July 20, 2005

MEMORANDUM TO: Rebecca L. Karas, Chief
Component Integrity and Testing Section
Mechanical and Civil Engineering Branch
Division of Engineering, NRR

FROM: David C. Fischer **/RA/**
Component Integrity and Testing Section
Mechanical and Civil Engineering Branch
Division of Engineering, NRR

SUBJECT: COMMENTS ON THE MARCH 25, 2005, PROPOSED REVISION TO ASME SECTION XI RISK-INFORMED CODE CASE –660, "RISK-INFORMED SAFETY CATEGORIZATION FOR REPAIR/REPLACEMENT ACTIVITIES"

In April 2005 the ASME forwarded to the NRC a proposed revision to ASME Section XI Code Case –660, "Risk-Informed Safety Classification For Repair/Replacement Activities" (reference ADAMS Accession number ML051880307). The proposed revision to Code Case –660, and the staff's comments on it (attached), will be discussed during a public meeting to be held at NRC Headquarters on July 26, 2005 (reference ADAMS Accession number ML051890500).

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ADAMS ACCESSION NUMBER: ML052010679

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DATE	07/20/2005

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NRC Comments on –660-2, April 22, 2005 (Disapproved)

1. The White Paper provides a redline/strikeout comparison of the current version of the code case with an unofficial version of the code case (–658). The proper comparison would be to the last approved version of the code case (–660, Rev 0).

2. Section 1200 (a) states that this classification is to be used only for repair/replacement activities. This is in conflict with the white paper which states that this code case can be used in conjunction with a 10CFR50.69 application or for an individual repair/replacement activity.

3. Section 1200 (a) states that failure potential is conservatively assumed to be 1.0 in performing the consequence evaluation. In fact, in risk evaluations, the failure potential is always assumed to be 1.0 in performing the consequence evaluation. It is then combined with the appropriate value of failure potential to determine risk. In this code case, it is questionable whether failure potential is being assumed to be 1.0 such as in justifying small leaks.

4. Section 1300 (PRA Scope and Technical Adequacy): As risk-informed modifications are being extended to encompass more and more requirements and SSCs, the staff is continuing to refine acceptable approaches for demonstrating adequate PRA quality for individual applications. ASME code cases present a particular challenge because, upon approval, the code case can be implemented by licensees without further staff involvement. This contrasts with most risk-informed changes where each proposed change must be submitted to the staff for prior review and approval.

   The non-mandatory Appendix R to Section XI and code case N 60 both require input from a PRA and both include PRA quality requirements. If approved as part of the next Section XI update, licensees could begin to apply Appendix R to implement risk-informed in service inspection programs without submitting request for relief from the ASME requirements in 10CFR50.55a. Licensees may implement the approved code case N 60 without prior staff review and approval. The staff accepted a determination of sufficient PRA quality for use in Appendix R that corresponds to the requirements applied during the staff reviews of requests to use an RI-ISI program as an alternative to the ASME program. The PRA quality requirements are based on experience from reviewing requests from more than 2/3 of the operating fleet.

   The quality requirements of N 60 are at a higher level and less specific than the Appendix R PRA quality requirements. The categorization in N 60, Rev 0, assumed a rupture probability of one, and develops the safety significance category based on the CCDP/CLERP. This was considered sufficiently conservative such that the high level requirement on the quality of the PRA was sufficient. Very great errors in the PRA would be necessary to yield non-conservative results.

   The PRA quality requirements for code cases must be sufficiently well defined to provide confidence that the PRA used is capable of supporting the application. The revised N 60 allows a small leak to be used instead of a rupture. Although not discussed, based on the guidelines provided it appears that a small leak could be used for a large fraction of the piping. Modeling the sometimes complex consequences of a small leak could place a greater demand on the technical adequacy of the PRA than loss of full flow and related environmental effects caused by a pipe rupture. The current language in the proposed N 60-2 states that, "PRA technical adequacy shall be assessed against a standard or set of acceptance criteria that is endorsed by the regulatory agency having jurisdiction over the plant site."

At this time, it cannot be concluded that this level of detail is sufficient for this application because the results will be used to guide the repair and replacement of piping and the categorization of the piping may rely upon the use of a technically adequate PRA. One method to assure the required technical adequacy that may be of use in these code cases is being pursued in support of the mitigating system performance index (see NRC ADAMS ML043510095). The technique involves first characterizing the elements of the PRA that have the greatest impact on the analysis needed to support the decisions on the acceptability of the proposed change. Next, the important characteristics are compared to the elements in the appropriate standard and the elements that are most important for that application identified. Finally, acceptable attributes for the important elements are defined.

