

ASME NTB-5-2022

Guidance for Determination of
Risk-Informed Safety Classification
for Light Water Reactor
Nuclear Facility Pressure
Retaining Components



ASME NTB-5-2022

GUIDANCE FOR DETERMINATION OF RISK-INFORMED SAFETY CLASSIFICATION FOR LIGHT WATER REACTOR NUCLEAR FACILITY PRESSURE RETAINING COMPONENTS

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FOREWORD

Over the last forty years there have been significant gains in the understanding of pressure boundary component integrity, factors that impact component reliability, the impact of inspections and the type of inspection, as well as risk assessment insights related to operating nuclear power reactors. This experience has brought about changes related to operating and inspection requirements including changes to ASME Section XI and Operation and Maintenance (OM) requirements, augmented inspection programs mandated by the regulator as well as plant specific actions taken by individual owners.

For ASME Section XI programs, these efforts have included new and revised code cases (e.g. N560, N577, N578, N660, N716 and N752) and the development of pilot and follow-on plant specific applications. For U.S. Nuclear Regulatory Commission (NRC) mandated programs, these efforts have included integration with risk-informed in-service inspection (ISI) programs, performance based initiatives as well as extension to new areas including break exclusion region/high energy line break BER/HELB requirements.

The action discussed in this NTB takes advantage of the aforementioned work and proposes a balanced action that reduces undue burden while ensuring plant safety. This action was in part spurred on by the NRC's Advisory Committee on Reactor Safeguards (ACRS), who in 1999, chided the industry as being "overly timid" in implementing risk-informed technology.

Thus, this action represents the next step in the use of risk-informed technology for defining ASME Section III requirements. This action builds upon the work done at ASME Sections III and XI and OM Code, the industry and the NRC in developing and implementing risk-informed classification, in-service and pre-service inspection activities as well as in-service testing activities. This action provides a balanced and reasonable alternative to existing requirements for pressure boundary classification and applicable "treatment" activities. This approach provides for transparency, reproducibility and stability to the code, code users, as well as regulatory bodies.

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ABSTRACT

Code Case N660, Revision 0 “Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities,” was published by ASME and was developed to expand the breadth of risk informed ASME Section XI requirements to pressure boundary components. This effort was conducted in conjunction with NRC and industry efforts dealing with risk informing Title 10 Code of Federal Regulations Part 50 (10CFR50) as outlined in SECY-98-300 (NRC, 1998).

Since N660, Rev 0 was published several important accomplishments have occurred. The South Texas Project (STP) exemption request was approved and implementation is underway, N660, R0 was tested on a number of systems, a final 10CFR50.69 rule was published, NEI00-04 was developed and updated and trial applications were conducted.

Based upon lesson learned from the above, an alternative to N660 was developed (draft code case N752) and used by ANO, Unit 2 in their risk-informed repair/replacement application. Draft code case N752 was ultimately updated to reflect lessons learned from this NRC approved application and was published in 2019.

Since that time, Vogtle Units 1 and 2 have been approved to use the draft code case N752 methodology in their 10CFR50.69 pilot plant application which was approved by the NRC in 2014. More recently, the US industry is moving quickly forward with site specific 10CFR50.69 license amendment requests (thirty three additional units approved to date), each using code case N752 methodology for categorization of the pressure boundary.

Further, Regulatory Guide 1.26, has been updated (revision 5) to reflect the use of risk-informed classification processes. In particular, it references regulatory positions for the acceptable use of processes to determine the safety significance of SSCs and place them into the appropriate risk-informed safety class (RISC) categories

This document provides guidance for determination of risk-informed safety classification for light water reactor (LWR) nuclear facility pressure retaining components and their welded attachments and supports, ASME Section III, Division 1, Subsections NCA, NB, NC, ND, and NF.

ABBREVIATIONS AND ACRONYMS

AISC	American Institute of Steel Construction
ANSI	American National Standards Institute
API	American Petroleum Institute
ASME	American Society of Mechanical Engineers
CCDP	Conditional Core Damage Probability
CFR	U.S. Code of Federal Regulations
CLERP	Conditional Large Early Release Probability
FMEA	Failure Modes and Effects Analysis
HSS	High-Safety Significant
IE	Initiating Event
LSS	Low-Safety Significant
NRC	U.S. Nuclear Regulatory Commission
O&M	Operations and Maintenance
PDF	Portable Document Format
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
QA/QC	Quality Assurance / Quality Control
RISC	Risk Informed Safety Classification
SCADA	Supervisory Control and Data Acquisition
Section III	Section III of the ASME Boiler and Pressure Vessel Code, Rules for Construction of Nuclear Facility Components
SME	Subject Matter Expert

