ATTACH. 2

ASME Meeting with U.S. Nuclear Regulatory Commission

ASME Code Sections III & XI Risk-Informed Code Cases

Kenneth R. Balkey, P.E.
Vice President-Elect, ASME Nuclear Codes and Standards

C. Wesley Rowley
Vice President, ASME Nuclear Codes and Standards

Kevin Ennis
Director, ASME Nuclear Codes and Standards
(Via Teleconference)

Rockville, MD
February 10, 2005

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Cognizant ASME Members & Staff and Supporting Industry Participants

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- *Gary Park* – Nuclear Mgmt Co. Chair, ASME Section XI
- *Shannon Burke* – ASME Staff Secretary, ASME Section XI
- ASME XI Working Group on Implementation of Risk-Based Exam
  - *Scott Kulat* –Inservice Engring (Chair)
  - *Robin Graybeal* –Inservice Engring
  - *Alex McNeill* –Dominion Energy VA
  - *Pat O'Regan* –Electric Power Res Inst*
- *Chris Sanna* – ASME Staff Secretary, ASME Section III*
- ASME III Working Group on Probabilistic Methods for Design
  - *Ralph Hill* -Golder Associates (Chair)*
  - *Graybeal, Kulat, McNeill, O'Regan**

INDUSTRY

- *Biff Bradley* – Nuclear Energy Institute
- 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

*Via Teleconference

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Purpose of Meeting

- To address recent NRC comments on ASME Code Cases N-660-1, N-660-2, N-720, N-716, N-711 from letter ballots at ASME Section XI & III working groups
- This open dialogue is being used to address key NRC technical issues early in the consensus standards development process so that NRC exceptions are eliminated in the regulatory endorsement process once the action is approved by ASME
- While NRC is a key stakeholder, comments from all ASME stakeholders will be addressed per the consensus standards development process to support worldwide use of these Code Cases

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*ASME Code Case N-660 – "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities," Section XI, Division 1**

*(*Note: Case applies to pressure-retaining items)*

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ASME Code Case N-660 – Status

- Code Case N-660 approved by Board on Nuclear Codes and Standards June 2002; Early WOG 50.69 pilot results used for support
- Use of N-660 in 2003 for service water systems (SWS) showed overly conservative results, Code Case N-660-1 developed for SWS only
- N-660 also used for trial use at Surry and Wolf Creek as part of WOG 50.69 pilot program (Fall of 2003 and Summer and Fall of 2004)
  - Wolf Creek systems – Containment Spray and Control Building Ventilation
  - Surry systems – Chemical & Volume Control System
- Feedback generated from trial use resulted in N-660-2 (i.e., Revision 2)
  - Main feedback involved allowance of small vs. large pipe failures, more appropriate evaluation for not-modeled components, and need for better guidance in the evaluation of additional considerations for low safety significant items

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ASME Code Case N-660 – Status (continued)

- N-660-1 approved by ASME XI WG in early 2004; being held depending on outcome of N-660-2 that may obviate need for N-660-1
- N-660-2, issued to Section XI working group level for letter ballot November 2004
- Comments discussed during ASME Boiler Code meetings in December 2004
- Additional resolution required from feedback
- ASME/NRC meeting scheduled for 02/10 to discuss proposed comment resolutions for N-660-2 before going forward with another ASME vote
- Depending on outcome of ASME/NRC, appropriate actions will take place at February 28, 2005 Code meeting

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Key N-660-2 Comments to be Addressed

1. Demonstrating technical adequacy of PRA
2. Need amplification of how plant configuration and design insights preclude possibility of a large pressure boundary failure
3. Additional considerations for low safety significant segments provide too much leeway to not consider large failures
4. Questions related to safety significant classification are not the same as the questions in Section 9 of NEI-00-04 (October 2004) as augmented by Regulatory Guide 1.201

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*ASME Code Case N-720 – "Risk-Informed Safety Classification for Construction of Nuclear Facility Components," Section III, Division 1**

*(*Note: Case applies to pressure-retaining items)*

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ASME Code Case N-720 History

- ASME Section III Working Group on Probabilistic Methods in Design was formed in 2003
- Code Case initiated following development and approval of N-660 in ASME Section XI in 2002
- 1st letter ballot of N-720 to the ASME Section III WG completed in August 2004
- Numerous comments addressed via telecons during Fall 2004
- Key issues discussed at Dec. 2004 Boiler Code meetings; 2nd letter ballot issued Jan. 2005

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Process for Development of Code Case N-720

- Process implemented for relating developments on N-660-2 as they pertain to N-720 and vice versa
  - Same cognizant ASME volunteers support the development of N-660 and N-720
  - Same RISC process is being applied, e.g., distinction in application of process to high energy and low energy systems
  - As N-660-2 gets finalized, appropriate information is evaluated for applicability and incorporation into N-720
  - Need team of NRC Staff to interface with ASME team to develop N-720 and to ensure that the final products for both Code Cases meet expectations of stakeholders

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Code Case N-720 – Unique Challenges

1. Level of maturity of PRA to support RISC during the design process
2. Limited or no operational experience in evaluating other considerations not addressed in the PRA
3. Availability of operating and emergency response procedures for consideration
4. Risk metrics are currently limited to core damage frequency and large early release frequency
5. PRA model and RISC process potentially changes hands during the design and construction phase

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Code Case N-720 – Unique Challenges (Cont)

6. Addressing classification from standpoint of not having alternating classifications within a system
7. Maintaining RISC throughout life of plant within the design specifications
8. Interface with ASME Section XI risk-informed applications in transitioning from design to inservice, particularly for treatment requirements
9. No process currently exists for defining RISC for "active" components
10. Need to find trial application

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ASME Code Case N-716

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ASME Code Case N-716 Risk-Informed / Safety Based (RIS_B)

- GOAL: Develop a consistent, generic and streamlined process for implementing and maintaining RI-ISI programs
  - ASME Whitepaper 2002-02-01
    - Reviewed over fifty plant-specific RI-ISI applications,
    - Thirty, of which, were Class 1 & 2 or full scope applications
    - Reviewed a number of industry and USNRC risk assessments
  - Change in risk assessment conducted for eight plants
    - N716 (RIS_B) provides a risk reduction or at worst, risk neutrality
    - Looked at BWRs and PWRs
    - Looked at plants that used the EPRI and WOG RI-ISI methods

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ASME Code Case N-716 - RIS_B

- Safety Significant (SS)
  - Reactor Coolant Pressure Boundary (e.g. Class 1)
  - Shutdown Decay Heat Removal (out to containment isolation)
  - Break Exclusion Region (BER)
  - Main Feedwater from S/Gs to BER
  - Segments with > 1E-6 CDF (from internal flooding study)

- Low Safety Significant (LSS)
  - Remaining items (i.e.
    - other Class 2,
    - all Class 3,
    - all NNS

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ASME Code Case N-716 - RIS_B

Inspection Requirements

- Safety Significant
  - NDE population equal to 10%, plus augmented programs
  - NDE locations selected based upon postulated susceptibility to degradation mechanisms

- Low Safety Significant
  - NDE per augmented programs (e.g. FAC, LC, IGSCC)
  - pressure/leakage testing continues

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ASME Code Case N-716 – RIS_B

Response to NRC Input

- Increased Safety Significant scope to include:
  - Main Feedwater from S/G to Break Exclusion Region
  - Segments with CDF greater than 1 E-06
- Added requirement for a plant-specific change in risk assessment, per RG1.174
- Added plant-specific PRA quality requirements consistent with NRC approved EPRI TR-1006937
- Feedback
  - ASME letter ballot
    - + Written responses, including changes to the Code Case, provided Jan 2004 & May 2004
  - NRC email 11-3-04
    - + Response provided 11-8-04, including changes to the Code Case
  - NRC e-mail 1-27-05
    - + Response provided 2-1-05, clarifying information provided

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ASME Code Case N-711

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ASME Code Case N-711

Alternative Examination Coverage Requirements

- Defines required examination volume based upon component configuration (e.g. pipe to valve) and susceptibility to degradation
- Plants that have implemented RI-ISI have existing expertise in-house (methodology independent)
- Plants that have not implemented RI-ISI would need to develop similar expertise

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ASME Code Case N-711

Response to NRC Input

- Added Table 3 which must be submitted to NRC within 90 days of outage
  - Table 3 identifies every weld that uses N711, amount of coverage and basis for limitation
  - Existing SXI process does not require relief requests to be submitted until one year after the ten year interval is completed
- "overly complicated scheme has been developed"
  - Most plants (80 – 90 percent) have existing capabilities
- "the proposed Code Case only uses certain ingredients of a risk process to eliminate welds from the examination process"
  - The case only addresses examination volume, therefore, no welds / inspections eliminated.
  - Note: the case does discuss there may be conservatisms in the original RI-ISI analysis and the owner has the option to review these conservatisms as part of RI-ISI living programs.

