ASME Meeting with U.S. Nuclear Regulatory Commission

ASME Code Section XI
Code Case N-660b, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities"

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Rockville, MD
July 26, 2005

ASME

1

Cognizant ASME Members & Staff and Supporting Industry Participants

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- *Ken Balkey* – Westinghouse
  Vice President, ASME Nuclear Codes and Standards
- *Kevin Ennis* – Director, ASME Nuclear Codes and Standards
- *Gary Park* – Nuclear Mgmt Co.
  Chair, ASME Section XI
- ASME XI Working Group on Implementation of Risk-Based Exam
  - *Robin Graybeal* – Inservice Engrng
  - *Pat O'Regan* – Electric Power Res Inst
  - *Mark Pyne* – Duke
  - *Alex McNeill* - Dominion
- *Ralph Hill* – Golder Associates
  Chair, ASME Board on Nuclear Codes and Standards Risk Management Task Group

INDUSTRY

- Nuclear Energy Institute
  - *Biff Bradley*
  - *Tony Pietrangelo*
- 10 CFR 50.69 WOG Pilot Plant Application Team Members
  - *Jason Brown* – Westinghouse
  - *Bob Lutz* – Westinghouse
  - *Glen Schinzel* – South Texas
  - *Barry Sloane* – Dominion Energy VA
  - *Jack Winebrenner* – Dominion Energy VA
  - *Mo Dingler* – Wolf Creek Nuclear Operating Company

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2

Attachment 5

Topics For Discussion

- Purpose of Meeting
- Background and Status of Code Case
- Overview of Key NRC Comments
- Discussion and Proposed Resolution of Key NRC Comments
- Future Actions

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3

Purpose of Meeting

- To address outstanding NRC comments on ASME Code Case N-660b
- This open dialogue is being used to address key NRC technical issues early in the consensus standards development process so that NRC exceptions are minimized in the regulatory endorsement process once the action is approved by ASME
- While NRC is a key stakeholder, comments from all ASME stakeholders are being addressed per the consensus standards development process to support worldwide use of this Code Case

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4

ASME Code Case N-660b - Status

- Code Case N-660 approved by ASME Board on Nuclear Codes and Standards June 2002; NRC expected to endorse Code Case N-660 in Regulatory Guide 1.147, Revision 14, soon to be published.
- Early use of N-660 in 2003 for service water systems (SWS) showed overly conservative results; Code Case N-660a developed for SWS only
- N-660 used for trial use at Surry and Wolf Creek as part of WOG 50.69 pilot program (Fall of 2003 and Summer/Fall of 2004)
  - Wolf Creek systems – Containment Spray and Control Building Ventilation
  - Surry systems – Chemical & Volume Control System and Component Cooling Water System
- Feedback from IDP during use of N-660 Rev. 0 resulted in N-660b; the IDP was prevented from assigning a LSS to pressure boundary items even though the collective knowledge and experience supported a LSS.

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5

ASME Code Case N-660b - Status (cont)

- Code Case N-660b issued to ASME Section XI Working Group for letter ballot November 2004
- Comments discussed during ASME Boiler Code meetings in December 2004; Additional resolution required
- ASME/NRC meeting held February 10, 2005 to discuss proposed comment resolutions for N-660b before going forward with another ASME vote
- Issued to ASME Section XI Working Group for letter ballot following February 28, 2005 ASME Boiler Code meeting with comments also being sought from –
  - ASME Section XI WG on Repair/Replacement/Modifications
  - ASME Section XI Subgroup Water-Cooled Systems
  - ASME Committee on Nuclear Risk Management
- Comments discussed during ASME Boiler Code meetings in May 2005; Additional resolution required from feedback
- Additional comments received from NRC, July 12, 2005

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6

Overview of Key NRC Comments

Comments Specific to Code Case N-660b

- Demonstrating technical adequacy of PRA
- Technical basis for how plant configuration and design insights preclude possibility of a large pressure boundary failure
- Criteria for crediting operator actions
  - Reliance on non-safety related equipment
  - Need to not assume single failure

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7

Overview of Key NRC Comments (Cont)

Recent Comments on Code Case N-660b with Implications on Approved Code Case N-660

- Limiting scope of Code Case to plants designed and constructed to ASME Section III
- Ability to credit operator actions
- Requirements for dose limits as a result of unintentional releases
- Definition of end points of piping segments as anchor to anchor

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8

Discussion and Proposed Resolution of Key NRC Comments

Summary of Key Proposed Changes to Case

- Tied PRA adequacy to PRA requirements of 10CFR50.69
- Removed Appendix II - clarifying criteria provided
- Removed criterion for large leak failure probability
- Provided consistency between NEI 00-04 and N-660b LSS considerations (including unintentional releases)
- Provided consistency between ASME PRA Standard and N-660b glossary terms
- Paragraph 1200, Classifications, returned to original text – items within BER shall be HSS

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9

Future Actions

- All results from discussion at today's meeting will be presented at August 2005 ASME Section XI Working Groups and discussed with respective management committees
- NRC will be informed of ASME Boiler and Pressure Vessel Code discussions on these actions via participation of NRC Staff in Code meetings per normal process
- ASME Board on Nuclear Codes and Standards will continue to monitor the development of these standards actions as part of the Regulatory Endorsement Task Group efforts
- Efforts will continue to be made to successfully resolve any outstanding comments in order to show progress and to encourage additional trial applications of Code Case N-660, particularly in support of 10 CFR 50.69

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10

LETTER BALLOT COMMENT FORM

COMMITTEE: BPV SC-XI WG IRBE SUBMITTED BY: Syed Ali (NRC) DATE: April 25, 2005

LETTER BALLOT # BC04-1505 SUBJECT: N-660b

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
General	D	The White Paper provides a redline/strikeout comparison of the current version of the code case with an unofficial version of the code case (N-658). The proper comparison would be to the last approved version of the code case (N-660, Rev 0).	The White Paper will be replaced with a redline/strikeout comparison to N-660, Rev 0.
-1200(a)	D	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.	Reference to 50.69 in the White Paper will be removed. This Case is a stand-alone action that can be used with or without 50.69.
-1200(a)	D	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.	Failure potential remains to be 1.0 for large leaks as well as lesser leaks. The justification for assuming a lesser leak rather than large pressure boundary failure is not based on failure potential but the operating condition coupled with a well engineered design of the piping system.
-1300	D	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–660 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 inservice inspection programs without submitting request for relief from the ASME requirements in 10CFR50.55a. Licensees may implement the approved code case N–660 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.	Because of the obligation ASME has with international users, the PRA quality section must remain at a high level and not refer to documentation that may not be applicable for users outside of the United States. However, the footnote has been revised to include 50.69 as an example of an acceptable means of prescribed acceptance criteria for PRA technical adequacy.

