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Present Activity in ASME Section XI Regarding Risk-Informed Maintenance

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Since 1996 Section XI of the ASME Boiler and Pressure Vessel Code has actively incorporated risk-informed concepts. The risk-informed process provides a framework for allocating inspection resources in a cost-effective manner and helps focus inspections where most critical for plant safety. Based on the success of the risk-informed ISI piping applications at US and non-US plants, Section XI has refined existing Code Cases and expanded the use of the risk-informed process to a variety of high-risk components and systems. The risk informed approach started in the area of inspection and is now being expanded to other plant maintenance activities. This article summarizes the Section XI actions and the continued development of the risk-informed process to improve nuclear plant maintenance.

Keywords: Risk-informed, ASME, Section XI, Probabilistic Risk Assessment, Maintenance, Inspection

1. Introduction

Since 1996 Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code [1] has actively incorporated risk-informed maintenance concepts. The ASME Boiler and Pressure Vessel Committee (BPVC) uses Code Cases [2] to introduce new technology and alternatives to existing requirements in the Code.

The initial risk-informed Code Cases addressed inservice inspection programs for selection of areas of piping at nuclear power plants. The three initial Nuclear Components Code Cases that have provided the foundation of the risk-informed efforts in recent years are:

- N-560, rules for examination programs for full-penetration similar metal Class 1 piping welds;
- N-577, rules for examination programs for all Class 1, 2, and 3 piping welds; and
- N-578, an alternate set of rules for examination programs for all Class 1, 2, and 3 piping welds.

Code Cases N-577 and N-578 are commonly referred to as the Westinghouse Owners Group (WOG) Code Case and the EPRI Code Case, respectively.

Recent risk-informed Code Cases by ASME Section XI have continued to assist licensees improve the safe and efficient maintenance of their plants. These Code Cases provide risk-informed safety classification for use in repair/replacement activities, rules for repair/replacement activities on these selected components, and application of risk-informed insights to increase the inspection interval for pressurized water reactor (PWR) vessels from 10 years to 20 years.

There are also a number of proposed risk-informed Code Cases currently under consideration by Section XI. These proposed Code Cases address visual examination during leak tests in lieu of non-destructive examination (NDE) requirements for specific designs of residual and regenerative PWR heat exchangers, development of a common basis for evaluation of examination locations where interference, access limitation, high radiation exposure, etc. make “essentially 100% examination” impractical, and new classification criteria for new pre-service and inservice inspection rules.

The following provides an overview of Section XI risk-informed Code Case activities as well as related risk-informed actions of the United States Nuclear Regulatory Commission (NRC).

2. The Importance of PRA

Probabilistic risk assessment (PRA) has been a critical component of all the risk-informed Code Cases approved to date by Section XI. In 1988 the ASME Center for

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Research and Technology Development (CRTD) began sponsoring programs to apply PRA and related technologies to inspection guidelines for nuclear power plants and operating industrial plants. Shortly thereafter Section XI committees began to incorporate the CRTD techniques into Code Cases. In 1996 the Section XI Subcommittee received BPVC acceptance of Code Case N-560 to utilize the risk-informed approach. The approval of Code Cases N-577 and N-578 followed. All three of these Code Cases use plant PRA information as a foundation for the risk-informed analysis.

NRC management has strongly endorsed the concept of the risk-informed approach and the utilization of the PRA as a foundation for the Code Cases [3]. In 1995 the NRC prepared a policy statement on probabilistic risk assessment that encouraged greater use of this analysis technique to improve safety decisionmaking and improve regulatory efficiency [4]. Since then the NRC has published several bulletins, Regulatory Guides, NUREGs, and Regulations addressing use of PRA, including Regulatory Guide 1.174, *"An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"* and NUREG-0800 [5, 6]. The ASME Code Cases all use the risk acceptance criteria provided by Regulatory Guide 1.174.

All three Code Cases, N-560, N-577, and N-578, have been accepted by NRC for use on a case-by-case basis at US nuclear power plants.

