Training Course on Civil/Structural Codes and Inspection

BMA Engineering, Inc.

Overall Outline

1000. Introduction

2000. Federal Regulations, Guides, and Reports

3000. Site Investigation

4000. Loads, Load Factors, and Load Combinations

5000. Concrete Structures and Construction

6000. Steel Structures and Construction

7000. General Construction Methods

8000. Exams and Course Evaluation

9000. References and Sources

2000. Federal Regulations, Guides, and Reports

- Objective and Scope
  - Introduce and highlight selected documents and contents relating to civil & structural inspection
  - Present and discuss
    - Federal regulations
    - Regulatory guides
    - Nuclear regulatory reports (NRC reports)
    - ANSI-ASME Codes and standards that detail these documents

2000. Federal Regulations, Guides, and Reports

CFR
- 10 CFR 50.34(b)(6)(ii)
- Appendix A and Appendix B to Part 50

RGs
- RG 1.28, RG 1.29, RG 1.94, RG 1.132
- RG 1.136, RG 1.138, RG 1.142, RG 1.149

NUREG
- NUREG 0800, NUREG/CR-6486
- NUREG/CR-6927

ASME
- ASME NQA-1 Quality assurance requirements

2000. Federal Regulations, Guides, and Reports

- Code of Federal Regulations (CFRs)
  - 10 CFR 50.34(b)(6)(ii) Contents of Applications; Technical Information
  - Appendix A to Part 50--General Design Criteria for Nuclear Power Plants
  - Appendix B to Part 50--Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

2000. Federal Regulations, Guides, and Reports

- Regulatory Guides (RGs) – CE/ST focus
  - RG 1.28: Quality Assurance Program Requirements (Design and Construction) [Revision 3, August 1985] – covered in Section 2000
  - RG 1.29: Seismic Design Classification [2006] – covered in NUREG0800
  - RG 1.94: Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants [Revision 1, April 1976] – covered in Appendix B

2000. Federal Regulations, Guides, and Reports

- Regulatory Guides (RGs) – CE/ST focus (cont.)
  - RG 1.132: Site Investigation for Foundations of Nuclear Power Plants [Revision 2, October 2003] – covered in Section 3000
  - RG 1.136: Materials, Construction, and Testing of Concrete Containments (Articles CC-1000, -2000, and -4000 Through -6000 of the “Code of Concrete Reactor Vessels and Containments”) [Revision 2, June 1981] – covered in Section 5000

2000. Federal Regulations, Guides, and Reports

- Regulatory Guides (RGs) – CE/ST focus (cont.)
  - RG 1.138: Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants [Revision 2, December 2003] – covered in Section 3000

2000. Federal Regulations, Guides, and Reports

- Regulatory Guides (RGs) – CE/ST focus (cont.)
  - RG 1.142: Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments) [Revision 2, November 2001] – covered in Section 5000
  - RG 1.199: Anchoring Components and Structural Supports in Concrete [November 2003] – covered in Section 5000

2000. Federal Regulations, Guides, and Reports

- Nuclear Regulatory Commission (NRC) Reports (NUREGs) – CE/ST focus
  - NUREG 0800 Section 3.0, Design of Structures, Components, Equipment, and Systems
  - NUREG/CR-6486, Assessment of Modular Construction for Safety-Related Structures at Advanced Nuclear Power Plants [1977] – covered in Sections 5000 and 6000

2000. Federal Regulations, Guides, and Reports

- Nuclear Regulatory Commission (NRC) Reports (NUREGs) – CE/ST focus (cont.)
  - NUREG/CR-6927, Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures - A Review of Pertinent Factors – covered in Section 5000

2000. Federal Regulations, Guides, and Reports

- Nuclear Regulatory Commission (NRC) Reports (NUREGs) – CE/ST focus (cont.)
  - ASME NQA-1- 1994 Quality assurance requirements for nuclear facility applications

2000. Federal Regulations, Guides, and Reports

CFR
- 10 CFR 50.34(b)(6)(ii)
- Appendix A and Appendix B to Part 50

RGs
- RG 1.28, RG 1.29, RG 1.94, RG 1.132
- RG 1.136, RG 1.138, RG 1.142, RG 1.149

NUREG
- NUREG 0800, NUREG/CR-6486
- NUREG/CR-6927

ASME
- ASME NQA-1 Quality assurance requirements

Code of Federal Regulations (CFRs)

- 10 CFR 50.34(b)(6)(ii) Contents of Applications; Technical Information
- Appendix A to Part 50--General Design Criteria for Nuclear Power Plants
- Appendix B to Part 50--Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

10 CFR 50.34(b)(6)(ii)

- Title: Contents of Applications; Technical Information
- Requirements:
  - **a) Preliminary safety analysis report**
  - **b) Final safety analysis report**
  - c) Physical security plan
  - d) Safeguard contingency plan
  - e) Protection Against Unauthorized Disclosure
  - f) Additional Three Mile Island (TMI)-Related Requirements

10 CFR 50.34(b)(6)(ii)

- Title: Contents of Applications; Technical Information (cont.)
  - g) Combustible Gas Control
  - h) Conformance with the Standard Review Plan (SRP)
  - i) A description and plans for implementation of the guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with the loss of large areas of the plant due to explosions or fire

10 CFR 50.34(b)(6)(ii)

(a) Preliminary safety analysis report

- Preliminary safety analysis report required
  - For each application for a construction permit
  - For stationary power reactor applications for construction permits applying on or after January 10, 1997

10 CFR 50.34(b)(6)(ii)

(a) Preliminary safety analysis report

- Two aspects:
  - A description and safety assessment of the site
  - A safety assessment of the facility
- Reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products

10 CFR 50.34(b)(6)(ii)

(a) Preliminary safety analysis report

- Design characteristics and proposed operation to be considered by the Commission:
  - Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials
  - Generally accepted engineering standards are applied to the design of the reactor
  - Unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials
  - Safety features engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur including consequence mitigation
  - Safety issues relating to operations

10 CFR 50.34(b)(6)(ii)

(a) Preliminary safety analysis report

- Preliminary design including:
  - Principal design criteria for the facility
  - Design basis
  - Materials, general arrangements and approximate dimensions)
- Preliminary analyses and evaluations
- Identification and justification for selection of variables, conditions based on preliminary safety analysis

10 CFR 50.34(b)(6)(ii)

(a) Preliminary safety analysis report

- Preliminary plan f organization, personnel, training, and operations
- **Description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility – Appendix B to Part 50: Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants**

10 CFR 50.34(b)(6)(ii)

(a) Preliminary safety analysis report

- Identification of
  - Structures, systems, or components of the facility, if any, which require research and development
  - Research and development program which will be conducted to resolve any safety questions
  - Schedule of the research and development program to resolve at or before the latest date stated in the application for completion of construction of the facility

10 CFR 50.34(b)(6)(ii)

(a) Preliminary safety analysis report

- Technical qualifications of the applicant
- Plan to cope with emergencies
- Potential hazards relating to multi-unit sites
- Compliance with earthquake engineering criteria
- Information needed to meet other CFR sections

10 CFR 50.34(b)(6)(ii)

(b) Final safety analysis report

- Information on the facility, design bases and limits on its operation, and a safety analysis of the structures, systems, and components and of the facility as a whole including:
  1) Current information on environmental and meteorological monitoring programs
  2) Description and analysis of the structures, systems, and components of the facility

10 CFR 50.34(b)(6)(ii)

(b) Final safety analysis report

3) Kinds and quantities of radioactive materials
4) Final analysis and evaluation of the design and performance of structures, systems, and components
5) Description and evaluation of the results of the applicant's programs, including research and development, if any, to demonstrate that any safety questions identified at the construction permit stage have been resolved

10 CFR 50.34(b)(6)(ii)

(b) Final safety analysis report

6) The following information concerning facility operation:
   i. The applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements
   ii. Managerial and administrative controls to be used to assure safe operation. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the requirements for such controls for nuclear power plants and fuel reprocessing plants. The information on the controls to be used for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of appendix B will be satisfied
   iii. Plans for preoperational testing and initial operations

10 CFR 50.34(b)(6)(ii)

(b) Final safety analysis report

6) The following information concerning facility operation:
   iv. Plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components
   v. Plans for coping with emergencies, which shall include the items specified in appendix E
   vi. Proposed technical specifications prepared in accordance with the requirements of § 50.36
   vii. On or after February 5, 1979, for operating licenses for nuclear powerplants to be operated on multiunit sites shall include an evaluation of the potential hazards

