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Director,
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CNRO-2007-00015

April 17, 2007

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Request for Alternative ANO2-R&R-004, Revision 1
Request to Use Risk-Informed Safety Classification and Treatment for Repair / Replacement Activities in Class 2 and 3 Moderate Energy Systems

Arkansas Nuclear One, Unit 2
Docket No. 50-368
License No. NPF-6

Dear Sir or Madam:

Pursuant to 10 CFR 50.55a(a)(3)(i), Entergy Operations, Inc. (Entergy) requests the NRC staff authorize the use of a risk-informed safety classification and treatment for repair/replacement activities in Class 2 and 3 moderate energy systems at Arkansas Nuclear One, Unit 2 (ANO-2). This request is provided in Enclosure 1 as Request for Alternative ANO2-R&R-004, Revision 1. A risk-informed safety classification process for Class 2 and 3 moderate energy systems is included in the request.

Entergy requests the NRC staff approve ANO2-R&R-004 by December 1, 2007. Should you have any questions regarding this submittal, please contact Guy Davant at (601) 368-5756.

This letter contains one commitment identified in Enclosure 2.

Very truly yours,

for

JFM/GHD/ghd
Enclosures: 1. Request for Alternative ANO2-R&R-004, Revision 1
2. Licensee-Identified Commitment

A047

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ENCLOSURE 1

CNRO-2007-00015

ANO2-R&R-004, Revision 1

ENTERGY OPERATIONS, INC.
ARKANSAS NUCLEAR ONE, UNIT 2

REQUEST FOR ALTERNATIVE
ANO2-R&R-004, Revision 1

I. COMPONENTS

Component Numbers:	Class 2 and 3 pressure boundary components in moderate energy systems
Code Classes:	2 and 3
References:	1. ASME Section XI, IWA-4000 2. 10 CFR 50.69, *Risk-Informed Categorization and Treatment of Structures, Systems, and Components of Nuclear Power Reactors* 3. EPRI TR-112657, Rev B-A, *Revised Risk-Informed Inservice Inspection Evaluation Procedure*, dated December 1999 4. NRC letter to Entergy Operations, Inc. (2CNA129805), *Request to Use a Risk-Informed Alternative to the Requirements of ASME Code Section XI, Table IWX-2500 at Arkansas Nuclear One, Unit No. 2 (TAC NO. M99756)*, dated December 29, 1998
Examination Category:	Various
Item Number:	Various
Description:	Alternative Requirements for Repair / Replacement Activities in Class 2 and 3 Moderate Energy Systems
Unit / Inspection Interval Applicability:	Arkansas Nuclear One, Unit 2 (ANO-2) – 3rd and future intervals

II. CODE REQUIREMENT(S)

ASME Section XI, IWA-4000 (Ref. 1) provides requirements for repair / replacement activities in Class 1, 2, and 3 components.

III. REQUESTED AUTHORIZATION

Pursuant to 10 CFR 50.55a(a)(3)(i), Entergy Operations, Inc. (Entergy) proposes to use the categorization process contained in Attachment 1 of this request at ANO-2 for components scoped within Class 2 and 3 moderate energy systems. Specifically, this process is used to determine the risk-informed safety classification (RISC) for repair/replacement activities applied to Class 2 and 3 pressure-retaining items or their associated supports (exclusive of Class CC and MC items) in moderate energy systems.

This process shall be applied on a system basis, including pressure-retaining items and their associated supports within the selected systems.

Upon completing the categorization process, components are ranked as either high safety significant (HSS) or low safety significant (LSS). Those components that are HSS will continue to meet existing ASME requirements for repair/replacement activities. Those components that are LSS will be exempt from ASME Section XI repair/replacement requirements. This approach is consistent with the process defined in 10 CFR 50.69 (Ref. 2).

Entergy requests that the NRC grant this request for the life of the facility.

IV. BASIS FOR THE PROPOSED ALTERNATIVE

ASME Code Case N-660, *Risk-Informed Safety Categorization for Repair/Replacement Activities*, was developed to support NRC and industry promulgation of 10 CFR 50.69. Since the time N-660 was developed, trial application of the code case has been conducted and §50.69 has transitioned from an Advanced Notice of Public Rulemaking (ANPR) to a final rule. Lessons learned from these trial applications have resulted in various attempts to revise N-660 and, ultimately, have resulted in developing the RISC process specified in Attachment 1, which is limited in application to Class 2 and 3 moderate energy systems.

