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# Question and Topic No. (1) – Requirements To Use Code Editions/Addenda Within 3 Years of Contract

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**ASME – NRC Workshop on New Reactors**

**April 9, 2008**

**Rockville, Maryland**

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# Requirements To Use Code Editions / Addenda Within 3 Years of Contract

**Question and Topic No. (1) -**

Per current ASME Boiler and Pressure Vessel Code requirements, designs must be done to Code Editions within 3 years of the contract date. Efforts are underway within ASME Subcommittee III, Subgroup on General Requirements to address this matter.

# Summary of Issue

The 10 CRF Part 52 licensing process and the application of the ASME Code are in conflict in specifying the applicable ASME Code Edition and Addenda to be used for plant design and construction.

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# Current Code Requirement

ASME Code (NCA-1140) requires that –

In no case shall the Code Edition and Addenda dates established in the Design Specifications be earlier than:

*(a)* 3 years prior to the date that the nuclear power plant construction permit application is docketed;

or

*(b)* the latest edition and addenda endorsed by the regulatory authority having jurisdiction at the plant site at the time the construction permit application is docketed.

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# 10 CFR Part 52 Regulation Requirement

- Under the 10CFR52 regulation, licensees will not apply for a construction permit (the licensee will be applying for a COL – Combined Construction and Operating License)
- Under the 10CFR52 regulation, the Code edition and addenda that is endorsed by the regulatory authority (NRC) for the design licensed by NRC is identified in the Certified Design
- Endorsement and application of later Code editions and addenda by the NRC under 10CFR50.55 a) is exempted under 10CFR52
- In applying for a COL, licensees will be referencing a Certified Design which has been licensed by the NRC

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# Proposal for ASME BPV Code Section III Consideration

Add item c) to this paragraph that reads:

*(c)* the edition and addenda that has been endorsed by the regulatory authority in a design licensed by the regulatory authority

# Questions

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# Question and Topic No. (2) – Status Of Requirements For Seismic Developments


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# Status of Requirements For Seismic Developments

**Question and Topic No. (2) -**

Seismic Rules: ASME is working to address the last remaining topic on seismic requirements. Which Code Edition is to be used in the new reactor licensing initiatives as the last requirement is being established?

# Background

- 1994 Addenda to the ASME Code, Section III, adopted revised criteria for piping seismic design. This effort was industry supported and funded to the amount of $1 million. NRC and others in industry had concerns with these rules and a new Task Group was established to discuss the issues raised.

- 2002 Addenda contained additional changes based on the efforts of an ASME Task Group that consisted of personnel from the Code and experts from outside of the Code activity.  In particular, this effort was heavily supported by the Japanese who conducted their own experimental and analytical studies.

# Background

- NRC staff did not endorse the revised criteria because of concerns with the technical basis to establish these criteria

- There were six issues raised and a new ASME Task Group was established. This activity was also heavily supported by the Japanese colleagues.

- The ASME Seismic Project Team resolved 5 of the 6 issues

# Last Open Piping Seismic Issue

- NRC proposed to place restriction in 10 CFR 50.55a endorsement of ASME Code piping rules that $B_2' = 3/4B_2$ for carbon steel elbows and tees at temperatures greater than 300°F instead of the *2/3rd* factor.

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# Background

- April 2006 meeting with ACRS reported on the progress of the activity to date
- The recommendation to the NRC Staff and ASME was that they work together to resolve the issue

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

ACRSR-2184

April 14, 2006

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REVIEW OF THE 1994 ADDENDA TO THE ASME CODE FOR CLASS 1, 2, AND 3 PIPING SYSTEMS AND THE RESOLUTION OF THE DIFFERENCES BETWEEN THE NRC STAFF AND ASME

Dear Mr. Reyes:

During the 531st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2006, we reviewed the resolution of the differences between the NRC staff and the American Society of Mechanical Engineers (ASME) regarding the 1994 Addenda to Section III of the ASME Boiler and Pressure Vessel Code for Class 1, 2, and 3 piping systems. During our reviews, we had the benefit of discussions with representatives of the NRC staff and ASME. We also had the benefit of the documents referenced.