KEY: D - SIGNIFIES NEGATIVE COMMENTS C - SIGNIFIES COMMENTS OTHER THAN NEGATIVES

Attachment 6

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
		The quality requirements of N–660 are at a higher level and less specific than the Appendix R PRA quality requirements. The categorization in N–660, 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–660 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–660-2 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.	
I-1.0	D	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.	The RISC process continues to be based on conditional consequence of failure. The probability of failure is not considered in justifying small leaks. The reference to failure probabilities in Method A programs has been removed. Rather than assuming a large pressure boundary failure, a smaller leak is assumed based on the operating conditions coupled with a well engineered design of the piping system.
I-3.0	D	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.	"common" replaced with "similar"

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I-3.0	D	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).	Text revised to include 3.1 and 3.2.
I-3.1.1(a)	D	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.	It was not the intent of the previous revision of this Code Case to allow pipe segments to be ranked as LSS solely based on the conclusion that the potential for a large break is negligible. Furthermore, it was not intended to rely heavily on programmatic activities in direct contradiction of the defense in depth principles articulated in RG 1.174. The basis for postulating small leaks in moderate energy systems is that the operating conditions (e.g., temperature, pressure) coupled with a well engineered design (i.e., national standards) assures incredibly low likelihood of large pressure boundary failures for moderate energy systems. All segments are still subject to the consequence evaluation and deterministic evaluation (if not already HSS by the consequence evaluation). In an effort to more clearly define the intent of this Code Case, Appendix II will be removed and paragraph I-3.1.1(a)(4) will be rewritten to state, "*applied to moderate energy systems that have been designed and constructed to the requirements (i.e. administrative and technical) of one of the following codes or standards applicable to that item: ASME, ANSI,*

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			AWS AISC, AWWA, API-650, API-620, MSS-SPs, TEMA, and those standards referenced within these documents." Furthermore, terminology of large and small leak has been replaced with system disabling leak and lesser leak. Also, a footnote has been added , "The relative size of a lesser leak shall be a function of the location-specific system design and operating characteristics (i.e., operating temperature, pressure, flowrate, material, wall thickness) and shall be benchmarked against plant-specific operating experience."
I-3.1.1(a)	D	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 Inservice 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.	This statement referring to probability of large leak will be removed from Case based on additional comments and reconsideration of the technical justification.
I-3.1.1(a)(3)	D	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.	Will add "flow restricting" before "orifice" to clarify type of orifice.
I-3.1.1(e)	D	Section I-3.1.1(c) 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.	The Case states that automatic and operator actions shall be evaluated consistent with assumptions made in the PRA (see paragraph -1330). The

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			statement stating "Automatic actions need not be safety related nor subject to single failure, however, the equipment relied on the perform automatic actions shall be of the same level of quality as the equipment being supported or protected" has been removed because of confusion and redundancy.
I-3.1.2	D	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.	Clarification made to I-3.1.2 to state that there are three "core damage" impact groups and in addition, failures shall also be evaluated for their importance relative to containment performance.
I-3.1.2	D	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 Wolf Creek pilot applications, and if so, how was it considered.	This information was considered in the Surry pilot application. The fire and seismic PRAs were reviewed to see if those assessments identified any risk outliers not already done so by the internal events PRA. None were identified so the RISC process relied on the conditional consequence assessment from the internal PRA model. Explanation of this will be added to the supporting Whitepaper.
I-3.1.2	D	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.	The PRA update would ensure technical assumptions are maintained. Section 4.0 states the update requirements.
I-3.2.2(b)(2)	D	Section I-3.2.2(b)(2) should specify which plant procedures (i.e., normal operating procedures) are being referred to in this evaluation.	The current words which do not refer to any specific plant procedures are consistent with NEI 00-04. To keep consistency no change will be made.
I-3.2.2	D	Section I-3.2.2 lists several conditions that would cause an item to be categorized as	WG voted to make the change as

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		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."	suggested and keep consistency between this Case and NEI 00-04.
I-4.0	D	Section I-4.0 states that plant design changes shall be screened prior to implementation .... Similarly, Section I-4.0 (a), (b), and (c) 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.	Agree – modification made.
I-4.0	D	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.	Section I-4.0 states that the risk-informed safety classification "shall be reevaluated on the basis of inspection periods and inspection intervals that coincide with the inspection program requirements for Inspection Program A or B of IWA-2431 or IWA-2432, as applicable." The WG feels that the current text is sufficient and clearly states the requirements for reevaluation. No change will be made.
Appendix II	D	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.	Appendix II has been removed. See response to comment on paragraph I-3.1.1(a) above.

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Additional comments from Gene Imbro – dated 3/25/05, received 7/12/05

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.

RESPONSE: Need additional basis for comment – no change has been made to Rev 0 – Code Case has been tested successfully at Surry (B31.1 plant design)

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.

RESPONSE: This was not the intent – paragraph has been replaced with original Rev 0 text – clarifier added that "all items within the break exclusion region... shall be classified as HSS."

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.
RESPONSE: Section I-3.1(b) allows credit to be taken for automatic as well as operator actions – no change

b. Core damage - ~~uncovery and~~ heatup 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.
RESPONSE: glossary term is consistent with ASME PRA Standard – no change

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
RESPONSE: change made

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).
RESPONSE: change made

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 1F, and
b. maximum operating pressure exceeds 275 psi

RESPONSE: *change made*

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.

