RIC 2006
Session W3BC
Risk-Informed Regulatory Structure For Future Reactors

ASME Related Nuclear Codes & Standards Developments

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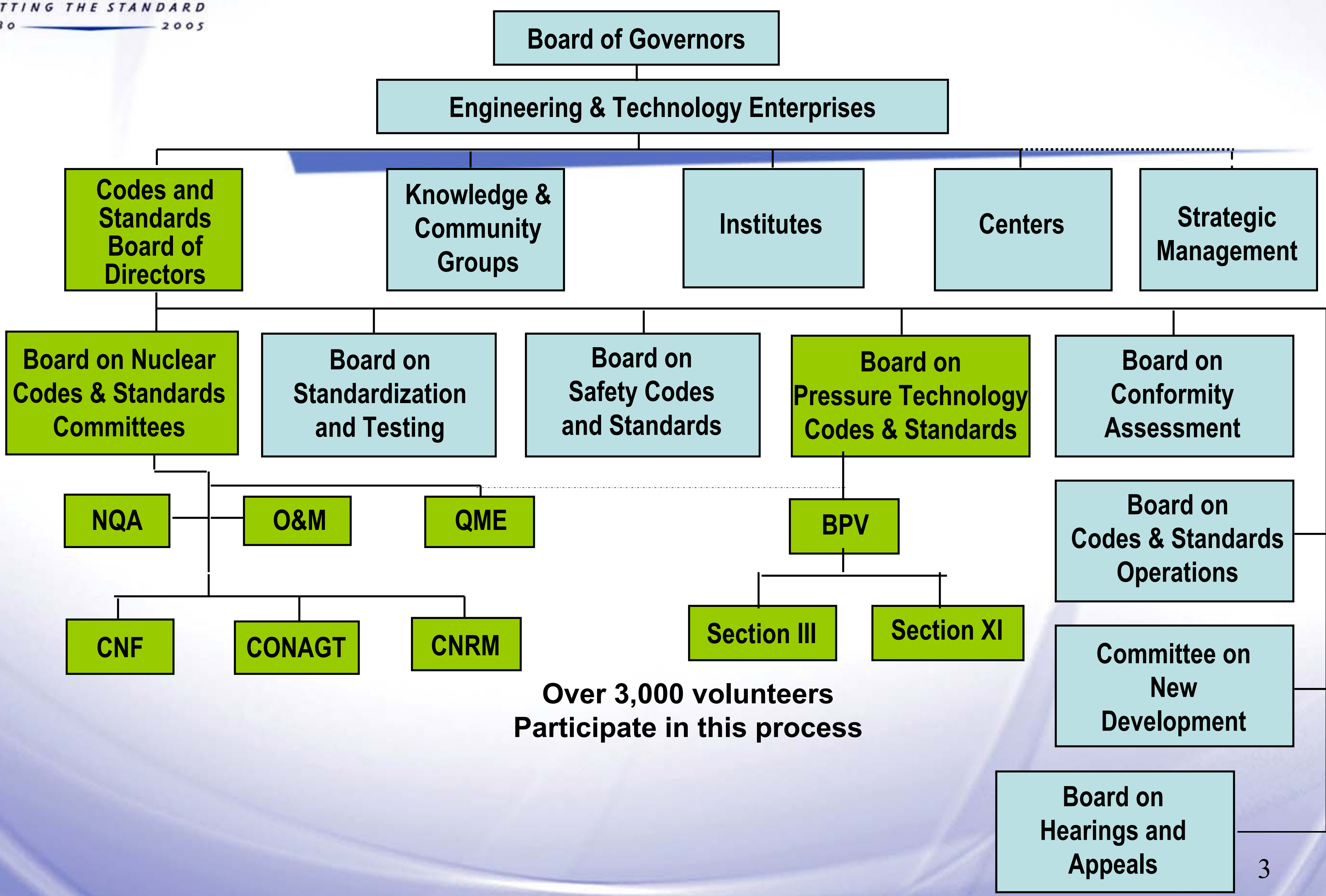
Topics

- ASME Nuclear Codes and Standards Overview
- Probabilistic Risk Assessment Standards
- Risk-Informed Safety Classification
- Risk-Informed Design Methods
- Challenges and Positive Steps Forward
- Summary

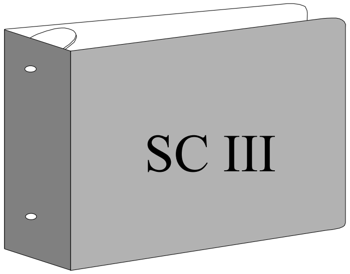


Over 3,000 volunteers Participate in this process

Board on Nuclear Codes & Standards

Charter: To manage all ASME activities related to codes, standards, and accreditation programs directly applicable to nuclear facilities and technology.