10 CFR 50.34(b)(6)(ii)

(b) Final safety analysis report

7) Technical qualifications
8) Description and plans for implementation of an operator requalification program
9) Description of protection provided against pressurized thermal shock events
10) Conformance to General Design Criterion, shall comply with the earthquake engineering criteria
11) Description and safety assessment of the site and of the facility
12) Applying for an operating license

10 CFR Appendix A to Part 50

- Title: General Design Criteria for Nuclear Power Plants
- Requirements:
  - 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, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public

Title 10, Appendix A to Part 50

- Appendix A consists of 64 Criteria
- Criterion 1. – Quality Standards and Records
  - Structures, systems, and components (SSCs) 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 SSCs will satisfactorily perform their safety functions
  - Appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit

10 CFR Appendix B to Part 50

- Title: Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants
- Requirements:
  - Every applicant for a construction permit is required (§50.34) to include in its preliminary safety analysis report a description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility
  - Every applicant for an operating license is required to include, in its final safety analysis report, information pertaining to the managerial and administrative controls to be used to assure safe operation

10 CFR Appendix B to Part 50

- **Quality Assurance** comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service
- Quality assurance includes **quality control**, which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements

10 CFR Appendix B to Part 50

I. Organization
II. Quality Assurance Program
III. Design Control
IV. Procurement Document Control
V. Inspections, Procedures, and Drawings
VI. Document Control
VII. Control of Purchased Materials, Equipment, and Services
VIII. Identification and Control of Materials, Parts, and Components

10 CFR Appendix B to Part 50

IX. Control of Special Processes
X. Inspection
XI. Test Control
XII. Control of Measuring and Test Equipment
XIII. Handling, Storage and Shipping
XIV. Inspection, Test, and Operating Status
XV. Nonconforming Materials, Parts, and Components
XVI. Corrective Action
XVII. Quality Assurance Records
XVIII. Audits

10 CFR Appendix B to Part 50

I. Organization

- Establish and execute the quality assurance program
- Can be delegated or use an independent internal unit with access to management to attain authority and organizational freedom
- Shall have independence from cost and schedule when opposed to safety considerations

10 CFR Appendix B to Part 50

II. Quality Assurance Program

- Written policies, procedures, or instructions and shall be carried out throughout plant life
- Structures, systems, and components to be covered by the program and the major organizations participating in the program with the designated functions
- Control over activities affecting the quality to the extent consistent with their importance to safety accomplished under suitably controlled conditions (appropriate equipment, adequate cleanness, etc.)
- Special controls, processes, test equipment, tools, and skills to attain the required quality, and the need for verification of quality
- Indoctrination and training of personnel performing activities affecting quality
- Regular review the status and adequacy of the quality assurance program

10 CFR Appendix B to Part 50

III. Design Control

- Correctly translate applicable regulatory requirements and the design basis for those structures, systems, and components into specifications, drawings, procedures, and instructions
- Specify appropriate quality standards and control deviations
- 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

10 CFR Appendix B to Part 50

III. Design Control

- Identify and control of design interfaces and for coordination among participating design organizations
  - Establish procedures for the review, approval, release, distribution, and revision of documents involving design interfaces

10 CFR Appendix B to Part 50

III. Design Control

- Provide for verifying 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
  - Use individuals or groups other than those who performed the original design, but who may be from the same organization
  - Use test programs to verify the adequacy with suitable qualifications testing of a prototype unit under the most adverse design conditions
  - Apply to 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

10 CFR Appendix B to Part 50

III. Design Control

- 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

10 CFR Appendix B to Part 50

IV. Procurement Document Control

- Reference applicable regulatory requirements, design bases, and other requirements for procurement of material, equipment, and services
- Require contractors or subcontractors to provide a quality assurance program consistent with the pertinent provisions of this appendix

10 CFR Appendix B to Part 50

V. Instructions, Procedures, and Drawings

- Prescribe activities affecting quality by documented instructions, procedures, or drawings
- Include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished

10 CFR Appendix B to Part 50

VI. Document Control

- Control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality
- Review for adequacy and release approval
- Distribute to and use at the location where the prescribed activity is performed
- Review changes to documents

10 CFR Appendix B to Part 50

VII. Control of Purchased Material, Equipment, and Services

- Assure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents
  - Source evaluation and selection based on objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source, and examination of products upon delivery
  - Documentary evidence retained at the nuclear powerplant or fuel reprocessing plant site and shall be sufficient to identify the specific requirements, such as codes, standards, or specifications, met by the purchased material and equipment
  - The effectiveness of the control assessed at intervals consistent with the importance, complexity, and quantity of the product or services

10 CFR Appendix B to Part 50

VIII. Identification and Control of Materials, Parts, and Components

- Identify and control materials, parts, and components, including partially fabricated assemblies
  – Identification of the item heat number, part number, serial number, or other appropriate means, either on the item or on records traceable to the item, as required throughout fabrication, erection, installation, and use of the item
  – Designed to prevent the use of incorrect or defective material, parts, and components

10 CFR Appendix B to Part 50

IX. Control of Special Processes

- 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

10 CFR Appendix B to Part 50

X. Inspection

- Establish an inspection program of activities affecting quality
- Perform inspection by individuals other than those who performed the activity being inspected
- Perform for each work operation where necessary to assure quality
- If inspection is impossible or disadvantageous, use indirect control by monitoring processing methods, equipment, and personnel
- Use both inspection and process monitoring when control is inadequate without both
- If mandatory inspection hold points, which require witnessing or inspecting by the applicant's designated representative and beyond which work shall not proceed without the consent of its designated representative are required, the specific hold points shall be indicated in appropriate documents

10 CFR Appendix B to Part 50

XI. Test Control

- Assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed per requirements
- Use, as appropriate, proof tests prior to installation, preoperational tests, and operational tests during nuclear power plant or fuel reprocessing plant operation, of structures, systems, and components
- Define test procedures
- Document and evaluate to assure that test requirements have been satisfied

10 CFR Appendix B to Part 50

XII. Control of Measuring and Test Equipment

- 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

10 CFR Appendix B to Part 50

XIII. Handling, Storage and Shipping

- 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, specify and provide special protective environments, such as inert gas atmosphere, specific moisture content levels, and temperature levels

10 CFR Appendix B to Part 50

XIV. Inspection, Test, and Operating Status

- 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 or fuel reprocessing plant
- Identify items which have satisfactorily passed required inspections and tests
- Indicate operating status of structures, systems, and components of the nuclear power plant or fuel reprocessing plant, such as by tagging valves and switches, to prevent inadvertent operation

10 CFR Appendix B to Part 50

XV. Nonconforming Materials, Parts, or Components

- Control materials, parts, or components which do not conform to requirements in order to prevent their inadvertent use or installation
  - Procedures for identification, documentation, segregation, disposition, and notification to affected organizations
  - Review of nonconforming items for acceptance, rejection, repair or rework in accordance with documented procedures

10 CFR Appendix B to Part 50

XVI. Corrective Actions

- Assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected
  - Determine the cause of the condition and take corrective actions
  - Document and report to appropriate levels of management

10 CFR Appendix B to Part 50

XVII. Quality Assurance Records

- Maintain sufficient records to furnish evidence of activities affecting quality
  - Operating logs and the results of reviews, inspections, tests, audits, monitoring of work performance, and materials analyses
  - Closely-related data such as qualifications of personnel, procedures, and equipment
  - Inspection and test records to identify the inspector or data recorder, the type of observation, the results, the acceptability, and the action taken in connection with any deficiencies noted. Records shall be identifiable and retrievable
- Consistent with applicable regulatory requirements, Establish requirements concerning record retention, such as duration, location, and assigned responsibility

10 CFR Appendix B to Part 50

XVIII. Audit

- Carry out a comprehensive system of planned and periodic audits to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program
  - Written procedures or check lists by appropriately trained personnel not having direct responsibilities in the areas being audited
  - Documented and reviewed by management having responsibility in the area audited
  - Follow-up action, including re-audit of deficient areas, where indicated

2000. Federal Regulations, Guides, and Reports

CFR
- 10 CFR 50.34(b)(6)(ii)
- Appendix A and Appendix B to Part 50

RGs
- RG 1.28, RG 1.29, RG 1.94, RG 1.132
- RG 1.136, RG 1.138, RG 1.142, RG 1.149

NUREG
- NUREG 0800, NUREG/CR-6486
- NUREG/CR-6927

ASME
- ASME NQA-1 Quality assurance requirements

Regulatory Guides (RGs)