Code Case N-660, as well as Attachment 1, is founded on the EPRI risk-informed ISI (RI-ISI) methodology documented in EPRI Report TR-112657 (Ref. 3). Entergy, through its active participation in EPRI, has been instrumental in the research and development that formed the EPRI methodology as well as its application and use within the industry. Additionally, a number of Entergy units served significant roles in attaining NRC acceptance of the technology and actual plant application. These include:

- The first-ever approved RI-ISI application [Vermont Yankee (VY)];
- The first full-scope application to a CE nuclear steam supply system (NSSS) design (ANO-2);
- The first partial-scope applications to a B&W NSSS design (ANO-1);
- The first full-scope follow-on BWR application (James A. FitzPatrick).

As part of the above activities, Entergy submitted, and NRC staff reviewed, the supporting calculations and analyses for the VY, ANO-1, and ANO-2 applications. As such, the staff is intimately familiar with the EPRI RI-ISI methodology and, in particular, its application at Entergy.

The diligent effort put forth by EPRI, Entergy, and the NRC staff has identified the robustness of the EPRI methodology to the point that now 80% of plants with approved RI-ISI programs, or with programs being implemented, are using the EPRI RI-ISI methodology or related products (e.g., ASME Code Case N-663).

As stated above, the classification process in Attachment 1 is founded on the EPRI RI-ISI methodology. In essence, it consists of the consequence assessment portion of the RI-ISI methodology, supplemented with the "additional considerations" contained in NEI 00-04, *10 CFR 50.69 SSC Categorization Guideline*.

Upon approval of ANO2-R&R-004, Revision 1, Entergy will conduct the evaluation of the "additional considerations" at ANO-2 and, as applicable, revise the consequence ranking assignments for Class 2 and 3 moderate energy components when used in risk-informed repair/replacement activities. (This will not impact/change the existing approved RI-ISI programs).

Consistent with 10 CFR 50.69, for those components identified as LSS, Entergy will replace the existing Section XI requirements with owner-defined periodic inspection and testing activities to confirm with reasonable confidence that each LSS item remains capable of performing its safety-related function(s) under design basis conditions. Any condition identified that would prevent a LSS component from performing its safety-related function(s) under design basis conditions will be corrected in a timely manner. For significant conditions adverse to quality that may be identified, measures will be taken via Entergy's Appendix B corrective action program to provide reasonable confidence that the cause of the condition is determined and corrective action taken to preclude repetition.

Entergy shall review changes to the plant, operational practices, applicable plant and industry operational experience, and, as appropriate, update the probabilistic risk assessment (PRA) and the categorization and treatment processes. Entergy shall perform this review in a timely manner but no longer than once every two refueling outages.

IV. <u>CONCLUSION</u>

10 CFR 50.55a(a)(3) states:

"Proposed alternatives to the requirements of (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that:

(i) The proposed alternatives would provide an acceptable level of quality and safety, or

(ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."

As discussed above, Entergy believes Request for Alternative ANO2-R&R-004, Revision 1 provides a level of safety and quality consistent with Code requirements. Additionally, the alternative is consistent with the NRC's risk-informed classifications and treatment specified in 10 CFR 50.69. Therefore, Entergy requests the NRC authorize the proposed alternative in accordance with 10 CFR 50.55a(a)(3)(i) for the third and future intervals at ANO-2.

REQUEST FOR ALTERNATIVE
ANO1-R&R-004, Revision 1

ATTACHMENT 1

RISK-INFORMED SAFETY CLASSIFICATION (RISC) PROCESS FOR CLASS 2 AND 3 MODERATE ENERGY SYSTEMS

RISK-INFORMED SAFETY CLASSIFICATION (RISC) PROCESS FOR CLASS 2 AND 3 MODERATE ENERGY SYSTEMS

I-1.0 INTRODUCTION

This attachment describes the risk-informed process that will be used by Entergy to determine the risk-informed safety classification (RISC) for Class 2 and 3 moderate energy systems. This RISC evaluation process, which is founded upon EPRI Report TR-112657 (Ref. 3) and the NRC-approved risk-informed inservice inspection (RI-ISI) application for ANO-2 (Ref. 4), is based on the conditional consequence of failure, given that the postulated failure has occurred.

