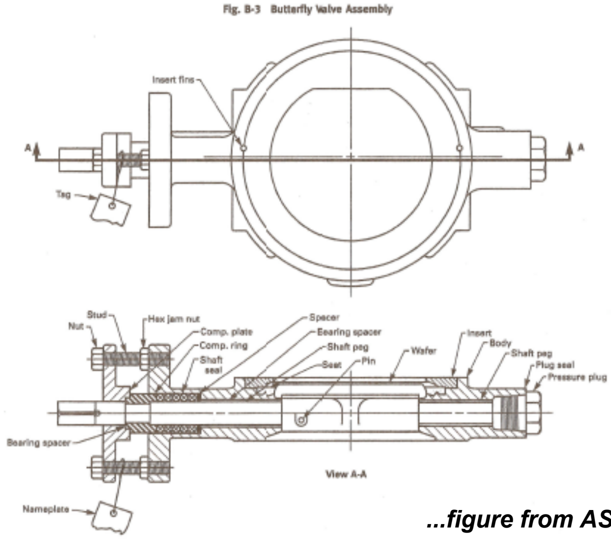
Key Aspects of Valve Assembly

Fig. B-3 Butterfly Valve Assembly



...figure from ASME QME-1-2007

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IEEE-344 Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations

- **Scope**
  - This document describes recommended practices for establishing seismic qualification procedures that...demonstrate that the Class 1E equipment can meet its performance requirements during and/or following one safe shutdown earthquake (SSE) event preceded by a number of operating basis earthquake (OBE) events.

IEEE-344 Seismic Qualification of Class 1E Equipment

- **Qualification Methods**
  - Predict the equipment's performance by analysis.
  - Test the equipment under simulated seismic conditions.
  - Qualify the equipment by a combination of test and analysis.
  - Qualify the equipment through the use of experience data.

IEEE-344 Seismic Qualification of Class 1E Equipment

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- **Vibrational aging**
  - The purpose of the vibrational aging is to show that the lower levels of normal and transient vibration, associated with plant operation and the lower intensity earthquake that has a higher probability of occurrence, will neither adversely affect an equipment's performance of its safety function nor cause any condition to exist that, if undetected, would cause failure of such performance during a subsequent SSE.

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IEEE-344 Seismic Qualification of Class 1E Equipment

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- **Vibrational Aging (cont'd)**
  - Vibrational aging is required to be performed before the SSE and OBE tests
  - The number of OBEs selected shall be justified for each site or shall produce the equivalent of 5 OBEs.
- **Loading**
  - Seismic qualification testing is performed at normal operating conditions (at load, at pressure)

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Reg Guide 1.100

- **Endorses ASME QME -2007 and IEEE 344-2004**
  - IEEE- 344 Exceptions
    - Use of Seismic Experience Data must be approved by the staff.
    - Staff does not agree with the 33Hz cutoff
    - OBE defined as 5 events equivalent to ½ the SSE
    - Use damping values from Reg Guide 1.61
    - Fatigue failure at low-cycle loads must be addressed
    - In structure response spectra per NUREG-0800

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Reg Guide 1.100

- **Exceptions to ASME QME-2007**
  - Use of Seismic Experience Data must be approved by the staff.
  - Staff does not agree with the 33Hz cutoff
  - OBE defined as 5 events equivalent to ½ the SSE
  - Use damping values from Reg Guide 1.61
  - Committing to ASME QME makes the Non-Mandatory Appendices mandatory
  - Mechanical Equipment must meet ASME Sec III

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Reg Guide 1.60, Design Response Spectra for Seismic Design of Nuclear Power Plants

- **Developed from a review of earthquake data from California earthquakes**
- **Defines a generic earthquake response spectra gleaned from existing data**
- **Legacy Reg Guide not used for new reactors**
- **New reactor develop site specific response spectra using Probabalistic Seismic Hazard Analysis Techniques (See RG 1.208)**

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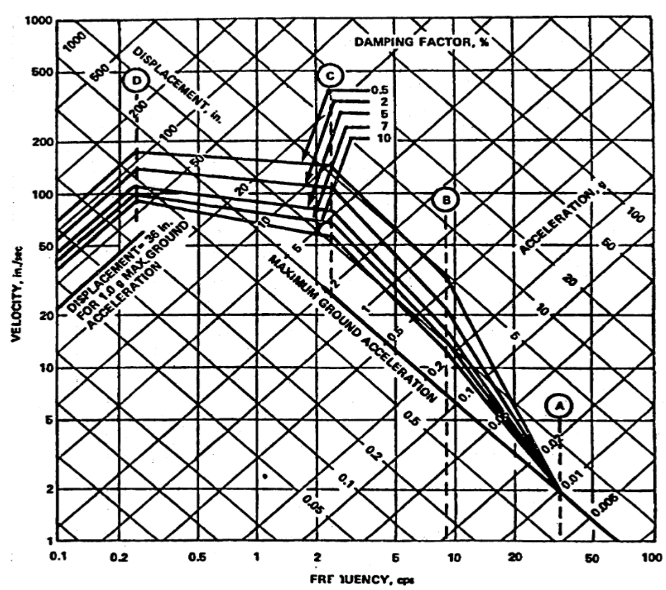

FIGURE 1. HORIZONTAL DESIGN RESPONSE SPECTRA – SCALED TO 1g HORIZONTAL GROUND ACCELERATION

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Reg Guide 1.61, Damping Values for Seismic Design of NPPs

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- **This Reg Guide provides values acceptable to the staff for:**
  - Structural Damping
  - Piping Damping
  - Cable tray and conduit damping
  - HVAC Duct damping
  - Mechanical and Electrical Component damping

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Reg Guide 1.61, Damping Values for Seismic Design of NPPs

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Table 1. SSE Damping Values

Structural Material	**Damping** (% of Critical Damping)
Reinforced Concrete	7%
Reinforced Masonry	7%
Prestressed Concrete	5%
Welded Steel or Bolted Steel with Friction Connections	4%
Bolted Steel with Bearing Connections	7%
Note: For steel structures with a combination of different connection types, use the lowest specified damping value, or as an alternative, use a "weighted average" damping value based on the number of each type present in the structure.	

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Reg Guide 1.61, Damping Values for Seismic Design of NPPs

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Table 2. OBE Damping Values

Structural Material	Damping (% of Critical Damping)
Reinforced Concrete	4%
Reinforced Masonry	4%
Prestressed Concrete	3%
Welded Steel or Bolted Steel with Friction Connections	3%
Bolted Steel with Bearing Connections	5%

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Reg Guide, 1.92 Combining Modal Responses & Spatial Components

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- **The Reg Guide presents various methods acceptable to the staff for combining modal and spatial components in seismic analysis for the design of Seismic Category 1 SSCs.**

- **Methods in the guide apply only when using the Uniform Support Motion (USM) method for response spectra analysis of multi-supported systems, e.g., piping.**

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Case Study
Seismic Response of MOV

- **What are the technical issues that may affect the seismic qualification of a valve?**
  - Valve configuration
    - Type of operator
    - Center of mass of valve and operator
    - Loads imposed on the yoke.
  - Method of valve mounting or support.
  - Piping end loads
  - Distortion of the valve body $\Rightarrow$ valve binding.

Learning Questions

- **How does equipment respond to seismic activity?**
- **What are the general seismic qualifications methods for active equipment?**
- **List some of the common terms and definitions used in seismic qualification.**

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Note - the ASME QME-1 Code is the primary reference for seismic qualification of active mechanical components (e.g., pumps, valves, and snubbers).

UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 38
Separation of Electrical Equipment

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Module 38
Separation of Electrical Equipment

Instructor: Gene Imbro, P.E.

