Entergy

Entergy Nuclear South
Entergy Operations, Inc.
17265 River Road
Killona, LA 70057-3093
Tel 504-739-6715
Fax 504-739-6698
rmurill@entergy.com

Robert J. Murillo
Licensing Manager
Waterford 3

W3F1-2008-0013

February 14, 2008

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

SUBJECT: Revision to Request for Alternative W3-ISI-005
Request to Use ASME Code Case N-716
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES
1. Entergy letter dated October 18, 2007, *Request for Alternative W3-ISI-005 Request to Use ASME Code Case N-716* (W3F1-2007-0046)
2. Entergy letter dated November 30, 2007, *Supplement to Request for Alternative W3-ISI-005 Request to Use ASME Code Case N-716* (W3F1-2007-0061)
3. Entergy letter dated December 12, 2007, *Supplement 2 to Request for Alternative W3-ISI-005 Request to Use ASME Code Case N-716* (W3F1-2007-0062)

Dear Sir or Madam:

Per Reference 1, Entergy requested NRC review and approval to implement a risk-informed Inservice Inspection (ISI) program based on ASME Code Case N-716 at Waterford 3 (W3). On November 30, 2007, W3 submitted a letter (Reference 2) to the NRC documenting how each Grand Gulf Nuclear Station (GGNS) Request for Additional Information (RAI) was addressed in the W3 ISI Code Case N-716 Alternative request. On December 12, 2007, W3 submitted a letter (Reference 3) to the NRC requesting a change in the Request for Alternative W3-ISI-005 approval date from March 14, 2008 to April 18, 2008 and establishing a regulatory commitment to update the internal flooding portion of the W3 PRA to conform to

A047
NRR

ASME Standard RA-Sb-2005 and to provide written confirmation to the NRC by February 18, 2008 of completion.

The purpose of this letter is to provide the NRC a revised Request for Alternative W3-ISI-005 that reflects the updated internal flooding portion of the W3 PRA which conforms to ASME Standard RA-Sb-2005 and to incorporate W3 specific responses to the GGNS RAI questions. The revised Request for Alternative W3-ISI-005 will replace Reference 1 in its entirety and is contained in Enclosure 1. Revision bars in the right hand margin identify the changes made to Reference 1. The Waterford 3 specific responses to GGNS RAI questions that were not incorporated into the revised Request for Alternative W3-ISI-005 template are contained in Enclosure 2. The Cross Reference list of GGNS RAI questions to location of W3 specific responses within the revised Request for Alternative W3-ISI-005 is contained in Enclosure 3.

This letter retains the original commitments made in Reference 1 and are identified in Enclosure 4. Should you have any questions regarding this submittal, please contact Ron Williams at (504) 739-6255.

Sincerely,

Oscar P. Pipkins for
R.J. Murillo

RJM/RLW

Enclosures:
1. Revised Request for Alternative W3-ISI-005
2. Waterford 3 Specific Responses to GGNS RAI Questions Not Incorporated into the Revised Request for Alternative W3-ISI-005
3. GGNS RAI Question Cross Reference List to location of W3 Specific Responses within Enclosures 1 and 2
4. Licensee-Identified Commitments

cc: Mr. Elmo E. Collins, Jr.
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

NRC Senior Resident Inspector
Waterford Steam Electric Station Unit 3
P.O. Box 822
Killona, LA 70066-0751

U.S. Nuclear Regulatory Commission
Attn: Mr. Kaly Kalyanam
MS O-7D1
Washington, DC 20555-0001

Louisiana Department of Environmental Quality
Office of Environmental Compliance
Surveillance Division
P. O. Box 4312
Baton Rouge, LA 70821-4312

American Nuclear Insurers
Attn: Library
95 Glastonbury Blvd.
Suite 300
Glastonbury, CT 06033-4443

Wise, Carter, Child & Caraway
Attn: J. Smith
P.O. Box 651
Jackson, MS 39205

Winston & Strawn
ATTN: N.S. Reynolds
1700 K Street, NW
Washington, DC 20006-3817

Morgan, Lewis & Bockius LLP
ATTN: T.C. Poindexter
1111 Pennsylvania Avenue, NW
Washington, DC 20004

ENCLOSURE 1

W3F1-2008-0013

REVISED REQUEST FOR ALTERNATIVE
W3-ISI-005

ENTERGY OPERATIONS, INC.
WATERFORD STEAM ELECTRIC STATION, UNIT 3

REVISED REQUEST FOR ALTERNATIVE
W3-ISI-005

Application of ASME Code Case N-716

RISK-INFORMED / Safety-based INSERVICE INSPECTION Program Plan

Table of Contents

REVISED REQUEST FOR ALTERNATIVE
W3-ISI-005

1. INTRODUCTION

Waterford Steam Electric Station Unit 3 (W3) is currently in the second inservice inspection (ISI) interval as defined by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Section XI Code for Inspection Program B. W3 plans to complete the current (second) ISI interval by implementing a risk-informed / safety-based inservice inspection (RIS_B) program during the third inspection period of the interval. Entergy will also implement 100% of the RIS_B program in the third interval.