3. Risk-Informed Code Cases

3. 1 Piping Inspection

Code Case N-560 (1996) provides alternative programs for examination for full-penetration similar metal Class 1 piping welds. This was the first Section XI risk-informed Code Case to include nuclear power plant PRA in technical basis. In this new approach, piping systems are divided into "segments" having similar consequence of failure and common degradation mechanisms, and, within segments, into "elements" selected for inspection based on degradation mechanism.

Code Case N-560 includes specific use of operating conditions, industry service experience, nondestructive examination (NDE) results, and presence of repairs in a segment. This Case lists many conditions to be assessed, evaluated, and graded, to arrive at selection of element locations and methods of examination. Although Code Case N-560 is much more complex than Section XI Table IWB-2500-1 Examination Category B-J, which was based upon non-nuclear ASME Code design and fabrication operating experience, the advantage of N-560 is that it focuses the examinations upon the most critical piping elements. The analysis using this Case usually results in significant reduction in the number of elements for which examination is required and subsequent reductions in critical path examination time and personnel radiation exposure. Based on lessons learned during plant applications, the Section XI committees have revised not only Code Case N-560, but Code Cases N-577 and N-578 as well. The current version of this Code Case is N-560-2, published in 2000.

Code Cases N-577 and N-578 were updated in 2000. The current versions, N-577-1 and N-578-1, provide alternative rules for examination programs for all Class 1, 2, and 3 piping welds. Code Case N-577-1 is called "Method A" or the WOG Code Case to distinguish it from N-578-1, called "Method B" or the EPRI Code Case. Both Code Cases extend the basic technology of Code Case N-560 to Classes 1 and 2 Examination Categories B-F, B-J, C-F-1, or C-F-2. Where shown by analysis, it may extend NDE to Class 3 piping elements, and even to non-nuclear class piping (service water, etc.) systems that are determined to be more risk-important than some Class 3 systems. Also, piping in systems evaluated as part of plant PRA but outside current Section XI examination boundaries may be included.

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In both Code Cases N-577-1 and N-578-1 the concept of High Safety Significant (HSS) or Low Safety Significant (LSS) piping structural elements within systems is introduced. The HSS elements comprise the primary basis for examination programs.

In general, N-577-1 is considered more PRA-dependent than Code Case N-578-1. Often N-577-1 is referred to as the "quantitative" method while Code Case N-578-1 is considered to be a more "qualitative" method.

Specific differences between these two Code Cases are found in their respective Appendix I. One such difference is the requirement in N-577-1 for an "expert panel". N-577-1 also requires determination of the adequacy of the applicable PRA.

A unique feature of Code Case N-578-1 is its specific use of operating conditions, industry service experience, non-destructive examination results, and presence of repairs in a segment. N-578-1 also lists many conditions to be assessed, evaluated, and graded, in order to arrive at selection of element locations and methods of examination. In addition to the examinations called for in Code Case N-560, N-578-1 also calls for all examinations included in the existing plant flow-assisted corrosion (FAC) Inspection Program and inter-granular stress corrosion cracking (IGSCC) Inspection Program. N-578-1 also specifies examination of at least 25% of the locations in Risk Categories 1, 2, and 3.

In addition to the three original risk-informed Code Cases, Section XI has continued to expand the application of the risk-informed process to other plant applications. These include the approved Code Cases N-660, N-691, and the proposed Code Cases N-706, N-711, and N-716. Each of these Code Cases is summarized below.

Proposed Code Case N-711 provides a common basis for evaluation of examination locations in Class 1 and 2 piping where such factors as interference, access limitation, and high radiation exposure, make "essentially 100% examination" impractical. This proposed Code Case may be applied when examination coverage at a location does not meet requirements of Section XI for Class 1 or Class 2 piping welds, or examination coverage requirements of Code Cases N-560-2, N-577-1, or N-578-1. An extensive table in Code Case N-711 provides various alternatives to address examinations in question.