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Learning Objectives

- **Learn the technical requirements for separation of equipment (electrical and mechanical).**
- **Learn the guidance provided in RG 1.75, IEEE standards, and ANS standards.**
- **Learn the NRC expectations for the new build NPPs.**

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Significant Sub-Topics

- 10 CFR 50 Appendix A / GDC 21 & 22
- Reg Guide 1.75
- IEEE-279 / 379 / 384 / 603
- Safety-related vs associated circuits
- Criteria for independence of Electrical Safety Systems
- Physical versus functional separation

Major References

- **10 CFR 50 Appendix A General Design Criteria for Nuclear Power Plants**
  - GDC 17 Electrical Power Systems
  - GDC 21 Protection System Reliability and Testability
  - GDC 22 Protection System Independence
  - GDC 24 Separation of Protection and Control Systems
- **Reg Guide 1.75 Criteria for Independence of Electrical Safety Systems**

Major References (cont'd)

- **IEEE Std. 279 – 1971 (1978) IEEE Standard Criteria for Protection Systems for Nuclear Power Generating Stations**
- **IEEE Std. 379 – 2000 IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems**
- **IEEE Std. 603 – 2009 IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations**

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10 CFR 50, Appendix A - Single Failure

- **Single failure**
  - A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions.
  - Multiple failures resulting from a single occurrence are considered to be a single failure.

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10 CFR 50, Appendix A - Single Failure

- **Single failure (cont'd)**
  - Fluid and electric systems are considered to be designed against an assumed single failure if neither:
    1. a single failure of any active component (assuming passive components function properly) nor
    2. a single failure of a passive component (assuming active components function properly),
  - Results in a loss of the capability of the system to perform its safety functions.

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GDC -17 Electric Power Systems

- **An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety.**
- **The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have <u>sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.</u>**

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GDC 21 Protection system reliability and testability.

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- **The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed.**
- **Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated.**

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GDC 22 Protection system independence

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- **The protection system shall be designed to assure that the effects of natural phenomena,…and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis.**

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U.S. NUCLEAR REGULATORY COMMISSION

Revision 3
February 2005

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.75
(*Draft was issued as DG-1129, dated December 2003*)

CRITERIA FOR
INDEPENDENCE OF ELECTRICAL SAFETY SYSTEMS

A. INTRODUCTION

Section 50.55a, "Codes and Standards," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires in 10 CFR 50.55a(h) that protection systems for plants with construction permits issued after January 1, 1971, but before May 13, 1999, must meet the requirements stated in either IEEE Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations,"[1] or IEEE Std. 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations."[1] For nuclear power plants with construction permits issued before January 1, 1971, protection systems must be consistent with their licensing basis or may meet the requirements of IEEE Std. 603-1991. The safety systems for plants with construction permits issued after May 13, 1999, must meet the requirements of IEEE Std. 603-1991.

Section 4.6 of IEEE Std. 279-1971 requires, in part, that channels that provide signals for the same protective function must be independent and physically separated. Section 5.6.1 of IEEE Std. 603-1991 states, "Redundant portions of a safety system provided for a safety function shall be independent of, and physically separated from, each other to the degree necessary to retain the capability of accomplishing the safety function during and following any design basis event requiring that safety function." General Design Criterion (GDC) 17, "Electric Power Systems," in Appendix A to 10 CFR Part 50 requires, in part, that electric power

[1] Standards promulgated by the Institute of Electrical and Electronics Engineers (IEEE) may be purchased from the IEEE Service Center, 445 Hoes Lane, Piscataway, NJ 08854 (800-678-4333).

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Regulatory Guide 1.75 CONTENTS

A. Introduction
B. Discussion
C. Regulatory Position
D. Implementation
Regulatory Analysis

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IEEE Std608-1991
(Revision of
ANSI/IEEE Std 603-1980)

IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations

Sponsor
Nuclear Power Engineering Committee
of the
IEEE Power Engineering Society

Approved June 27,1991
IEEE Standards Board

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United States Nuclear Regulatory Commission
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IEEE Std. 603-1991 CONTENTS

1. Scope
2. Definitions
3. References
4. Safety system designation
5. Safety system criteria
6. Sense and Command Features – Functional and Design Requirements
7. Executive Features – Functional and Design Requirements
8. Power source requirements

Annex A Illustrations

Annex B LOCA Safety Functions

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IEEE-603 Paragraph 5.6.1

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- **Section 5.6.1 of IEEE Std. 603-1991 states,**
- **"Redundant portions of a safety system provided for a safety function shall be <u>independent of, and physically separated from</u>, each other to the degree necessary to retain the capability of accomplishing the safety function during and following any design basis event requiring that safety function."**

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IEEE Std384-1992
(Revision of IEEE Std 384-1981

IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits

Sponsor
Nuclear Power Engineering Committee of the IEEE Power Engineering Society

Approved June 18,1991
IEEE Standards Board

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IEEE Std. 384-1992 CONTENTS

1. Scope
2. Purpose
3. References
4. Definitions
5. General Independence Criteria
6. Specific Separation Criteria
7. Specific Electrical Isolation Criteria

Annex A - Relationship of Cable Testing Programs to IEEE Std 384-1992

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IEEE-384 Definitions

- **<u>Class IE</u>: The safety classification of the electrical equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or are otherwise essential in preventing a significant release of radioactive material to the environment.**
- **<u>Associated circuits</u>: Non-Class IE circuits that are not physically separated or are not electrically isolated from Class IE circuits by acceptable separation distance, safety class structures, barriers, or isolation devices. Circuits include the interconnecting cabling and the connected loads.**

IEEE-384 Definitions (cont'd)

- **<u>Independence</u>: The state in which there is no mechanism by which any single design basis event, such as a flood, can cause redundant equipment to be inoperable.**
- **<u>Isolation device</u>: A device in a circuit that prevents malfunctions in one section of a circuit from causing unacceptable influences in other sections of the circuit or other circuits.**

IEEE-384 Definitions (cont'd)

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- **Redundant equipment or system: Equipment or system that duplicates the essential function of another piece of equipment or system to the extent that either may perform the required function regardless of the state of operation or failure**

- **Separation distance: Space that has no interposing structures, equipment, or materials that could aid in the propagation of fire or that could otherwise disable Class 1E systems or equipment.**

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Independence Criteria
(from IEEE 384)

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- **5.0 General Independence Criteria**
  - 5.1 Required independence
  - 5.2 Methods of achieving independence
  - 5.3 Equipment and circuits requiring independence
  - 5.4 Compatibility with auxiliary supporting features
  - 5.5 Associated circuits
  - 5.6 Non-Class IE circuits – General criteria
  - 5.7 Mechanical systems
  - 5.8 Structures and equipment
  - 5.9 Fire protection systems
  - 5.10 Fire

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5.1 Required Independence

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- **Physical separation and electrical isolation shall be provided to maintain the independence of Class IE circuits and equipment for proper safety functions so that the safety functions required during and following any design basis event can be accomplished.**

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5.2 Methods of Achieving Independence

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- **The physical separation of circuits and equipment shall be achieved by the use of safety class structures, separation distance, or barriers or any combination thereof. Electrical isolation shall be achieved by the use of separation distance, isolation devices, shielding and wiring techniques, or combinations thereof.**

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5.5 Associated circuits

- **Circuits can be come associated in several ways some of these include:**
  - Electrical connection to a Class 1E power supply without the use of an isolation device.
  - Electrical connection to an associated power supply without the use of an isolation device.
  - Proximity to associated circuits and equipment without the required physical separation or barriers

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5.5 Associated Circuits (cont'd)