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ASME Code Case N-711

Response to NRC Input

- "one of the criteria for granting relief is that safety will be maintained for that specific application of the relief"
  - Examinations done to PDI requirements,
  - N711 requires that the volume of interest is captured by the examination
  - If not, relief requests are still required (e.g. PWSCC)

- "In this regard, technical limitations or supplemental measures may be imposed for certain coverage situations, such as requiring an alternative examination method or examinations on a different frequency, for example"
  - Per Table 3 of the Case, NRC will be receiving on a per outage basis (versus once per 11 years), examinations that do not obtain full coverage
    - + Weld number
    - + Exam category
    - + Weld description
    - + Percent coverage
    - + Description of the limitation

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Future Actions

- All results from discussion at today's meeting will be presented at appropriate ASME Section III and 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

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ATTACH. 3

Case N-660-2
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 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). Because this classification is to be used only for repair/replacement activities, failure potential is conservatively assumed to be 1.0 in performing the consequence evaluation per I-3.0 in Appendix I. These classifications might not be directly related to other risk-informed applications.

(b) Class 1 items connected to the reactor coolant pressure boundary, as defined in paragraphs 10 CFR 50.55a (c)(2)(i) and (c)(2)(ii), are within the scope of the RISC evaluation process. All other Class 1 items and items that are within the break exclusion region [>NPS 4 (DN 100)] for high-energy piping systems and their associated supports (NB, NC and NF) shall be classified as HSS and shall meet the full requirements of NCA, NB, NC and NF and are not part of this case. Break exclusion region shall be defined as applicable high-energy piping crediting alternatives to single failure criteria as approved by the regulatory agency having jurisdiction at the plant site.

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

-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.

(a) probabilistic risk assessment (PRA)
(b) plant operations
(c) system design
(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

The Owner is responsible for demonstrating the technical adequacy of any PRA used as the basis for this RISC process. PRA technical adequacy shall be assessed against a standard[2] or set of acceptance criteria that is endorsed by the regulatory agency having jurisdiction over the plant site.

-9000 GLOSSARY

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

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 an undesired consequence of core damage given a specific failure (e.g., piping segment failure)

conditional large early release probability (CLERP) – an estimate 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 such as 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 is anticipated and involving enough of the core to cause a significant release

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

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[2] ASME RA-S-2002, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications*, 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 specific applications.

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 component
high-energy systems – 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 °F, and
b. maximum operating pressure exceeds 275 psi

high-safety-significant (HSS) function – a function that has been determined to be safety significant from 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
level 1 analysis – identification and quantification of the sequences of events leading to the onset of core damage
low-energy systems – those systems that are 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 low-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.
low-safety-significant (LSS) function – a function not determined to be safety significant from traditional plant risk-assessment evaluations of core damage or large early release events or from other relevant information
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)
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, loss of inventory due to draining of a tank, 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, to ensure that the safety functions are satisfied
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
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 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. 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.

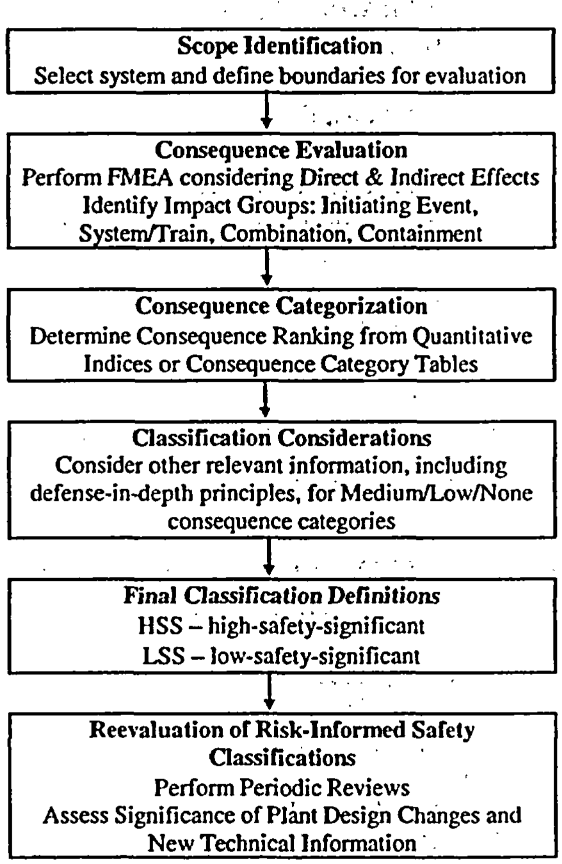

Figure I-1
Risk-Informed Safety Classification Process

I-2.0 SCOPE IDENTIFICATION

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

I-3.0 CONSEQUENCE EVALUATION

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). To accomplish this grouping, direct and indirect effects shall be assessed for each piping segment in a high-energy system and those segments in low-energy systems that have been modeled in the plant PRA, if applicable. For these high-energy systems a Consequence Category is determined from the Consequence Evaluation as defined in I-3.1.1 and I-3.1.2. The plant-specific Probabilistic Risk Assessment (PRA) (level 1 analysis with internal initiating events, as a minimum) shall be used. Segments in low-energy systems not modeled in the plant PRA shall be evaluated per I-3.2.2(b), (c), and (d). 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. 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.

I-3.1 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 failure for piping segments. Alternatively, the consequence analysis can be performed assuming a smaller leak, when
  (1) a smaller leak is more conservative; or
  (2) a small 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 failure (e.g., design guard pipes); or
  (4) applied to Class 2 and 3 low-energy systems that meet the requirements of Appendix I.

(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 effects such as pipe whip, jet impingement, 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.

(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. Automatic actions need not be safety related nor subject to single failure.

(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 impact groups: initiating event, system, or combination. 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 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 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 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 affected function of the system is called upon. This corresponds to the frequency of 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 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. 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. 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.

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 their importance relative to 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.

I-3.2 ClassificationI-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 shall be considered HSS.

(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 and segments in low-energy systems shall be determined HSS or LSS by considering the other relevant information for determining classification. Under the same conditions of I-3.1.1(a), a large pressure boundary failure 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 releases of radioactive material that would result in the implementation of off-site 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.

(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.

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. 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)

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		None
II	Anticipated Event	$\geq 10^{-1}$	Reactor Trip, Turbine Trip, Partial Loss of Feedwater	Low/ Medium
III	Infrequent Event	10^{-1} to 10^{-2}	Excessive Feedwater or Steam Removal	Low/Medium
			Loss of Off Site Power	Medium/High
IV	Limiting Fault or Accident	$<10^{-2}$	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 CT (≤ 1 week)	HIGH	HIGH	MEDIUM	MEDIUM	LOW*	LOW	LOW	LOW
	Short 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 CT (≤ 1 week)	HIGH	MEDIUM	MEDIUM	LOW*	LOW	LOW	LOW	LOW
	Short 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 CT (≤ 1 week)	HIGH	MEDIUM	LOW*	LOW	LOW	LOW	LOW	LOW
	Short 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
$>10^{-4}$	$>10^{-5}$	High
$10^{-6} < \text{value} \leq 10^{-4}$	$10^{-7} < \text{value} \leq 10^{-5}$	Medium
$\leq 10^{-6}$	$\leq 10^{-7}$	Low
No change to base case	No change to base case	None

ATTACH. 4

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Case N-660-2
Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities
Section XI, Division 1

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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 2004 Edition]

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-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). Because this classification is to be used only for repair/replacement activities, failure potential is conservatively assumed to be 1.0 in performing the consequence evaluation per I-3.0 in Appendix I. These classifications might not be directly related to other risk-informed applications.

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(b) Class 1 items connected to the reactor coolant pressure boundary, as defined in paragraphs 10 CFR 50.55a (c)(2)(i) and (c)(2)(ii), are within the scope of the RISC evaluation process. All other Class 1 items and items that are within the break exclusion region [>NPS 4 (DN 100)] for high-energy piping systems and their associated supports (NB, NC and NF) shall be classified as HSS and shall meet the full requirements of NCA, NB, NC and NF and are not part of this case. Break exclusion region shall be defined as applicable high-energy piping crediting alternatives to single failure criteria as approved by the regulatory agency having jurisdiction at the plant site.

Deleted: 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 applied.