5. Section I-1.0 states that this RISC process is based on conditional consequence of failure. This is not quite accurate as probability of failure is considered in justifying small leaks.

6. Section I-3.0: Terminology for defining segments is not consistent in the code case. This section defines segments as having common consequences while Section I-1.0 defines segments as having similar consequences.

7. Section I-3.0 states that "Throughout the evaluation of I-3.0, credit may be taken for plant features and operator actions to the extent these would not be affected by failure of the segment under consideration." It then goes on to list several features which must be satisfied in order to take credit for operator action. Similar features should be satisfied when taking credit for operator action in the Analysis and Assessments (Section I-3.1) and Classification (Section I-3.2).

8. Section I-3.1.1(a) lists several situations when large pressure boundary leaks need not be assumed (i.e., when a smaller leak can be assumed). One of the situations is "Class 2 and 3 moderate-energy systems that meet the requirements of Appendix II." We do not endorse the use of licensee programs (e.g., erosion/corrosion program, corrective action program, program to protect against localized corrosion) as the primary basis for concluding that the potential for a large break is negligible and thereby categorizing certain pipe segments as LSS. This is a significant departure from the bounding consequence-based concept on which the Code Case was based and previously found acceptable to the staff. If industry wants to distinguish between large and small breaks as part of a risk-informed Code Case then the different break sizes should be modeled as separate initiating events in a plant-specific PRA (supported by failure rate data) to arrive at the appropriate categorization. Categorizing based in large part on programmatic activities (i.e., over-reliance on programmatic activities as proposed) is not consistent with the defense in depth principle as articulated in RG 1.174. Furthermore, there is nothing in the proposed Code case to ensure that these various programs, that potentially make the potential for a large break negligible, remain effective.

9. Section I-3.1.1(a) states large pressure boundary leaks need not be considered when "the probability of a large leak at end of normal operation (e.g., 40 years), as calculated per Method A in the Non-mandatory Appendix R for Risk-Informed In service Inspection for Piping is less than or equal to 10^{-4}." This is also a significant departure from the

bounding consequence-based concept on which the Code Case was based and previously found acceptable to the staff. The referenced methodology is the Westinghouse RI-ISI methodology which uses probabilistic fracture mechanics to categorize weld/elements for the purpose of determining which welds should be inspected. This methodology should not be expanded for use in repair/replacement unless the materials used in the repair/replacement activities have the exact same fracture mechanics properties as materials currently installed. It just doesn't make any technical sense to expand the RI-ISI methodology to repair/replacement activities. This methodology is also not consistent with 10 CFR 50.69 (whereby repair/replacement requirements are relaxed) where SSCs are categorized in part based on the consequence of their failure (e.g., RAW). In addition, the basis for 10^{-4} is not provided in the associated White Paper and 10^{-4} appears to be much too high of a threshold.

10. Section I-3.1.1(a) (3) needs further amplification and examples of physical configurations that preclude the possibility of a large leak. In the cited example of an orifice, it is not clear as to what would be the purpose of the orifice. For example, if the orifice is for flow measurements, it may not preclude large leak.

11. Section I-3.1.1(e) states that automatic actions need not be safety related nor subject to single failure. In that case, they may not be reliable and should not be considered in classifying the segment.

12. Section I-3.1.2 states that each piping system, or portion thereof, shall be classified into one of three impact groups: initiating event, system, or combination. The Section then goes on to introduce a fourth impact group, Containment Performance Impact Group.

13. Section I-3.1.2 states that available risk information related to the mitigation of fire, seismic, shutdown, and other external events shall be considered. Please amplify on how this information will be considered. For example, was this considered in the Surry and Wolfcreek pilot applications, and if so, how was it considered.

14. The Code Case defines a "train" as "a set of equipment (e.g., pump, piping, associated valves, motor, and control power) that individually fulfills a safety function (e.g., high pressure safety injection) with an unavailability of 1E-02 as credited in Table I-2 and I-3." Section I-3.1.2 states that "The Owner or his designee shall ensure that the quantitative basis of Table I-2 (e.g., one full train unavailability approximately 10-2) is consistent with the failure scenario being evaluated." The Code case offers no practical approach to ensure that this underlying technical assumption (i.e., unavailability of 1E-02 per train) is maintained for all the different trains of safety systems.

15. Section I-3.2.2(b)(2) should specify which plant procedures (i.e., normal operating procedures) are being referred to in this evaluation.