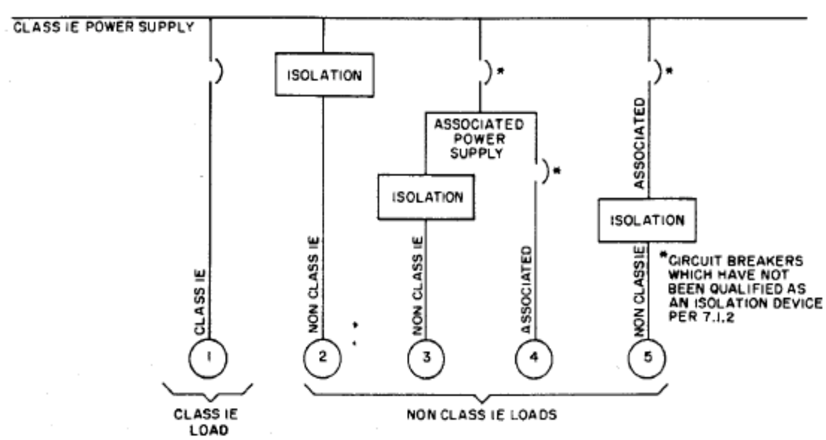

Figure 1—Examples of Association by Connection and Application of Isolation Devices

5.5 Associated Circuits (cont'd)

- **Associated circuits**
  - Shall remain with (traceable to the associated Class 1E division), or be physically separated the same as, those Class 1E circuits with which they are associated.
- **Associated circuits, including their isolation devices or the connected loads without the isolation devices, shall be subject to the qualification requirements placed on Class 1E circuits…**

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5.6 Non-Class IE circuits – General criteria

- **The independence of non-Class 1E circuits from Class 1E circuits or associated circuits shall be achieved by complying with the following requirements:**
  - Non-Class 1E circuits shall be physically separated from Class 1E circuits and associated circuits by the minimum separation requirements…
  - Non-Class 1E circuits shall be electrically isolated from Class 1E circuits and associated circuits by the use of isolation devices, shielding, and wiring techniques or separation distance,…

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5.7 Mechanical Systems

- **Class IE circuits shall be routed or protected so that the failure of the mechanical equipment of one division cannot disable Class IE circuits or equipment essential to the performance of the safety function by the systems of the redundant division(s).**
- **The effects of pipe whip, jet impingement, water spray, flooding, radiation, pressurization, elevated temperature, or humidity on redundant electrical systems caused by failure…shall be considered.**
- **The potential hazard of missiles resulting from failure of rotating equipment or high energy systems shall be considered.**

6. Specific Separation Criteria

- **Physical**
  - Proximity distance
  - Protective “shielding” (e.g., missile barriers)
  - Different rooms
- **Electrical**
  - Isolation devices

NRC Exceptions to IEEE-384

- **Sections 7.1.2.1, 7.1.2.4, and 7.2.2.3 of IEEE Std. 384-1992 should be supplemented as follows:**
  - *The breaker or fuse that is automatically opened by fault current may be used as an isolation device, provided that*
    - *(a) the fault current under bolted and arcing fault conditions … will cause the nearest circuit breaker or fuse to interrupt the fault current prior to initiation of a trip of any upstream protection device,*

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NRC Exceptions to IEEE-384 (cont'd)

- *(b) periodic testing of circuit breakers (visual inspection of fuses and fuse holders) during every refueling must demonstrate that the overall coordination scheme under multiple faults of non-safety-related loads remains within the limits specified in the design criteria for the nuclear power plant.*

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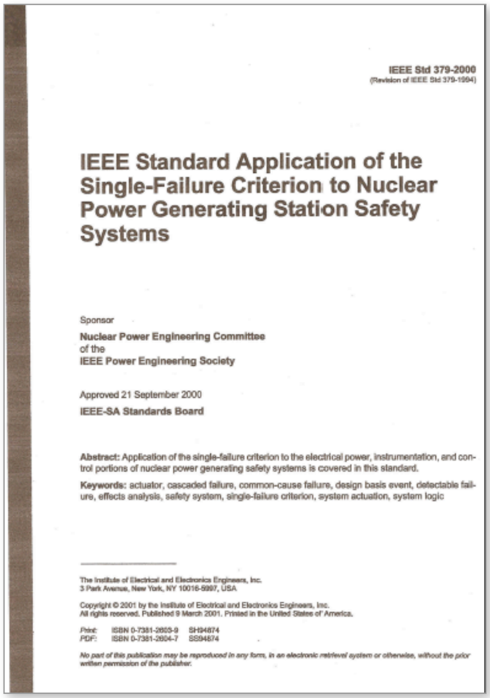

IEEE Std 379-2000
(Revision of IEEE Std 379-1994)

IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems

Sponsor
Nuclear Power Engineering Committee
of the
IEEE Power Engineering Society

Approved 21 September 2000
IEEE-SA Standards Board

Abstract: Application of the single-failure criterion to the electrical power, instrumentation, and control portions of nuclear power generating safety systems is covered in this standard.

Keywords: actuator, cascaded failure, common-cause failure, design basis event, detectable failure, effects analysis, safety system, single-failure criterion, system actuation, system logic

The Institute of Electrical and Electronics Engineers, Inc.
3 Park Avenue, New York, NY 10016-5997, USA

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Protecting People and the Environment

IEEE Std. 379 – 2000
CONTENTS

Learning Questions

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- **What is an associated circuit?**
- **Why is the concept of associated circuits important?**
- **What are the NRC requirements for the new build NPPs?**

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Module 39
Environmental Qualification of Equipment

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Module 39

Environmental Qualification of Equipment

Instructor: C. Wesley Rowley, P.E.

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Learning Objectives

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- **Learn how equipment is designed for specific environmental conditions.**
- **Learn how various equipment responds to various environmental conditions.**
- **Learn the common terms and definitions used in environmental qualification.**

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Significant Sub-topics

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- Major references
- Applicable Terms
- Environment Categorization
- Qualification Principles
- Qualification Methods
- Qualification Program

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Major References

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- **Title 10 Code of Federal Regulations**
  - Part 50.49 Environmental qualification …
  - Appendix A / General Design Criteria 1, 2, 4, & 23
- **NRC Regulatory Guide 1.89, Rev 1, June 1984, Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants**
- **IEEE Std 323-2003, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Stations**
- **NUREG-0588, Rev 1, Interim Staff Position on Environmental Qualification of Electrical Equipment**

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Title 10 CFR Part 50.49

U.S.NRC
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§ 50.49 Environmental qualification of electric equipment important to safety for nuclear power plants.

(a) Each holder of or an applicant for an operating license issued under this part, or a combined license or manufacturing license issued under part 52 of this chapter … shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section …

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10 CFR 50.49 (cont'd)

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(b) Electric equipment important to safety covered by this section is:

1. Safety-related electric equipment.

1. Nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions … by the safety-related equipment.

1. Certain post-accident monitoring equipment.

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10 CFR 50.49 (cont'd)

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(c) Requirements for

1. dynamic and seismic qualification of electric equipment important to safety,
2. protection of electric equipment important to safety against other natural phenomena and external events, and
3. environmental qualification of electric equipment important to safety located in a mild environment are not included within the scope of this section.

   A mild environment is an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences.

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10 CFR 50.49 (cont'd)

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(d) The applicant or licensee shall prepare a list of electric equipment important to safety covered by this section. In addition, the applicant or licensee shall include the information in paragraphs (d)(1), (2), and (3) of this section for this electric equipment important to safety in a qualification file. The applicant or licensee shall keep the list and information in the file current and retain the file in auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment is important to safely meet the requirements of paragraph (j) of this section.

1. The performance specifications under conditions existing during and following design basis accidents.
2. The voltage, frequency, load, and other electrical characteristics for which the performance specified in accordance with paragraph (d)(1) of this section can be ensured.
3. The environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the location where the equipment must perform as specified in accordance with paragraphs (d)(1) and (2) of this section.