DEFINITIONS

basic safety function – one of the key safety functions of the plant; reactivity control, core cooling, heat sink, and reactor coolant system (RCS) inventory [Note: loss of a single train would typically not constitute a loss of a function]

completion time (CT) – the amount of time allowed for returning a component or function to service. In the context of this document, 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 the probability of core damage given a specific failure (e.g., piping segment failure)

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

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 automatically close when containment isolation is required

core damage – uncover and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in offsite public health effects

failure – an event involving leakage, rupture, or other condition that would prevent an item from performing its intended safety function

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

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

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

high-energy piping system – a system that is either in operation or maintained pressurized under conditions where either or both of the following are met:

- a. operating temperature exceeds 200 °F (95 °C), or
- b. operating pressure exceeds 275 psig (1.90 MPa)

high-safety-significant (HSS) function – a function that has been determined to be safety significant from an approved risk-informed categorization process using a plant-specific probabilistic risk assessment and / or other relevant deterministic information (e.g., defense-in-depth philosophy¹ considerations) as described in I-3.2.

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 an earthquake or a transient or loss of coolant accident (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

¹ U.S. NRC Regulatory Guide 1.174 provides a definition for defense-in-depth philosophy

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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 such that there is a potential for early health effects

loop – a subset of a system or a train (e.g. many emergency core cooling systems in PWR plants contain four injection paths also known as injection loops)

low-safety-significant (LSS) function – a function not determined to be high-safety significant from an approved risk significance categorization process using a plant-specific plant probabilistic risk assessment and / or other relevant deterministic information (e.g., defense-in-depth philosophy¹ considerations) as described in I-3.2.

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

piping segment – a continuous portion of piping, components, or a combination thereof in which a failure (i.e., loss of its pressure-retaining function) 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)

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 used in this document, 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 a mean unavailability of 1E-02 as credited in Table I-2 and Table I-3. A half train (0.5 trains) should have a mean unavailability of 1E-01, 1.5 trains should have a mean unavailability of 1E-03, etc.

unaffected backup train – a train 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.

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1 SCOPE

This document provides a process for determining the Risk-Informed Safety Classification (RISC) of light water reactor nuclear facility Class 1, 2 and 3 pressure-retaining components and their welded attachments and supports. Pressure retaining components may include passive components (e.g., piping, vessels, tanks, etc.) and the pressure-retaining portion of active components (e.g., valve bodies, pump casings, etc.) and their welded attachments and supports.

2 APPLICABILITY

Guidance in this document is intended to be used by the Owner², or his designee, to conduct risk classification for the purpose of specifying appropriate Codes and requirements for the construction of an LWR system or component. Having used this guidance, the Owner then specifies the Risk-informed Safety Classification, applicable Codes and all appropriate requirements for construction in the Design Specification as required by Section III of the ASME Boiler and Pressure Vessel Code, Paragraph NCA-3250.

² Owner: the organization legally responsible for the construction and/or operation of a nuclear facility including but not limited to one who has applied for, or has been granted, a construction permit or operating license by the regulatory authority having lawful jurisdiction. (NCA-9000)

3 RISK-INFORMED SAFETY CLASSIFICATIONS

- (a) The RISC process is described in Appendix I of this document. Pressure-retaining items should be classified high-safety significant (HSS) or low-safety significant (LSS), except as noted in 3(b) below.
- (b) (1) Class 1 portions of the reactor coolant pressure boundary (RCPB) that do not meet (i) or (ii) below:
 - (i) in the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system.
 - (ii) the component is or can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and, assuming the other valve is open, its closure time must be such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.
- (2) ASME Boiler and Pressure Vessel Code, Section III, Class 1 items, that portion of the Class 2 feedwater system [$> \text{NPS } 4 \text{ (DN } 100\text{)}$] of Pressurized Water Reactors (PWRs) from the steam generator to the outer containment isolation valve, and
- (3) items that are within a break exclusion region³ [$> \text{NPS } 4 \text{ (DN } 100\text{)}$] for high-energy piping systems and their associated supports (NB, NC and NF) should be classified high-safety significant (HSS).

³ Break exclusion region should 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. NUREG-0800, Sections 3.6.1 and 3.6.2 provide a definition of break exclusion region.