Piping segments are categorized based on the conditional consequence of failure. This process divides each selected system into piping segments that are determined to have similar consequences of failure. Once categorized, the safety significance of each piping segment is identified. Figure I-1 illustrates the RISC methodology presented in the following sections.

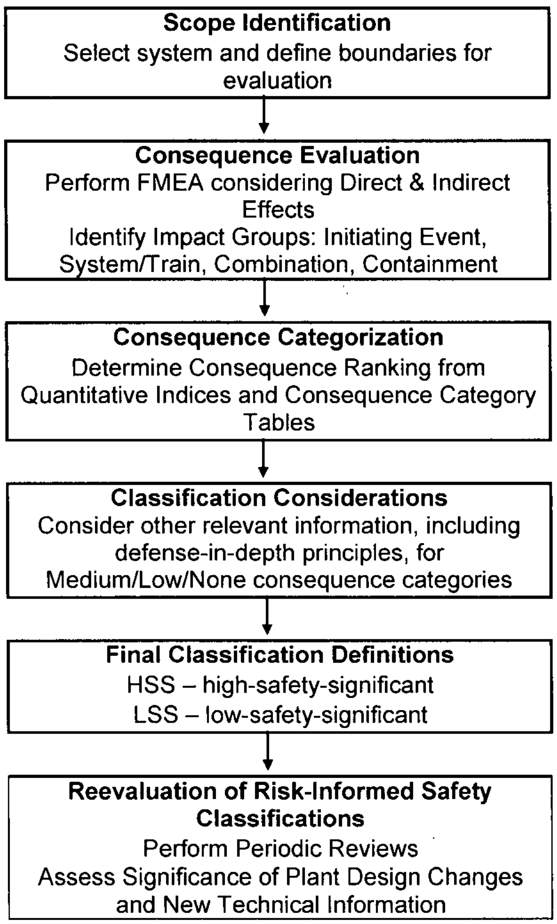

FIGURE I-1

Risk-Informed Safety Classification Process

I-2.0 SCOPE IDENTIFICATION

Entergy will define the boundaries included in the scope of the RISC evaluation process consistent with the previously approved RI-ISI application.

I-3.0 CONSEQUENCE EVALUATION

I-3.0.1 Introduction

Pressure-retaining items shall be evaluated by defining piping segments that are grouped based on similar conditional consequences (i.e., given failure of the piping segment). To accomplish this grouping, direct and indirect effects shall be assessed for each piping segment. A consequence category for each piping segment is determined from the failure modes and effects analysis (FMEA) and impact group assessment as defined in Sections I-3.1.1 and I-3.1.2, respectively. The failure consequence can be quantified using the available probabilistic risk assessment(s) (PRA) to support the impact group assessment of Section I-3.1.2. Throughout the evaluations specified in Sections 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 segment under consideration. When crediting operator action, the likelihood for success and failure will be determined consistent with ANO-2's NRC-approved RI-ISI application (Ref. 4). The scenario that results in the highest consequence ranking shall be used.[1] As an example, to take credit for operator actions, the following features shall be provided:

- An alarm or other system feature provides 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 define operator actions; and
- Operators are trained on the procedures.

I-3.0.2 PRA Scope and Technical Adequacy

The technical adequacy of the PRA used to support the evaluations required by this attachment shall be assessed. The PRA technical adequacy basis for the ANO-2 RI-ISI program application shall be reviewed to confirm it is applicable to the safety significant categorization of this application, including verifying assumptions on equipment reliability for equipment not within the scope of this request.

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[1] Further details on evaluating operator actions and their impact on the consequence ranking are provided in NRC Safety Evaluation Report dated October 28, 1999 pertaining to EPRI Report TR-112657, Rev B-A.

I-3.1 ANALYSIS AND ASSESSMENTS

I-3.1.1 Failure Modes and Effects Analysis (FMEA) [2]

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

For moderate energy systems that have been designed and constructed to the requirements (i.e., administrative and technical) of one of the following codes or standards applicable to that item -- ASME, ANSI, AWS, AISC, AWWA, API-650, API-620, MSS-SPs, TEMA, and those standards referenced within these documents -- the consequence evaluation may be performed assuming a small leak that is:

(1) Determined by analytical evaluations[3] that include relevant design basis conditions (e.g., pressure, temperature, SSE); or

(2) Ensured due to a documented physical configuration that precludes the possibility of a large pressure-boundary failure (e.g., flow restricting orifice); or

(3) In lieu of (1) or (2), evaluated for a spectrum of pressure-boundary failure sizes (i.e., small to large).