RESPONSE: *change made*

g. Train - As defined in this appendix, Aa 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~~

RESPONSE: *Revised term clarified to define 0.5 train unavailability, "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. A half train (0.5 trains) shall have a mean unavailability of 1E-01, 1.5 trains shall have a mean unavailability of 1E-03, etc."*

4. **I-3.0 Consequence Evaluation**

a. This paragraph states: AAll 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.

RESPONSE: *Further discussion and clarification of comment is required. Approved version of N-660 follows RI-ISI methodology of grouping pipe segments based on similar consequences. While the design may change, the design basis does not change with repair/replacement.*

b. This paragraph states: AThroughout 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≅.

> 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.

RESPONSE: *The process, as is, is consistent with 1.174 and does not allow "over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided." Crediting operator actions is standard practice for risk-informed applications.*

i. Operator action in this Code Case is assumed to be 100% successful in all cases as long as the criteria of 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.

RESPONSE: *Operator action is not assumed to be 100% successful. If operator actions are credited, the likelihood for success and failure needs to be determined and both cases evaluated and used for the final categorization and the more limiting case is used for final categorization. Clarification provided in Section I-3.0.*

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

RESPONSE: *The PRA credits actions and equipment that are not safety-related and is maintained to best reflect the plant design and operations. The PRA is updated to reflect changes in model assumptions and updates are evaluated to determine any changes in the risk categorization. Reference has been added to the PRA technical adequacy of paragraph 1330.*

5. **I-3.1.1 Failure Modes and Effects Analysis (FMEA).** This paragraph states: APotential 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 Alarge 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.

RESPONSE: *In paragraph I-3.1.1(a), "large pressure boundary leak" replaced with "system disabling leak" and "smaller leak" replaced with "lesser leak". Also, the following will be added as a footnote to "lesser leak", "The relative size of a lesser leak shall be a function of the location-specific system design and operating characteristics (i.e., operating temperature, pressure, flowrate, material, wall thickness) and shall be benchmarked against plant-specific operating experience."*

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.

RESPONSE: *Appendix II has been removed from the Code Case. Paragraph (a)(4) has been replaced with "applied to moderate energy systems that have been designed and constructed to the requirements (i.e. administrative and technical) of one of the following codes or standards applicable to that item: ASME, ANSI, AWS AISC, AWWA, API-650, API-620, MSS-SPs, TEMA, and those standards referenced within these documents."*

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 <u>automatically</u> on a given signal ~~or by operator action~~.

RESPONSE: *Statement unchanged from approved version of N-660. Task group feels it's unnecessary to make change based on fact that operator action is credible in other risk-informed applications. Also, most low energy systems do not have automatic isolation features.*

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.
RESPONSE: The point made of risk-informed applications is that by not assuming single failure we assess "true risk" as well as don't guarantee success like in a solely deterministic application. Last statement has been removed to avoid confusion and reference has been made to the PRA technical adequacy section, -1330.

i. AShall be evaluated≅ is meaningless unless there are a specified set of criteria for the item to be evaluated against. No criteria are provided.
RESPONSE: Criteria established in PRA and Table I-2.

e. System Redundancy. The existence of redundancy for accident mitigation purposes shall be considered.AShall be considered≅ is meaningless.
RESPONSE: Criteria established in PRA and Table I-2.

6. I-3.2.2(b)(2) Classification Considerations states that AFailure 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.

RESPONSE: The intent is NOT to allow reliance on operator actions outside plant procedures to achieve or maintain safe shutdown. The user should assume that the plant would be unable to reach or maintain safe shutdown once outside plant procedures.

7. I-3.2.2(b)(5) Classification Considerations states that AFailure 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.

RESPONSE: Considerable effort has been made to provide consistency between NEI 00-04 (50.69 guidance) and N-660. Since the March version of N-660, this statement has been modified as such, "Failure of the pressure boundary function will not result in an unintentional release of radioactive material that would result in the implementation of offsite radiological protective actions." This change appropriately answers a similar comment from SAli made on 4/25/05.

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

RESPONSE: *See revised definition for "train"*

Case N-~~658~~660b
Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities
Section XI, Division 1

Inquiry: What additional classification criteria may be used as a supplement to the group classification criteria of IWA-1320 to determine Risk-Informed Safety Classification for use in risk-informed repair/replacement activities?

Reply: It is the opinion of the Committee that as a supplement to the group classification criteria of IWA-1320, the following requirements may be used to determine the Risk-Informed Safety Classification for risk-informed repair/replacement activities.

[Applicability: 1980 Edition with Winter 1981 Addenda through ~~2001~~ 2004 Edition]

-1000 SCOPE AND RESPONSIBILITY

-1100 Scope

This Case provides a process for determining the Risk-Informed Safety Classification (RISC) for use in risk-informed repair/replacement activities. The RISC process of this Case may be applied to any of Class 1, 2, 3, or non-class[1] pressure-retaining items or their associated supports, except core supports, in accordance with the risk-informed safety classification criteria established by the regulatory authority having jurisdiction at the plant site.

-1200 Classifications

(a) The RISC process is described in Appendix I of this Case. Pressure retaining and component support items shall be classified High Safety Significant (HSS) or Low Safety Significant (LSS). ~~However, because~~Because this classification is to be used only for repair/replacement activities, failure potential is conservatively assumed to be 1.0 in ~~determining a~~performing the consequence ~~category~~ evaluation per I-3.0 in Appendix I. These classifications might not be directly related to other risk-informed applications.

(b) Class 1 items that are part of the reactor coolant pressure boundary except as provided in paragraphs (c)(2)(i) and (c)(2)(ii) of Title 10 of the U.S. Code of Federal Regulations (10 CFR), Part 50.55a shall be classified High Safety Significant (HSS). For items that are connected to the reactor coolant pressure boundary, as defined in paragraph 10 CFR 50.55a (c)(2)(i) and (c)(2)(ii), the RISC process of (a) should be

[1] Non-class items are items not classified in accordance with IWA-1320.

Attachment 7

applied. All items that are within the break exclusion region[2] [>NPS 4 (DN 100)] shall be classified as HSS.