# ACRSR-2184 April 14, 2006 (cont'd)

**RECOMMENDATION**

Most of the differences between the staff and ASME are resolved. The staff proposes to address the one remaining issue related to dynamic strain aging of certain carbon steels at temperatures greater than 300 °F by placing a restriction on the endorsement of the ASME Code in 10 CFR 50.55a. This approach is practical; however, we encourage the staff to work with ASME to resolve the one remaining issue.

**DISCUSSION**

The NRC staff initially did not endorse the revised seismic design criteria in the 1994 Addenda to the ASME Code because of concerns with the technical basis used to establish these criteria. Since that time, the ASME has initiated changes to the Code to address the staff's concerns. These changes include eliminating the application of the seismic rules to flow-transient loads, eliminating the NB-3200 strain criteria, modifying the Class 2 and 3 Level B limits to be consistent with the Level D limits, eliminating changes specifying the methods to generate seismic loads in the evaluation of reversing dynamic loads, and adding provisions to address potential strain concentrations. The staff agrees with these changes.

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# ACRSR-2184 April 14, 2006 (cont'd)

The remaining unresolved issue between ASME and the staff relates to the effects of dynamic strain aging on the ultimate tensile capacity of certain carbon steels at temperatures greater

than 300 °F. The staff proposes to address this issue by placing a restriction in the 10 CFR 50.55a endorsement of the ASME Code. This approach is practical; however, we encourage the staff to work with ASME to resolve the one remaining issue.

Sincerely,

/RA/

Graham B. Wallis
Chairman


References:

1. U.S. Nuclear Regulatory Commission, "Seismic Analysis of Piping," NUREG/CR-5361, June 1998.
2. Letter to G.M. Eisenberg, Director, Nuclear Codes and Standards, ASME, from Brian W. Sheron, NRR, "ASME Code Revisions to the Design Rules for Piping Systems," May 24, 1995.
3. Presentation by John R. Fair, NRR, to the ACRS Subcommittee on Materials and Metallurgy, "Piping Seismic Design Criteria," March 25, 1999.
4. Presentation by John R. Fair, NRR, to William J. Shack, ACRS, "Status of ASME Code Piping Seismic Design Criteria," October 3, 2003.

# What Has Been Done

Section III established another Task Group, and it has presented its conclusions to Section III at the last Boiler Code meeting in February 2008. Once again, this included the Japanese team.

# Next Step

- ASME and USNRC staff should meet to review the latest conclusions
- The difference between the two positions is approximately a 12% impact on the allowables
- This difference is well within the accuracy of the calculations and the seismic load predictions
- There is also a need to understand the basis for the 3/4 factor proposed

# Summary

- This efforts has been an extremely long process, and the commitment to the activity by the participants has been extraordinary
- Much has been learned
- Cooperation between industry and the regulator over the period has been good
- This last issue can be resolved in a timely manner

# Questions

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# Question and Topic No. (3) – Environmental Fatigue Qualification

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# Environmental Fatigue Qualification

**Question and Topic No. (3) -**

Environmental Fatigue Qualification – The ASME Subgroup on Design of ASME Subcommittee III is reviewing alternative methods to address fatigue; especially the effects of environment on fatigue.

The question is “What is the best approach to addressing fatigue using these alternative approaches in new reactor licensing initiatives?

# Background on ASME Efforts to Address the Impact of Environmental Fatigue

| | |
|---|---|
| 1991 | PVRC Steering Committee formed by ASME BNCS in response to NRC Branch Draft Technical Position on Environmental Effect on Fatigue |
| 1993/1994 | ASME members participate with NRC on fatigue issues (GSI-166 and SECY-94-191) |
| 1995 | WRC Bulletin 404 Environmentally Assisted Cracking Fatigue Crack Growth Curves; Bettis Studies |
| 1996 | ASME Section XI Appendix L added to address operating plant fatigue issues |
| 1999 | ASME Section XI Code Case N-643 FCG Rate Curves issued |
| 2000 | Section XI Task Group Appendix L formed to adopt PVRC recommendations |
| 2003 | Section III Task Group on Environmental Effects on Fatigue formed |
| 2006 | ASME Section III Task Group closed with recommendations for Section III Subgroup Design to evaluate if design rules should be changed |

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# Overall Plan

It is the intent that Section III Subgroup Design will develop a non-mandatory appendix that is effectively a handbook on environmental fatigue methods.

The sections of the proposed appendix will be developed as individual Code Cases.

Upon issuing and gaining experience and use of the Code Cases, an Appendix incorporating the Cases including lessons learned will be issued.