Proposed Code Case N-716 for Class 1, 2, and 3 piping welds is based on "lessons learned" in application of risk-informed technology, including operating experience. It provides new classification criteria, leading to new preservice and inservice inspection rules. To determine weld classifications as HSS or LSS, the use of PRA data is permitted but not required.

3. 2 Repair/Replacement

Code Case N-660 (2002) provides a risk-informed safety classification (RISC) process for use in repair/replacement activities. The RISC process that may be applied to any Class 1, 2, 3, or non-class pressure-retaining items or their supports, in accordance with criteria established by regulatory authorities having jurisdiction at the plant site. Although items are classified either as HSS or LSS, these classifications might not be directly related to other risk-informed applications, such as in previously described Code Cases. For instance, most Class 1 items are considered HSS.

Code Case N-660 also requires that core damage frequency (CDF) and large early release frequency (LERF) must be included as risk metrics in the RISC process. Documentation of adequacy of any PRA used in the process is required. Appendix I of Code Case N-660 shows methodology similar to that of Code Case N-560. Emphasis is on piping systems, because that is where most failures occur.

The first revision of this Code Case addressed the Service Water System. The proposed second revision, N-660-2, is undergoing trial application at the Wolf Creek and Surry Power Stations and is the subject of discussions between Section XI's Working Group on Implementation of Risk-Based Examination and the NRC.

Code Case N-662 (2002) is a companion to Code Case N-660 and provides rules for repair/replacement activities on components classified using Code Case N-660. Repair/replacement activity requirements are determined using the RISC classification (HSS or LSS), the safety-related or non-safety-related classification, and the Class 1, 2, 3 or no-class ASME Code classification. For instance, for RISC category 1, and Classes 1, 2, or 3, all repair/replacement activity rules apply. For HSS safety-related non-class, structural integrity requirements specified in the Code Case N-662 apply. There are no requirements for LSS safety-related non-class systems.

3. 3 Inspection Interval

Code Case N-691 applies risk-informed insights to increase the inspection interval for pressurized water reactor (PWR) vessels from 10 years to 20 years. This increase applies only to reactor vessel pressure-retaining full-penetration welded seams. These are Examination Category B-A shell and head welds, and welds to the head flanges, Examination Category B-D nozzle-to-vessel welds, and Examination Category B-J welds to those nozzles. In this Code Case, probabilistic fracture mechanics (PFM) and risk analyses are applied. As basis for determining change in risk, inputs to a reactor vessel PFM and risk analysis must include:

- Accident transients and frequencies
- Operational transients
- Initial flaw distribution
- Fluence distribution
- Material fracture toughness
- Crack growth rate correlation
- Cladding and residual stress
- Effectiveness of inservice inspection

Integration of results of PFM and risk analyses with frequency of postulated events determine change in risk. Risk acceptance criteria are provided in NRC Regulatory Guide 1.174 [5].

3. 4 Heat Exchanger Inspection

Proposed Code Case N-706 for specific designs of residual and regenerative heat exchangers provides for visual examination during leak tests in lieu of NDE requirements. Risk assessment of this change shows negligible change in risk of core damage. In the technical basis document for this Code Case [7] several designs of regenerative and residual heat exchangers used in US PWRs and BWRs were considered.

The risk-informed assessment has shown that:

- Limiting case for evaluation was determined by design having highest service temperature and pressure, and thermal stress.
- Fatigue is only damage mechanism of concern, and cumulative fatigue usage factor is less than 0.1, which is very low.
- Failure probability analysis, employing Monte Carlo method, conducted on three most important regions of regenerative heat exchanger, showed very low probability of failure.
- Calculated change in risk of core damage compares favorably with acceptability criteria of NRC Regulatory Guide 1.174.

These results established acceptability of examination changes provided in Code Case N-706.