- RG 1.28: Quality Assurance Program Requirements (Design and Construction) [Revision 3, August 1985] – covered in Section 2000
- RG 1.94: Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants [Revision 1, April 1976] – covered in Appendix B

RG 1.28: Quality Assurance Program Requirements

- RG 1.28: Quality Assurance Program Requirements (Design and Construction) [Revision 3, August 1985]
- A Typical Outline of RGs
  - A. Introduction
  - B. Discussion Including Historical Perspectives
  - C. Regulatory Position
    1. Qualifications of Inspection and Test Personnel
    2. Quality Assurance Records
    3. Audits
  - D. Implementation
  - Regulatory Analysis

RG 1.28: Quality Assurance Program Requirements

- A. RG 1.28 describes a method acceptable to the NRC staff for complying with the provisions of Appendix B with regard to establishing and implementing the requisite quality assurance program for the design & construction of nuclear power plants
- B. Discussion Including Historical Perspectives
  - RG 1.28 (Safety Guide 28) issued in 1972 that endorsed ANSI N45.2-1971
  - Revised as ANSI/ASME N45.2-1977
  - Evolved to ANSI/ASME NQA-1-1979
  - Revised in ANSI/ASME NQA-1-1983

RG 1.28: Quality Assurance Program Requirements

- C. Regulatory Position
  1. Qualifications of Inspection and Test Personnel
  2. Quality Assurance Records (Table 1)
  3. Audits (internal and external)
- D. Implementation (rationale)
- Regulatory Analysis (bases for changes)

RG 1.28: Quality Assurance Program Requirements

- Table 1 (record types and retention times)
  - Design records (mostly lifetime, 3 yr, 10 yr)
  - Procurement records (mixed)
  - Manufacturing records (mostly lifetime)
  - Installation construction records (mixed)
    - Receiving, storage, civil, mechanical, welding, electrical and instrumentation and control, general
  - Preoperational and startup test records (mixed)

RG 1.94: Quality Assurance Requirements for …

- A. RG 1.94 describes a method acceptable to the NRC staff for complying with the Commission's regulation with regard to quality assurance requirements for installation, inspection, and testing of structural concrete and structural steel during the construction phase of nuclear power plants
- It refers to ANSI N45.2.5-1974 (ANSI/ASME NQA-1-1983)

2000. Federal Regulations, Guides, and Reports

CFR
- 10 CFR 50.34(b)(6)(ii)
- Appendix A and Appendix B to Part 50

RGs
- RG 1.28, RG 1.29, RG 1.94, RG 1.132
- RG 1.136, RG 1.138, RG 1.142, RG 1.149

NUREG
- NUREG 0800, NUREG/CR-6486
- NUREG/CR-6927

ASME
- ASME NQA-1 Quality assurance requirements

Nuclear Regulatory Commission (NRC) Reports (NUREGs) – CE/ST focus

- **NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems**
- NUREG/CR-6486, Assessment of Modular Construction for Safety-Related Structures at Advanced Nuclear Power Plants [1977] – covered in Sections 5000 and 6000
- NUREG/CR-6927, Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures - A Review of Pertinent Factors – covered in Section 5000

NUREG 0800 Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition

- Table of contents
  - Safety Review by NRC
    - Construction Permit (CP) Application
    - Operating License (OL) Application
    - Early Site Permit (ESP) Application
    - Design Certification (DC) Application
    - Combined License (COL) Application
    - Standard Design Approval (SDA) Application
    - Manufacturing License (ML) Application
    - Requests for Amendments

NUREG 0800 Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition

- Table of contents (cont.)
  - NUREG 0800 Sections
    1. Introduction and General Description of Plant
    2. *Site Characterization (partially covered in Section 3000)*
    3. **Design of Structures, Components, Equipment, and Systems**
    4. Reactor
    5. Reactor Coolant System and Connected Systems
    6. Engineering Safety Features
    7. Instrumentation and Control

NUREG 0800 Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition

- Table of contents (cont.)
  - NUREG 0800 Sections
    8. Electrical Power
    9. Auxiliary Systems
    10. Steam and Power Conversion System
    11. Radioactive Waste Management
    12. Radiation Protection
    13. Conduct of Operation

NUREG 0800 Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition

- Table of contents (cont.)
  - NUREG 0800 Sections
    14. Initial Test Program and ITAAC-Design Certification (ITAAC – Inspections, Tests, Analyses, and Acceptance Criteria)
    15. Accident Analysis
    16. Technical Specifications
    17. Quality Assurance
    18. Human Factor Engineering
    19. Severe Accidents

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Loads
  - Seismic classification (3.2.1)
  - Wind loading (3.3.1)
  - Tornado loads (3.3.2)
  - Flood protection (3.4.1 & 3.4.2)
  - Missiles (3.5.1.1 to 3.5.1.6, 3.5.2)
    - Barrier design (3.5.3)
  - Design against pipe rupture (3.6.1 to 3.6.3)

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Loads (cont.)
  - Seismic (3.7.1 to 3.7.4)
- Containment, other structures and foundation (3.8.1 to 3.8.5)
- Special topics for mechanical components (3.9.1 to 3.9.8)
- Seismic & Dynamic qualification of Mechanical & electrical equipment (3.10)

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Environmental qualification of Mechanical & electrical equipment (3.11)
- ASME code class 1, 2, and 3 piping systems (3.12)
- Threaded fasteners - ASME code class 1, 2, and 3 (3.13)
- System quality
  - System quality group classification (3.2.2)

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Seismic classification (RG 1.29)
  - Applicable to Structures, Systems and Components (SSCs)
  - Two earthquake intensities
    - Safe-shutdown earthquake ground motion (SSE)
    - Safe-operation earthquake ground motion (SOE)

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Seismic classification (RG 1.29) (cont.)
  - Category I SSCs are SSCs designed to remain functional if the SSE occurs
  - RG 1.29 lists all activities affecting the safety-related functions (mechanical, electrical, coolant, pressure boundaries, etc.)
  - Covering the seismic-nonseismic Category I SSCs interface and extend inside the nonseismic Category I SSCs

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Wind loading (3.3.1)

  Use ASCE/SEI 7-05 (See also Section 4000)

For a design wind speed, V, the velocity pressure, q_z, evaluated at height, z, is given by:

$q_z = 0.00256\ K_z K_{dt} K_d V^2 I$ (lb/ft^2)

K_z = velocity pressure exposure coefficient evaluated at height, z, as defined in ASCE/SEI 7-05, Table 6-3, but not less than 0.87
K_{dt} = topographic factor equal to 1.0
K_d = wind directionality factor equal to 1.0
V = design wind speed in miles per hour (mi/h) as stated in SRP Section 2.3.1
I = importance factor equal to 1.15

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Tornado loads (3.3.2)
  - For wind effects use ASCE/SEI 7-05 (3.3.1)
  - Load combinations

$W_t = W_p$ Eq. 1
$W_t = W_w + 0.5\ W_p + W_m$ Eq. 2

where:
W_t = total tornado load
W_w = load from tornado wind effect
W_p = load from tornado atmospheric pressure change effect
W_m = load from tornado missile impact effect

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Flood protection (3.4.1 & 3.4.2)
  - Protection of SSCs that are necessary for safe shutdown or uncontrolled release of significant radioactivity
  - Types of flooding
    - Internal due to pipe breaks, tank failure, etc.
    - External by plant systems, such as exterior tanks
    - External to the plant, such as probable maximum flood, tsunami, etc.