Nuclear Codes and Standards



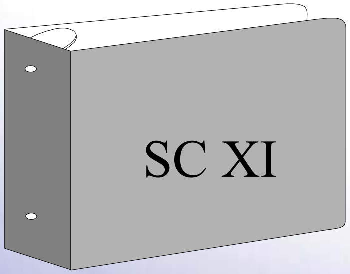
Boiler and Pressure Vessel Code, Section III

Rules for Nuclear Components

Division 1 – Eight Subsections

Division 2 – Concrete Containments

Division 3 – Transport Packaging

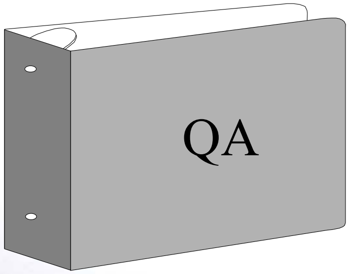


Boiler and Pressure Vessel Code, Section XI

Inservice Inspection

Nuclear Codes and Standards

Probabilistic Risk Assessment – RA-S



Nuclear Quality Assurance – NQA -1

Qualification of Mechanical Equipment – QME-1

Nuclear Codes and Standards

Cranes

Overhead and Gantry Cranes – NOG-1
Underhung and Monorail – NUM-1

Gas Treatment

Air and Gas Treatment – AG-1

OM

Operation and Maintenance – OM
Standards and Guides – SG

Probabilistic Risk Assessment Standards

PRA Standards Needs

PRA Standards activities will be needed to support current and new reactor developments –

1. Integration of ASME RA-S standard with Internal Fire, Low Power/Shutdown, and External Event PRAs consistent with NRC Plan for PRA quality expectations and requirements
2. Continue to build PRA Standards for Levels 1, 2 and 3 to support current light water reactors (LWRs)
3. Extend PRA Standards for current LWRs to address Advanced Reactor Designs – e.g.,
   - ABWR
   - AP-1000
   - EPR
4. New PRA Standards for Gen IV Reactor Designs

PRA Standards Needs for New Reactor Designs

- PRA is integrated from the beginning of the design
- Licensing of new reactors should continue as PRA Standards are under development
- Key initiatives will be coordinated with Nuclear Risk Management Coordinating Committee (NRMCC) –
  - Comprised of leaders from ASME, ANS, NRC, NEI, DOE, IEEE, EPRI, Owners Groups
  - Develop overall structure and plan for all needed PRA Standards

Risk-Informed Safety Classification

NRC Risk-Informed Safety Classifications

(From NRC Rule, §50.69, November 22, 2004)

1 **"RISC-1" SSCs** Safety-Related Safety Significant	2 **"RISC-2" SSCs** Nonsafety-Related Safety Significant
3 **"RISC-3" SSCs** Safety-Related Low Safety Significant	4 **"RISC-4" SSCs** Nonsafety-Related Low Safety Significant

←Risk-Informed→

← Deterministic →

Risk-Informed Safety Classification

- Today's reactors can use 10 CFR 50.69
- Advanced reactor designs are using PRA results to support definition of safety-related and non-safety-related SSCs –
  - 10 CFR 50.69 could be applied "after-the-fact" by licensee to further refine the classification – but this would be a cost-benefit decision
- For new Gen IV reactor designs, 10 CFR 50.69 does not apply –
  - An iterative process
  - Need to develop criteria

Risk-Informed Safety Classification

- ASME Activities
  - Several Code Cases have been developed and approved to address pressure boundary function for ISI and repair/replacement consistent with 10 CFR 50.69
  - ASME Section XI, Division 2 has efforts underway to develop risk-informed ISI approach for high temperature gas-cooled reactors
  - ASME Code Case N-720 under development for Risk-Informed Safety Classification for Construction of Nuclear Facility Components, Section III, Division 1
- ANS 28 Subcommittee is developing ANS 53.1 – Nuclear Safety Criteria for the Design of Modular Helium Cooled Reactor Plants

Risk-Informed Design Methods

Risk-Informed Design Methods

- Example – Needs for Gen IV - Nuclear Graphite Core Components Code
- Key Differences in Properties/Behavior from Steel
  - Non-linear materials behavior
  - Yield stress is not definable
  - Large scatter of strength data
  - Strength increases with increasing temperature
  - Fast neutron flux changes all material properties and induces dimensional change and creep