-1300 OWNER'S RESPONSIBILITY

-1310 Determination of Classification

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

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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.

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

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

-1330 Adequacy of the PRA

The Owner is responsible for demonstrating the technical adequacy of any PRA used as the basis for this RISC process. PRA technical adequacy shall be assessed against a standard[2] or set of acceptance criteria that is endorsed by the regulatory agency having jurisdiction over the plant site.

Deleted: 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

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

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 an undesired consequence of core damage given a specific failure (e.g., piping segment failure)

conditional large early release probability (CLERP) – an estimate 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 such as 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 is anticipated and involving enough of the core to cause a significant release

failure – an event involving leakage, rupture, or a condition that would disable the ability of an item to perform 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

[2] ASME RA-S-2002, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications*, 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 specific applications.

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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 component

high-energy systems – 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 °F, and
b. maximum operating pressure exceeds 275 psi

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high-safety-significant (HSS) function – a function that has been determined to be safety significant from plant probabilistic risk assessment or from other relevant information (e.g., defense in depth considerations)

Deleted: 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).

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

level 1 analysis – identification and quantification of the sequences of events leading to the onset of core damage

low-energy systems – those systems that are 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 low-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.

low-safety-significant (LSS) function – a function not determined to be safety significant from traditional plant risk-assessment evaluations of core damage or large early release events or from other relevant information.

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)

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

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

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spatial effect – a failure consequence affecting other systems or components, such as failures due to pipe whip, jet impingement, jet spray, loss of inventory due to draining of a tank, 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, to ensure that the safety functions are satisfied

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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

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

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I-1.0 INTRODUCTION

This Appendix 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. 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.

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Scope Identification
Select system and define boundaries for evaluation

↓

Consequence Evaluation
Perform FMEA considering Direct & Indirect Effects
Identify Impact Groups: Initiating Event, System/Train, Combination, Containment

↓

Consequence Categorization
Determine Consequence Ranking from Quantitative Indices or Consequence Category Tables

↓

Classification Considerations
Consider other relevant information, including defense-in-depth principles, for Medium/Low/None consequence categories

↓

Final Classification Definitions
HSS – high-safety-significant
LSS – low-safety-significant

↓

Reevaluation of Risk-Informed Safety Classifications

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Perform Periodic Reviews
Assess Significance of Plant Design Changes and
New Technical Information

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 EVALUATION

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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). To accomplish this grouping, direct and indirect effects shall be assessed for each piping segment in a high-energy system and those segments in low-energy systems that have been modeled in the plant PRA, if applicable. For these high-energy systems a Consequence Category is determined from the Consequence Evaluation as defined in I-3.1.1 and I-3.1.2. The plant-specific Probabilistic Risk Assessment (PRA) (level 1 analysis with internal initiating events, as a minimum) shall be used. Segments in low-energy systems not modeled in the plant PRA shall be evaluated per I-3.2.2(b), (c), and (d). 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. To take credit for operator actions, the following features shall be provided:

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- 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.

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To determine that the consequence evaluation and considerations are sufficient for the RISC process, the requirements of the following subparagraphs shall be met.

Deleted: 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 Analysis and Assessments

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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 failure for piping segments. Alternatively, the consequence analysis can be performed assuming a smaller leak, when

(1) a smaller leak is more conservative; or

(2) a small 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

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(3) it can be documented that a physical configuration precludes the possibility of a large pressure boundary failure (e.g., design guard pipes); or

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(4) applied to Class 2 and 3 low-energy systems that meet the requirements of Appendix I.

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(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 effects such as pipe whip, jet impingement, jet spray, and loss-of-inventory effects (e.g., draining of a tank).

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(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.

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(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. Automatic actions need not be safety related nor subject to single failure.

(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 impact groups: initiating event, system, or combination. 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 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 shall be included.

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(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 is the ratio of the core damage frequency due to the initiating event to the initiating event frequency.

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(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 affected function of the system is called upon. This corresponds to the frequency of events that require the system operation.

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(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 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. 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. 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.

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 their importance relative to 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.

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 shall be considered HSS.

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Deleted: Allowed Outage Time

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<#>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.
<#>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 ... [1]

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(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, and segments in low-energy systems shall be determined HSS or LSS by considering the other relevant information for determining classification. Under the same conditions of I-3.1.1(a), a large pressure boundary failure 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.

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(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 releases of radioactive material that would result in the implementation of off-site 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.

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(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,

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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.

Deleted: 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.

Deleted: 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. 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)

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Deleted: 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.

02/03/05

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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		None
II	Anticipated Event	$\geq 10^{-1}$	Reactor Trip, Turbine Trip, Partial Loss of Feedwater	Low/ Medium
III	Infrequent Event	10^{-1} to 10^{-2}	Excessive Feedwater or Steam Removal	Low/Medium
			Loss of Off Site Power	Medium/High
IV	Limiting Fault or Accident	$<10^{-2}$	Small LOCA, Steam Line Break, Feedwater Line Break, Large LOCA	Medium/ High

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

02/03/05

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BC01-364
03/05/02

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 CT (≤ 1 week)	HIGH	HIGH	MEDIUM	MEDIUM	LOW*	LOW	LOW	LOW
	Short 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 CT (≤ 1 week)	HIGH	MEDIUM	MEDIUM	LOW*	LOW	LOW	LOW	LOW
	Short 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 CT (≤ 1 week)	HIGH	MEDIUM	LOW*	LOW	LOW	LOW	LOW	LOW
	Short CT (≤ 1 day)	HIGH	LOW*	LOW	LOW	LOW	LOW	LOW	LOW

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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).

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BC01-3641
03/05/02

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
$\geq 10^{-4}$	$\geq 10^{-5}$	High
$10^{-6} \leq$ value $\leq 10^{-4}$	$10^{-7} <$ value $\leq 10^{-5}$	Medium
$\leq 10^{-6}$	$\leq 10^{-7}$	Low
No change to base case	No change to base case	None

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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.

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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.

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The Owner may further refine the classification ranking by more extensive application of the process defined in these requirements. These analyses shall be documented.

ATTACH. 5

LETTER BALLOT COMMENT FORM

COMMITTEE: WG / RI SUBMITTED BY: Syed Ali/Gene Imbro (ASME XI SG on RRA) DATE: Dec 07, 2004

LETTER BALLOT # BC04-1505 SUBJECT: N-660 Rev. 2

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
Inquiry/Reply	D	It is not clear if this code is a supplement to the group classification criteria of IWA-1320 or replaces the current classification criteria.	Consistent with the intent of Code Case N-662, this Case is a supplement to the group classification criteria of IWA-1320.
-1330	D	What is the mechanism of the Owner demonstrating the technical adequacy of the PRA and to whom is it demonstrated?	See response to next comment.
-1330	D	The section references the ASME PRA Standard (RA-S-2002 and addenda) but does not state that the Owner shall comply with the standard.	The Owner is not required to comply with the PRA Standard. Reference to the PRA Standard is now made in the footnote to the following sentence of -1330, "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."
-9000	D	In the definitions of HSS and LSS functions, the use of the term "traditional plant risk-assessment" is not clear.	"Traditional plant risk-assessment" replaced with "plant probabilistic risk assessment."
-9000	D	Change the definitions of plant mitigative features to: "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."	Change made
-9000	D	In the definitions of success criteria, change the last word from "satisfied" to "accomplished."	Current definition is consistent with ASME PRA Standard (RA-S-2002) – for consistency, no change made

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

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
I-3.0, last sentence	D	Provide clarification or reference to the section as to how the additional relevant information is considered.	Reference made to Section I-3.2.2(b)
I-3.1.1(a)(3)	D	Needs amplification on how plant configuration (examples) precludes the possibility of a large pressure boundary failure.	Plant configuration removed as a consideration for assuming small leak rather than large leak.
I-3.1.1(a)(4)	D	Needs amplification on what design insights are considered (examples) and how they are considered.	Item clarified to now read, "Ambient fluid temperature or pressure in pipe segment precludes the possibility of a large pressure boundary failure."
I-3.1.1(e)	D	Under what conditions is credit allowed for operator action? Auto isolation? Single failure considerations?	The term "operator action" has been added to the glossary with credible conditions. Also, the following sentence has been added to (e), "Automatic actions need not be safety related nor subject to single failure."
I-3.1.2, last sentence before (a)	D	How are fire and seismic risk considered? Under what conditions would a fire cause a piping failure?	Clarification provided in replacement sentence for use of other risk information.
I-3.1.2(b)(3)	D	Is completion time defined in T/S?	Yes, also referred to as allowed outage time.
I-3.1.2	D	Notation in parenthesis should be "CCDF."	Change made
I-3.1.2(d)(2)	D	Last sentence does not make sense.	Clarification made
I-3.2.2(b)	D	The questions related to safety significant classification are not the same as the questions in Section 9 of NEI-00-04 (October 2004) as augmented by RG 1.201 (June 2004). We note that the NEI and RG questions are more directed toward active SSCs failure while the code case questions are related to passive SSC failure so identical text may not be the best solution and may be undesirable. However, we have no evaluation detail to demonstrate that the two sets of questions both represent a consistent set of necessary and sufficient questions.	See attached file for comparison of N-660 and NEI 00-04 considerations with proposed changes. The table shows for each issue the old and new N-660 and NEI 00-04 considerations along with a comment to describe any difference between the two. See also revised text in I-3.2.2(b).