# Code Case 1

**NUREG $F_{EN}$ method for Environmental Fatigue Assessment**

This Case would place NUREG-6909 $F_{EN}$ methodology into Code format. The approach would include the use of the air curves with factors of 2 and 12.

# Code Case 2

**Conservative Environmental Fatigue Curves**

This Case will consist of bounding Environmental Fatigue Curves for Carbon, Stainless and Low Alloy steels.

The Subgroup on Fatigue Strength would determine if we should use the Curves from the NUREG with a margins of 2 on stress and 12 on cycles or use the previously proposed O'Donnell Curves with factors of 2 and 20 on air and then adjust for worst case oxygen and strain rate.

# Code Case 3

**Flaw Tolerance Approach**

This Case would incorporate Section XI Appendix L methods of evaluation when U exceeds 1.

This appendix also should include a screening criteria to reduce the number of subject locations.

# Code Case 4

**PVRC $F_{EN}$ Method**

This Code Case was proposed previously. A determination needs to be made if the PVRC method is needed due to the NRC NUREG.

# Required Support for Appendix

**The following items would be developed to support an appendix that combined all of the elements developed above:**

**Methods for determining strain rate**

This article would incorporate the methods proposed by the Japanese under their simplified, moderate, and aggressive strain rate models as previously presented. Alternatively, the task force team could provide other insights or rules. The intent would be to document acceptable approaches for determining the stain rate to be used for $F_{EN}$ calculations.

**Sample Calculations:**

This article would provide examples for the application of all methods listed above.

# Impact Assessment

ASME needs help from industry to determine the anticipated impact of environmental effects on components. The area of largest impact is expected to be in piping where simplified fatigue analysis methods of NB-3600 are utilized. For NB-3600 analysis, it would be good to have a complete, before and after analysis, for both carbon and stainless steels utilizing the piping designs in new plants.

It is necessary to determine the attainable usage factors for typical piping systems using the new curves for a sample of pipe anchor groups including containment penetrations. The ideal approach would be to get the NSSS vendors of the new plants to do detailed environmental fatigue analyses for both typical RCS loop piping for stainless and carbon steel systems using the NUREG approach and the enveloping fatigue curves.

Task 1: Stainless Steel with new Air Curve, NUREG FEN, and simplified curves.

Task 2: Carbon Steel NUREG FEN and simplified curves.

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# Other Tasks Required

1. Investigate fatigue resistant fitting designs that provide reduced stress indices due to imposed geometric controls.

2. Develop a management approach to fatigue. This effort would combine the Design and Operational program for Fatigue Management of critical RCS components.

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# Required Regulatory Changes

Significant items that need to be addressed include inspection and the issue of pipe break postulation.

It is expected that changes will be required to the current regulatory application of the results from ASME analysis outputs for the new environmental fatigue rules.

# Questions

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# Question and Topic No. (4) – New ASME Section III And Section XI Task Group

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# New ASME Section III And Section XI Task Group

**Question and Topic No. (4) -**

Inspection: A new joint ASME Section III and Section XI Task Group will be addressing inspection requirements and incorporation of operational experience into design requirements. What is the best approach to incorporate these emerging requirements in new reactor licensing initiatives?

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# New ASME Section III And Section XI Task Group

- The task group will consist of both designers, manufacturers, regulators and owners
- This effort will provide all perspectives of the end users
- The key in addressing operating experience is participation
- Some of the issues already being considered are materials used, accessibility, and documentation requirements

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# New ASME Section III And Section XI Task Group

- When an issue is identified, a code change and code case will be developed
- It would be important to get a fast response both from the committee process and the adoption by regulators

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# Summary

- It is important to get the issues out on the table to be resolved as things are quickly moving forward with the new designs
- It is also important to get the committee and regulatory process finalized on how changes will be adopted to support the combined operating license approvals
- The only identified vehicle that is available for adoption on an expedited schedule would be the issuance of a Code case

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# Questions

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# Question and Topic No. (5) – Performance Demonstration For Radiographic Testing (RT)

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# Performance Demonstration For Radiographic Testing (RT)

## Question and Topic No. (5) -

Performance Demonstration may be required for radiographic testing (RT) and ultrasonic testing (UT) examinations performed during fabrication. If these requirements are incorporated into ASME Section III, what is the best approach to incorporate these requirements in new reactor licensing initiatives?