Risk-Informed Design Methods

- Main Processes in Probabilistic Design –
  - Establish loads / load combinations & probability of occurrences
  - Characterization of material strength distribution
  - Prediction of stress state in parts for load cases
  - Application of probabilistic failure criterion
  - Comparison of probability of failure to required reliability
- ASME Research Project underway for application of load & resistance factor design (LRFD) methods for nuclear plant piping
- Effort underway by ASME Board on Nuclear Codes & Standards to determine how to incorporate these methods into ASME Section III

Challenges and Positive Steps Forward

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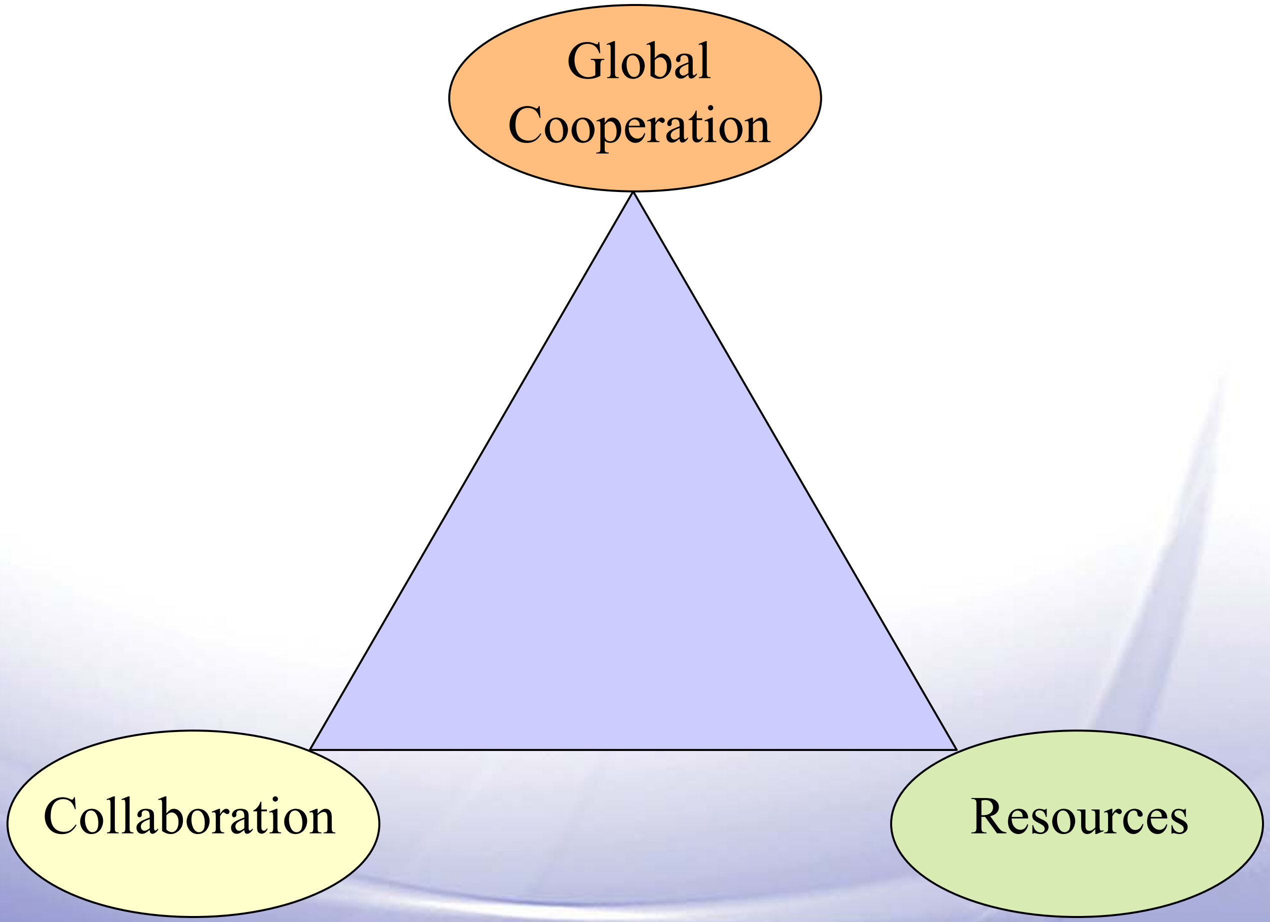

Summary

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- ASME looks forward to continuing to work with NRC on development of a Risk-Informed Regulatory Structure for Future Reactors as well as other issues to address NRC needs
- ASME has a number of related initiatives underway in both Nuclear Codes and Standards Actions as well as supporting research activities
- ASME would like to expand collaborative efforts underway with other Standards Development Organizations, industry groups, and government agencies to support new reactor needs
- Relevant global cooperative initiatives are underway and can be built upon to address future new reactor needs