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

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
I-3.2.2(b)	D	This section gives too much leeway to not consider large failures.	Based on new text in I-3.0, this section is applied to medium/low piping in high-energy systems and all piping in low-energy systems. Other revisions to design insights and historical data reference as seen in the resolution of comments below and revised text in I-3.2.2(b).
I-3.2.2(b)	D	Design insights need to be defined and have associated acceptance criteria along with possibly, examples.	Item clarified to now read, "Ambient fluid temperature or pressure in pipe segment precludes the possibility of a large pressure boundary failure." Examples will be provided in White Paper.
I-3.2.2(b)	D	It is not clear how the historical data is considered and what is the acceptance criteria associated with it.	Statement regarding historical data has been removed from the consideration of this Code Case.
I-3.2.2, footnote	D	Requirements should not be placed in the footnote.	Requirements for "operator action" have been moved to I-3.0.
I-3.2.2, footnote	D	Add these two items: • Plant procedures to define operator action • Operator training in the procedures.	Additions made and moved to I-3.0.
I-3.2.2, footnote, 2nd bullet	D	Does the equipment referenced need to be safety-related and what are the reliability and availability requirements for the equipment.	No, equipment does not need to be safety related. The reliability and availability requirements are assumed by the personnel included in the categorization process. Changes to those assumptions are evaluated as part of the reevaluation process in I-4.0.
I-3.2.2, footnote, 3rd bullet	D	What are the criteria for evaluation.	Clarification provided in revised statement now located in I-3.0.

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

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
I-3.2.2(6)	D	Last part of sentence "that would result.......actions' should be deleted as it provides too high a threshold.	The Protective Action Guide is more limiting than the 10 CFR 100 limits and therefore provide additional margin to account for uncertainties in the modeling of the releases, such as those in best estimate models. Current text will remain unchanged.
I-3.2.2(c), last sentence	D	We do not agree that only one of the two criteria in this section is sufficient. Both criteria should be satisfied to maintain LSS category.	Statement revised as stated in original N-660 version. Intent was not to have two criteria but only one. See revised text in I-3.2.2(c).
I-3.2.2(d), last sentence	D	This needs to be defined and clarified. We do not agree with this exception.	Sentence deleted as it provides no use. Original intention of sentence is captured in Section I-3.2.2(b).
I-3.2.2(b)(7) to (b)(11)	D	It is not clear how these 5 conditions relate to the particular segment being evaluated.	These 5 conditions have been taken directly from RG 1.174. Their consideration ensures the maintaining of defense in depth.
I-4.0	D	Add the following statement at the beginning: "Plant design changes shall be screened prior to implementation to determine if they would result in a significant change to the plant risk profile.	Change made

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

LETTER BALLOT COMMENT FORM

COMMITTEE: ASME XI SG on RRA SUBMITTED BY: Ed Gerlach DATE: Dec 06, 2004

LETTER BALLOT # BC04-1505 SUBJECT: N-660 Rev. 2

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
Applicability	C	The applicability statement needs to be revised to reflect the 2004 Edition.	Change made
-1330	C	It is not clear whether use of the ASME PRA Standard is a requirement or a recommendation.	Use of ASME PRA Standard is not required. However, it is now referenced in footnote to -1330.

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

LETTER BALLOT COMMENT FORM

COMMITTEE: WG / RI SUBMITTED BY: Robin Graybeal DATE: Nov 12, 2004

LETTER BALLOT # BC04-1505 SUBJECT: N-660 Rev. 2

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
-1320	C	My copy printed the sub paragraphs as (c), (d), (e), (f). Should they not be (a), (b), (c), (d)?	Change made
-1320	C	I'm not sure it is appropriate to place desirable conditions in a Code action. If we want RI-ISI discipline represented, it should be stated as a requirement.	Feedback from WOG pilot program suggested including RI-ISI personnel on engineering team. This is noted as only a suggestion and not a requirement; therefore, an RI-ISI engineer will not be made a requirement.
I-3.2.2(b)(5)	C	Change to: A plant condition monitoring program (e.g., flow accelerated corrosion program, etc.).....	Statement actually deleted. See attached file, "Comparison of Code Case N-660 and NEI 00-04.doc", for clarification.
Table I-5	C	It would be more clear if CLERP with the appropriate ranges was added to Table I-5	Change made

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

LETTER BALLOT COMMENT FORM

COMMITTEE: ASME XI SG on RRA SUBMITTED BY: Mark Herrera DATE: Dec 05, 2004

LETTER BALLOT # BC04-1505 SUBJECT: N-660 Rev. 2

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
-1320	C	Is the intent to require a degreed engineer, Professional engineer certification, etc????	Reference to RI-ISI engineer has been removed – not a requirement.
I-3.1.2(c), last line	C	CDF already defined	Change made
I-3.1.2(d), 6th line	C	CCDP already defined (do a global check on definition of acronyms).	Change made
I-3.1.2(d)(2)	C	Should the paragraph following this section, also be part of (d)(2) since it appears to illustrate the required margin?	Yes, paragraph will be indented to line up with (d)(2).

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

LETTER BALLOT COMMENT FORM

COMMITTEE: WG / RI SUBMITTED BY: Alex McNeill DATE: Nov 30, 2004

LETTER BALLOT # BC04-1505 SUBJECT: N-660 Rev. 2

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
Applicability	C	..... through 2004 Edition	Change made
-1200(b)	C	Should "that portion of the Class 2 feedwater system [> NPS 4 (DN 100)] of Pressurized Water Reactors (PWRs) from the steam generator to the outer containment isolation valve, and items that are within the break exclusion region[1] [> NPS 4 (DN 100)] for high energy piping systems and their associated supports be classified HSS. Note 1: Break exclusion region shall be defined as applicable high energy piping crediting alternatives to single failure criteria as approved by the regulatory agency having jurisdiction at the plant site.	Added from "..., and items that are within" through to the end of the suggested text. Note 1 was also added. SG piping not candidate for LSS.
-9000	C	Add definitions for the following terms (recommend coordination with N-720 in Section III): 1) Containment Barrier 2) Failure Mode 3) Failure Potential 4) High Energy Systems (only if consideration above accepted) 5) Low Energy Systems (only if consideration above accepted) 6) Train 7) Unaffected Backup Trains	Definitions reviewed against those in Code Case N-720 as well as the PRA Standard (RA-S-2002) and all additions have been made as suggested. Also, minor changes to HSS were made. The term "level 1 analysis" was added as defined in the PRA Standard.
I-3.1.1(2) & I-3.2.2(b)(6)	C	Recommend removal of references to USNRC documents and replace with "document acceptable to the regulatory authority (or agency) having jurisdiction."	Reference to USNRC documents have been removed in both sections and replaced with text similar to that suggested.

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

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
I-3.1.2(d)	C	A rewrite is planned for code case N-720 for their corresponding paragraph by Dr. Ian Wall, Pat O'Regan and Barry Sloane. Please review that change for applicability to this case.	Corresponding paragraph in Code Case N-720 was reviewed and changes were made to I-3.1.2(d).