The ASME Section XI code of record for the second ISI interval at W3 is the 1992 Edition for Examination Category B-F, B-J, C-F-1, and C-F-2 Class 1 and 2 piping components. The ASME Section XI code of record for the third ISI interval at W3 is the 2001 Edition through 2003 Addenda for these welds.

The objective of this submittal is to request the use of the RIS_B process for the inservice inspection of Class 1 and 2 piping. The RIS_B process used in this submittal is based upon ASME Code Case N-716, *Alternative Piping Classification and Examination Requirements, Section XI Division 1*, which is founded in large part on the RI-ISI process as described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A, *Revised Risk-Informed Inservice Inspection Evaluation Procedure.*

1.1 Relation to NRC Regulatory Guides 1.174 and 1.178

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis,*" and Regulatory Guide 1.178, *An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping.* Additional information is provided in Section 3.4.2 relative to defense-in-depth.

1.2 Probabilistic Safety Assessment (PSA) Quality

The W3 Level 1 PSA was initially developed in response to the NRC Generic Letter 88-20 on Individual Plant Examinations. The Individual Plant Examination (IPE) was submitted to the NRC in August 1992. The W3 IPE consisted of the Level 1 PSA and back-end analysis (Level 2) consistent with the requirements of NRC Generic Letter (GL) 88-20, *Individual Plant Examination for Severe Accident Vulnerabilities – 10 CFR 50.54(f)*. The NRC responded with a Safety Evaluation Report (SER) in a letter dated March 4, 1997 and approved the W3 IPE results. The letter concluded that the W3 IPE met the intent of GL 88-20; that is, the W3 IPE process was capable of identifying the most likely severe accidents and severe accident vulnerabilities for W3.

The IPE was subjected to a number of reviews. In addition to normal engineering and cross-discipline reviews, the IPE received a peer review by PRA experts from a PRA consultant, and comments were addressed prior to its August 1992 submittal to NRC. The NRC review of the IPE, transmitted to W3 in March 1997, identified several weaknesses. All but one of the weaknesses in the Level 1 analysis (with one exception noted below) were addressed by the June 2003 model update. The exception had to do with a lack of simulator exercises for in-control room operator response times and walkdowns for ex-control room times. Current PRA quality standards identify either walkthroughs, talkthroughs (detailed procedure reviews with operators), or simulator observations as acceptable bases for operator response times (ASME PRA Standard, Supporting Requirement HR-G5, Categories II & III). The W3 PRA used operator talkthroughs for all of the post-initiator operator actions.

Several PRA model updates have been completed on the W3 PSA since the IPE was submitted. These were done in order to maintain the PSA model reasonably consistent with the as-built, as-operated plant. The scope of the updates was based on review of results, plant input to the model, updated plant failure and initiating event data as well as model enhancements.

An industry peer review of the W3 PSA was conducted in January 2000 on the Revision 2 PSA and the report was subsequently published in April 2000. The peer review concluded that there were several areas where the W3 model was very weak and needed improvement. The W3 PSA model update, completed in June 2003, addressed most of the significant Facts and Observations (F&O's) from this certification.

In June 2003, Revision 3 of the W3 Level 1 PSA was issued. The scope of this revision included the incorporation of new methodologies in addition to revisions to various elements of the model. The modeling changes were made as a result of changes to the plant, revised plant procedures, revisions to system success criteria, more detailed system models and the addition of systems to the model. New methodologies for various tasks necessary for the PSA update were also utilized. These include the following:

- Utilized a more accepted methodology (alpha factor method) for the common cause analysis. In addition, the common cause analysis was much more extensive (applied to more components) than the analysis in the previous revision.

- Updated the human reliability analysis (HRA) with a more comprehensive and thorough methodology. This analysis was also much more extensive and took into account dependencies between multiple human error events when they occurred within a single cut set.

- Incorporated a new method for accounting for recovery of losses of offsite power. This method uses a convolution approach to account for time dependencies in individual cut sets. A plant-specific offsite power recovery curve was also developed utilizing only those loss-of-offsite-power events that are applicable to W3.

- Utilized more detailed fault trees to determine the frequency for certain support system initiating events.

- Utilized updated data to determine basic event probabilities and initiating event frequencies. There was more extensive use of plant-specific data (primarily major components of risk significant maintenance rule systems).