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- Flood protection (3.4.1 & 3.4.2) (cont.)
  - Structural assessment of seismic Category I SSCs under the following flood and groundwater related loads:
    - Currents
    - Waves
    - Hydrodynamic effects
  - Definition of flood design bases using frequency and probabilistic analysis, and site characteristics

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Missiles (3.5.1.1 to 3.5.1.6 , 3.5.2)
  - Internally generated missiles
    - Outside containment
    - Inside containment
  - Turbine missiles
  - Missiles generated by tornados and extreme winds
  - Site proximity missiles (except aircraft)
  - Aircraft missiles

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Missiles (3.5.1.1 to 3.5.1.6 , 3.5.2) (cont.)
  - Internally generated missiles: Sources
    - Component overspeed failures
    - Originating from high-energy fluid systems failures
    - As a consequence of gravitational effects
    - Examples: valve bonnets, hardware-retaining bolts, relief valve parts, turbine blades, instrument wells, etc.
  - SSCs necessary for the safe shutdown of the reactor facility and the failure of SSCs that could cause a significant release of radioactivity

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Missiles (3.5.1.1 to 3.5.1.6, 3.5.2) (cont.)
  - Probabilistic analysis (cont.)
    - Acceptable risk (statistically significant):
      - Probability of missile occurrence (P_1) less than 10^{-7} per year, the missile
      - Product of P_1 and the probability of impact (P_2) on a significant target less than 10^{-7} per year
      - Product of P_1, P_2, and the probability of significant damage (P_3) less than 10^{-7} per year
    - Otherwise, provide by one or more of six missile mitigation methods

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Missiles (3.5.1.1 to 3.5.1.6, 3.5.2) (cont.)
  - Missile mitigation methods (3.5.2; cont.)
    - Locating SSCs in a missile-proof structure
    - Separating redundant SSCs from the missile path or range
    - Providing local shields and barriers
    - Designing to withstand impact of the most damaging missile
    - providing design features to prevent the generation of missiles
    - Orienting missile sources to prevent strikes

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Missiles (3.5.1.1 to 3.5.1.6 , 3.5.2) (cont.)
  - Turbine missiles (same limits on $P_1P_2P_3$)
  - P_1 (table), Product P_2P_3:
    - 10^{-3} to 10^{-4} per year per plant if favorable oriented (10^{-2} to 10^{-3} otherwise)
    - Cases A, B, C & D for methods to compute probabilities

TABLE 3.5.1.3-1
PROBABILITY OF TURBINE FAILURE RESULTING IN THE EJECTION OF TURBINE ROTOR (OR INTERNAL STRUCTURE) FRAGMENTS THROUGH THE TURBINE CASING (P_1) AND RECOMMENDED LICENSEE ACTIONS

Case	PROBABILITY PER YEAR FOR A FAVORABLY ORIENTED TURBINE	PROBABILITY PER YEAR FOR AN UNFAVORABLY ORIENTED TURBINE	RECOMMENDED LICENSEE ACTION
A	$P_1 < 10^{-4}$	$P_1 < 10^{-5}$	This condition represents the general, minimum reliability requirement for loading the turbine and bringing the system on line.
B	$10^{-4} < P_1 < 10^{-3}$	$10^{-5} < P_1 < 10^{-4}$	If this condition is reached during operation, the turbine may be kept in service until the next scheduled outage, at which time the licensee must take action to reduce P_1 to meet the appropriate Case A criterion before returning the turbine to service.
C	$10^{-3} < P_1 < 10^{-2}$	$10^{-4} < P_1 < 10^{-3}$	If this condition is reached during operation, the turbine must be isolated from the steam supply within 60 days, at which time the licensee must take action to reduce P_1 to meet the appropriate Case A criterion before returning the turbine to service.
D	$10^{-2} < P_1$	$10^{-3} < P_1$	If this condition is reached during operation, the turbine must be isolated from the steam supply within 6 days, at which time the licensee must take action to reduce P_1 to meet the appropriate Case A criterion before returning the turbine to service.

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Missiles (3.5.1.1 to 3.5.1.6 , 3.5.2) (cont.)
  - Air-driven missiles
    - High-speed wind, tornado, hurricane, etc.
    - Design for strike probability of 10^{-7} per year
    - Types: (2) massive high-kinetic-energy missile that deforms on impact, (2) a rigid missile to test penetration resistance, and (3) a small rigid missile of a size sufficient to just pass through any openings in protective barriers.
    - Acceptable missiles and their associated wind speeds are identified in Table 2 of RG 1.76 (see next slides)

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Missiles (3.5.1.1 to 3.5.1.6 , 3.5.2) (cont.)
  - Air-driven missiles (cont.)

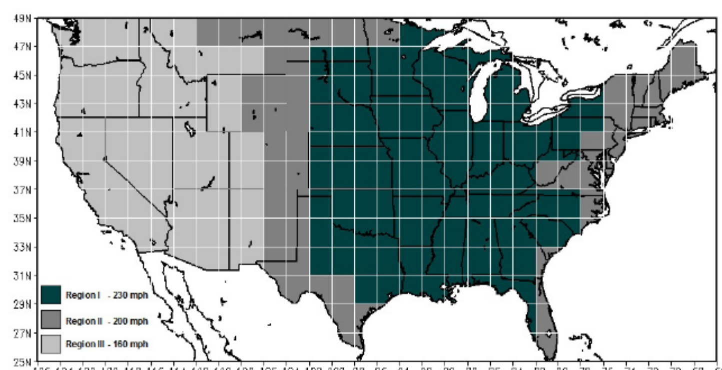

Figure 1. Tornado intensity regions for the contiguous United States for exceedance probabilities of 10^{-7} per year

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

Table 2. Design-Basis Tornado Missile Spectrum and Maximum Horizontal Speeds

Missile Type		Schedule 40 Pipe	Automobile	Solid Steel Sphere
Dimensions		0.168 m dia × 4.58 m long (6.625 in. dia × 15 ft long)	Region I and II: 5 m × 2 m × 1.3 m (16.4 ft x 6.6 ft x 4.3 ft); Region III: 4.5 m x 1.7 m x 1.5 m (14.9 ft x 5.6 ft x 4.9 ft)	2.54 cm dia (1 in. dia)
Mass		130 kg (287 lb)	Region I and II: 1810 kg (4000 lb); Region III: 1178 kg (2595 lb)	0.0669 kg (0.147 lb)
$C_D A/m$		0.0043 m^2/kg (0.0212 ft^2/lb)	Region I and II: 0.0070 m^2/kg (0.0343 ft^2/lb); Region III: 0.0095 m^2/kg (0.0464 ft^2/lb)	0.0034 m^2/kg (0.0166 ft^2/lb)
V_{Mh}^{max}	Region I	41 m/s (135 ft/s)	41 m/s (135 ft/s)	8 m/s (26 ft/s)
	Region II	34 m/s (112 ft/s)	34 m/s (112 ft/s)	7 m/s (23 ft/s)
	Region III	24 m/s (79 ft/s)	24 m/s (79 ft/s)	6 m/s (20 ft/s)

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Missiles (3.5.1.1 to 3.5.1.6 , 3.5.2) (cont.)
  - Site proximity missiles (except aircraft)
    - Offsite activities with the potential for missile production (e.g., explosion)
    - An order of magnitude of 10^{-6} per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower
    - Computation of total probability per year (next slide)

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

The total probability of the missiles striking a vulnerable critical area of the plant is estimated. The total probability per year (P_T) may be estimated using the following expression:

$$P_T = P_E \times P_{MR} \times P_{SC} \times P_p \times N$$

where:

P_E = probability per year of design-basis event obtained from the review performed under SRP Section 2.2.3

P_{MR} = probability of missiles reaching the plant

P_{SC} = probability of missiles striking a vulnerable critical area of the plant

P_p = probability of missiles exceeding the energies required to penetrate to vital areas (e.g., based on wall thickness provided for tornado missiles) or producing secondary missiles that could damage vital equipment

N = number of missiles generated by the design-basis event

P_p may be assumed to be equal to 1 as a first step in the analysis. If P_T thus calculated is greater than an order of magnitude of 10^{-7} per year, then site proximity missile impact effects should be estimated, on request, by the organizational unit responsible for reviewing specific SSC. The request should be accompanied by a specified missile description, including missile size, shape, weight, energy, material properties, and trajectory.