# Performance Demonstration For Radiographic Testing (RT) Cont'd

## Recognizing The Current State

- ASME Section XI requires volumetric examination for most welds (RT or UT)
- Appendix VIII of ASME Section XI requires performance demonstration for UT
- Radiography (RT) does not have any performance demonstration included in ASME Section XI
- Some new designs do not provide access for performing Ultrasonic Examination (UT)
  - This situation is due to the length of the valve and/or nozzle extension
- Position being taken is all the Code requires is a volumetric examination
  - Concern is that RT may not be able to detect some of the degradation that has occurred in operating plants

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# Performance Demonstration For Radiographic Testing (RT) Cont'd

## Challenges For New Reactors

- New designs should include the accessibility requirements for the appropriate NDE being applied
  - Longer extensions on valves or nozzles
- Performance demonstration for RT can be established
  - Issue becomes implementation on operating plants with the ALARA concern when performing RT
- Performance demonstration for RT to be assigned to the new ASME Section III and XI Joint Task Group on inspection requirements

# Summary

- Requirements in both Section III and Section XI already exist to make items accessible for inspection
- If NDE methods for construction and inservice inspection are going to be different RT verses UT, then those differences must be accounted by performance demonstration
- The most useful product form that ASME could provide to implement these requirements into new reactor licensing initiatives would be a Code Case.

# Questions

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# Question and Topic No. (6) - Risk-Informed ISI For New Reactors

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# Risk-Informed ISI For New Reactors

**Question and Topic No. (6) -**

There is interest in implementing risk-informed inservice inspection (RI-ISI) programs at the start of the operation of new reactors. Discussion will be held on approach for developing RI-ISI programs without traditional ASME Section XI ISI program to serve as a baseline.

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# Risk-Informed ISI For New Reactors

## Recognizing The Current State

- RI-ISI programs for piping are being implemented at over 95% of the operating plants in the USA
- Programs are based on technical reports in conjunction with ASME Code Cases N-560, N-577, N-578, their revisions, and N-716 approved by NRC on a plant-specific basis
- Nonmandatory Appendix R is published in Section XI, 2005 addenda, but is awaiting regulatory approval by rulemaking; Uses method A (PWROG) and method B (EPRI)

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# Risk-Informed ISI For New Reactors

## Recognizing The Current State

- Delta risk evaluation is required to show risk neutrality or risk reduction, but requires an existing traditional ISI program for comparison
- Risk = consequence x failure probability and failure probability is based on operating experience
- RI-ISI programs have resulted in reductions in selected piping examinations from essentially 25% under A traditional ISI program to ~10% for Class 1 systems and less than 7.5 % for Class 2 systems
- RI-ISI programs for piping at operating plants are cost effective and provide defense-in-depth with acceptable levels of quality and safety

# Risk-Informed ISI For New Reactors

## RI-ISI Challenges For New Reactors

- How to determine the impact of risk without a delta risk evaluation that would require the development of a traditional ISI program for piping before one could show how a RI-ISI program would be acceptable
- How to determine failure probabilities without operating experience
- What new reactor PRA quality level will be needed to meet the application requirements for current industry initiatives covered in U.S. NRC Reg Guide 1.200

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# Risk-Informed ISI For New Reactors

## Section XI Working Group Activities

- Action Item Opened WG/IR# 04-01 – Application of Nonmandatory Appendix R To New Plants

## Lessons Learned So Far

- Risk–informed safety classification efforts have shown that one can make determinations by consequence using a conservative failure probability of 1.0; Result would not be cost effective because one ends up with essentially the same percentage of Class 1 welds being selected as a traditional ISI program
- Development of ISI under Section XI - Division 2 for high temperature gas-cooled reactors has pilot study underway for pebble bed modular reactor using reliability integrity management (RIM) program that takes advantage of risk insights for design and operation; Possible approach for RI-ISI applications for new reactors

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# Summary

- PRA personnel resources within the industry are stretched as well as other needed professionals in supporting current reactor needs
- ASME is doing what it can to make progress in this area;  However, for the first set of new reactors, it is a challenge to have a timely, viable solution in place
- If vendors, licensees, and regulator are willing to help with this effort, your volunteer support would be greatly appreciated;  ASME would welcome your active participation

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# Questions