(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 Section I-3.0.1.

(c) Indirect Effects

Indirect effects include spatial effects (e.g., spray) 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 systems or piping segments that are not modeled in the plant-specific PRA, either explicitly or implicitly, analysis may be required to identify applicable initiating events. This analysis shall be conducted in accordance with this attachment.

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[2] Further details on evaluating and ranking the consequence impact groups and configurations are discussed in NRC Safety Evaluation Report dated October 28, 1999 and EPRI Report TR-112657, Rev B-A.

[3] Generic Letter 90-05, *Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping,* provides an example of acceptable guidance in determining alternate leak sizes.

(e) System Impact or Recovery

System impact or recovery involves the means of detecting a failure and the Technical Specification actions associated with the system and other affected systems. Possible automatic and operator actions to prevent a loss-of-system function shall be evaluated.

(f) System Redundancy

The existence of redundancy for accident mitigation purposes shall be considered.

(g) System Configuration

The consequence evaluation and ranking is organized into four basic consequence impact groups as discussed in Section 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 [4]

The results of the FMEA for each system, or portion thereof, shall be classified into one of the following three core damage impact groups: (1) Initiating Event, (2) System, or (3) Combination. In addition, failures shall also be evaluated for their importance relative to containment performance. Each system, or portion thereof, shall be partitioned into postulated piping failures that cause an initiating event, disable a system/train/loop without causing an initiating event, or cause an initiating event and disable a system/train/loop. 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 (f) below.

(a) Initiating Event 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: (1) HIGH, (2) MEDIUM, (3) LOW, or (4) 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. Applicable design basis events previously analyzed in the Owner's updated Final Safety Analysis Report (FSAR) 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 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. Differences in the consequence rank between the use of Tables I-1 and I-5 shall be reviewed, justified, and documented or the higher consequence rank will be assigned. The quantitative

[4] Further details on evaluating and ranking the consequence impact groups and configurations are discussed in USNRC Safety Evaluation Report dated October 28, 1999 and EPRI Report TR-112657, Rev B-A.

index for the initiating event impact group is the ratio of the core damage frequency (CDF) 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/train/loop essential for preventing 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* - This 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 system.

(3) *Exposure time* - This determines the amount of 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 Specifications.

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. Quantitative indices may be used to assign consequence categories in accordance with Table I-5 in lieu of Table I-2 provided 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 Tables I-2 and I-5 shall be reviewed, justified, and documented or the higher consequence rank will be assigned. The quantitative index for the system impact group is the product of the change in conditional core damage frequency (CCDF) and the exposure time. Additionally, for defense-in-depth purposes, postulated failures leading to "zero defense" (i.e., no backup trains) shall be assigned a HIGH consequence.

(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 consequence category is a function of two factors:

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

(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 provided the quantitative basis of Table I-3 (e.g., one full-train unavailability approximately 10^{-2}) is consistent with the failure scenario being evaluated. Differences in the consequence rank between the use of Tables I-3 and I-5 shall be reviewed, justified, and documented or the higher consequence rank will be assigned.

(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 that do not result in a loss-of-coolant accident (LOCA) that bypasses containment, the quantitative indices of Table I-5 for conditional large early release probability (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 that bypasses containment.

(e) Shutdown Operation Evaluation

The previously established consequence rank shall be reviewed and adjusted to reflect the pressure boundary failure's impact on plant operation during shutdown.[5]

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. If a shutdown PRA is not available, the effect of pressure-boundary failures on core damage and containment performance shall be evaluated. The major characteristics to be considered are defined as follows:

- The system operations, safety functions, and success criteria change in different stages of other modes of operation.
- The exposure time for the majority of the piping associated with shutdown operation is typically less than 10% 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 evaluated.
- 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.[6]
- Recovery time may be longer, thus allowing for multiple operator actions.

[5] Further details are discussed in NRC Safety Evaluation Report dated October 28, 1999 and EPRI Report TR-112657, Rev B-A.

[6] A standard that provides an acceptable method for determining PRA scope, technical adequacy, and peer review requirements is ASME RA-S-2002, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications* with the RA-Sa-2003 and RA-Sb-2005 Addenda. This standard sets forth requirements for PRAs used to support risk-informed decisions for commercial nuclear power plants, peer review and PRA review processes and requirements, and prescribes a method for applying these requirements for various categories of applications.