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Missiles (3.5.1.1 to 3.5.1.6 , 3.5.2) (cont.)
  - Aircraft hazards (need not to be a design basis)
    - Considerations include

      A. Airports (distance and traffic limits)

      B. Federal airways (distance limit)

      C. Holding and approach patterns (volume limit)

      D. Military airports, training routes, and training areas (distance and traffic limits)
    - Sum of probabilities from these sources not to exceed an order magnitude of 10^{-7} per year

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Barrier design procedures (3.5.3)
  - Concrete (National Defense Research Council formula)

Minimum Acceptable Barrier Thickness Requirements
For Local Damage Prediction Against Tornado Generated Missiles

Regions*	Concrete Strength MPa (psi)	Wall Thickness cm (inches)	Roof Thickness cm (inches)
Region I	20.7 (3000) 27.6 (4000) 34.5 (5000)	46.2 (18.2) 42.9 (16.9) 40.6 (16.0)	33.5 (13.2) 31.2 (12.3) 29.7 (11.7)
Region II	20.7 (3000) 27.6 (4000) 34.5 (5000)	39.1 (15.4) 36.3 (14.3) 34.5 (13.6)	28.4 (11.2) 26.4 (10.4) 25.1 (9.9)
Region III	20.7 (3000) 27.6 (4000) 34.5 (5000)	30.2 (11.9) 28.2 (11.1) 26.7 (10.5)	22.1 (8.7) 20.6 (8.1) 19.6 (7.7)

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Barrier design procedures (3.5.3)
  - Steel:
    - Stanford Research Institute (SRI) "U.S. Reactor Containment Technology" (ORNL/NSIC-5, Vol.1, Chapter 6, Oak Ridge National Laboratory, 1965) by W.B. Cottrell and A.W. Savolainen
    - Ballistic Research Laboratory formula described in, "Reactor Safeguards," by C. R. Russell, published by MacMillan, New York, 1962

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- Barrier design procedures (3.5.3)
  - Composite sections:
    - "Ballistic Perforation Dynamics," Journal of Applied Mechanics, Transactions of the ASME, Vol. 30, Series E, No. 3, September 1963 by R. F. Recht and T. W. Ipson

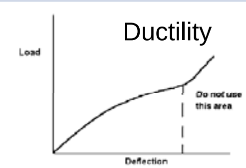
NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Design against pipe rupture (3.6.1 to 3.6.3)

Ensure that the environmental effects of failures do not cause loss of safety-related functions

1. Protection against piping failure
2. Rupture location & dynamic effects
3. Leak-before break analysis

Ductility



Fatigue cracking
Fracture mechanics
Leak rate computation
Applied loads on piping
Material/weld properties
Flaw sizes

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

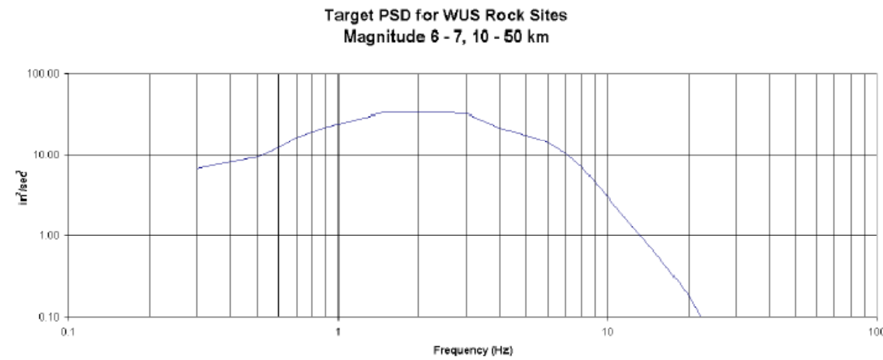
- Seismic (3.7.1 to 3.7.4)

Earthquake Design parameters	Seismicity & geologic conditions	1. Operating basis earthquake (OBE) 2. Safe shutdown earthquake (SSE)	Three spatial components of ground motion (two horizontal and one vertical)
Structures, systems & components (SSCs)	Ground motion response spectra (GMRS)	Site-specific GMRS	Design response spectra for seismic category I SSCs at top of first surface of soil materials
Design time histories (if design spectra unavailable)	Use 3-compent cancelation time history if available; otherwise generate using design spectra	1. Percentages of critical damping 2. Support/foundation media	Seismic analysis of category I SSCs

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Seismic (3.7.1 to 3.7.4)

Power Spectral Density (PSD) in in^2/sec^3 for Western US (WUS)



NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Seismic (3.7.1 to 3.7.4)

Same Level:
1. Ground motion response spectra (GMRS)
2. certified seismic design response spectra (CSDRS)

Develop foundation level spectra consistent with CSDRS for each Category I SSC

Check if minimum spectrum is met

If CSDRS > GMRS and site & design condition are met, CSDRS are adequate; otherwise evaluate using an approved method*

Not Same Level:
1. Ground motion response spectra (GMRS)
2. certified seismic design response spectra (CSDRS)

Develop foundation level spectra and foundation input response spectra (FIRS) consistent with CSDRS for each Category I SSC

Check if minimum spectrum is met

If CSDRS > FIRS and site & design condition are met, CSDRS are adequate; otherwise evaluate using an approved method*

* Approved method accounts for soil-structure interaction (SSI), torsion, rocking, in-structure response, redesign, re-analysis as needed until limits are met. Cases without a certified design require developing GMRS, smoothed response spectra, foundation-level response spectra, analysis, redesign & reanalysis until meeting limits.

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Seismic (3.7.1 to 3.7.4)
  - Analysis methods:
    - Response spectrum method
    - Time history analysis method
    - Equivalent static load analysis method
  - Natural frequencies and response
  - Material, damping, stiffness and mass properties
  - Hydrodynamic effects

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Seismic (3.7.1 to 3.7.4)
  - Soil-structure interaction (SSI):
    - The random nature of the soil and rock configuration and material characteristics
    - Uncertainty in soil constitutive modeling (soil stiffness, damping, etc.)
    - Nonlinear soil behavior
    - Coupling between the structures and soil
    - Lack of uniformity in the soil profile, which is usually assumed to be uniformly layered in all horizontal directions

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Seismic (3.7.1 to 3.7.4)
  - Soil-structure interaction (SSI) (cont.):
    - Effects of the flexibility of soil/rock.
    - Effects of the flexibility of basemat
    - The effect of pore water on structural responses, including the effects of variability of ground-water level with time
    - Effects of partial separation or loss of contact between the structure (embedded portion of the structure and foundation mat) and the soil during the earthquake
    - Strain-dependent soil properties

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Seismic (3.7.1 to 3.7.4)
  - Interaction of non-category I structures with Category I SSCs (decoupling criteria based on mass ratios and fundamental frequency ratios)
  - Effects of parameter variation on floor responses
  - Use of equivalent vertical static factors
  - Methods used to account for torsional effects
  - Procedures for damping
  - Overturning moments and sliding forces

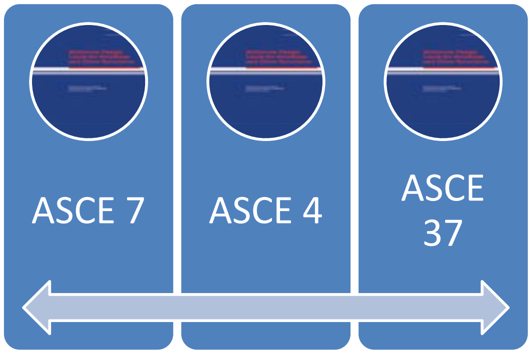
NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Seismic (3.7.1 to 3.7.4)
  - Seismic analysis to cover Category 1 subsystems using similar approaches deployed for SSCs:
    - Platforms
    - Support frame structures
    - Yard structures
    - Buried piping, tunnels and conduits
    - Concrete dams
    - Atmospheric tanks

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Seismic (3.7.1 to 3.7.4)
  - Seismic analysis to cover instrumentation (per RG 1.12 and RG 1.66)

Covered in Section 4000



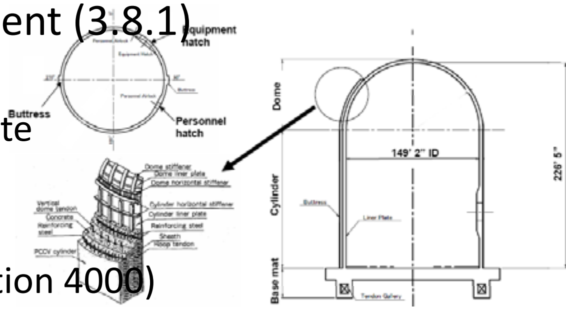
NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

Containment, other structures and foundations

- Containment
  - Concrete 3.8.1
  - Steel 3.8.2
  - Internal structures 3.8.3
- Other structures 3.8.4
- Foundations 3.8.5

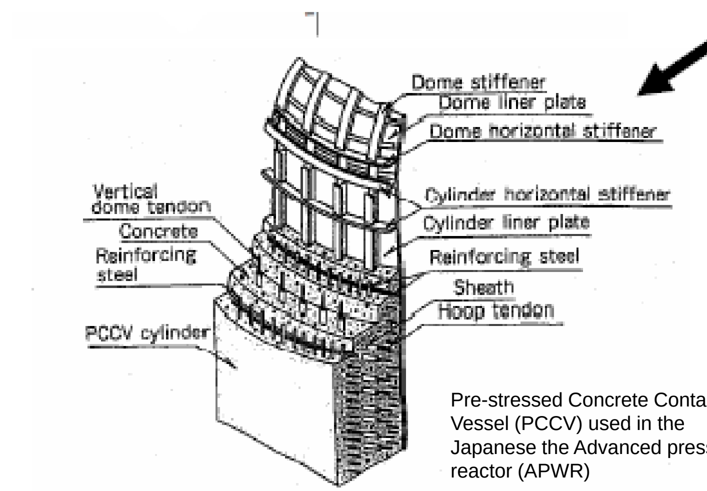
NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Concrete containment (3.8.1)
  - Reinforced and prestressed concrete (Section 5000)
  - Loads and load combinations (Section 4000)
  - Analysis methods (Section 5000)
  - Failure modes, ultimate strength, working stresses (Section 5000)
  - Material properties and strength (Section 5000)



NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Concrete containment (3.8.1)



Pre-stressed Concrete Container Vessel (PCCV) used in the Japanese the Advanced pressurized water reactor (APWR)

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Steel containment (3.8.2)
  - Steel and composite sections (Section 6000)
  - Loads and load combinations (Section 4000 and next slide)
  - Analysis methods (Section 6000)
  - Failure modes, ultimate strength, working stresses (Section 6000)
  - Material properties and strength (Section 6000)

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Steel containment (3.8.2) load combinations:
  - Testing conditions
  - Design conditions
  - Service condition A: Operating condition, OBEarthquake (elastic)
  - Service condition B: Operating condition, LOCA, OBEarthquake (towards yield)
  - Service condition C: Operating condition, LOCA, SSEarthquake (inelastic)
  - Service condition D: LOCA, SSEarthquake, local dynamic effects (inelastic)

LOCA = loss of coolant accident)

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Steel containment (3.8.2) load combinations (cont.):
  - Construction loads
  - External environmental loads

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Internal structures (3.8.3)
  - PWR dry containment
    - Concrete supports for reactor
    - Concrete support for steam generator
    - Primary shield wall and reactor cavity
    - Secondary shield walls
    - Other interior structures

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

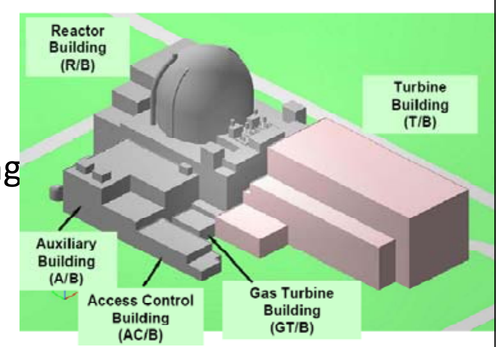
- Internal structures (3.8.3)
  - PWR ice-condensed containment
    - The divider barrier
    - Ice condenser

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Internal structures (3.8.3)
  - BWR containment
    - Drywell
    - Weir wall
    - Refueling pool and operating floor
    - Concrete support for reactor and recirculation pump
    - Reactor pedestal
    - Reactor shield wall
    - Other interior structures

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Other Category I and safety-related structures (3.8.4)
  - Containment enclosure building
  - Auxiliary building
  - Fuel storage building
  - Control building
  - Diesel generator building
  - Other structures
  - Masonry walls



NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Foundations (3.8.5)
  - Containment structure foundation
  - Containment enclosure building foundation
  - Auxiliary building foundation
  - Other Category I foundations

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Foundations (3.8.5): analysis
  - Soil-structure interaction
  - Concrete and steel (Sections 5000 and 6000)
  - Failure modes to include overturning, sliding and flotation

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Special topics for mechanical components (3.9.1 to 3.9.8)
  - Dynamic testing (vibration stresses, transient forces, etc.)
  - ASME code class 1, 2 and 3
  - Control rod driver systems
  - Reactor pressure vessel internals
  - Pumps, valves and dynamic restraints
  - Risk-informed in-service testing

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Seismic & dynamic qualification of Mechanical & electrical equipment (3.10)
  - Structural Integrity and Functionality of Mechanical and Electrical Equipment:
    - In the event of a safe-shutdown earthquake (SSE), after a number of postulated occurrences of the operating-basis earthquake (OBE)
    - In combination with other relevant dynamic and static loads

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Environmental qualification of Mechanical & electrical equipment (3.11)
  - All items of equipment that are important to safety (mechanical, electrical, and instrumentation and control (I&C), including digital I&C) are capable of performing their design safety functions under all normal environmental conditions, anticipated operational occurrences, and accident and post-accident environmental conditions

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- ASME code class 1, 2, and 3 piping systems (3.12)
  - The design of piping systems should include seismic Category I, Category II, and non-safety systems

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- Threaded fasteners - ASME code class 1, 2, and 3 (3.13)
  - Selection of materials, design, inspection and testing of its threaded fasteners (i.e., threaded bolts, studs, etc.) prior to initial service and during service
  - Scope limited to the review of threaded fasteners in ASME Boiler and Pressure Vessel Code (Code) Class 1, 2 or 3 systems
  - Covered in Section 6000

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

- System quality group classification (3.2.2)
  - Fluid systems (per tables on next slides) according to Quality Groups A, B, C and D for
    - Pressurized water reactor (PWR) plants
    - Boiling water reactor (BWR) plants
  - Risk-informed categorization process, such as probabilistic risk analysis (PRA) methods
  - Risk is the potential loss due to an adverse event measured using loss-exceedence probability curves (see next slide)

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems

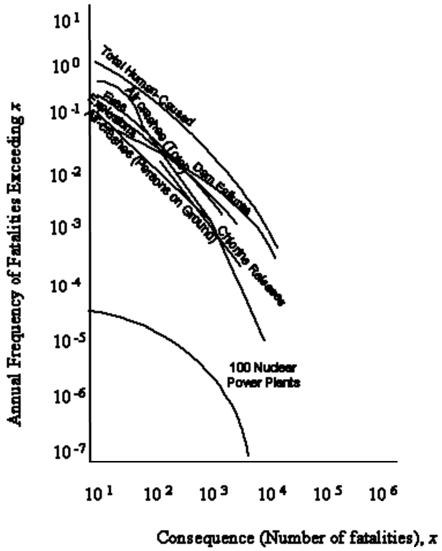
FLUID SYSTEMS IMPORTANT TO SAFETY FOR BWR PLANTS

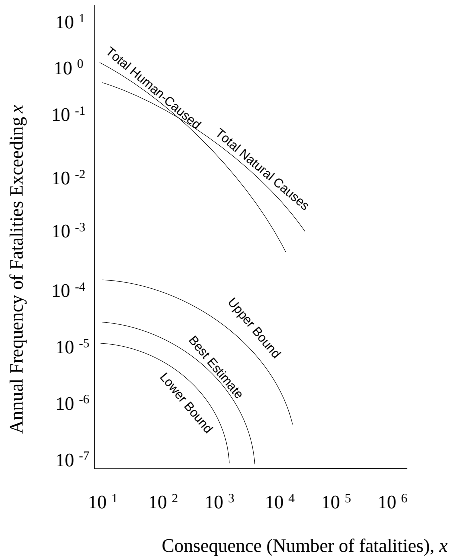
Combustible Gas Control System[6]
Compressed Air System[1,2,6]
Condensate Storage System[2]
Control Rod Drive Hydraulic System[2]
Containment Cooling System
Containment Isolation System[6]
Emergency Core Cooling Systems
Emergency Diesel Engine Fuel Oil Storage and Transfer System[6]
Emergency Diesel Engine Cooling Water System
Emergency Diesel Engine Starting System
Emergency Diesel Engine Lubrication System
Emergency Diesel Engine Combustion Air Intake and Exhaust System
Equipment and Floor Drainage System[2,6]
Feedwater System (up to outermost containment isolation valve or shutoff valve, as applicable)
Fuel Pool Cooling and Cleanup System[2,4]
Main Steam System (up to but not including the turbine)
Main Steam Isolation Valve Leakage Control System[6]
Nuclear Boiler System
Process and Post-Accident Sampling Systems[3]
Reactor Auxiliary Cooling Water Systems (e.g., Essential Cooling Water and Chilled Water Systems)[2]
Reactor Core Isolation Cooling System
Reactor Recirculation System
Reactor Water Cleanup System
Relief Valve Discharge Piping[5]
Residual Heat Removal (RHR) System
RHR Service Water System
Standby Gas Treatment System[6]
Standby Liquid Control System
Station Service Water System[2]
Ultimate Heat Sink and Supporting Systems[6]
Ventilation Systems for Areas such as Control Room and Engineered Safety Features Rooms[6]