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

LETTER BALLOT COMMENT FORM

COMMITTEE: WG / RI SUBMITTED BY: Pat O'Regan DATE: Dec 03, 2004

LETTER BALLOT # BC04-1505 SUBJECT: N-660 Rev. 2

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
General	D	Although I approve this item, it seems necessary to provide additional clarification/criteria for use in determining when the following sections are applicable. Adding examples of when they have been applied during the pilot effort would be helpful. I.3.1.1(a)(4) "design insights do not support a large break based on pressure/temperature/flow in the pipe segment" I.3.2.2(b) "As in the consequence evaluation of Section I-3.1.1(a), large pipe failure need not be assumed when design insights do not support a large break based on pressure/temperature/flow in the pipe segment." I.3.2.2(b) "Historical data may also be considered regarding whether the failure is unlikely to occur or could be detected in a timely manner." I.3.2.2(b)(5) "The plant condition monitoring program would identify any known active degradation mechanisms in the pipe segment prior to its failure in test or an actual demand event (e.g., flow accelerated corrosion program), or materials resistant to the known damage mechanisms have been used for construction of the component."	"Design insights" and "historical data" statements have been removed to avoid confusion. Clarification has been made to consider high and low energy systems separately (see I-3.0). Examples of this will be provided in the White Paper. I-3.2.2(b)(5) has been removed for reasons documented in the attached Word table, "Comparison of N-660 and NEI 00-04.doc".
General	D	Also, as the "additional considerations" were originally developed for SSCs not modeled in the PRA, the case is overly restrictive or duplicative. One option is to have the level of analysis a function of system energy (i.e. temp/pressure). That is, high energy systems would be required to be analyzed per the FMEA of 3.1.1. Low energy systems would be analyzed per the "additional considerations" of I.3.2.2(b).	This suggestion has been accepted. See revised text in I-3.0 for distinction between high and low energy systems.

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

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
-1200(b)	D	Depending on the resolution of the above, it may be prudent to add the following to -1200(b): "that portion of the Class 2 feedwater system [> NPS 4 (DN 100)] of Pressurized Water Reactors (PWRs) from the steam generator to the outer containment isolation valve, and items that are within the break exclusion region1 [> NPS 4 (DN 100)] for high energy piping systems"	Added from "..., and items" through to the end of the suggested text. SG piping not candidate for LSS.

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

LETTER BALLOT COMMENT FORM

COMMITTEE: WG / RI   SUBMITTED BY: Mark Pyne   DATE: Dec 06, 2004

LETTER BALLOT # BC04-1505   SUBJECT: N-660 Rev. 2

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
Applicability	D	The applicability should be through the 2003 Addenda not Edition.	Based on numerous comments, Applicability changed to read 2004 Edition.
-1320	D	Clarify if use of an RI-ISI engineer is required or not. If required, change to a requirement for using RI-ISI expertise rather than an RI-ISI engineer to be consistent with other areas of expertise required.	No reference will be made to an RI-ISI engineer as it was only a suggestion and not a requirement.
-1330	D	Clarify if use of the PRA standard is required. It should be consistent with 50.69	ASME PRA Standard is not required but is referenced in footnote to -1330. See new text for assessing PRA technical adequacy.
-9000	D	The definitions for CCDP and CLERP repeat use of the word conditional. This is not good practice for definitions. I suggest using wording similar to that for conditional consequence.	Change made
I-3.2.2(5)	D	This section could be very confusing to a new user. Is the intent to merely identify the mechanism or identify and take some credit for being able to prevent failure? Also, the phrase "identify any known" is awkward.	Intent is to prevent failure. Statement actually deleted for reasons documented in attached Word file, "Comparison of N-660 and NEI 00-04.doc".
I-4.0	D	Consider moving first sentence to the end of the paragraph. This would result in a discussion of routine or normal assessment of the program followed by the non-routine.	Normal assessment moved to beginning of paragraph.

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

LETTER BALLOT COMMENT FORM

COMMITTEE: ASME XI SG on RRA SUBMITTED BY: Rick Swayne DATE: Nov 12, 2004

LETTER BALLOT # BC04-1505 SUBJECT: N-660 Rev. 2

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
Applicability	C	Applicability should be through the 2004 Edition. There is no 2003 Edition.	Change made
-1200	C	-1200(b) and I-2.0 assume that classification of an item as Class 1 is mandatory or optional. This is incorrect. There is no requirement to classify anything as Class 1. Therefore, "optional," in this context, is meaningless.	To avoid confusion, discussion of "optional" classification removed.
-1320	C	-1320(f) is unclear. "Desirable" is not good Code language. What is a Risk-informed Inservice Inspection engineer? Who certifies them?	Reference to RI-ISI engineer removed and no longer a requirement.
-1330	C	-1330 is not clear with regard to whether or not RA-S-2002 is mandatory. Which addenda are required?	ASME PRA Standard is not required, however, it is referenced in the footnote to -1330.
-9000	C	In -9000, the definition of condition monitoring implies that you can detect when failure is likely to occur. This seems to be poor language.	Term removed from glossary because I-3.2.2(b)(5) removed. See attached Word table, "Comparison of N-660 and NEI 00-04.doc." for justification.
I-3.0	C	In I-3.0, what criteria can be used to determine if the required evaluation and consideration are sufficient?	The following statement has been added to the end of I-3.0, "To determine that the consequence evaluation and considerations are sufficient for the RISC process, the requirements of the following subparagraphs shall be met."
I-3.1.1.1(a)(4) & I-3.2.2(b)	C	What "design insights" are used to satisfy I-3.1.1.1(a)(4) and I-3.2.2(b)?	Reference to design insights deleted to avoid confusion. Distinction now made between high and low energy systems in I-3.0 to address issue of small vs. large pipe failures.

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

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
I-3.1.2	C	The new last sentence of I-3.1.2 is vague and does not specify any requirement. What is its purpose?	Clarification provided in revised statement regarding use of other risk information.
I-3.1.2(d)	C	The changes in the 2nd sentence of I-3.1.2(d) do not improve the sentence. Evaluated and effect are better than assessed and impact.	Sentence has been revised to be consistent with 1st sentence.
I-3.1.2(d)(2)	C	In I-3.1.2(d)(2), how is a "margin of a factor" of 0.1 applied?	Entire section revised and discussion on margin has been removed
I-3.2	C	I-3.2 is too vague. What is meant by "considering" and by "other relevant information?"	Sentence removed because it was not necessary. I-3.2.2(b) provides the "other relevant information" to be evaluated (or considered) in determining the final RISC group.
I-3.2.2(b)	C	In I-3.2.2(b), delete the added "Section." In Note 2, "must" should be "shall."	"Section" has been removed. Footnote describing requirements for operator actions has been moved to I-3.0 and the "must" has been replaced with "shall".
I-3.2.2(c)	C	In I-3.2.2(c), what are "sufficient safety margins?" How are they determined? How is uncertainty accounted for?	**TBD**
I-4.0	C	In I-4.0, "should" should be "shall."	Change made
I-4.0	C	In I-4.0, if the third period reevaluation serves as the next interval reevaluation, when is a subsequent reevaluation required? May an interval be skipped?	Intervals may not be skipped. Sentence removed to avoid confusion.
I-4.0	C	What are the requirements for evaluation of the conditions in (a), (b), and (c)? What is a sufficient evaluation?	Added reference to risk-informed process described in I-3.0 for the evaluation requirements.

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

LETTER BALLOT COMMENT FORM

COMMITTEE: WG / RI SUBMITTED BY: Ray West DATE: Dec 01, 2004

LETTER BALLOT # BC04-1505 SUBJECT: N-660 Rev. 2

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
General	D	With the issuance of the new 10 CFR 50.69 Rule, I believe that this Code Case needs to be revised where appropriate to now be in compliance with the Rule. It would be remise of the ASME to issue a Code Case without giving due consideration to what an Owner is going to have to comply with in order to use the Case. It is for that reason that I am voting negative at this time, but I do have comments that I have outlined below which I feel would improve the Case overall.	Noted, see comment dispositions below
-1200(a)	D	The designator for paragraph (a) should be removed and –1200(b) should be deleted. The requirements of –1200(b) are repeated in I-2.0 and there should be no reason to have these requirements in two places.	-1200(b) remains in place but with revisions. Requirements of -1200(b) have been deleted in I-2.0 since they were not there in original version of Code Case.
-1200(a)	D	The reference to I-3.1 should be I-3.0.	Change made
-1320	D	A requirement to have an Integrated Decision Making Panel (IDP) with responsibilities to perform the classification process needs to be revisited. The reference to suggesting that a RI-ISI engineer be involved in the process is not needed. An Owner can decide if that person would add to the success of the process and that may be the case if the plant has a Class 2 or 3 RI-ISI program, but to have that person on the IDP should not be a requirement.	Reference to RI-ISI engineer requirement has been removed. The Working Group decided to leave section as is without reference to IDP requirements.
-1330	D	Should be reworded to make the use of the ASME PRA Standard a requirement as follows: The Owner is responsible for demonstrating the technical adequacy of any PRA used as the basis for the RISC process. The PRA used shall meet the capability requirements of the ASME PRA Standard (RA-S-2002 with the RA-Sa-2003 Addenda) to the extent required to support this process.	The ASME PRA Standard is listed for clarification but not as a requirement. See new text in -1330 and footnote.