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10 CFR 50.49 (cont'd)

(e) The electric equipment qualification program must include and be based on the following:

1. *Temperature and pressure*. The time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design basis accident during or following which this equipment is required to remain functional.
2. *Humidity*. Humidity during design basis accidents must be considered.
3. *Chemical effects*. The composition of chemicals used must be at least as severe as that resulting from the most limiting mode of plant operation (e.g., containment spray, emergency core cooling, or recirculation from containment sump). If the composition of the chemical spray can be affected by equipment malfunctions, the most severe chemical spray environment that results from a single failure in the spray system must be assumed.
4. *Radiation*. The radiation environment must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects.

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10 CFR 50.49 (cont'd)

(e) The electric equipment qualification program (cont'd)

5. *Aging*. Equipment qualified by test must be preconditioned by natural or artificial (accelerated) aging to its end-of-installed life condition. Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment. If preconditioning to an end-of-installed life condition is not practicable, the equipment may be preconditioned to a shorter designated life. The equipment must be replaced or refurbished at the end of this designated life unless ongoing qualification demonstrates that the item has additional life.
6. *Submergence* (if subject to being submerged).
7. *Synergistic effects*. Synergistic effects must be considered when these effects are believed to have a significant effect on equipment performance.
8. *Margins*. Margins must be applied to account for unquantified uncertainty, such as the effects of production variations and inaccuracies in test instruments. These margins are in addition to any conservatisms applied during the derivation of local environmental conditions of the equipment unless these conservatisms can be quantified and shown to contain appropriate margins.

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10 CFR 50.49 (cont'd)

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(f) Each item of electric equipment important to safety must be qualified by one of the following methods:

1. Testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.
2. Testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable.
3. Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.
4. Analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

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10 CFR 50.49 (cont'd)

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(j) A record of the qualification, including documentation in paragraph (d) of this section, must be maintained in an auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment important to safety covered by this section:

1. Is qualified for its application; and
2. Meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life.

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Appendix A GDC 1

- **Criterion 1 – Quality standards and records.**
  - Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Appendix A GDC 2

- **Criterion 2 – Design bases for protection against natural phenomena.**
  - Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Appendix A GDC 4

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- **Criterion 4 – Environmental and dynamic effects design bases.**
  - Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.
  - These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.
  - However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

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Appendix A GDC 23

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- **Criterion 23 – Protection system failure modes.**
  - The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

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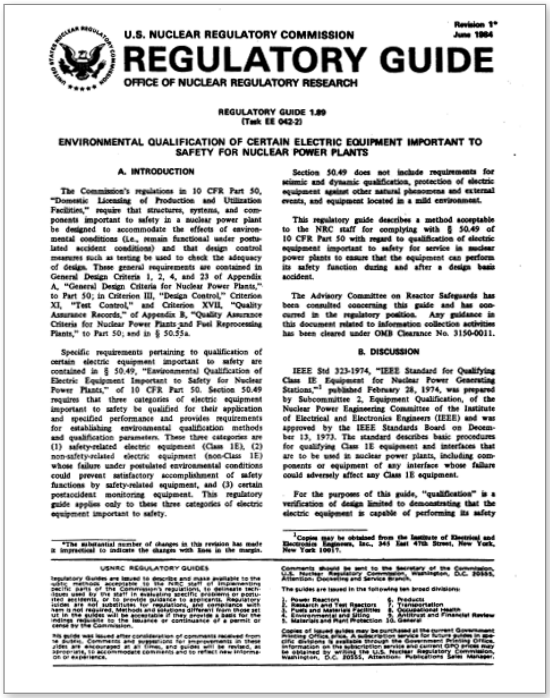

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NRC RG 1.89, Rev 1 June 1984 CONTENTS

A. Introduction
B. Discussion
C. Regulatory Position
D. Implementation

Appendix A – Typical Safety-Related **Electrical Equipment** or Systems
Appendix B – Typical Examples of Non-Safety-Related Equipment
Appendix C – Methods for Calculating Mass and Energy Release
Appendix D – Methodology and Sample Calculation for Quantification Radiation Dose
Appendix E – Qualification Documentation for **Electric Equipment**

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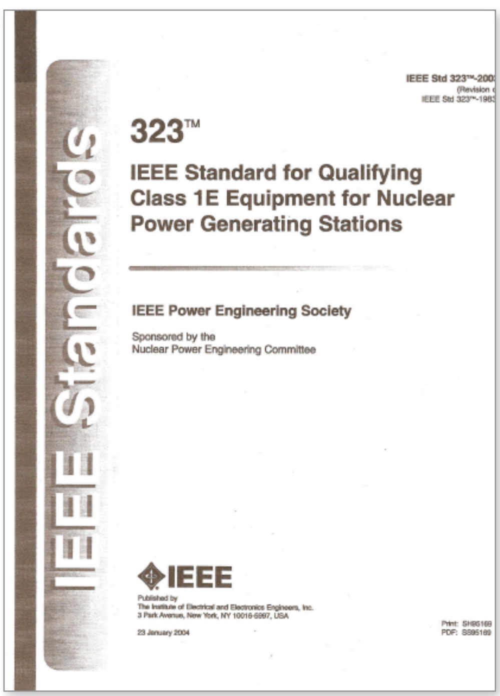

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IEEE Std. 323 - 1974 CONTENTS

1. Scope
2. Purpose
3. Definitions
4. Introduction
5. Principles of Qualification
6. Qualification Procedures & Methods
7. Simulated Service Condition Test Profile
8. Documentation

Appendix A - In-Containment DBE Simulation for PWR and BWR
Appendix B - In-Containment DBE Simulation for HTGR

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Definitions

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- **Class IE. The safety classification of the electric equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise are essential in preventing significant release of radioactive material to the environment.**

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Definitions (cont'd)

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- **Design Life - The time during which satisfactory performance can be expected for a specific set of service conditions.**

- **Installed Life - The interval from installation to removal, during which the equipment or component thereof may be subject to design service conditions and system demands.**

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Definitions (cont'd)

- **Qualified Life - The period of time for which satisfactory performance can be demonstrated for a specific set of service conditions.**
  - NOTE: The qualified life of a particular equipment item may be changed during its installed life where justified.
- **Type Tests - Tests made on one or more sample equipments to verify adequacy of design and the manufacturing processes.**

Principles of Qualification

- **Principles and procedures for demonstrating the qualification of Class 1E equipment shall include:**
  - Assurance that the severity of qualification exceeds the maximum anticipated service requirements and conditions
  - Assurance that extrapolation of data is justified for known potential failure modes and mechanisms leading to them
  - On-going qualification testing of installed equipment whose qualified life is less than the design life of the equipment
  - Documentation providing the basis for qualification
  - Qualification test data as required for on-going qualification

Qualification Methods

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- **Initial Qualification**
  - Type testing
  - Operating experience
  - Analysis
  - Combined methods
- **On-Going Qualification**
  - The qualification methods described above may yield a qualified life of equipment that is less than the anticipated installed life of the equipment. When this occurs, an on-going qualification program may be implemented.

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Qualification Program

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- **The qualification of Class 1E Equipment includes identification of :**
  - Performance characteristics during a DBA
  - Range of voltage, frequency, load, etc.
  - Installation and mounting configuration
  - Preventive maintenance schedule
    - Replacement of seals and lubricants
  - Range and duration of harsh environment

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Test Sequence

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- **The equipment test sequence is as follows:**
  1. Operated at to the extremes of performance specified;
  2. Aged to simulate end-of-life conditions including DBA radiation levels;
  3. Subjected to the mechanical vibration seen in service and in addition a simulated SSE
  4. Operated during a simulated DBE (defined in Std)
  5. Operated while exposed to simulated post-accident conditions.

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Reg Guide 1.89

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- **Endorses IEEE-323-1974 as an acceptable way to meet 10 CFR 50.49**
- **It contains additions guidance in a number of areas, for example:**
  - Provides methods for calculating temperature and pressure inside containment for a LOCA and MSLB
  - The effects of Sprays and Chemicals
  - Determination of radiation conditions inside and outside containment
  - Determination of environmental conditions outside the containment.