As part of the Revision 3 update to the PSA, most of the important observations resulting from the peer review were also addressed. Following Revision 3 to the Level 1 update, a decision was made to develop a Large Early Release Frequency (LERF) model rather than update the W3 IPE Level 2 model. The LERF model was developed using the methods described in NUREG/CR-6595, Rev. 1, *An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events*, and is directly linked to the internal events model. Because of the different method, most of the Level 2 peer review observations are not applicable and have not been addressed. The W3 LERF model was completed and issued in June 2004.

Recently, in preparation for W3's transition to NFPA 805, a gap assessment of the W3 PSA model has been completed. Gaps to the ASME PSA standard and Reg Guide 1.200 Revision have been identified. The gaps impacting the fire PRA are being closed in the near term in order to meet the NFPA 805 transition schedule. HRA interviews are needed with Plant Operations personnel and have not been able to be scheduled because of unavailability of operators. It is expected that all of the significant model gaps to the ASME Standard impacting the Fire PRA will be closed with the Revision 4 Model Update that is slated to be completed in mid 2008. We believe W3 will also close remaining significant model gaps with this update. Irrespective of the above, a review of the open A&B F&Os for impact on the RIS_B application was conducted and identified that they would not have a significant impact on the RIS_B results.

Finally in support of the RIS_B application, the internal flooding study portion of the W3 PSA has been updated to the ASME PRA Standard (RA-Sb-2005). The updated W3 PSA for internal flooding meets capability category II. Results from this update confirm that there is no additional piping that needs to be added to the high safety significant scope (see paragraph 2(a)(5) of N716) of the W3 RIS_ B program. Additionally, inputs used to support the change in risk assessment (e.g. CCDPs/CLERPs) for LSS piping remain unchanged.

Revised Request for Alternative W3-ISI-005 is based on the W3 PSA Internal Events Revision 3 model and the W3 LERF model. The base case Internal Events Core Damage Frequency (CDF) is 1.69E-5/year and the base case LERF is 2.47E-7/year.

Based on the above, Entergy believes that the current PSA model, used in the RIS_B evaluation, has an acceptable level of quality to support this application.

2. PROPOSED ALTERNATIVE TO CURRENT ISI PROGRAMS

2.1 ASME Section XI

ASME Section XI Examination Categories B-F, B-J, C-F-1, and C-F-2 currently contain requirements for the nondestructive examination (NDE) of Class 1 and 2 piping components, except as amended by application of ASME Code Case N-663 (Request for Alternative CEP-ISI-007) that was approved for use at W3 by the NRC on August 26, 2003.

The alternative RIS_B Program for piping is described in Code Case N-716. The RIS_B Program will be substituted for the current program for Class 1 and 2 piping (Examination Categories B-F, B-J, C-F-1 and C-F-2) in accordance with 10 CFR 50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety. Other non-related portions of the ASME Section XI Code will be unaffected.

2.2 Augmented Programs

The impact of the RIS_B application on the various plant augmented inspection programs listed below were considered. This section documents only those plant augmented inspection programs that address common piping with the RIS_B application scope (e.g., Class 1 and 2 piping).

- The original plant augmented inspection program for high-energy line breaks outside containment, implemented in accordance with W3 Final Safety Analysis Report (FSAR) Section 6.6.8, "Augmented Inservice Inspection to Protect against Postulated Piping Failures," is being revised in accordance with the risk-informed break exclusion region methodology (RI-BER) described in EPRI Report 1006937, *Extension of EPRI Risk Informed ISI Methodology to Break Exclusion Region Programs*. EPRI Report 1006937 was approved by the NRC in 2002. The results of the RI-BER application demonstrated that the volumetric examination requirement for this scope of piping could be reduced from 100% to approximately 13%. As a result, 13% of the BER population will be examined during the course of each ten-year interval which exceeds the 10% requirement imposed by Code Case N-716.

- A plant augmented inspection program has been implemented at W3 in response to NRC Bulletin 88-08, *Thermal Stresses in Piping Connected to Reactor Coolant Systems*. The thermal fatigue concern addressed by this bulletin was explicitly considered in the application of the RIS_B process and is subsumed by the RIS_B Program.

- The plant augmented inspection program for flow accelerated corrosion (FAC) per GL 89-08, *Erosion/Corrosion-Induced Pipe Wall Thinning*, is relied upon to manage this damage mechanism but is not otherwise affected or changed by the RIS_B Program.

- A plant augmented inspection program is being implemented at W3 in response to MRP-139, *Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines*. The requirements of MRP-139 will be used for the inspection and management of PWSCC susceptible welds and will supplement the RIS_B Program selection process. The RIS_B Program will not be used to eliminate any MRP-139 requirements.

- W3 is in the process of evaluating MRP-146, *Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines*, and these results will be incorporated into the RIS_B Program, if warranted.