FLUID SYSTEMS IMPORTANT TO SAFETY FOR PWR PLANTS

Auxiliary Feedwater System
Boron Thermal Regeneration System[1,2]
Boron Recycle System[1,2]
Chemical and Volume Control System
Combustible Gas Control System[1,6]
Compressed Air System[1,2,6]
Condensate Storage System[1]
Containment Cooling System
Containment Isolation System[6]
Containment Purge System
Containment Spray System
Emergency Core Cooling System
Emergency Diesel Engine Fuel Oil Storage and Transfer System[6]
Emergency Diesel Engine Cooling Water System
Emergency Diesel Engine Starting System
Emergency Diesel Engine Lubrication System
Emergency Diesel Engine Combustion Air Intake and Exhaust System
Equipment and Floor Drainage System[2,6]
Feedwater System[3]
Main Steam System[3]
Pressurizer Power-Operated Relief Valves (PORVs) (including associated components and block valves)[6]
Process and Post-Accident Sampling Systems[3]
Reactor Auxiliary Cooling Water Systems (e.g., Component Cooling Water and Essential Chilled WaterSystems)[2]
Reactor Coolant System
Refueling Water Storage System[2]
Residual Heat Removal System
Spent Fuel Pool Cooling and Cleanup System[2,4]
Station Service Water System[2]
Steam Generator Blowdown System[3]
Ultimate Heat Sink and Supporting Systems[6]
Ventilation Systems for Areas such as Control Room and Engineered Safety Features Rooms

NUREG 0800 Section 3.0: Design of Structures, Components, Equipment, and Systems





Risk Analysis in Engineering and Economics, Ayyub, 2003

2000. Federal Regulations, Guides, and Reports

- CFR
  - 10 CFR 50.34(b)(6)(ii)
  - Appendix A and Appendix B to Part 50
- RGs
  - RG 1.28, RG 1.29, RG 1.94, RG 1.132
  - RG 1.136, RG 1.138, RG 1.142, RG 1.149
- NUREG
  - NUREG 0800, NUREG/CR-6486
  - NUREG/CR-6927
- ASME
  - ASME NQA-1 Quality assurance requirements

Title 10, Appendix B to Part 50

ASME NQA-1- 1994

- **Quality Assurance** comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service
- Quality assurance includes **quality control**, which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements

Title 10, Appendix A to Part 50

ASME NQA-1- 1994

- Criterion 1. – Quality Standards and Records
  - Structures, systems, and components (SSCs) 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 SSCs will satisfactorily perform their safety functions
  - Appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit

ASME NQA-1 Quality Assurance Requirements

- The ASME NQA Standard sets forth requirements and nonmandatory guidance for the establishment and execution of quality assurance programs for nuclear facility applications

ASME NQA-1 Quality Assurance Requirements

- The ASME NQA Standard consists of the following parts:
  - Part I (formerly NQA-1): quality assurance program requirements for the siting, design, construction, operation, and decommissioning of nuclear facilities (Basic Requirements & Supplementary Requirements)
  - Part II (formerly NQA-2): quality assurance program requirements for the planning and execution of identified tasks during the fabrication, construction, modification, repair, maintenance, and testing of SSCs for nuclear facilities
  - Part III (from NQA-1 and NQA-2): nonmandatory appendices
  - Part IV: reserved for future NQA position papers and application matrices

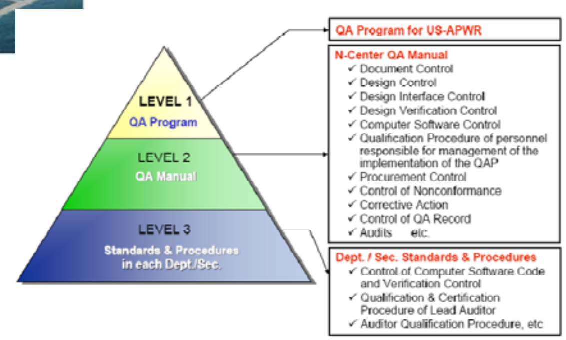
ASME NQA-1 Quality Assurance Requirements

- Incorporated into NQA-1-1983
  - N45.2.1 Fluid Systems
  - N45.2.2 Packaging & Handling
  - N45.2.3 Housekeeping
  - N45.2.5 Civil / Structural Inspection
  - N45.2.8 Mechanical Equipment Inspection
  - N45.2.15 Hoisting & Rigging
  - N45.2.20 Subsurface Investigations

ASME NQA-1 Quality Assurance Requirements

Example: Tsuruga Japan with 2007 design to meet US requirements (US-APWR) by Mitsubishi

Advanced pressurized water reactor (APWR)



ASME NQA-1 Quality Assurance Requirements

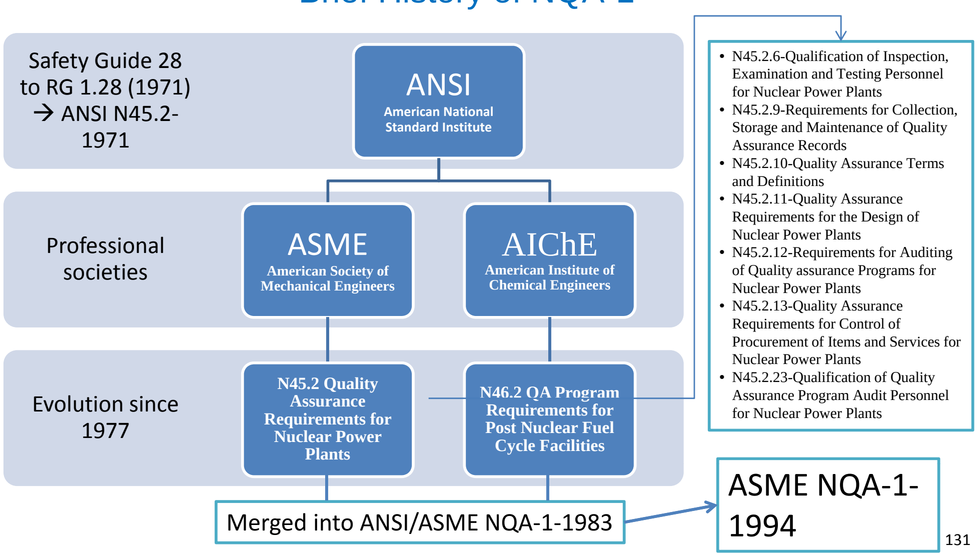
- ASME NQA-1- 1994 Quality assurance requirements for nuclear facility applications
  - Atomic Energy Commission (AEC) → DOE Standards and Directives
  - AEC → NRC Regulations, Regulatory Guides, Standard Review Plans (SRPs)
  - ANSI → ASME N45.2 → NQA-1/2/3 Standards
  - Other related NQA Standard (ANS, IEEE, etc.)

ASME NQA-1 Quality Assurance Requirements

- Selected precursor documents leading to QA requirements of NQA-1
  - Mil - Q-9858 Quality System
  - Mil - I-45208 Inspection System
  - Mil - C-45662 Calibration System
  - NASA NPC 200-2
  - BUSHIPS BSI 4410.17
  - 10 CFR 50, Appendix B
  - 10 CFR 50, Appendix A, Criterion 1

ASME NQA-1 Quality Assurance Requirements

Brief History of NQA-1



ASME NQA-1 Quality Assurance Requirements

NQA Chronology

1950s	1960s
AEC SFO QC-1 **MIL-Q-1958**	**MIL-Q-1958A** **10 CFR50 Appendix A** **AEC → DOE RDT F2-2T**

ASME NQA-1 Quality Assurance Requirements

NQA Chronology

1970s	1980s
10 CFR50 Appendix A 10 CFR50 Appendix B ANSI N45-2-1971 AEC Safety Guide 28 ANSI N18.7(ANS 3.2) ANSI/ASME N45.2-1977 ASME N45.2.9 – 2.23 NRC Rainbow series ANSI N46.2-1978 ASME NQA-1-1979 (N45.2.6, 2.9, 2.10, 2.11, 2.12, 2.13, 2.23)	DOE O 5700.6, 6A, 6B ASME NQA-1-1983 (N45.2.1, 2.2, 2.3, 2.5, 2.8, 2.15, 2.20) REG Guide 1.28, Rev 3 ASME NQA-2-1983 ASME NQA-1-1986 ASME NQA-1-1989 ASME NQA-3-1989