4 DETERMINATION OF RISC

In accordance with NCA-2110(c), the Owner, or his designee, should be responsible for providing system safety criteria to classify equipment in the nuclear power plant. This responsibility includes providing appropriate RISC classification in accordance with Appendix I of this document. In addition to the requirements of NCA-3252, the Design Specification should contain or reference the RISC Evaluation Report (I-4.0) and this document.

5 APPLICABLE DISCIPLINES

Personnel with expertise in the following disciplines should be included in addressing I-3.2.2, I-3.2.3 and I-4.2 of the RISC classification process.

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

Other disciplines may be added, such as materials engineering, chemistry, or nondestructive examination, relevant to the specific system or equipment issues. Personnel may be experts in more than one discipline, but are not required to be experts in all disciplines. For new system or equipment designs, where there is limited service experience, personnel with expertise from similar plant designs (e.g., earlier or same versions or models) should be used.

To qualify as an expert, personnel should be experienced in the applicable discipline and related nuclear power plant requirements, and in the application of the requirements of the Code relating to the applicable discipline. Personnel selected for their expertise should have a minimum of four years of varied nuclear application experience, including 2 years in the applicable discipline for which they are serving as an expert. This experience should indicate that the expert has sufficient knowledge of anticipated plant and system operating and test conditions and their relationship to Code design criteria pertinent to the applicable Code item. In addition, the expert should be knowledgeable of the specific Code requirements pertaining to his specialty field. Guidelines reflecting the appropriate degree of Code knowledge are contained in ASME Section III, Division 1, Appendix XXIII “Guide B - Nonmandatory Guidelines for Establishing Code Knowledge.”

6 PRA SCOPE AND TECHNICAL ADEQUACY

The PRA should be of sufficient scope and level of detail to support the RISC process, including verification of assumptions on equipment reliability from equipment not within the scope of this document. The PRA should be subjected to a review process where it is assessed against a standard⁴ or set of acceptance criteria that is accepted by the regulatory agency having jurisdiction over the plant site. All deficiencies identified that impact the RISC process should be reconciled during the analysis to support the RISC process. The resolution of all PRA issues that impact the RISC process should be documented. EPRI report 1021467-A (Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-informed In-Service Inspection Programs, Palo Alto, CA: 2011. 1021467) provides one example of addressing PRA completeness.

⁴ A standard that may be used for this application is ASME/ANS RA-Sa-2009, *Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications*. This standard sets forth requirements for PRAs used to support risk-informed decisions for commercial nuclear power plants and prescribes a method for applying these requirements for various categories of applications.

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

I-1.0 INTRODUCTION

This Appendix describes the risk-informed process that should be 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 (i.e., failure is assumed to occur with a probability of 1.0). This process divides each selected piping system⁵ into piping segments that are determined to have similar consequences of failure. These piping segments consist of piping, components, or a combination thereof, and their supports, and 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.