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NUREG-0588
Rev. 1

Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment

Including Staff Responses to Public Comments
Resolution of Generic Technical Activity A-24

A. J. Szukiewicz, Task Manager
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

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NUREG-0588, Rev 1 CONTENTS

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Case Study

- **Case Study on determination of the NPP containment environment during different operating and accident modes for the Containment Fan Coil Units:**
  - Containment environment: harsh or mild?
  - Environmental conditions?
  - Environmental qualification method?

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Learning Questions

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- **Identify the common terms and definitions used in environmental qualification.**

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Module 40

Case Studies of Construction Issues Related to Mechanical Components

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Module 40

Case Studies of Construction Issues Related to Mechanical Components

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Module 41
International Codes and Standards

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Module 41

International Codes and Standards

Instructor: Gene Imbro, P.E.

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Learning Objectives

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- Discuss international code and standards activities important to the NRC
- Review how NRC considers international codes and standards in the regulatory process
- Review IAEA activities including NRC role
- Discuss MDEP activities
- Review some significant international codes and standards (e.g. ISO)

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Major Sub-topics

- NRC review process and new reactor designs
- MDEP and applicable issues
- IAEA activities
- International Codes and Standards

3

References for the Module

- **Considerable international activity underway and status is evolving**
  - MDEP
  - International exchange of information
  - Presentation based on recent status reports
- **USNRC New Reactor Applications**
- **Review of IAEA and International Standards Website status**

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International Standards NRC Regulatory Process

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- NRC participates in International Standards Development with the IAEA
- NRC considers new information, including International Standards, in development of rules and regulations, and as new Generic Safety Issues
- NRC is a member of the Convention for Nuclear Safety
- IAEA IRRS (Integrated Regulatory Review Service) mission at NRC beginning in October 1998

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5

International Standards NRC Licensing Process

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- NRC reviews applications for Design Certification and COLs that may include International Standards as part of the proposed licensing basis

- Use of international standards is an acceptable alternative if justified by the applicant

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Multinational Design Evaluation Program (MDEP)

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- **Multinational Design Evaluation Program is a multinational initiative to develop innovative approaches to leverage resources and knowledge of national regulatory authorities who will be undertaking the review of new reactor power plant designs.**

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MDEP Objectives

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- **Enhancing multilateral co-operation within existing regulatory frameworks**
- **Increasing multinational convergence of codes, standards, and safety goals;**
- **Implementing MDEP regulatory practices and products to facilitate licensing reviews of new reactors**

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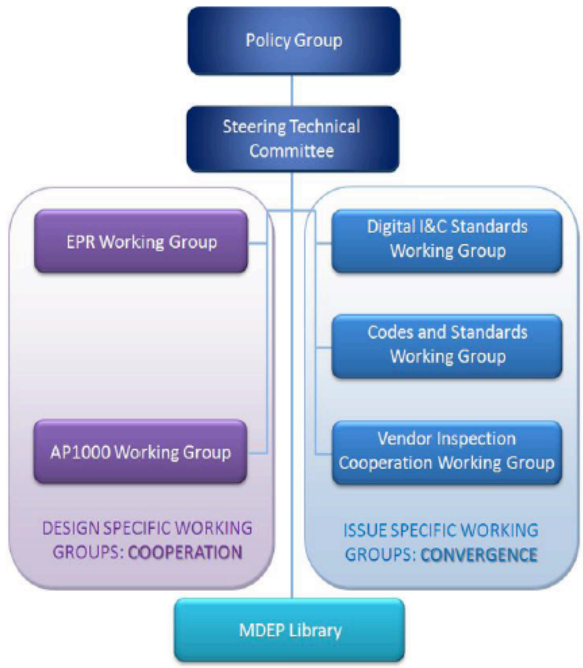
MDEP Organization



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MDEP Codes and Standards Working Group (CSWG)

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- **Objectives of CSWG**
  - Evaluate differences between major pressure boundary design codes
  - Identify most beneficial areas for convergence of codes
  - Examine potential paths for harmonization of pressure-boundary codes

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MDEP CSWG

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- **Pressure-boundary design codes**
  - ASME (U.S.) BPVC, Section III (2007)
  - AFCEN (France) RCC-M (2008)
  - JSME (Japan) S-NC-1 (2008)
  - KEA (Korea) KEPIC (2007)
  - CSA (Canada) N285A (2008E-09A)
  - RNO (Russia) PNAE G-7-002-86 SPiR-WWER-2012

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MDEP CSWG

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- **Current Status**
  - April 7-9, 2010 (Paris, France), SDO team met with MDEP/CSWG to discuss status and results of Code comparisons for Class 1 vessels, piping, pumps and valves. The MDEP/CSWG presented to the SDOs its conceptual plan to harmonize pressure boundary codes and standards on an international level.
  - Final Code-comparison tables and report to be issued for Class 1 vessels, piping, pumps and valves (except Russia) in CSWG November 2010 meeting

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MDEP Vendor Inspection Cooperation Working Group (VICWG)

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- **Established in 2008; currently participated by 10 countries**
- **Long-term objective: to establish a common framework and to organize multinational inspections**
- **Three-step program plan**
  - The 1st step (08/09)
  - The 2nd step (2010/2011)
  - The 3rd step (2011 forward)

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MDEP VICWG

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- **NRC's participation in VICWG**
- **Active and ongoing inspections are underway**
- **Lessons and experience are being shared**

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ISO Standards

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- **International Organization for Standardization**
  - World's largest developer and publisher of International Standards
  - Network of the national standards institutes of 163 countries, one member per country
  - Central Secretariat in Geneva, Switzerland
  - Non-governmental organization that forms a bridge between the public and private sectors

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ISO Standards

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- **ISO Standards related to nuclear and radiation safety**
  - ISO has published about 160 standards related to nuclear and radiation safety
  - NRC seldom endorses these ISO standards

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ISO Standards

- **Examples of ISO standards in nuclear and radiation safety**
  - ISO 6258, Nuclear Power Plants. Design Against Seismic Hazards;
  - ISO/NP 15690 Radiation Protection - Best practice for dealing with discrepancies between personal dosimeter systems used in paralle
  - ISO/ASTM 51204:2004 Practice for dosimetry in gamma irradiation facilities for food processing
  - ISO 26802:2010 Nuclear facilities - Criteria for the design and the operation of containment and ventilation systems for nuclear reactors

ISO Standards

- **Quality Assurance**
  - Canadian CSA Z 299 series of standards were issued in the mid-1970s.
  - The British standard BS 5750 was issued in 1979.
  - In December 1979, the USA issued ANSI/ASQC Z-1.15, Generic Guidelines for quality systems.
  - The ISO technical committee (TC) 176, Quality management and quality assurance, was established in 1979.