16. Section I-3.2.2 lists several conditions that would cause an item to be categorized as HSS (i.e., if answered FALSE). These conditions include: Failure of the passive pressure boundary function will not result in unintentional release of radioactive material in excess of plant offsite dose limits specified in 10 CFR Part 100. A threshold lower than the Part 100 limits should be specified. In order to maintain consistency with a similar NEI 00-04 consideration, this criterion should be modified to read: failure of the passive pressure boundary function will not result in an unintentional release of radioactive material that would result in the implementation of offsite radiological protective actions.

17. Section I-4.0 states that plant design changes shall be screened prior to implementation Similarly, Section I-4.0 (a), (b), and ©) provide evaluations that shall be made to determine if changes to safety classifications need to be made. Please state that the plant procedures shall be modified to require that these evaluations will be performed.

18. Section I-4.0 states that safety classifications shall be reevaluated on the basis of inspection periods and inspection intervals. Please clarify that reevaluations shall be performed on the basis of inspection periods to be consistent with RI-ISI procedures.

19. Appendix II purports to "provide one method of determining credible lead [presumably leak] size for a considered piping segment." However, the Appendix II only provides guidance for when the potential for large (complete segment) breaks can be considered negligible. It offers no guidance for determining what size break is credible.

Imbro Comments on Code Case N–660b dated 03/25/05

1. **Inquiry -** In the last line, after *risk-informed repair replacement activities* insert ....for facilities that are design and constructed in accordance with ASME Section III.

2. **-1220 Classifications -** Class 1 items and items within the break exclusion region [>NPS 4] and their support are classified as HSS and are not part of the Code Case. What is the technical basis for excluding Class 1 lines #NPS 4? A break in a NPS 4 Class1 line would likely be beyond the makeup capability of the normal makeup systems of a PWR and would be considered a small break loss of coolant accident. I don't agree that licensees/owners should be given the option to categorize such lines as having low safety significance. All lines required by NRC regulation to be Class 1 should be considered as HSS and be excluded from the Code Case.

3. **Definitions -**

   a. Containment Barrier - a component(s) that provides a containment boundary/isolation function such as normally closed valves or valves that are designed to ~~go closed upon actuation~~ automatically close when containment isolation is required.

   b. Core damage - ~~uncovery and~~ heat up of the reactor core to the point at which ~~prolonged oxidation and severe fuel damage is anticipated and involving enough of the core to cause a significant release~~ the maximum fuel element cladding temperature exceeds 2200 ^{0}F or the maximum fuel element cladding oxidation exceeds 17% of the total cladding thickness.

   c. Failure - an event involving leakage, rupture, or a condition that would ~~disable the ability of~~ prevent an item ~~to perform~~ from performing its intended safety function

   d. Failure potential - likelihood of ruptures or leakage that result in a reduction or loss of the pressure-retaining capability of the ~~component~~ item or the likelihood of a condition that would prevent an item from performing its safety function (e.g., fails to start, fails to run).

e. High Energy system - those systems that ~~for the major operational period~~ are either in operation or maintained pressurized under conditions where either or both of the following are met:

a. maximum operating temperature exceeds 200 ºF, and

b. maximum operating pressure exceeds 275 psi

f. Moderate-energy systems - those systems that during normal plant conditions are either operated or maintained at conditions below that specified for high energy systems. For the purposes of break postulation, systems that qualify as high energy systems for only a short period of time but qualify as moderate-energy systems for the major operational period may be treated as moderate-energy systems. ~~not high-energy systems and systems that meet the temperature/pressure thresholds of high-energy systems but only for short operational periods.~~ Short operational periods are defined as about 2 percent of the time that the system operates as a moderate-energy system (e.g., reactor decay heat removal); however, systems such as auxiliary feedwater systems operated during PWR reactor startup, hot standby, or shutdown qualify as high-energy systems.

g. Train - As defined in this appendix, "a train" consists of a set of equipment (e.g., pump, piping, associated valves, motor, and control power) that individually fulfills a safety function (e.g., high pressure safety injection) ~~with an unavailability of 1E-02 as credited in Tables I-2 and I-3~~

4. **I-3.0 Consequence Evaluation**

a. This paragraph states: "All pressure retaining items and their supports shall be evaluated by defining piping segments that are grouped based on common conditional consequence (i.e., given failure of the piping segment)". This would permit change of classification within the scope of a piping analysis problem. That is an Owner could transition from ASME Section III piping and components to, for example, B31.1 or AWWA piping and components in the same piping design run. I don't this is good engineering practice and raises technical issues regarding mixing design criteria. Piping segments should as a minimum be from anchor to anchor. While this thinking with regard to piping segments not being defined as from anchor to anchor may be acceptable for Risk-Informed ISI, it is not acceptable for design changes.