(f) External Events Evaluation

The previously established consequence rank shall be reviewed and adjusted to reflect the pressure boundary failure's impact on the mitigation of external events.[7]

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

- External events that can cause a pressure-boundary failure (e.g., seismic events); and
- External events that do not affect 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 RISC

Piping segments may be grouped together within a system if the analysis and assessment performed in Section I-3.1 determine the effect of the postulated failures to be the same. The RISC definitions shall be:

- 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 fall into the HIGH consequence category in any table by the analysis and assessment in Section I-3.1 shall be considered HSS.

(b) Piping segments determined to fall into the MEDIUM, LOW, or NONE (no change to base case) consequence category in any table by the consequence evaluation in Section I-3.1 shall be determined to be HSS or LSS by considering the information in (1) through (10), below. Under the same conditions of Section I-3.1.1(a), a large pressure-boundary leak does not need to be assumed. Also, credit may be taken for plant features and operator actions to the extent these would not be affected by failure of the segment under consideration. If plant features and operator actions are credited, they shall be consistent with those credited in Section I-3.1. The following conditions shall be evaluated and answered TRUE or FALSE:

(1) Failure of the pressure-retaining function of the segment will not directly or indirectly (e.g., through spatial effects) fail a basic safety function.

(2) Failure of the pressure-retaining function of the segment will not prevent the plant from reaching or maintaining safe shutdown conditions; the pressure-retaining 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

[7] Further details are discussed in NRC Safety Evaluation Report dated October 28, 1999 and EPRI Report TR-112657, Rev B-A.

pressure-boundary failure results in the need for actions outside of plant procedures or available backup plant mitigative features.

(3) The pressure-retaining function of the segment is not called out or relied upon in the plant emergency/abnormal operating procedures or similar guidance as the sole means for successfully performing operator actions required to mitigate an accident or transient.

(4) The pressure-retaining function of the segment 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 of the segment will not result in an unintentional release of radioactive material that would result in implementing offsite radiological protective actions.

The RISC process shall demonstrate that the defense-in-depth philosophy is maintained. Defense-in-depth is maintained if:

(6) Reasonable balance is preserved among preventing core damage, preventing containment failure or bypass, and mitigating an offsite release.

(7) There is no over-reliance on programmatic activities and operator actions to compensate for weaknesses in 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 Section I-3.2.2(b), then the RISC process shall verify that there are sufficient margins to account for uncertainty in the engineering analysis and in the supporting data. 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 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 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.

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

[1] Refer to Section 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

[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

[1] An example of "active protection" is a valve that needs to close on demand.
[2] An example of "passive protection" is a valve that needs to remain closed.

TABLE I-5

QUANTITATIVE INDICES FOR CONSEQUENCE CATEGORIES

CCDP (no units)	CLERP (no units)	Consequence Category
$>10^{-4}$	$>10^{-5}$	High
$10^{-6} <$ 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

TABLE I-6

DEFINITION OF CONSEQUENCE IMPACT GROUPS AND CONFIGURATIONS

CONSEQUENCES		
Impact Group	**Configuration**	**Description**
Initiating Event	Operating	A PBF* occurs in an operating (pressurized) system resulting in an initiating event
Loss of Mitigating Ability	Standby	A PBF occurs in a standby system and does not result in an initiating event, but degrades the mitigating capabilities of a system or train. After failure is discovered, the plant enters the applicable Allowed Outage Time defined in the Technical Specification
	Demand	A PBF occurs when system/train operation is required by an independent demand
Combination	Operating	A PBF causes an initiating event with an additional loss of mitigating ability (in addition to the expected mitigating degradation due to the initiator)
Containment	Any	A PBF, in addition to the above impacts, also affects containment performance

* PBF – pressure-boundary failure

ENCLOSURE 2

CNRO-2007-00015

LICENSEE-IDENTIFIED COMMITMENT

LICENSEE-IDENTIFIED COMMITMENTS

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Entergy shall review changes to the plant, operational practices, applicable plant and industry operational experience, and, as appropriate, update the probabilistic risk assessment (PRA) and categorization and treatment processes. Entergy shall perform this review in a timely manner but no longer than once every two refueling outages.		✓	Upon implementation of ANO2-R&R-004, Rev. 1