ISO Standards

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- **Quality Assurance (cont'd)**
  - It was followed in 1987 by ISO 9001, ISO 9002 and ISO 9003, which provided the requirements for quality management systems operated by organizations with varying scopes of activity, from those including an R&D function, to those uniquely carrying out service and maintenance.
- **ISO 9001 and Appendix B to 10 CFR 50**
  - SECY-03-0117

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National Standards Used as International Standards

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- **API – American Petroleum Institute**
- **ASME – American Society of Mechanical Engineers**
- **ASTM – American Society of Testing & Materials**
- **IEEE – Institute of Electrical & Electronic Engineers**

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IAEA

- **NRC is an active member of the IAEA**
- **The IAEA standards process is a vital part of the international nuclear safety program**

IAEA

Documents

- **Fundamentals**
- **Standards**
- **Guides**
- **Practices**

Organization

- **Members States**
- **Executive Director**
- **Commission of Safety Standards**
- **Committees**
  - Nuclear Safety Standards
  - Radiation Safety Standards
  - Transport Safety Standards
  - Waste Safety Standards

IAEA Safety Standards
for protecting people and the environment

Safety Assessment for Facilities and Activities

General Safety Requirements Part 4
No. GSR Part 4

IAEA
International Atomic Energy Agency

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

General Safety Requirements No. GSR Part 4 Contents

1. Introduction
2. Basis for Requiring a Safety Assessment
3. Graded Approach to Safety Assessment
4. Safety Assessment
5. Management, Use, and Maintenance of the Safety Assessment
6. References
7. Contributors to Drafting and Review
8. Bodies for the Endorsement

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IAEA Documents

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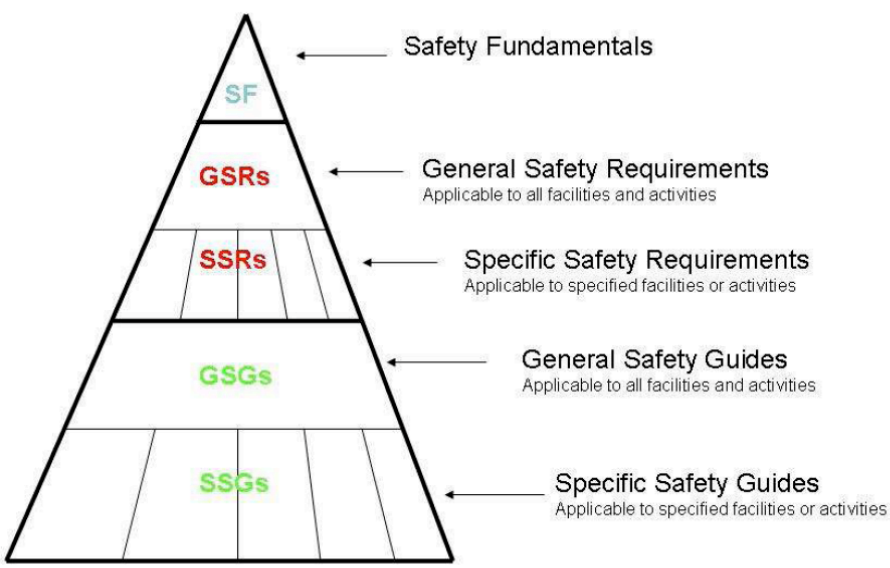

Overall Process for the Development of the IAEA Safety Standards

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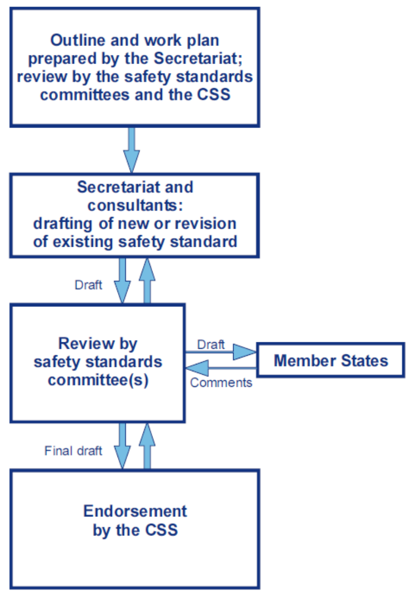

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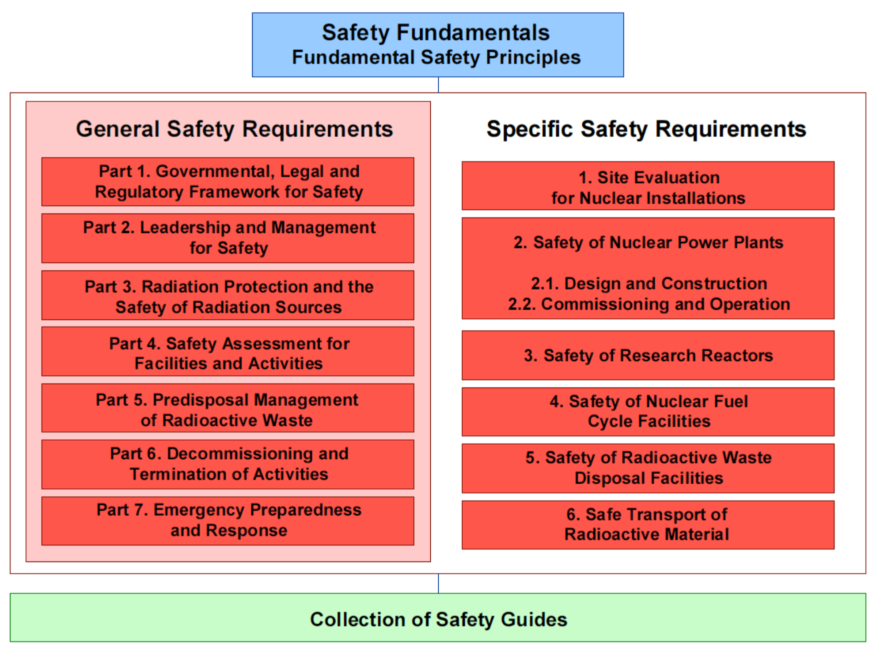

The long term structure of the IAEA Safety Standards Series

International Construction Experience

- **NRC is actively engaged with countries currently constructing nuclear power plants**
- **Inspection personnel are assigned to foreign reactor construction sites to gain insight**
- **Olkiliuto 3 construction experience has been shared openly by STUK and TVO at periodic workshops held in Helsinki and the site**

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Olkiliuto Lessons

- **Both the licensee and the vendor must have:**
  - project management and quality management skills
  - experience from management of a large construction project
  - knowledge and experience in all technical areas relevant for nuclear safety: civil, mechanical, electrical, and I&C engineering, and nuclear technologies (water chemistry, nuclear fuel, reactor physics, thermohydraulics, safety analysis)
  - skills and arrangements to verify achievement of required quality arrangements to control and correct quality non-conformances

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Olkiliuto Lessons

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- **For ensuring good management of the subcontractor chains, it is important that in each call for tender for sub-contracts the vendor clearly indicates and emphasizes the nuclear specific practices, such as:**
  - a requirement to provide design documentation well in advance of planned manufacturing,
  - multiple quality controls and regulatory inspections to beconducted during manufacturing, and
  - expectations on safety culture.

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Learning Questions

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- **How do international standards affect the NRC Licensing Process?**
- **What is the development process for ISO standards, especially those that relate to NPPs?**
- **What are some of the important international issues being discussed with other countries?**

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 42
ANSI/ANS-56.4

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Module 42
ANSI/ANS-56.4

Instructor: Gene Imbro, P.E.

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U.S.NRC United States Nuclear Regulatory Commission *Protecting People and the Environment*

Learning Objectives

- **Purposes, methods, criteria for conducting various types of the containment pressure and temperature transient analyses for the dry primary containment, pressure suppression pool primary containment and the secondary containment**

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ANSI/ANS 56.4

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- **Title:** **Pressure and temperature transient analysis for light water reactor containments**

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Major Sub-topics

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- Mass and energy release
- Dry primary containment pressure and temperature transient analysis
- Water pressure suppression primary containment pressure and temperature transient analysis
- Secondary containment pressure and transient analysis

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Introduction

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1. **Purpose and Scope**
2. **Mass and energy release**
3. **Dry primary containment pressure and temperature transient analysis**
4. **Water pressure suppression primary containment pressure and temperature transient analysis**
5. **Secondary containment pressure and transient analysis**

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Purpose and Scope

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- **Purpose**
  - Providing methods and criteria necessary to perform the pressure and temperature transient analyses required for the primary and secondary containment design and equipment qualification

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Purpose and Scope

- **Scope**
  - LWR containments
  - ECCS minimum backpressure analysis
  - Conservative method

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Mass and Energy Releases

- **Design basis events for the analyses**
  - Loss of coolant accidents (LOCAs)
  - PWR MSLB or MFLB inside containment
- **Reactor coolant system releases**
  - Energy sources
    1. Reactor coolant system water and metal
    2. Steam generator secondary water and metal
    3. Core stored energy
    4. Fission heat