-1300 OWNER'S RESPONSIBILITY

-1310 Determination of Classification

The responsibilities of the Owner shall include determination of the appropriate classification for the items identified for each risk-informed repair/replacement activity, in accordance with Appendix I of this Case. The Owner shall ensure that core damage frequency (CDF) and large early release frequency (LERF) are included as risk metrics in the RISC process.

-1320 Required Disciplines

Personnel with expertise in the following disciplines shall be included in the classification process.

~~(c)~~(a)probabilistic risk assessment (PRA)
~~(d)~~(b) plant operations
~~(e)~~(c)system design
~~(f)~~(d)safety or accident analysis

Personnel may be experts in more than one discipline, but are not required to be experts in all disciplines.

-1330 ~~Adequacy of the PRA~~PRA Scope and Technical Adequacy

The PRA must at a minimum model severe accident scenarios resulting from internal initiating events occurring at full power operation. PRA or qualitative approaches that evaluate the plant for external events, low power, and shutdown must also be considered. The PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a review process assessed against a standard[3] or set of acceptance criteria[4] that is endorsed by the regulatory agency having jurisdiction over the plant site. ~~The Owner is responsible for demonstrating adequacy of any PRA used as the basis for this process. All deficiencies identified shall be reconciled during the analysis to support the RISC process. The resolution of all PRA issues shall be documented.~~

-9000 GLOSSARY

basic safety function – one of the key safety functions of the plant; reactivity control, core cooling, heat sink, and RCS inventory

completion time (CT) – the amount of time allowed for completing a required action. In the context of this Case, the required action is to restore operability (as defined in the technical specifications) to the affected system or equipment train

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[2] NUREG-0800, Sections 3.6.1 and 3.6.2 define an acceptable definition of break exclusion region.

[3] A standard that may serve as an example for this application is ASME RA-S-2002, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications* with the RA-Sa-2003 Addenda and the RA-Sb-2005 Addenda. This standard sets forth requirements for PRAs used to support risk-informed decisions for commercial nuclear power plants and prescribes a method for applying these requirements for various categories of applications.

[4] A set of acceptance criteria that may be considered as an example is prescribed in 10 CFR 50.69.

conditional consequence – an estimate of an undesired consequence, such as core damage or a breach of containment, assuming failure of an item~~,~~ (e.g., conditional core damage probability (CCDP))

conditional core damage probability (CCDP) – an estimate of the probability of an undesired consequence of core damage given a specific failure (e.g., piping segment failure)

conditional large early release probability (CLERP) – an estimate of the probability of an undesired consequence of large early release (i.e., breach of containment) given a specific failure (e.g., piping segment failure)

containment barrier – a component(s) that provides a containment boundary/isolation function including normally closed valves or valves that are designed to go closed upon actuation

core damage – uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage involving a large fraction of the core is anticipated~~and involving enough of the core to cause a significant release~~

failure – an event involving leakage, rupture, or ~~a~~other condition that would prevent ~~disable the ability of~~ an item ~~to perform~~ from performing its intended safety function

failure mode – a specific functional manifestation of a failure (i.e., the means by which an observer can determine that a failure has occurred) by precluding the successful operation of a piece of equipment, a component, or a system (e.g., fails to start, fails to run, leaks)

failure modes and effects analysis (FMEA) – a process for identifying failure modes of specific items and evaluating their effects on other components, subsystems, and systems

failure potential – likelihood of ruptures or leakage that result in a reduction or loss of the pressure-retaining capability of the 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)

high-energy systems – those systems that are either in operation or maintained pressurized under conditions where either or both of the following are met:

a. operating temperature exceeds 200 °F

b. operating pressure exceeds 275 psi

high-safety-significant (HSS) function – a function that has been determined to be safety significant from ~~traditional plant risk assessment evaluations of core damage or large early release events (e.g., evaluations performed to support the Maintenance Rule – 10 CFR 50.65).~~the plant probabilistic risk assessment or from other relevant information (e.g., defense in depth considerations)

initiating event (IE) – any event either internal or external to the plant that perturbs the steady state operation of the plant~~, if operating,~~ thereby initiating an abnormal event, such as a transient or LOCA within the plant. Initiating events trigger sequences of events that challenge plant control and safety systems whose failure could potentially lead to core damage or large early release

large early release – the rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of off-site emergency response and protective actions

low-safety-significant (LSS) function – a function not determined to be high-safety significant from the plant probabilistic risk assessment or from other relevant information (e.g., defense in depth considerations)

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. 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.

piping segment – a portion of piping, components, or a combination thereof, and their supports, in which a failure at any location results in the same consequence, e.g., loss of a system, loss of a pump train

plant mitigative features – systems, structures, and components that can be relied on to prevent an accident or that can be used to mitigate the consequences of an accident

probabilistic risk assessment (PRA) – a qualitative and quantitative assessment of the risk associated with plant operation and maintenance that is measured in terms of frequency of occurrence of risk metrics, such as core damage or a radioactive material release and its effects on the health of the public (also referred to as a probabilistic safety assessment, PSA)

~~recovery~~ **operator action** – a human action performed to regain equipment or system operability from a specific failure or human error in order to mitigate or reduce the consequences of the failure

risk metrics – a determination of what activity or conditions produce the risk, and what individual, group, or property is affected by the risk

spatial effect – a failure consequence affecting other systems or components, such as failures due to pipe whip, jet impingement, jet spray, harsh environment, debris generation or flooding

success criteria – criteria for establishing the minimum number or combination of systems or components required to operate, or minimum levels of performance per component during a specific period of time ~~(mission time)~~, to ensure that the safety functions are satisfied

train – As used 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. A half train (0.5 trains) shall have a mean unavailability of 1E-01, 1.5 trains shall have a mean unavailability of 1E-03, etc.

unaffected backup trains – a train(s) that is not adversely impacted (i.e., failed or degraded) by the postulated piping failure in the FMEA evaluation. Impacts can be caused by direct or indirect effects of the postulated piping failure.