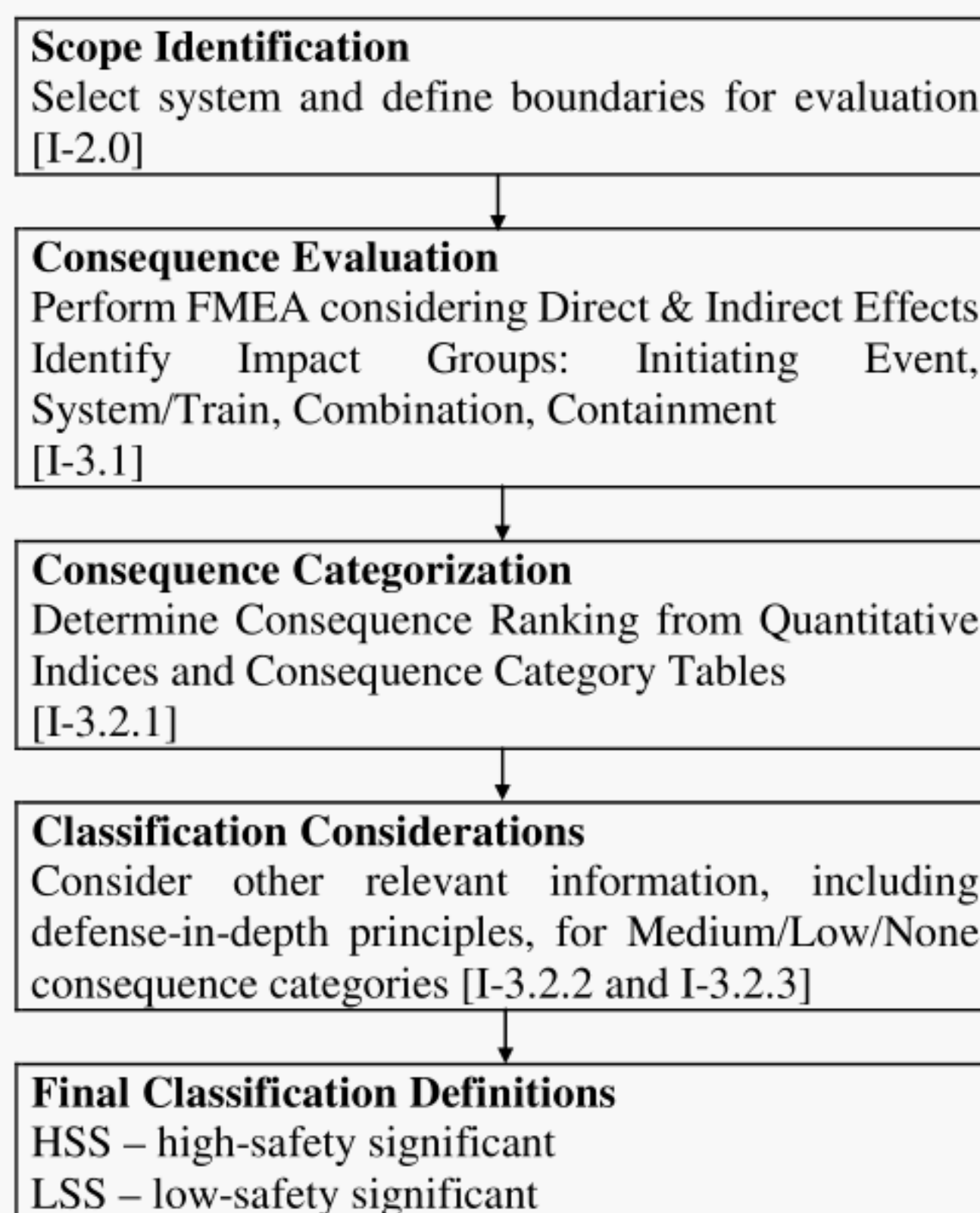


Figure I-1: Risk-Informed Safety Classification Process

I-2.0 SCOPE IDENTIFICATION

The Owner, or his designee, should define the boundaries included in the scope of the RISC evaluation process. This should include defining components boundaries that provide the interface between two systems (e.g. heat exchangers).

⁵ Note that “system” and “piping” system are used interchangeably throughout this document.

I-3.0 EVALUATION OF RISK-INFORMED SAFETY CLASSIFICATIONS

All pressure-retaining items in a piping system should be evaluated by defining piping segments that are grouped based on similar conditional consequence (i.e., given failure of the piping segment). To accomplish this grouping, the direct effects and indirect effects should be assessed for each piping segment. A Consequence Category for each piping segment is determined from the Failure Modes and Effects Analysis and Impact Group Assessment as defined in I-3.1.1, and I-3.1.2, respectively. The failure consequence can be quantified using a PRA(s) meeting the guidance of Section 5 of this document. Throughout the evaluations of I-3.0, I-3.1, and I-3.2, credit may be taken for plant features and operator actions to the extent these would not be affected by failure of the piping segment under consideration. When crediting operator action, the likelihood for success and failure needs to be determined and the scenario that results in the highest consequence ranking should be used⁶. To take credit for operator actions, the following features should be provided:

- (a) An alarm or other system feature to provide clear indication of failure,
- (b) Equipment activated to recover from the condition must not be affected by the failure,
- (c) Time duration and resources are sufficient to perform operator action,
- (d) Plant procedures to define operator actions, and
- (e) Operator training in the procedures.

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 should be identified, and their effects should be evaluated. This evaluation should consider the following:

- (a) Pressure Boundary Failure Size. The consequence analysis should be performed for a spectrum of pressure boundary failure sizes from small to large. The failure size that results in the highest consequence ranking should be used.