3. 5 Closure Head Flange Weld Inspection

A draft Code Case to reduce examination of PWR closure head-to-flange weld from volumetric and surface to visual examination is currently being developed by Section XI. One of the bases for the justification in proposing a reduction in method/extent of examination is that probabilistic fracture mechanics methods rank this weld quite low in risk.

4. NRC Risk Informed Activities

4. 1 Phased Approach to PRA Quality

The NRC has been actively working to stabilize the PRA quality expectations and requirements in order to achieve an appropriate level of PRA quality for risk-informed regulatory decision making. The NRC has prepared a plan for a phased approach to achieving their quality objectives. The plan, approved in October 2004, will assist both the NRC and industry by:

- moving towards improved and more complete PRAs
- increasing efficiencies in the NRC staff's review of risk informed applications
- clarifying expectations for 10CFR50.46 and 10CFR50.69 rulemakings
- near term progress in enhancing safety through the use of available risk informed methods while striving for increased effectiveness and efficiency in the longer term

4. 2 Risk-Informed Special Treatment Requirements of 10 CFR 50 – Option 2

The final version of 10 CFR 50.69, *Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors*, was published in late November 2004. A copy of the final rule and associated documents can be found at www.nrc.gov.

In this rule the NRC has provided an alternative approach for establishing the requirements for treatment of structures, systems and components (SSCs) using a risk-informed method of categorizing SSCs according to their safety significance. Treatment includes, but is not limited to, quality assurance, testing, inspection, condition monitoring, assessment, evaluation, and resolution of deviations. The amendment revises requirements with respect to ``special treatment,'' that is, those requirements that provide increased assurance (beyond normal industrial practices) that SSCs perform their design basis functions. This amendment permits licensees to remove SSCs of low safety significance from the scope of certain identified special treatment requirements and revise requirements for SSCs of greater safety significance.

The NRC has been actively tracking the progress of the industry efforts to develop Code Case N-660-2, the latest revision of risk-informed Code Case N-660 for repair/replacement activities. The pilot plant efforts at Wolf Creek and Surry are an important source of information for the NRC when reviewing the applicability of the Code Case. The Wolf Creek plant examined containment spray and control building ventilation systems. Chemical and volume control and component cooling water systems were examined in the Surry pilot.

4. 3 Risk-Informing 10 CFR 50.46, ECCS

The NRC has been working on a proposed rulemaking (10 CFR Part 50.46a) to provide an alternative risk-informed set of requirements for emergency core cooling systems (ECCS). This proposed rule would revise the double-ended guillotine break (DEGB). In certain cases, the "transition" break size (TBS) could be used in lieu of the DEGB of the largest pipe in the reactor coolant system.

ECCS requirements will be commensurate with the relative likelihood of breaks in one of two categories. Pipe breaks larger than the TBS, based on their lower likelihood, can be analyzed by the more realistic and less stringent methods established in the new § 50.46a.

The proposed rule will likely be available for public comment in the later part of 2005.

5. Conclusion

The risk-informed process provides a structured and systematic framework for allocating inspection resources in a cost-effective manner and helps focus inspections where they are most critical for plant safety. The risk-informed process has begun to indicate that many welds, components, and systems are not as important as the originally thought when the ASME Section III Classes 1, 2, and 3 categories were developed. The risk-informed procedures and rules as developed by the ASME Section XI take full advantage of PRA data, industry and plant experiences, information on specific damage mechanisms, and other available information.

Two important features of the risk-informed process are the requirement for clear documentation of the analysis and the need to make modifications and improvements as new information and insights become available. The risk-informed process is a “living program”.

To date almost all US plants and many plants in Europe and Asia have applied the risk-informed Code Cases N-560, N-575, or N-578 to their piping ISI program. Based on the lessons learned from these applications, Section XI has continuously refined and improved these risk-informed Code Cases.

The success of the risk-informed applications has lead Section XI to expand the use of the risk-informed process to other high-risk components and systems. There is a growing base of support by the nuclear industry and the regulatory organizations worldwide to continue the development and use of the risk-informed process to improve plant maintenance.

Acknowledgement

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