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

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
Figure I-1		The RISC process steps are incomplete as shown in Figure I-1 and should be revised to include a final step identifying the requirement for Reevaluation of Risk-Informed Safety Classifications and the titles and numbering of the major Sections (e.g., I-2.0, I-3.0, I-4.0, etc.) should reflect what is in this Table and not mixed up as they are now and thus I-3.0 should be re-titled as CONSEQUENCE EVALUATION and I-3.1 should be "Analysis and Assessments," etc.	Change made to Figure I-1 and Section titles (I-3.0 and I-3.1).
I-2.0		I believe that this requirement would be much clearer if The following words were deleted from the beginning of the second sentence: "Items optionally classified to Class 1 and," and then just start the sentence as: "Class 1 items connected to the ...," and the reference to 10 CFR 50.55a (c)(2)(i) and (c)(2)(ii) will sufficiently cover the exceptions.	Change made to requirement in - 1200(b).

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

LETTER BALLOT COMMENT FORM

COMMITTEE: ASME XI SG on RRA	SUBMITTED BY: Ron Yonekawa	DATE: Nov 19, 2004
LETTER BALLOT # BC04-1505	SUBJECT: N-660 Rev. 2	

PAGE & PARAGRAPH REFERENCE	D/C	COMMENTS AND/OR RECOMMENDATIONS	DISPOSITION OF COMMENTS
-1320	C	I think that the desirability of having a Risk Informed ISI engineer involved in 1320 is not good case wording. Perhaps it could be a requirement with some exceptions listed.	Reference to an RI-ISI engineer as a requirement has been removed.
-1200(b)	C	In 1200 b, where we talk about items optionally classified as class 1, is this referring to optionally constructed to class 1 rules, or is this referring to optionally classified as class 1 in the ISI program? If it is the former, I can see why an item would be constructed to class 1 rules even though the final application is not class 1. But it seems to me that if it is the latter, it should be treated as HSS because there would have been some reason that it was optionally classified as class 1 in the ISI program.	Discussion on "optionally" deleted to remove confusion.

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

ATTACH. 6

Case N-720, Risk Informed Safety Classification for Construction of Nuclear Facility Components, Section III, Division 1, Subsections NCA, NB, NC, ND, and NF.

Inquiry: What alternative classification and quality assurance rules may be used in the construction of nuclear facility components when using risk-informed safety classification?

Reply: It is the opinion of the Committee that the following requirements, describing application of the Risk Informed Safety Classification process, may be used as an alternative to the requirements of NCA-2110(c), NCA-3253, and NCA-3800.

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

PART A Supplemental Risk-Informed Classification Requirements

1. Scope

This part provides a process for determining the Risk-Informed Safety Classification (RISC) of nuclear facility items having only a pressure retaining function (also referred to as passive components) or the passive function of active components. The classification process provided by this Case should be used in combination with a classification process for active components using standards and guidance endorsed by the regulatory authority having jurisdiction over the nuclear facility. The RISC process of this Case may be applied to supplement the classification required by NCA-2110 and NCA-3253 for NB, NC, ND, and NF components.

2. Classifications

(a) The RISC process is described in Appendix I of this Case. Pressure retaining, component support and piping support items shall be classified High Safety Significant (HSS) or Low Safety Significant (LSS). Failure potential for this classification is conservatively assumed to be 1.0 in performing the initial consequence evaluation per I-3.1 in Appendix I. The classifications from this case are intended for new plant construction.

(b) Class 1 items, that portion of the Class 2 feedwater system [> NPS 4 (DN 100)] of Pressurized Water Reactors (PWRs) from the steam generator to the outer containment isolation valve, and items that are within the break exclusion region[1] [> NPS 4 (DN 100)] for high energy piping systems and their associated supports (NB, NC and NF) shall be classified High Safety Significant (HSS) and shall meet the full requirements of NCA, NB, NC and NF and are not part of this case.

3. Determination of Classification

In addition to the classification requirements of NCA-3220(g) the Owner shall provide the appropriate RISC classification in accordance with Appendix I of this Case (Note: this may be pre-prepared by the plant designer but needs to be approved by the Owner during the construction phase.). The Owner shall ensure that core damage frequency (CDF) and large early release frequency (LERF) are included as risk metrics in the RISC process. In addition to the requirements of NCA-3252, the Design Specification shall

[1] Break exclusion region shall be defined as applicable high energy piping crediting alternatives to single failure criteria as approved by the regulatory agency having jurisdiction at the plant site.

contain or reference the RISC Evaluation Report (I-4.0), and include references to the facility PRA and applicable technical specifications supporting the RISC classification of the component(s)

4. Required Disciplines

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

(a) probabilistic risk assessment (PRA)

(b) plant operations

(c) system design

(d) safety or accident analysis

(e) Other disciplines, such as materials engineering, chemistry, or nondestructive examination, relevant to the specific system or equipment issues. For new construction expertise from similar plant designs (e.g., earlier or same versions or models) may be used. For new construction involving experimental or a first of a kind model plant, expertise used shall be in agreement with the regulatory authority having jurisdiction at the plant. Personnel may be experts in more than one discipline, but are not required to be experts in all disciplines.

5. Adequacy of the PRA

The Owner is responsible for demonstrating adequacy of any PRA used as the basis for this process. The ASME PRA Standard (RA-S-2002 and addenda) provides requirements regarding PRA capability.

6. Glossary (used within the context of this code case)

conditional core damage probability (CCDP) - the conditional probability of a core damage event given a core damage event

conditional large early release probability (CLERP) – the conditional probability of large early release given a specific failure (e.g., piping segment failure)

Completion Time (CT) - The amount of time allowed for completing a required action. In the context of this code case, the required action is to restore operability (as defined in the Technical Specifications) to the affected system or equipment train

condition monitoring – monitoring methods used to measure the performance or functional condition of equipment to detect or predict when failure is likely to occur based on a planned time interval

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

containment barrier – containment barrier is defined as 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

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

failure mode – correspond to the size of postulated failures (e.g., large break vs. small leak) as well as the configuration assumed during the failure (e.g., operating, standby, demand)

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 component

high-energy systems - 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°F, and

b. maximum operating pressure exceeds 275

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 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 (LER) – 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 such that there is a potential for early health effects

low-energy systems - those systems that are 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 low 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

low-safety-significant (LSS) function – a function not determined to be safety significant from traditional plant risk-assessment evaluations of core damage or large early release events or from other relevant information.

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 used to prevent or mitigate an accident

probabilistic risk assessment (PRA) – 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 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, loss of inventory due to draining a tank, 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 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

unaffected backup trains - A train(s) as defined above, 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 the risk-informed process used to determine Risk-Informed Safety Classification (RISC) for use in risk-informed construction 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.

Scope Identification
Select system and define boundaries for evaluation

Consequence Evaluation
Perform FMEA considering Direct & Indirect Effects
Identify Impact Groups: Initiating Event, System/Train, Combination, Containment

Consequence Categorization
Determine Consequence Ranking from Quantitative Indices or Consequence Category Tables

Classification Considerations
Consider other relevant information, including defense-in-depth principles, for Medium/Low/None consequence categories

Final Classification Definitions
HSS – high-safety-significant
LSS – low-safety-significant

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. All Class 1 items , that portion of the Class 2 feedwater system [> NPS 4 (DN 100)] of Pressurized Water Reactors (PWRs) from the steam generator to the outer containment isolation valve, and items that are within the break exclusion region [> NPS 4 (DN 100)] for high energy piping systems shall be classified High Safety Significant (HSS) and the provisions of the RISC evaluation shall not apply.

I-3.0 EVALUATION OF RISK-INFORMED SAFETY CLASSIFICATIONS

All pressure retaining items and their supports for a piping system, shall be evaluated by defining piping segments that are grouped based on common 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 in high-energy systems. For these high energy systems a Consequence Category is determined from the Consequence Evaluation as defined in I-3.1.1, and I-3.1.2. Low-energy systems shall be evaluated per I-3.2.2(b), (c) and (d).

I-3.1 Consequence Evaluation

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 failure for piping segments. Alternatively, the consequence analysis can be performed assuming a smaller leak, when

(1) a smaller leak is more conservative; or

(2) a small leak can be justified through a leak-before-break analysis in accordance with the criteria acceptable to the regulatory authority having jurisdiction; or

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

(4) design insights do not support a large break based on pressure/temperature/flow in the pipe segment.