ASME NQA-1 Quality Assurance Requirements

NQA Chronology

1990s	2000s
NRC Standard Review Plan (NQA-1 & NQA-2) DOE O 5700.6C DOE 10CFR Part 830 (sub-part 830-122) ASME NQA-1-1994 ASME NQA-1-1997 DOE O 414.1A	DOE O 414.1B DOE O 414.1C DOE 10CFR Part 830 (sub-part 830-122) ASME NQA-1-2000 ASME NQA-1-2004

ASME NQA-1 Quality Assurance Requirements

- Developmental milestones
  - Performance based versus compliance
  - Graded approach
  - Risk informed

ASME NQA-1 Quality Assurance Requirements

Example

Quality related regulations, rules, orders, codes, and standards

Civilian Nuclear	Department of Energy	Environmental	Commercial or International
10 CFR 50, Appendix B (NRC Programmatic) 10CFR50, Part 21 (Reporting Defects) 10 CFR50.55 (Codes & Standards) R.G. 1.28 (NRC Adoption of ANSI/ASME Standards) ASME Section III (BPVC – Nuclear) ASME NQA-1 (Quality Program)	10CFR 830, Subpart A (Nuclear – Programmatic) 10CFR820, Appendix A (Price Anderson) DOE O 414.A (Quality Assurance) DOE G 414.1-2 (QA Management System) DOE G 414.I-1 (Independent & Management Assessment) DOE P 450.4 (Safety Management System)	EPA QA/R-2 (EPA Requirements for Quality Management Plans) EPA QA/R- 5 (EPA Requirements for QA Project Plans) ANSI/ASQC E4-1994 (Specifications and Guidelines for Quality Systems for Environmental Data Collection and Environmental Technology Programs) RW-0333P OCRWM-YMP (Waste Repository QA Program)	ISO-9001-2000 (QA Program) ISO-14001 (Environmental Management System) ASME VIII (Non-nuclear QA/QC) Note Adoption / use of an ISO program does not require Certification.

ASME NQA-1 Quality Assurance Requirements

- Typical users of NQA-1:
  - Utilities / power generators
  - ASME Section III users
  - Department of Energy
  - National laboratories, repositories, and fuel fabrication facilities
  - Trans-Alaska pipeline
  - Offshore nuclear power projects
  - Others

ASME NQA-1 Quality Assurance Requirements

18 Basic Requirements (BR):

1. Organization
2. Quality Assurance Program
3. Design Control
4. Procurement Document Control
5. Instructions, Procedures, and Drawings
6. Document Control
7. Control of Purchased Items and Services
8. Identification and Control of Items

ASME NQA-1 Quality Assurance Requirements

18 Requirements (cont.)

9. Control of Processes
10. Inspection
11. Test Control
13. Handling, Storage, and Shipping
14. Inspection, Test, and Operating Status
15. Control of Nonconforming Items
16. Corrective Action
17. Quality Assurance Records
18. Audits

ASME NQA-1 Quality Assurance Requirements

Basic Requirement (BR) 1 – Organization: • Identify quality problems • Provide solutions • Verify implementation • Control process including disposition of nonperformance	Senior management: – Defines policies and objectives – Establishes and communicates expectations for quality and continual improvement – Identifies and allocates resources to achieve expectation – Specifies roles, responsibilities and authorities – Ensures NQA principles are understood, accepted and followed Quality of work is: – Achieved and maintained by performers – Verified by those not directly responsible for performing the work

ASME NQA-1 Quality Assurance Requirements

NQA-1 BR 2 QA Program – 2S-1. Qualification of inspection and testing personnel – 2S-2. Qualification of nondestructive testing personnel – 2S-3. Qualification of quality assurance program audit personnel – 2S-4. Personnel indoctrination and training	Management: – Ensures proper development and implementation of the organization's Quality Assurance Program (QAP) – Ensures people are competent to perform their assigned quality-affecting work – Provides training to achieve and maintain worker proficiency and qualifications – Assesses QAP and management systems to ensure compliancy, adequacy effectiveness and efficiency – Seeks and uses relevant experience

ASME NQA-1 Quality Assurance Requirements

BR 3 Design Control	– Items are designed and changes controlled using sound engineering practices, analyses and configuration management – Designs are analyzed and independently verified
BR 4 & 7 Procurement	– Items and services are purchased, accepted and controlled to specified requirements
BR 5 Instructions, procedures and drawings	– Work processes are planned and controlled – Activities are performed in accordance with prescribed documentation – Management ensures the right people have the right information at the right time

ASME NQA-1 Quality Assurance Requirements

BR 6 & 17 Documents and records	– Quality documents and records are developed, identified and controlled in accordance with specified requirements
BR 8 & 13 Item identification and control	– Items are identified and controlled during shipping, handling, installation or use to assure their quality and prevent damage, loss or deterioration
BR 9 Process control	– Special processes for achieving quality are performed under controlled conditions to specified requirements – Use of correct materials, tools and processes and control changes is assured

ASME NQA-1 Quality Assurance Requirements

BR 10 & 11 Inspection and test control	– Items are inspected and tested to verify conformance to specified requirements including computer program testing
BR 12 Measuring and test equipment control	– Measuring & Testing Equipment used for activities affecting quality are controlled to specified accuracy requirements
BR 14 Inspection, test and operating status	– Item inspection, test and operating status are identified to ensure required activities have been performed successfully

ASME NQA-1 Quality Assurance Requirements

BR 15 Control of nonconforming items	– Items that do not meet specified requirements are identified and controlled
BR 16 Corrective action	– Conditions adverse to quality are identified, controlled and corrected to prevent recurrence, and reviewed for lessons learned
BR 18 Audits	– The QAP is assessed using compliance and performance-based methods using independent personnel

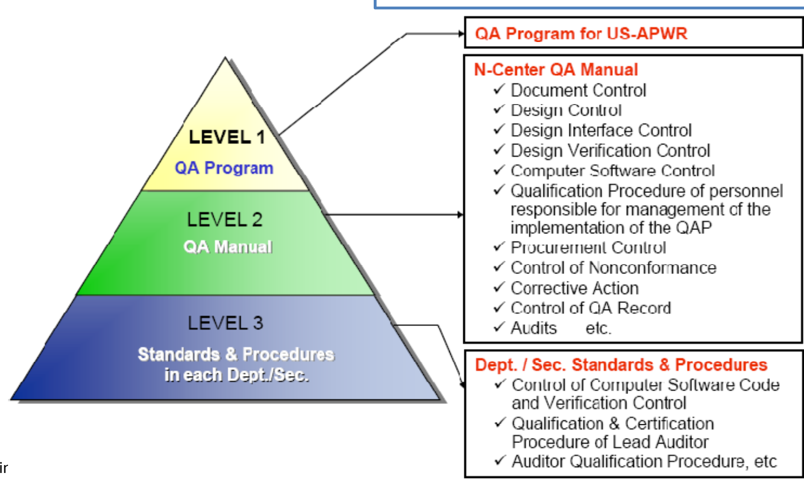
ASME NQA-1 Quality Assurance Requirements – Appendices

- Nonmandatory Appendices:
  - Appendix 1A-1. Organization
  - Appendix 2A-1. Qualifications of Inspection and Test Personnel
  - Appendix 2A-2. Quality Assurance Programs
  - Appendix 2A-3. Education and Experience of Lead Auditors
  - Appendix 3A-1. Design Control
  - Appendix 4A-1. Procurement Document Control
  - Appendix 7A-1. Control of Purchased Items and Services
  - Appendix 16A-1. Corrective Actions
  - Appendix 17A-1. Quality Assurance Records
  - Appendix 18A-1. Audits

ASME NQA-1 Quality Assurance Requirements – Appendices

An Example implementation:

Example: Tsuruga Japan with 2007 design to meet US requirements (US-APWR) by Mitsubishi



2000. Federal Regulations, Guides, and Reports

- CFR
  - 10 CFR 50.34(b)(6)(ii)
  - Appendix A and Appendix B to Part 50
- RGs
  - RG 1.28, RG 1.29, RG 1.94, RG 1.132
  - RG 1.136, RG 1.138, RG 1.142, RG 1.149
- NUREG
  - NUREG 0800, NUREG/CR-6486
  - NUREG/CR-6927
- ASME
  - ASME NQA-1 Quality assurance requirements

2000. Federal Regulations, Guides, and Reports

- Objective and Scope Met
  - Introduced and highlighted selected documents and contents relating to civil & structural inspection
  - Presented and discussed
    - Federal regulations
    - Regulatory guides
    - Nuclear regulatory reports (NRC reports)
    - ANSI-ASME Codes and standards that detail these documents

Completed Items of Overall Outline