APPENDIX I RISK-INFORMED SAFETY -CLASSIFICATION (RISC) PROCESS

I-1.0 INTRODUCTION

This Appendix ~~provides~~ describes the risk-informed process used to determine Risk-Informed Safety Classification (RISC) for use in risk-informed repair/replacement

activities. This RISC process is based on conditional consequence of failure.~~The process provides a conservative assessment of the importance of an item.~~ This process divides each selected system into piping segments that are determined to have similar consequence of failure. These piping segments are categorized based on the conditional consequence. Once categorized, the safety significance of each piping segment is identified. Figure I-1 illustrates the RISC methodology presented in the following sections.

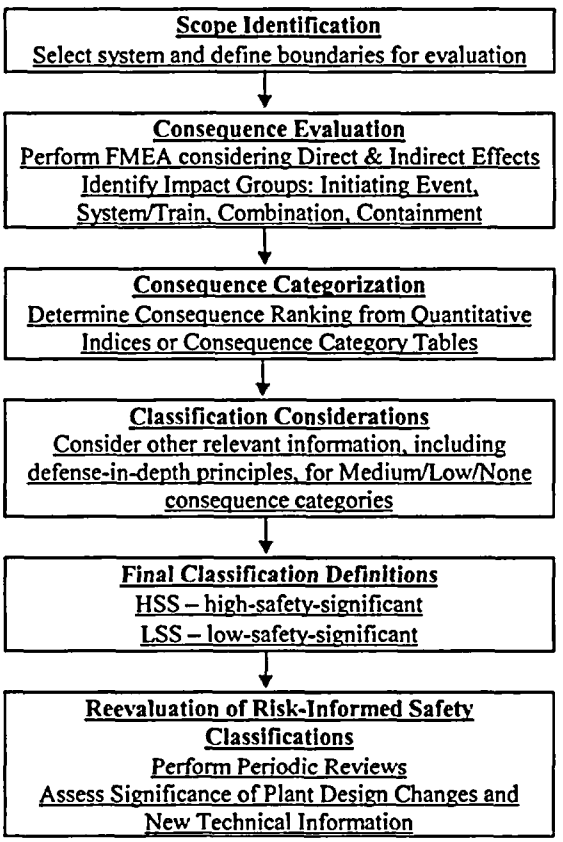

Figure I-1
Risk-Informed Safety Classification Process

I-2.0 SCOPE IDENTIFICATION

The Owner shall define the boundaries included in the scope of the RISC evaluation process.

I-3.0 CONSEQUENCE ~~ASSESSMENT~~EVALUATION

All pressure retaining items and their supports shall be evaluated by defining piping~~Piping segments can be~~ segments that are grouped based on ~~common~~ similar conditional consequence (i.e., given failure of the piping segment). To accomplish this grouping, ~~the~~ direct ~~effects,~~ and indirect effects shall be assessed for each piping segment. A Consequence Category for each piping segment is determined from the

Consequence Evaluation as defined in I-3.1.1 and I-3.1.2. The failure consequence can be quantified using the available PRA(s). Throughout the evaluations of I-3.0, 3.1, and 3.2, 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. When crediting operator action, the likelihood for success and failure needs to be determined and both cases evaluated and used for the final categorization. To take credit for operator actions, the following features shall be provided:

- an alarm or other system to provide clear indication of failure,
- equipment activated to recover from the condition must not be affected by the failure,
- time duration and resources are sufficient to perform operator action,
- plant procedures to define operator actions, and
- operator training in the procedures.

To determine that the consequence evaluation and considerations are sufficient for the RISC process, the requirements of the following subparagraphs shall be met. ~~Additionally, information shall be collected for each piping segment that is not modeled in the PRA, but considered relevant to the classification (e.g., information regarding design basis accidents, shutdown risk, containment isolation, flooding, fires, seismic conditions).~~

I-3.1 ~~Consequence Evaluation~~Analysis and Assessments

I-3.1.1 Failure Modes and Effects Analysis (FMEA). 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~~system disabling ~~failure~~ leak for piping segments. Alternatively, the consequence analysis can be performed assuming a ~~smaller~~ lesser leak[5], when

(1) a ~~smaller~~ lesser leak is more conservative; or

(2) ~~when~~ a ~~small~~ lesser leak can be justified through a leak-before-break analysis in accordance with the criteria specified in appropriate documentation acceptable to the regulatory agency having jurisdiction over the plant site; or

(3) it can be documented that a physical configuration precludes the possibility of a large pressure boundary leak (e.g., flow restricting orifice); or

(4) applied to moderate energy systems that have been designed and constructed to the requirements (i.e. administrative and technical) of one of the following codes or standards applicable to that item: ASME, ANSI, AWS AISC, AWWA, API-650, API-620, MSS-SPs, TEMA, and those standards referenced within these documents. ~~1061, Volume 3; 10CFR50, Appendix A, General Design Criterion 4; or~~

~~(3)it can be documented that plant configuration precludes the possibility of a large pressure boundary failure.~~

[5] The relative size of a lesser leak shall be a function of the location-specific system design and operating characteristics (i.e., operating temperature, pressure, flowrate, material, wall thickness) and shall be benchmarked against plant-specific operating experience.

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

(c) Indirect Effects. These include spatial ~~interactions~~ effects ~~such as pipe whip, jet spray,~~ and loss-of-inventory effects (e.g., draining of a tank).

(d) Initiating Events. For systems or piping segments that are modeled either explicitly or implicitly in any existing plant-specific PRA, any applicable initiating event is identified using a list of initiating events from that PRA and the plant design basis.~~These are identified using a list of initiating events from any existing plant-specific Probabilistic Risk Assessment (PRA) or Individual Plant Examination (IPE) and the Owner's Requirements.~~

(e) 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 consistent with the PRA information as required in paragraph – 1330.

(f) System Redundancy. The existence of redundancy for accident mitigation purposes shall be considered.