(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 effects such as pipe whip, jet impingement, 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 Probabilistic Risk Assessment (PRA), any applicable initiating event is identified using a list of initiating events from that PRA.

(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.

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

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 the impact groups as defined below. Each system, or portion thereof, shall be partitioned into postulated failures that cause an initiating event, disable a system without causing an initiating event, or cause an initiating event and disable a system. In assessing the appropriate consequence category, risk information for all initiating events, including fire and seismic, should be considered. The final consequence category assignment (high, medium, low, or none) for each segment shall be the highest category as determined in accordance with (a) through (d) below.

(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 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 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:

-modeled in a PRA that degrades or fails a high-safety-significant function but does not cause an initiating event, or

-that results in failure of another high-safety-significant piping segment, e.g. through indirect effects, or

-that will prevent or adversely affect the plant's capability to reach or maintain safe shutdown conditions,

shall be based on the following:

(1) Frequency of challenge that determines how often the affected function of the system is called upon. This corresponds to the frequency of 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 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. 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.

For failures modeled in a PRA, 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 plant core damage frequency (CDF) resulting from the pressure boundary failure] and the exposure time, which equates to a change in core damage probability. That is CDF due to the failed segment multiplied by 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.

For failures modeled in a PRA, quantitative indices (CCDP) 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 assessed for their impact on containment performance. This shall be evaluated as follows.

(1) For postulated failures which do not result in a LOCA which bypass 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.

I-3.2 Classification

Risk Informed Safety Classification is determined by considering the Consequence Category in conjunction with other relevant information.

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 (I-3.1) 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) Low-energy systems shall be determined HSS or LSS by considering relevant information for determining classification. As in the consequence evaluation of Section I-3.1.1(a), large pipe failure need not be assumed when design insights do not support a large break based on pressure/temperature/flow in the pipe segment. Also, credit may be taken for plant features and recovery actions[2]. Historical data may also be considered regarding whether the failure is unlikely to occur or could be detected in a timely manner. Historical data should be restricted to items procured to a specification no more stringent than the minimum specification that could be imposed on a similar item determined to be LSS by this process.

- The following conditions shall be evaluated and answered true or not true: Failure of the piping segment will not directly fail a high safety-significant function.
  (1) Failure of the piping segment will not result in failure of a high safety-significant piping segment, e.g., through indirect effects.
  (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 action outside of plant procedures or available backup plant mitigative features.
  (4) The pressure boundary function is not called out or relied upon in the Emergency/Abnormal Operating Procedures or the Severe Accident Mitigation Guidelines as the sole means for the successful performance of operator actions required to mitigate or diagnose an accident or transient.
  (5) The plant condition monitoring program would identify any known active degradation mechanisms in the pipe segment prior to its failure in test or an actual demand event (e.g., flow accelerated corrosion program), or materials resistant to the known damage mechanisms have been used for construction of the component.
  (6) Failure of the piping segment will not result in releases of radioactive material that would result in the implementation of off-site radiological protective actions.

  The RISC process shall demonstrate that the defense-in-depth philosophy is maintained. Defense-in-depth may be demonstrated by following the guidelines acceptable to the regulatory authority having jurisdiction. Defense-in-depth is maintained if:
  (7) A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation
  (8) Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.
  (9) System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).
  (10) Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed.

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[2] To take credit for recovery actions, the following features must be provided:
- An alarm or other system providing clear indication of failure
- Equipment activated to recover from the condition must not be affected by the failure, and
- Time duration and resources required for recovery activities are evaluated.

(11) Independence of fission-product barriers is not degraded.

If any of the above eleven (11) conditions are anwered FALSE, 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:
 (1) Ensuring that safety analysis acceptance criteria in the plant licensing basis are met, or
 (2) Ensuring that proposed revisions account for analysis and data uncertainty.

 If LSS has been assigned from I-3.2.2(b) and at least one of the above safety margin conditions are TRUE, then LSS should be assigned; if both of the above safety margin conditions are FALSE, then HSS shall be assigned.

(d) A support, hanger, or snubber shall have the same classification as the highest-ranked piping segment within the piping analytical model in which it 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 RISC Evaluation Report

A report of the RISC evaluation of the component(s) shall be prepared. A copy of the completed report shall be provided to the Owner or his designee.

I-4.1 Contents of the RISC Evaluation Report

The RISC Evaluation Report shall contain the following, as a minimum:

(a) Introduction (includes background and purpose)
(b) Program Scope & Approach
 (1) Scope of Structures, Systems, and Components Selected for Risk-Informed Safety Classification (includes system and component selection)
 (2) Approach
  i Assembly of Plant-Specific Inputs (includes discussion of PRA models/deterministic insights used)
  ii Consequence Evaluation (includes segment/consequence definition and impact group assessment)
  iii Classification Considerations (includes defense in depth and safety margin assessment)
(c) Categorization Basis
 (1) Plant-specific risk information (includes PRA data or other non-PRA risk insights)
 (2) Characterization of PRA quality (includes technical adequacy of PRA or other risk insights used)
 (3) Results of Consequence Evaluation (includes consequence category ranking results; High, Medium, Low, or None)
 (4) Results of Classification Considerations (includes TRUE/FALSE rationale for Medium, Low, and None consequence category segments)
(d) Documentation
 (1) Documentation of Categorization Process (includes documentation software and method of documenting relevant information)
 (2) Change Control Provisions (includes discussion of reevaluation process and update to the documentation) during the period of construction
(e) References
(f) Appendices
 (1) Exceptions to this case (if applicable)

(2) Other relevant information (e.g., additional plant-specific procedures/guidelines, exceptions to enforcement authority endorsing or governing documents, etc.)

I-4.2 Review of the RISC Evaluation Report

The Owner or his designee shall review the RISC Evaluation Report to determine that all plant and system operating and test conditions (NCA-2141) have been evaluated and that the requirements of this Code Case have been satisfied. Documentation shall be provided by the Owner or his designee to indicate that the review has been conducted. A copy of this documentation shall be filed as a lifetime record (NCA-4134.17) at the location of the installation and made available to the enforcement authorities having jurisdiction over the plant installation before components or supports are placed in service.

I-5.0 Reevaluation of Risk-Informed Safety Classifications

Any modification of the PRA, technical specifications, or any other document that impacts a component's RISC classification shall be reconciled with the RISC Evaluation Report (I-4.0) by the personnel or organization responsible for the evaluation. A revision or addenda to the RISC Evaluation Report shall be prepared and provided to the Owner or his designee for review and filing in accordance with I-4.2

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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		None
II	Anticipated Event	$\geq 10^{-1}$	Reactor Trip, Turbine Trip, Partial Loss of Feedwater	Low/ Medium
III	Infrequent Event	10^{-1} to 10^{-2}	Excessive Feedwater, Excessive Steam Removal	Low/Medium
			Loss of Off Site Power	Medium/High
IV	Limiting Fault or Accident	$<10^{-2}$	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 CT (≤ 1 week)	HIGH	HIGH	MEDIUM	MEDIUM	LOW*	LOW	LOW	LOW
	Short 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 CT (≤ 1 week)	HIGH	MEDIUM	MEDIUM	LOW*	LOW	LOW	LOW	LOW
	Short 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 CT (≤ 1 week)	HIGH	MEDIUM	LOW*	LOW	LOW	LOW	LOW	LOW
	Short CT (≤ 1 day)	HIGH	LOW*	LOW	LOW	LOW	LOW	LOW	LOW

Notes: (1) 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) and (2) CT = Completion Time, see 6. Glossary.

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

Consequence Category	Corresponding CCDP Range	Corresponding CLERP Range
High	$>10^{-4}$	$>10^{-5}$
Medium	$10^{-6} < \text{CCDP} \leq 10^{-4}$	$10^{-7} < \text{CLERP} \leq 10^{-5}$
Low	$\leq 10^{-6}$	$\leq 10^{-7}$
None	No change to base case	No change to base case

PART B Alternative Quality Assurance and Construction Requirements

1. Scope
Applies to Class 1, 2 and 3 components and their supports that have been classified using Part A of this Case.

2. Requirements
(a) Class 1, 2 and 3 HSS components and their supports shall meet the full requirements of subsections NCA, NB, NC and NF as applicable.
(b) Class 2, and 3 LSS components and their supports may use the alternative structural integrity requirements of this Case.

3. Structural Integrity Requirements
The functionality of the component will be maintained by meeting one of the following:
(a) construction codes or standards applicable to the items: ASME, ANSI, AWS, AISC, AWWA, API-650, API-620, MSS-SPs, TEMA and those standards referenced within these documents, or
(b) The requirements of NCA, NB, NC, ND, and NF as applicable, but substituting an Owner's Quality Assurance Program (NCA-8140) to verify supplied material conformance in lieu of a Material Organization's Quality System Program (NCA-3800).