b. This paragraph states: "Throughout the evaluation of I-3.0, credit may be taken for plant features and operator actions to the extent these would not be affected by failure of the segment under consideration".

i. The consequence evaluation can result in repairs and replacements of the plant systems and components, i.e., physical hardware changes, that are permitted to meet less stringent quality and design requirements than ASME Section III. I do not agree that operator action should be used as a basis to minimize design and quality requirements. This, in my opinion, would not be aligned with the Reg. Guide 1.174 guidance with respect to maintaining defense-in-depth. Operator action in this Code Case is assumed to be 100% successful in all cases as long as the criteria of I-3.0 are met. There may be instances where operator action may not be successful for whatever reason. Therefore, operator action should complement a robust system design and not be used as a surrogate for design requirements. While some of the defense-in-depth considerations are addressed in I-3.2.2(b)(3) and (4), I believe that crediting of operator action as a basis to reduce design and quality requirements is unacceptable.

ii. Credit for mitigation of the effects of a pipe break should rely only on redundant safety-related equipment.

5. **I-3.1.1 Failure Modes and Effects Analysis (FMEA).** This paragraph states: "Potential failure modes for each system or piping segment shall be identified, and their effects shall be evaluated. This evaluation shall consider the following:
(a) Pressure Boundary Failure Size. The consequence analysis shall be performed assuming a large pressure boundary leak for piping segments. Alternatively, the consequence analysis can be performed assuming a smaller leak, when......."

a. The terminology "large and small" with respect to breaks is undefined. Break sizes should be postulated in accordance with the guidance in the Standard Review Plan, NUREG-0800. NUREG-0800 is the guidance that the staff used review OL applications and to grant operating licenses to the majority of the operating plants.

b. Item (a)(4) - The programs specified in Appendix II are programs that probably all licensees have. Therefore, this is not an adequate discriminator for postulating break size. The result of using this Appendix would be that many, if not all, postulated breaks would be postulated as small breaks. This paragraph and Appendix II should be removed from the Code Case.

c. Item (b) - Isolability of a Break. A break can be automatically isolated by a check valve, ~~a closed isolation valve,~~ or an isolation valve that closes automatically on a given signal ~~or by operator action~~.

d. System Impact or Recovery. The means of detecting a failure, and the Technical Specifications associated with the system and other affected systems. Possible automatic and operator actions to prevent a loss of system function shall be evaluated. Automatic actions shall ~~need not~~ be safety related ~~nor~~ and subject to single failure.
   i. "Shall be evaluated" is meaningless unless there are a specified set of criteria for the item to be evaluated against. No criteria are provided.

e. System Redundancy. The existence of redundancy for accident mitigation purposes shall be considered."Shall be considered" is meaningless. Sounds like a helpful hint.

6. I-3.2.2(b)(2) Classification Considerations states that "Failure of the pressure boundary function will not prevent the plant from reaching or maintaining safe shutdown conditions; and the pressure boundary function is not significant to safety during mode changes or shutdown. Assume that the plant would be unable to reach or maintain safe shutdown conditions if a pressure boundary failure results in the need for actions outside of plant procedures or available backup plant mitigative features." This criteria is highly problematic. The failure of a pressure boundary function that would prevent a plant from achieving and maintaining safe shutdown would be outside any plant's licensing basis. It's not clear what the second sentence is trying to convey. If the plant needs to rely on operator actions outside plant procedures to achieve or maintain safe shutdown in the event of a pressure boundary failure, is the Code Case implying that this would be acceptable? This would be unacceptable to the NRC staff under any circumstances and would be indicative of significant design discrepancy. Again, this situation, if it existed, would be not be permitted by NRC regulations.

7. I-3.2.2(b)(5) Classification Considerations states that “Failure of the pressure boundary function will not result in unintentional release of radioactive material in excess of plant offsite dose limits specified in 10 CFR Part 100." This comment has been made several times by the NRC staff and has not been addressed by ASME. The NRC does not agree that Part 100 releases or anything close to Part 100 releases are an acceptable criteria for categorizing pipe breaks. For operating plants, it is not expected that a pipe break would cause any significant release of activity to the public in any operating reactor. The answer to this question should always be True. Therefore, this metric is not an adequate discriminator for categorization of piping segments. The requirements of Part 100 specified in the Code Case could be replaced with the requirements of 10CFR20.1301. This would assure that any increased break frequency resulting from the reduction of quality and design requirements would not impact the health and safety of the public.

8. Table I-2 - The Code Case should define what is meant by 0.5 Unaffected Backup Trains.