I-3.1.2 Impact Group Assessment. The results of the FMEA evaluation for each piping system, or portion thereof, shall be classified into one of three core damage impact groups: initiating event, system, or combination. In addition, failures shall also be evaluated for their importance relative to containment performance. Each piping system, or portion thereof, shall be partitioned into postulated piping failures that cause an initiating event, disable a system without causing an initiating event, or cause an initiating event and disable a system. The consequence category assignment (high, medium, low, or none) for each piping segment within each impact group shall be selected in accordance with (a) through (d) below. Available risk information related to the mitigation of fire, seismic, shutdown, and other external events shall be considered.

(a) Initiating Event (IE) Impact Group Assessment. When the postulated failure results in only an initiating event (e.g., loss of feedwater, reactor trip), the consequence shall be classified into one of four categories: high, medium, low, or none. The initiating event category shall be assigned according to the following:

(1) The initiating event shall be placed in one of the ~~categories~~ Design Basis Event Categories in Table I-1. All applicable design basis events previously analyzed in the Owner's updated final safety analysis report~~,~~ or PRA~~, or IPE~~ shall be included.

(2) Breaks that cause an initiating event classified as Category I (routine operation) need not be considered in this analysis.

(3) For piping segment breaks that result in Category II (Anticipated Event), Category III (Infrequent Event), or Category IV (Limiting Fault or Accident), the consequence category shall be assigned to the initiating event according to the conditional core damage probability (CCDP) criteria specified in Table I-5. The quantitative index for the initiating event impact group ~~(CCDP)~~ is the ratio of the core damage frequency due to the initiating event to the initiating event frequency.

(b) System Impact Group Assessment. The consequence category of a failure that does not cause an initiating event, but degrades or fails a system essential to prevention of core damage, shall be based on the following:

(1) Frequency of challenge that determines how often the ~~mitigating~~ affected function of the system is called upon. This corresponds to the frequency of ~~initiating~~ events that require the system operation.

(2) Number of backup systems (portions of systems, trains, or portions of trains) available, which determines how many unaffected systems (portions of systems, trains, or portions of trains) are available to perform the same mitigating function as the degraded or failed systems.

(3) Exposure time, which determines the time the system would be unavailable before the plant is changed to a different mode in which the failed system's function is no longer required, the failure is recovered, or other compensatory action is taken. Exposure time is a function of the detection time and ~~Allowed Outage Time~~completion time, as defined in the plant Technical Specification.

Consequence categories shall be assigned in accordance with Table I-2 as High, Medium, or Low. Frequency of challenge is grouped into design basis event categories II, III, and IV. ~~Exposure time shall be obtained from Technical Specification limits.~~ 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. ~~In lieu of Table I-2, quantitative~~ Quantitative indices may be used to assign consequence categories in accordance with Table I-5 in lieu of Table I-2. The quantitative index for the system impact group is the product of the change in conditional core damage frequency (CCDF) and the exposure time.

(c) Combination Impact Group Assessment. The consequence category for a piping segment whose failure results in both an initiating event and the degradation or loss of a system shall be determined using Table I-3. The Owner or his designee shall ensure that the quantitative basis of Table I-3 (e.g., one full train unavailability approximately 10^{-2}) is consistent with the pipe failure scenario being evaluated. The consequence category is a function of two factors:

(1) Use of the system to mitigate the induced initiating event;

(2) Number of unaffected backup systems or trains available to perform the same function.

~~In lieu of Table I-3, quantitative~~ Quantitative indices may be used to assign consequence categories in accordance with Table I-5 in lieu of Table I-3.

(d) Containment Performance Impact Group Assessment. The above evaluations determine failure importance relative to core damage. Failures shall also be evaluated for ~~its~~their importance relative to ~~effect on~~ containment performance. This shall be evaluated as follows.

(1) For postulated failures which do not result in a LOCA which bypasses containment, the quantitative indices of Table I-5 for CLERP shall be used.

(2) Table I-4 shall be used to assign consequence categories for those piping failures that can lead to a LOCA, which bypasses containment.

~~accomplished by addressing two issues, both of which are based on an approximate conditional probability value of not greater than 0.1 between the CCDP and the likelihood of large early release from containment. If there is no margin, i.e.,~~

~~conditional probability of a large early release due to core damage is greater than 0.1, the assigned consequence category shall be increased one level. The two issues are described as follows:~~

~~(1)CCDP values for initiating events and safety functions are evaluated to determine if the potential for large early release due to containment failure requires the consequence category to be increased.~~

~~(2)The effect on containment isolation is evaluated. If there is a containment barrier available, the consequence category from the core damage assessment is retained. If there is no containment barrier or the barrier failed in determining the consequence category from the core damage assessment, some margin in the core damage consequence category assignment must be present for it to be retained.~~

~~For example, if the CCDP for core damage is less than 10^{-5}, i.e., a Medium consequence assignment, and there is no containment barrier, the Medium consequence assignment is retained, because there is 0.1 margin to the High consequence category threshold, i.e., 10^{-4}. However, if the CCDP for core damage is $5x10^{-5}$, i.e., a Medium consequence assignment, and there is no containment barrier, the consequence category is increased to High, because the margin to the High consequence category threshold, i.e., 10^{-4}, is less than 0.1. Table I-4 shall be used to assign consequence categories for those piping failures that can lead to a LOCA outside containment. In lieu of using Table I-4, quantitative indices may be used to assign consequence categories in accordance with Table I-5 with each range lowered one order of magnitude, e.g., not less than 10^{-5} is High.~~

~~**I-3.1.3 Piping segments, Functions, and Design, Operational, or Risk Considerations Not Modeled in PRA.** If any of the conditions in (a) or (b) below are true, the piping shall be classified HSS.~~

~~(a)For piping segments, functions, and design, operational, or risk considerations that are not explicitly modeled in the PRA, the effects of the following shall be evaluated.~~

~~(1)Failure of the piping segment will significantly increase the frequency of an initiating event, including those initiating events originally screened out in the PRA, such that the CDF or large early release frequency (LERF) would be estimated to increase by more than 10^{-6}/yr or 10^{-7}/yr, respectively.~~

~~(2)Failure of the piping segment will compromise the integrity of the reactor coolant pressure boundary as defined in -1200(b).~~

~~(3)Even when considering operator actions used to mitigate an accident, failure of the piping segment will fail a high safety significant function.~~