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Mass and Energy Releases
Reactor Coolant System Releases

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- Energy sources (cont'd)
  5. Decay of actinides
  6. Fission product heat decay
  7. Metal water reaction rate
  8. Main steam lines, main/auxiliary feedwater lines
  9. ECCS flow
  10. Safety injection tank nitrogen expansion

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Mass and Energy Releases
Reactor Coolant System Releases

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- Initial conditions
  1. Time of life
  2. Power level – 102% of licensed power level
  3. Core inlet temperature
  4. Reactor coolant system pressure
  5. SG pressure
  6. SG water level
  7. Pressurizer level
  8. Core parameters
  9. Safety injection tank (PWR only)

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Mass and Energy Releases
Reactor Coolant System Releases

- **The worst active single failure shall be considered in determining energy and mass release**
- **Loss of non-emergency power shall be assumed if this leads to higher primary containment pressures**
- **Modeling**

Mass and Energy Releases
PWR Secondary System Releases

- **PWR Secondary System Releases**
  - Energy sources
    1. Reactor coolant system water and metal
    2. Steam generator secondary water and metal
    3. Core storage energy
    4. Fission heat
    5. Decay of actinides
    6. Fission product heat decay
    7. Main steam lines, main/auxiliary feedwater lines

Mass and Energy Releases
PWR Secondary System Releases

- Initial conditions
  1. Time of life
  2. Power level
  3. Core inlet temperature
  4. Reactor coolant system pressure
  5. SG pressure
  6. SG water level
  7. Pressurizer level
  8. Core parameters
  9. Control element assembly position
  10. Boron concentration

Mass and Energy Releases
PWR Secondary System Releases

- The worst active single failure shall be considered in determining energy and mass release

- Loss of non-emergency power shall be assumed if this leads to higher primary containment pressures

Dry Primary Containment P/T Transient Analysis

- **Purpose of analysis**
  - To determine those breaks yielding the maximum internal pressure and temperature for structural design of the primary containment
  - To determine the maximum duration at elevated pressure and temperature for equipment qualification
  - To determine the minimum containment pressure for specific LOCAs for the purpose ECCS performance evaluation

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Dry Primary Containment P/T Transient Analysis

- **Purposes of analysis (cont'd)**
  - Determine the minimum pressure for the purpose of designing containment structure to accommodate the worst-case negative pressure differential
- **Maximum pressure and temperature analysis**
  - Analysis shall be conducted for a spectrum of primary and secondary pipe ruptures

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Dry Primary Containment P/T Transient Analysis

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- **Duration of analysis**
  - Dry containment analysis model
    - Two distinct regions: atmosphere and sumps
    - Energy and mass transfer mechanisms
      - I. Pipe break blowdown
      - II. Energy source terms
      - III. Structural heat transfer

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Dry Primary Containment P/T Transient Analysis

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- Energy and mass transfer mechanisms
  - IV. Containment spray
  - V. Containment heat removal
  - VI. Atmosphere - sump interface
- Initial conditions
- Single failure criterion

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Dry Primary Containment P/T Transient Analysis

- **Minimum backpressure analysis**
  - Purpose - to provide input for evaluation of ECCS performance
  - Selection of a spectrum of breaks
  - Initial conditions chosen to yield a conservatively low minimum dry primary containment atmosphere region pressure
  - Structure heat sinks to be maximized
  - CHRS operating parameters assuming rated values

Dry Primary Containment P/T Transient Analysis

- **Minimum backpressure analysis (cont'd)**
  - Effect of ECCS water spillage
  - Blowdown phase separation
    - Venting through dry primary containment purge valve function
    - Energy/mass transfer across atmosphere-sump interface

Dry Primary Containment P/T Transient Analysis

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- **Minimum Dry Primary Containment Pressure Analysis**
  - Purpose: to determine the worst case negative pressure differential across the containment structure
  - Scenario: inadvertent actuation of the dry primary containment spray system
  - Initial and boundary conditions chosen to yield a conservatively low estimate of the containment pressure

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Dry Primary Containment P/T Transient Analysis

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- **Minimum Dry Primary Containment Pressure Analysis (cont’d)**
  - Duration of analysis – to ensure that absolute minimum dry primary containment atmosphere pressure has been found

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Water Pressure Suppression Primary Containment P/T Transient Analysis

- **Purpose of analysis**
- **Maximum pressure and temperature analysis**
- **Minimum WPS primary containment pressure analysis**
- **Bypass leakage**

Water Pressure Suppression Primary Containment P/T Transient Analysis

- **Maximum pressure and temperature analysis**
  - Purpose: to predict a conservatively high estimate of WPS primary containment pressure and temperature resulting from a postulated pipe break
  - Postulated accidents – a spectrum of breaks of different sizes, locations and orientations being analyzed to determine the maximum pressure/temperature in the primary containment

Water Pressure Suppression Primary Containment P/T Transient Analysis

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- **Duration of analysis**
  - Small steam line break
  - Large line breaks
- **WPS Primary Containment Models**
  - Three regions – drywell, wetwell atmosphere and wetwell pool
  - Various mass and energy transfer mechanisms, e.g., blowdown, energy sources not account for in determining mass/energy release rate, vent models, structural heat sinks, spray, CHRS, wetwell atmosphere-pool interface

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Water Pressure Suppression Primary Containment P/T Transient Analysis

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  - Initial conditions chosen in a conservative manner
  - Meeting single failure criterion
- **Minimum WPS Primary Containment Pressure Analysis**
  - Purpose – to determine the limiting negative pressure differential across the WPS primary containment structures
  - Initial conditions – credible and conservative conditions shall be assumed

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Water Pressure Suppression Primary Containment P/T Transient Analysis

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- **Bypass leakage**
  - The bypass leakage analysis is performed to determine the maximum allowable leakage area for direct transfer of steam from drywell to wetwell air space without passing through the vents and into the pool
  - The results serve to specify the performance of other systems and maximum allowable leakage area to prevent wetwell overpressurization

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Secondary Containment P/T Transient Analysis

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- **Introduction**
  - Analyzing the performance of the emergency ventilation system (EVS) of the secondary containments during a LOCA
  - For PWRs, the response of the secondary containment as a result of an MSLB inside primary containment is also addressed

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Secondary Containment P/T Transient Analysis

- **Initial conditions – being chosen in a conservative manner (e.g., pressure, temperature humidity, free volume, containment conditon)**
- **Single failure criterion – the worst single failure assumed**

Secondary Containment P/T Transient Analysis

- **Transient Conditions**
  - Primary containment transient
    - The pipe break inside the primary containment shall be the most sever LOCA
    - The heat transfer from the primary containment to the secondary shall be conservatively estimated by maximizing the heat transfer coefficient between them

Secondary Containment P/T Transient Analysis

- **Transient conditions (cont'd)**
  - Secondary containment transient
    - Mass transfer
    - Heat transfer
    - Primary containment pressure stress and thermal expansion and contraction
    - Determination of secondary containment thermodynamic conditions

Module 43
Part 52 ITAAC

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Module 43
Part 52 ITAAC

Instructor: Gene Imbro, P.E.

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Learning Objectives

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- Review information from previous module on ITAAC
- Discuss the inspection rationale for sampling ITAAC
- Go through several more ITAAC examples

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ITAAC

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- **There are typically hundreds of ITAAC in one Design Certification Document. They are found in the Tier 1 Section of the DCD along with all other Tier 1 information.**
- **ITAAC are primarily focused on the facility rather than operational programs**
- **Each ITAAC takes the tabular form of a detailed listing of:**
  - the specific design commitment
  - the associated inspection (I), test(T) or analysis(A), that needs to be performed to ensure the commitment has been fulfilled
  - the acceptance criterion(AC) applicable to the inspection or test.