ATTACH. 7

Comparison of Code Case N-660 and NEI 00-04 IDP considerations

Issue	Old N-660 Considerations [Section I-3.2.2(b)]	New N-660 Considerations	Old NEI 00-04 Considerations [Section 9.2.2]	New NEI 00-04 Considerations	Comment
Fail a basic function	1. Failure of the pressure boundary function will not directly fail a high-safety-significant function. 2. Failure of the pressure boundary function as a result of indirect effects will not result in failure of a high-safety-significant function.	1. Failure of the pressure boundary function will not directly or indirectly (e.g., through spatial effect) fail a basic function.	3. Failure of the function/SSC will not adversely affect the defense-in-depth remaining to perform the function. This is evaluated by confirming that failure of an active function/SSC will not directly or indirectly (e.g., through spatial effects) fail a basic function. This applies to any function/SSC under consideration, including functions/SSCs that are assumed to be inherently reliable (e.g., piping and tanks) or those that may not be explicitly modeled in the PRA (e.g., room cooling systems and instrumentation and control systems).	1. Failure of the active function/SSC will not directly or indirectly (e.g., through spatial effects) fail a basic function. This applies to any function/SSC under consideration, including functions/SSCs that are assumed to be inherently reliable (e.g., relief valves) or those that may not be explicitly modeled in the PRA (e.g., and instrumentation and control systems).	N-660: Direct and indirect effects combined into one consideration. NEI 00-04: Defense in depth is covered by other considerations below.

Comparison of Code Case N-660 and NEI 00-04 IDP considerations

Issue	Old N-660 Considerations [Section I-3.2.2(b)]	New N-660 Considerations	Old NEI 00-04 Considerations [Section 9.2.2]	New NEI 00-04 Considerations	Comment
Safe Shutdown	3. 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 action outside of plant procedures or available backup plant mitigative features.	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.	6. Failure of the function/SSC will not prevent the plant from reaching or maintaining safe shutdown conditions; and the function/SSC is not significant to safety during mode changes or shutdown. The IDP should assume that the plant would be unable to reach or maintain safe shutdown conditions if a function/SSC failure results in the need for actions outside of plant procedures or available backup functions/SSCs.	2. Failure of the active function/SSC will not prevent the plant from reaching or maintaining safe shutdown conditions; and the active function/SSC 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 function/SSC failure results in the need for actions outside of plant procedures or available backup functions/SSCs.	Practically no change to old text.

Next version of NEI 00-04 will have revised considerations as presented in this table.

Comparison of Code Case N-660 and NEI 00-04 IDP considerations

Issue	Old N-660 Considerations [Section I-3.2.2(b)]	New N-660 Considerations	Old NEI 00-04 Considerations [Section 9.2.2]	New NEI 00-04 Considerations	Comment
Operator actions to mitigate an accident or transient	4. The pressure boundary function is not called out or relied upon in the Emergency/Abnormal Operating Procedures or the Severe Accident Mitigation Guidelines as the sole means for the successful performance of operator actions required to mitigate or diagnose an accident or transient.	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 function/SSC is not called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means of achieving actions for the successful performance of operator actions required to mitigate an accident or transient. This also applies to instrumentation and other equipment associated with the required actions.	3. The active function/SSC 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. This also applies to instrumentation and other equipment associated with the required actions.	Δ – NEI states "achieving actions" and also includes instrumentation and other equipment associated with the required actions. SAMG removed because it is used only after core damage has occurred.

Deleted:

Comparison of Code Case N-660 and NEI 00-04 IDP considerations

Issue	Old N-660 Considerations [Section I-3.2.2(b)]	New N-660 Considerations	Old NEI 00-04 Considerations [Section 9.2.2]	New NEI 00-04 Considerations	Comment
Operator actions to assure long term containment integrity, monitoring post-accident conditions, or offsite emergency planning activities	NONE	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. The function/SSC is not called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means of achieving actions for assuring long term containment integrity, monitoring of post-accident conditions, or offsite emergency planning activities. This also applies to instrumentation and other equipment associated with the required actions.	4. The active function/SSC 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. This also applies to instrumentation and other equipment associated with the required actions.	Δ – Similar to comment above regarding differences between 660 and 00-04. N-660 did not address this consideration.

Comparison of Code Case N-660 and NEI 00-04 IDP considerations

Issue	Old N-660 Considerations [Section I-3.2.2(b)]	New N-660 Considerations	Old NEI 00-04 Considerations [Section 9.2.2]	New NEI 00-04 Considerations	Comment
Condition monitoring	5. The plant condition monitoring program (e.g., flow accelerated corrosion program) would identify any known active degradation mechanism in the pressure boundary segment prior to its failure in test or an actual demand event.	Statement removed entirely.	A plant condition monitoring program would identify the degradation of the SSC prior to its failure in test or an actual demand event. – (followed all other considerations)	Statement removed entirely.	Statement removed entirely from both sources because it deals mainly with treatment rather than categorization and if a component is determined to be RISC-3 then any condition monitoring program will likely be removed for that given component.
Radioactive release	6. Failure of the pressure boundary function will not result in releases of radioactive material that would result in the implementation of off-site radiological protective actions.	Renumbered to #5.	7. For a function/SSC that acts as a barrier to fission product release during plant operation or during severe accidents, failure of the function/SSC would not result in the implementation of off-site radiological protective actions.	5. Failure of the active function/SSC that acts as a barrier to fission product release during plant operation or during severe accidents would not result in the implementation of off-site radiological protective actions.	Practically no change.

Next version of NEI 00-04 will have revised considerations as presented in this table.

Comparison of Code Case N-660 and NEI 00-04 IDP considerations

Issue	Old N-660 Considerations [Section I-3.2.2(b)]	New N-660 Considerations	Old NEI 00-04 Considerations [Section 9.2.2]	New NEI 00-04 Considerations	Comment
Reasonable balance	7. A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.	6. Reasonable balance is preserved among prevention of core damage, prevention of containment failure or bypass, and mitigation of consequences of an offsite release.	1. Reasonable balance is preserved among prevention of core damage, prevention of containment failure or bypass, and mitigation of consequences of an offsite release.	Renumbered to #7.	No Δ
Programmatic activities	8. Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.	7. There is no over-reliance on programmatic activities and operator actions to compensate for weaknesses in the plant design.	2. There is no over-reliance on programmatic activities and operator actions to compensate for weaknesses in the plant design.	Renumbered to #8.	No Δ
System redundancy	9. System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).	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.	3. 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.	Renumbered to #9.	No Δ

Next version of NEI 00-04 will have revised considerations as presented in this table.

Comparison of Code Case N-660 and NEI 00-04 IDP considerations

Issue	Old N-660 Considerations [Section I-3.2.2(b)]	New N-660 Considerations	Old NEI 00-04 Considerations [Section 9.2.2]	New NEI 00-04 Considerations	Comment
Common cause	10. Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed (e.g., biofouling).	9. Potential for common cause failures is taken into account in the risk analysis categorization.	4. Potential for common cause failures is taken into account in the risk analysis categorization.	Renumbered to #10.	No Δ
Barriers	11. Independence of fission-product barriers is not degraded.	Renumbered to 10.	5. The overall redundancy and diversity among the plant's systems and barriers is sufficient to ensure that no significant increase in risk would occur.	11. Independence of fission-product barriers is not degraded.	No Δ

Next version of NEI 00-04 will have revised considerations as presented in this table.

Comparison of Code Case N-660 and NEI 00-04 IDP considerations

Issue	Old N-660 Considerations [Section I-3.2.2(b)]	New N-660 Considerations	Old NEI 00-04 Considerations [Section 9.2.2]	New NEI 00-04 Considerations	Comment
Cause an initiating event	NONE	No change	1. Failure of the function/SSC will not directly cause an initiating event, including those initiating events originally screened out of the PRA based on anticipated low frequency of occurrence.	Deleted.	N-660: Any impact on initiating events already addressed in risk assessment. NEI 00-04: Anything that could significantly increase initiating event frequencies would already be HSS from the risk assessment.
Loss of reactor coolant pressure boundary	NONE	No change	2. Failure of the active function/SSC will not cause a loss of reactor coolant pressure boundary integrity resulting in leakage beyond normal makeup capability.	Renumbered to #6.	N-660: The RCPB is not in the scope for N-660 classification.

Next version of NEI 00-04 will have revised considerations as presented in this table.