~~(4)Failure of the piping segment will result in failure of other safety significant piping segments, e.g., through indirect effects.~~

~~(5)Failure of the piping segment will prevent or adversely affect the plant's capability to reach or maintain safe shutdown conditions.~~

~~(b)In addition to being HSS in terms of their contribution to CDF or LERF, piping segments might also be HSS in terms of other risk metrics or conditions. Therefore, the following conditions shall be evaluated.~~

~~(1)The piping segment is a part of a system that acts as a barrier to fission product release during severe accidents.~~
~~(2)The piping segment supports a significant mitigating or diagnosis function addressed in the Emergency Operating Procedures or the Severe Accident Management Guidelines.~~
~~(3)Failure of the piping segment will result in unintentional releases of radioactive material in excess of plant offsite dose limits specified in 10 CFR Part 100.~~

~~**I-3.1.4 Maintain Defense-in-Depth.** When categorizing piping segments LSS, the RISC process shall demonstrate that the defense-in-depth philosophy is maintained. Defense-in-depth may be demonstrated by following the guidelines of U.S.N.R.C. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," dated July 1998.~~

~~**I-3.1.5 Maintenance of Adequate Safety Margins.** When categorizing piping segments LSS, the RISC process shall verify that there are sufficient safety margins to account for uncertainty in the engineering analysis and in the supporting data. Safety margin shall be incorporated when determining performance characteristics and parameters, e.g., piping segment, system, and plant capability or success criteria. The amount of margin should depend on the uncertainty associated with the performance parameters in question, the availability of alternatives to compensate for adverse performance, and the consequences of failure to meet the performance goals. Sufficient safety margins are maintained by ensuring that safety analysis acceptance criteria in the plant licensing basis are met, or proposed revisions account for analysis and data uncertainty.~~

I-3.2 Classification

I-3.2.1 Final Risk-Informed Safety Classification. Piping segments may be grouped together within a system, if the consequence evaluation (I-3.1) determines the effect of the postulated failures to be the same. The Risk-Informed Safety Classification shall be as follows:

Classification Definitions

HSS – Piping segment considered high-safety-significant
LSS – Piping segment considered low-safety-significant

I-3.2.2 Classification Considerations.

(a) Piping segments determined to be a High consequence category in any table by the consequence evaluation in I-3.1~~(I-3.1.1 and I-3.1.2)~~ shall be considered HSS.~~ The Owner may further refine the classification ranking by more extensive application of the process defined in these requirements. These analyses shall be documented.~~
(b) Piping segments determined to be a Medium, Low, or None (no change to base case) consequence category in any table by the consequence evaluation in I-3.1~~(I-3.1.1) and (I-3.1.2)~~ shall be determined HSS or LSS by considering the ~~RISC evaluation and the~~ other relevant information ~~(I-3.1.3, I-3.1.4, and I-3.1.5) provided~~ for determining classification.~~ Any piping segment initially determined to be a~~

~~Medium consequence category and that is subject to a known active degradation mechanism shall be classified HSS.~~ Under the same conditions of I-3.1.1(a), a large pressure boundary leak does not need to be assumed. Also, 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. The following conditions shall be evaluated and answered TRUE or FALSE.

(1) Failure of the pressure boundary function will not directly or indirectly (e.g., through spatial effects) fail a basic function.
(2) 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.
(3) The pressure boundary function is not called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means for the successful performance of operator actions required to mitigate an accident or transient.
(4) The pressure boundary function is not called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means for assuring long term containment integrity, monitoring of post-accident conditions, or offsite emergency planning activities.
(5) Failure of the pressure boundary function will not result in an unintentional release of radioactive material that would result in the implementation of offsite radiological protective actions.

The RISC process shall demonstrate that the defense-in-depth philosophy is maintained. Defense-in-depth may be demonstrated by following acceptable guidelines of the regulatory agency having jurisdiction. Defense-in-depth is maintained if:

(6) Reasonable balance is preserved among prevention of core damage, prevention of containment failure or bypass, and mitigation of an offsite release.
(7) There is no over-reliance on programmatic activities and operator actions to compensate for weaknesses in the plant design.
(8) System redundancy, independence, and diversity are preserved commensurate with the expected frequency of challenges, consequences of failure of the system, and associated uncertainties in determining these parameters.
(9) Potential for common cause failures is taken into account in the risk analysis categorization.
(10) Independence of fission-product barriers is not degraded.

If any of the above ten (10) conditions are answered FALSE, then HSS shall be assigned. Otherwise, LSS may be assigned.

(c) If LSS has been assigned from I-3.2.2(b), then the RISC process shall verify that there are sufficient safety margins to account for uncertainty in the engineering analysis and in the supporting data. Safety margin shall be incorporated when determining performance characteristics and parameters, e.g., piping segment,

system, and plant capability or success criteria. The amount of margin should depend on the uncertainty associated with the performance parameters in question, the availability of alternatives to compensate for adverse performance, and the consequences of failure to meet the performance goals. Sufficient safety margins are maintained by ensuring that safety analysis acceptance criteria in the plant licensing basis are met, or proposed revisions account for analysis and data uncertainty.

If sufficient safety margins are maintained then LSS should be assigned; if not, then HSS shall be assigned.~~Piping segments and their associated supports determined to be consequence category Low or None (no change to base case) by the consequence evaluation (I-3.1.1 and I-3.1.2) and not meeting (a) or (b) above in any table, or not modeled, shall be determined HSS or LSS using the other relevant information (I-3.1.3, I-3.1.4, and I-3.1.5).~~

(d) A component support, hanger, or snubber shall have the same classification as the highest-ranked piping segment within the piping analytical model in which the support is included.~~ The Owner may further refine the classification ranking by more extensive application of the process defined in these requirements. These analyses shall be documented.~~

I-4.0 Reevaluation of Risk-Informed Safety Classifications

The assessment of potential equipment performance changes and new technical information shall be performed during the normally scheduled periodic review cycle. Plant design changes shall be screened prior to implementation to determine if they would result in a significant change to the plant risk profile. Plant procedures shall be modified to require that these evaluations will be performed. If significant changes to the plant risk profile are identified, or if it is identified that a low-safety-significant SSC can (or actually did) prevent a safety significant function from being satisfied, an immediate evaluation and review shall be performed prior to the normally scheduled periodic review.