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10 CFR Part 52 and Role of ITAAC

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- **ITAAC are proposed by a vendor (e.g. Westinghouse) in their Design Certification Document (DCD) in accordance with the requirements of 52.47 (b) (1)-the content of the application to the NRC**
- **The application also includes the proposed design information, including applicable Codes and Standards for the design of the facility.**
- **The most important Codes and Standards and the ITAAC are included in the Tier 1 part of the application. Tier 1 contains the most important information associated with design and operation of the facility.**

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NRC Review

- **ITAAC are reviewed as part of the DC review by the NRC Staff**
- **Results are described in Section 14.3.2 of SER of AP 1000**
- **The staff performed a detailed review of the AP 1000 with many questions, investigating areas where an additional ITAAC or commitment was necessary. Some examples are:**
  - Containment wall thickness
  - CVCS instruments and controls
  - CRD penetration access and inspectability
  - Accessibility of components for inspection
  - Design Reliability Assurance Program DRAP component lists
  - Cable pulling procedures, standards, and ITAAC
  - Containment sump design change

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NRC Construction Inspection Program

- **The primary element of the Construction Inspection Program applicable to this course is the inspection of ITAAC associated with Mechanical Codes and Standards**
- **Inspection Strategies are Developed by the NRC staff to focus inspections on the most important areas of facility construction. These areas are determined by an integrated process that considers such attributes as safety significance (informed by risk assessments), potential for error, and opportunity to check the ITAAC later in the process through other inspection means.**
- **These inspections of "Targeted" ITAAC will ensure that design commitments, including those with Codes and Standards, have been carried out during the construction of the plant**
- **The "AC", or acceptance criteria in the ITAAC, will be tied to the Code information in Tier 1 and amplified in Tier 2 of the DCD.**

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ITAAC Inspection

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- **ITAAC grouped into approximately 70 families that are defined by a matrix that includes all plant equipment and all activities**
- **The ITAAC that have the most value in each family and that must be inspected are referred to as "Targeted" ITAAC.**
- **Judgment was used by the NRC to establish how many Targeted ITAAC were needed in each family. This is a balance between value and available resources.**
- **SECY 07 0047 goes through this in detail. Overall about 30-40% of ITAAC are inspected.**

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ITAAC MATRIX

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...excerpt slide from an NRC / Region II presentation in June 2009 by NRC staff (at the ASME NTS-1 in Atlanta)

	A) As-Built Inspectiion	B) Welding	C) Construction Testing	D) Operations Testing	E) Qualification Criteria	F) Design/ Fabrication Requirements
01) Foundations & Buildings	A01	B01	C01	D01	E01	F01
02) Structural Concrete	A02	B02	C02	D02	E02	F02
03) Piping	A03	B03	C03	D03	E03	F03
04) Pipe Support & Restraints	A04	B04	C04	D04	E04	F04
05) Reactor Pressure Vessel & Internals	A05	B05	C05	D05	E05	F05
06) Mechanical Components	A06	B06	C06	D06	E06	F06
07) Valves	A07	B07	C07	D07	E07	F07
08) Electrical Components & Systems	A08	B08	C08	D08	E08	F08
09) Electrical Cable	A09	B09	C09	D09	E09	F09
10) I&C Components & Systems	A10	B10	C10	D10	E10	F10
11) Containment Integrity & Pen's	A11	B11	C11	D11	E11	F11
12) HVAC	A12	B12	C12	D12	E12	F12
13) Equipment Handle & Fuel Racks	A13	B13	C13	D13	E13	F13
14) Complex Sys w/ Multi-Comp	A14	B14	C14	D14	E14	F14
15) Fire Protection	A15	B15	C15	D15	E15	F15
16) Engineering	A16	B16	C16	D16	E16	F16
17) Security	A17	B17	C17	D17	E17	F17
18) EP	A18	B18	C18	D18	E18	F18
19) Radiation Protection	A19	B19	C19	D19	E19	F19

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THE AP1000 ITAAC MATRIX

	A)As-Built Inspection	B)Welding	C)Construction Testing	D) Operational Testing	E)Qualification Criteria	F)Design/Fabrication Requirements
01)Foundations & Buildings	14				1	4
02)Structural Concrete			1			
03)Piping	10	10	10	4		17
04)Pipe Supports & Restraints						8
05)RPV & Internals	7	2	1	2	1	4
06)Mechanical Components	28	5	6	22	4	22
07)Valves	8	4	6	27	12	20
08)Electrical Components & Systems	15		5	24	8	8
09)Electrical Cable	10		1			11
10)I&C Components & Systems	61		35	63	16	9
11)Containment Integrity & Penetrations	6			1	1	1
12)HVAC	11	3	3	14	2	10
13)Equipment Handling & Fuel Racks	6			5	3	3
14)Complex Systems w/ Multiple Components	25			4	4	6
15)Fire Protection	7		1	2		
16)Engineering	5				2	10
17)Security	3				1	
18)Emergency Planning						
19) Radiation Protection	5				1	1

<u>NOTE</u>: The values specified in this table include ITAAC for the certified design only. Additional site specific ITAAC will be added when they are identified in the COL application.

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Enclosure 4

9

ITAAC Example

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… excerpt slide from an NRC / Region II presentation in June 2009 by NRC Staff (at the ASME NTS-1 in Atlanta)

TABLE 3.3-6 (BUILDINGS)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.a) Exterior walls and the basemat of the nuclear island have a water barrier up to site grade.	An inspection of the as-built exterior walls and the basemat of the nuclear island <u>up to floor elevation 100'-0"</u>, for application of water barrier will be performed during construction before the walls are poured	A report exists that confirms that a water barrier exists on the nuclear island exterior walls up to site grade.

- **ITAAC = ITA + AC**
- **Standard 3 column format**
  - Design commitment (from design documents)
  - **Inspections, Tests, Analyses** (method)
  - **Acceptance criteria** (if met shows meeting the design commitment)

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ITAAC Examples

… excerpt slide from an NRC / Region II presentation in June 2009 by Chuck Ogle (at the ASME NTS-1 in Atlanta)

ITAAC Number	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2.1.02.08a.ii	8.a) The pressurizer safety valves provide overpressure protection in accordance with Section III of the ASME Boiler and Pressure Vessel Code.	ii) Testing and analysis in accordance with ASME Code Section III will be performed to determine set pressure.	ii) A report exists and concludes that the safety valves set pressure is 2485 psig + 25 psi.
2.2.02.04a	4.a) The components identified in Table 2.2.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components identified in Table 2.2.2-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
2.2.04.04a	4.a) The components identified in Table 2.2.4-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components identified in Table 2.2.4-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
2.2.04.08a.ii	8.a) The SGS provides a heat sink for the RCS and provides overpressure protection in accordance with Section III of the ASME Boiler and Pressure Vessel Code.	ii) Testing and analyses in accordance with ASME Code Section III will be performed to determine set pressure.	ii) A report exists to indicate the set pressure of the valves is less than 1305 psig.
2.3.10.04a	4.a) The components identified in Table 2.3.10-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components identified in Table 2.3.10-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.

NRC Inspection of ITAAC

- **NRC is developing guidance for inspectors**
- **Guidance is expected to include:**
  - Which ITAAC to inspect
  - Applicable inspection procedures
  - What type of observations need to be made (e.g. measurement, walk-down, review)
  - What structures, components to inspect
  - Duration of the inspection
  - Particular planning considerations for the inspection
  - Need for interface with headquarters experts

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Learning Questions

- **What is the purpose of an ITAAC?**
- **What is a Targeted ITAAC?**
- **Why is a MATRIX of ITAAC families used?**
- **What fraction of ITAAC does a licensee have to inspect?**

Module 44
Course Evaluation and Summary

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Module 44
Course Evaluation and Summary

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- **Handout Course Evaluation Sheets to Students for Completion**

- **Question and Answer Period.**

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