In lieu of this, a small leak may be assumed provided it can be ensured that the possibility of a large pressure boundary failure has been precluded (e.g., presence of a flow restricting orifice).
- (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. In lieu of automatic isolation, operator action may be credited consistent with I-3.0.
- (c) Indirect Effects. These include spatial effects (e.g., pipe whip, flooding) and loss-of inventory effects (e.g., draining of a tank).
- (d) Initiating Events. Applicable initiating events are identified using a list of initiating events from the plant-specific PRA and the plant design basis. For system failures or piping segments failures that are not modeled, either explicitly or implicitly, in the plant-specific PRA, analysis might be required to identify applicable initiating events (i.e. a representative failure effect is not modeled in the PRA). This analysis should be conducted in accordance with this Appendix.
- (e) System Impact or Recovery. These are 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 should be evaluated consistent with I-3.0.

⁶ Further details on the evaluation of operator actions and its impact on the consequence ranking, the evaluation and ranking of the consequence impact groups and configurations and the evaluation of shutdown and external events are discussed in EPRI Report TR-112657, Rev B-A and the associated NRC Safety Evaluation Report dated Oct. 28, 1999.

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- (f) System Redundancy. The existence of redundancy for accident mitigation purposes should be considered.
- (g) System Configuration. The consequence evaluation and ranking is organized into four basic consequence impact groups as discussed in I-3.1.2. The three corresponding system configurations for these impact groups are defined in Table I-6.

I-3.1.2 Impact Group Assessment

The results of the FMEA evaluation for each piping system, or portion thereof, should be classified into one of three core damage impact groups: initiating event, system, or combination. In addition, failures should also be evaluated for their importance relative to containment performance. Each piping system, or portion thereof, should be partitioned such that failure of each partition can cause an initiating event, disable a system without causing an initiating event, or cause an initiating event and disable a system, train or loop. The consequence category assignment (high, medium, low, or none) for each piping segment should be the highest category as determined in accordance with (a) through (f) 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 should be classified into one of four consequence categories: high, medium, low, or none. The initiating event category should be assigned according to the following:
 - (1) The initiating event should be placed in one of the Design Basis Event Categories in Table I-1. All applicable design basis events previously analyzed in the preliminary or final safety analysis report (PSAR/FSAR) or PRA should 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 should be assigned to the initiating event according to the conditional core damage probability (CCDP) criteria specified in Table I-5. Differences in the consequence rank between the use of Table I-1 and Table I-5 should be reviewed, justified and documented or the higher consequence rank assigned. The quantitative index for the initiating event impact group is the ratio of the core damage frequency due to the initiating event to the frequency for that initiating event.
- (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, should be evaluated. This evaluation should include all safety functions supported by the piping segment as well as all safety functions impacted by failure of the piping segment. This evaluation should 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 loops) available, which determines how many unaffected systems (portions of systems, trains, or loops) 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 or allowed outage time, as defined in the plant Technical Specification.

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Consequence categories should 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. Quantitative indices may be used to assign consequence categories in accordance with Table I-5 in lieu of Table I-2 provided the Owner, or his designee, ensures that the quantitative basis of Table I-2 (e.g., one full train unavailability approximately 10^{-2} , exposure time) is consistent with the failure scenario being evaluated. Differences in the consequence rank between the use of Table I-2 and Table I-5 should be reviewed, justified, and documented or the higher consequence rank should be assigned. The quantitative index for the system impact group (i.e., CCDF) is the product of the change in conditional core damage frequency (CCDF) and the exposure time. The CCDF is the difference in the CDF (given loss of the system/train) and the CDF (from the base PRA). Additionally, all postulated failures leading to “zero defense” (i.e., no backup trains) should be assigned a high consequence rank.

- (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 should be determined using Table I-3. 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 (CCDF or CLERP) may be used to assign consequence categories in accordance with Table I-5 in lieu of Table I-3 provided the Owner, or his designee, ensures 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. Differences in the consequence rank between the use of Table I-3 and Table I-5 should be reviewed, justified and documented or the higher consequence rank should be assigned.

- (d) **Containment Performance Impact Group Assessment.** The above evaluations determine failure importance relative to core damage. Failures should also be assessed for their impact on containment performance. This should be evaluated as follows:
 - (1) For postulated failures which do not result in a LOCA that bypasses containment, the quantitative indices of Table I-5 for CLERP should be used.
 - (2) Table I-4 should be used to assign consequence categories for those piping failures that can lead to a LOCA that occurs outside of containment.
- (e) **Shutdown operation** should be evaluated. The previously established consequence rank should be reviewed and adjusted to reflect the pressure boundary failure’s impact on plant operation during shutdown⁷.