Risk-Informed Safety Classification made in accordance with the risk-informed process, described in I-3.0, shall be reevaluated on the basis of inspection periods and inspection intervals that coincide with the inspection program requirements for Inspection Program A or B of IWA-2431 or IWA-2432, as applicable. The performance of each inspection period or inspection interval reevaluation may be accelerated or delayed by as much as one year. The reevaluation shall determine if any changes to the risk-informed safety classifications need to be made, by evaluation of the following:

a) Plant design changes (e.g., physical; new piping or equipment installation; programmatic: power uprating / 18 to 24 month fuel cycle; procedural: pump test frequency changes, operating procedure changes)
b) Changes in postulated conditions or assumptions (e.g., check valve seat leakage greater than previously assumed, decrease in reliability of plant mitigative features)
c) PRA updates (e.g., new initiating events, new system functions, more detailed model used, initiating event and failure data changes)

~~New information may become available that alters the RISC for a piping segment. Such information may result from changes to the PRA, plant operation, or design of items.~~

~~The Owner shall identify and verify the effect of the new information on the RISC assigned to the piping segment.~~

~~When it is determined that the new information affects the RISC, the Owner shall reevaluate the classification, using the same approach originally used to establish the RISC.~~

TABLE I-1
CONSEQUENCE CATEGORIES FOR INITIATING EVENT IMPACT GROUP

Design Basis Event Category	Initiating Event Type	Representative Initiating Event Frequency Range (1/yr)	Example Initiating Events	Consequence Category (Note 1)
I	Routine Operation	>1		~~N/A~~None
II	Anticipated Event	$10^{-1}<x \leq$ ~~$\geq 10^{-1}$~~	Reactor Trip, Turbine Trip, Partial Loss of Feedwater	Low/ Medium
III	Infrequent Event	$10^{-2}<x \leq 10^{-1}$ ~~10^{-1} to 10^{-2}~~	Excessive Feedwater or Steam Removal	Low/Medium
			Loss of Off Site Power	Medium/High
IV	Limiting Fault or Accident	$< \leq 10^{-2}$	Small LOCA, Steam Line Break, Feedwater Line Break, Large LOCA	Medium/ High

Note 1: Refer to I-3.1.2(a)(3)

TABLE I-2
GUIDELINES FOR ASSIGNING CONSEQUENCE CATEGORIES TO FAILURES RESULTING IN SYSTEM OR TRAIN LOSS

Affected Systems		Number of Unaffected Backup Trains							
Frequency of Challenge	**Exposure Time to Challenge**	**0.0**	**0.5**	**1.0**	**1.5**	**2.0**	**2.5**	**3.0**	**≥3.5**
Anticipated (DB Cat II)	All Year	HIGH	HIGH	HIGH	HIGH	MEDIUM	MEDIUM	LOW*	LOW
	Between tests (1-3 months)	HIGH	HIGH	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW
	Long ~~AOT~~ CT (≤1 week)	HIGH	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW	LOW
	Short ~~AOT~~ CT (≤1 day)	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW	LOW	LOW
Infrequent (DB Cat. III)	All Year	HIGH	HIGH	HIGH	MEDIUM	MEDIUM	LOW*	LOW	LOW
	Between tests (1-3 months)	HIGH	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW	LOW
	Long ~~AOT~~ CT (≤1 week)	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW	LOW	LOW
	Short ~~AOT~~ CT (≤1 day)	HIGH	MEDIUM	LOW*	LOW	LOW	LOW	LOW	LOW
Unexpected (DB Cat. IV)	All Year	HIGH	HIGH	MEDIUM	MEDIUM	LOW*	LOW	LOW	LOW
	Between tests (1-3 months)	HIGH	MEDIUM	MEDIUM	LOW*	LOW	LOW	LOW	LOW
	Long ~~AOT~~ CT (≤1 week)	HIGH	MEDIUM	LOW*	LOW	LOW	LOW	LOW	LOW
	Short ~~AOT~~ CT (≤1 day)	HIGH	LOW*	LOW	LOW	LOW	LOW	LOW	LOW

Note: If there is no containment barrier and the consequence category is marked by an *, the consequence category should be increased (medium to high or low to medium).

TABLE I-3
CONSEQUENCE CATEGORIES FOR COMBINATION IMPACT GROUP

Event	Consequence Category
Initiating Event and 1 Unaffected Train of Mitigating System Available	High
Initiating Event and 2 Unaffected Trains of Mitigating Systems Available	Medium[1] (or IE Consequence Category from Table I-1)
Initiating Event and More Than 2 Unaffected Trains of Mitigating Systems Available	Low[1] (or IE Consequence Category from Table I-1)
Initiating Event and No Mitigating System Affected	N/A

Note 1: The higher classification of this table or Table I-1 shall be used.

TABLE I-4
CONSEQUENCE CATEGORIES FOR FAILURES RESULTING IN INCREASED POTENTIAL FOR AN UNISOLATED LOCA OUTSIDE OF CONTAINMENT

Protection Against LOCA Outside Containment	Consequence Category
One Active[1]	HIGH
One Passive[2]	HIGH
Two Active	MEDIUM
One Active, One Passive	MEDIUM
Two Passive	LOW
More than Two	NONE

Note 1: An example of Active Protection is a valve that needs to close on demand.
Note 2: An example of Passive Protection is a valve that needs to remain closed.

TABLE I-5
QUANTITATIVE INDICES FOR CONSEQUENCE CATEGORIES

CCDP or Quantitative Index, no units	CLERP or Quantitative Index, no units	Consequence Category
$\geq 10^{-4}$	$>10^{-5}$	High
$10^{-6} < \text{value} \leq 10^{-4}$	$10^{-7} < \text{value} \leq 10^{-5}$	Medium
$\leq 10^{-6}$	$\leq 0^{-7}$	Low
No change to base case	No change to base case	None