If the plant has a shutdown PRA, the important initiators and systems will have already been identified for shutdown operation and their effect on core damage and containment performance evaluated. If a shutdown PRA is not available, the effect of pressure-boundary failures on core damage and containment performance should be evaluated. The major characteristics to be considered are defined as follows:

- (1) The system operations, safety functions, and success criteria change in different stages of other modes of operation.
- (2) The exposure time for the majority of the piping associated with shutdown operation is typically less than 10 percent per year. The exposure time associated with being in a more risk-significant configuration is even shorter, depending on the function or system that is being

⁷ Further details are discussed in NUMARC 91-06, “Guidelines for Industry Actions to Address Shutdown Management,” dated 1991.

evaluated.

- (3) The unavailability of mitigating trains could be higher due to planned maintenance activities. Shutdown guidelines need to be evaluated to assure that sufficient redundancy is protected during different modes of operation.
- (4) Recovery time may be longer, thus allowing for multiple operator actions.
- (f) External events should be evaluated. The previously established consequence rank should be reviewed and adjusted to reflect the pressure boundary failure's impact on the mitigation of external events.

The effect of external events on core damage and containment performance should be evaluated from two perspectives, as follows:

- (1) External events that can cause a pressure boundary failure (e.g., seismic events), and
- (2) External events that do not affect the likelihood of pressure boundary failure but create demands that might cause pressure boundary failure and events (e.g. fires).

I-3.2 CLASSIFICATION

I-3.2.1 Final Risk-Informed Safety Classification

Piping segments may be grouped together within a system, if the analysis and assessment performed in I-3.1 determines the effect of the postulated failures to be the same. The Final Risk-Informed Safety Classification should 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 (Table I-1 through Table I-5) by the analysis and assessment in I-3.1 should 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 analysis and assessment in I-3.1 should be determined HSS or LSS by considering the information in (1) through (6) below. Under the same conditions of I-3.1.1(a), a large pressure boundary failure does not need to be assumed. Additionally, credit may be taken for plant features and operator actions to the extent that these would not be affected by failure of the piping segment under consideration. If plant features and/or operator actions are credited, they should be consistent with those credited in I-3.1. The following conditions should be evaluated and answered TRUE or FALSE:
 - (1) Failure of the pressure-retaining function will not directly or indirectly (e.g., through spatial effects) fail a basic safety function.
 - (2) Failure of the pressure-retaining 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.
 - (3) The pressure-retaining function is not called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means for the

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successful performance of operator actions required to mitigate an accident or transient.

- (4) The pressure-retaining 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-retaining function will not result in an unintentional release of radioactive material that would result in the implementation of off-site radiological protective actions.
- (6) The RISC process demonstrates that the defense-in-depth philosophy is maintained. Defense-in-depth is maintained if the following are met:
 - (i) reasonable balance is preserved among prevention of core damage, prevention of containment failure or bypass, and mitigation of an offsite release;
 - (ii) there is no over-reliance on programmatic activities and operator actions to compensate for weaknesses in plant design;
 - (iii) 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;
 - (iv) potential for common cause failures is taken into account in the risk analysis categorization;
 - (v) independence of fission-product barriers is not degraded.

If any of the above six (6) conditions are answered FALSE, then HSS should be assigned. Otherwise, LSS may be assigned.

- (c) If LSS has been assigned from I-3.2.2(b), then the RISC process should verify that there are sufficient safety margins to account for uncertainty in the engineering analysis and in the supporting data. Safety margin should 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 changes are proposed to the plant safety analyses they should be approved by the regulatory authority having jurisdiction at the plant site.

If sufficient safety margins are maintained, then LSS may be assigned; if not, then HSS should be assigned.

- (d) All supports should have the same classification as the highest-ranked piping segment in the portion of piping between piping system structural boundaries, as defined in I-3.2.3(b), in which the supports are included.

I-3.2.3 Additional Considerations

- (a) The piping segment boundaries should be established at piping components where pressure boundary isolation of the higher safety significant piping segment can be maintained. This boundary could include such piping components as isolation valves, check valves, normally closed manual valves, pressure reducing devices, break exclusion region, wall, etc.

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- (b) The piping system structural boundary for design and analysis purposes should be established at the first structural isolation location past the system isolation boundary in the lower safety significant piping. This boundary could include such items as substantial plant components (pumps, tanks, etc.), penetrations or pipe supports that are designed with the assumption that essentially zero piping deflection and rotation in three orthogonal directions is permitted or break exclusion region boundary restraints.

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A report of the RISC evaluation of the component(s) should be prepared. A copy of the completed report should be retained by the Owner.

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The RISC Evaluation Report should contain the following, as a minimum.

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