3. RISK-INFORMED / SAFETY-BASED ISI PROCESS

The process used to develop the RIS_B Program conformed to the methodology described in Code Case N-716 and consisted of the following steps:

- Safety Significance Determination
- Failure Potential Assessment
- Element and NDE Selection
- Risk Impact Assessment
- Implementation Program
- Feedback Loop

3.1 Safety Significance Determination

The systems assessed in the RIS_B Program are provided in Table 3.1. The piping and instrumentation diagrams and additional plant information including the existing plant ISI Program were used to define the piping system boundaries.

Per Code Case N-716 requirements, piping welds are assigned safety-significance categories, which are used to determine the treatment requirements. High safety-significant (HSS) welds are determined in accordance with the requirements below. Low safety-significant (LSS) welds include all other Class 2, 3, or Non-Class welds.

(1) Class 1 portions of the reactor coolant pressure boundary (RCPB), except as provided in 10 CFR 50.55a(c)(2)(i) and (c)(2)(ii);

(2) Applicable portions of the shutdown cooling pressure boundary function. That is, Class 1 and 2 welds of systems or portions of systems needed to utilize the normal shutdown cooling flow path either:

(a) As part of the RCPB from the reactor pressure vessel (RPV) to the second isolation valve (i.e., farthest from the RPV) capable of remote closure or to the containment penetration, whichever encompasses the larger number of welds; or

(b) Other systems or portions of systems from the RPV to the second isolation valve (i.e., farthest from the RPV) capable of remote closure or to the containment penetration, whichever encompasses the larger number of welds;

(3) That portion of the Class 2 feedwater system [> 4 inch nominal pipe size (NPS)] of pressurized water reactors (PWRs) from the steam generator to the outer containment isolation valve;

(4) Piping within the break exclusion region (> NPS 4) for high-energy piping systems as defined by the Owner. This may include Class 3 or Non-Class piping; and

(5) Any piping segment whose contribution to CDF is greater than 1E-06 (and per NRC feedback on the Grand Gulf and DC Cool RIS_B applications 1E-07 for LERF) based upon a plant-specific PSA of pressure boundary failures (e.g., pipewhip, jet impingement, spray, inventory losses). This may include Class 3 or Non-Class piping.

3.2 Failure Potential Assessment

Failure potential estimates were generated utilizing industry failure history, plant-specific failure history, and other relevant information. These failure estimates were determined using the guidance provided in EPRI TR-112657 (i.e., the EPRI RI-ISI methodology), with the exception of the deviation discussed below.

Table 3.2 summarizes the failure potential assessment by system for each degradation mechanism that was identified as potentially operative.

A deviation to the EPRI RI-ISI methodology has been implemented in the failure potential assessment for W3. Table 3-16 of EPRI TR-112657 contains criteria for assessing the potential for thermal stratification, cycling, and striping (TASCS). Key attributes for horizontal or slightly sloped piping greater than NPS 1 include:

1. The potential exists for low flow in a pipe section connected to a component allowing mixing of hot and cold fluids; or

2. The potential exists for leakage flow past a valve, including in-leakage, out-leakage and cross-leakage allowing mixing of hot and cold fluids; or

3. The potential exists for convective heating in dead-ended pipe sections connected to a source of hot fluid; or

4. The potential exists for two phase (steam/water) flow; or

5. The potential exists for turbulent penetration into a relatively colder branch pipe connected to header piping containing hot fluid with turbulent flow;

AND

➢ $\Delta T > 50°F$,

AND

➢ Richardson Number > 4 (this value predicts the potential buoyancy of a stratified flow)

These criteria, based on meeting a high cycle fatigue endurance limit with the actual ΔT assumed equal to the greatest potential ΔT for the transient, will identify locations where stratification is likely to occur, but allows for no assessment of severity. As such, many locations will be identified as subject to TASCS where no significant potential for thermal fatigue exists. The critical attribute missing from the existing methodology that would allow consideration of fatigue severity is a criterion that addresses the potential for fluid cycling. The impact of this additional consideration on the existing TASCS susceptibility criteria is presented below.

➢ **Turbulent Penetration TASCS**

Turbulent penetration typically occurs in lines connected to piping containing hot flowing fluid. In the case of downward sloping lines that then turn horizontal, significant top-to-bottom cyclic ΔTs can develop in the horizontal sections if the horizontal section is less than about 25 pipe diameters from the reactor coolant piping. Therefore, TASCS is considered for this configuration.

For upward sloping branch lines connected to the hot fluid source that turn horizontal or in horizontal branch lines, natural convective effects combined with effects of turbulence penetration will keep the line filled with hot water. If there is no potential for in-leakage towards the hot fluid source from the outboard end of the line, this will result in a well-mixed fluid condition where significant top-to-bottom ΔTs will not occur. Therefore, TASCS is not considered for these configurations. Even in fairly long lines, where some heat loss from the outside of the piping will tend to occur and some fluid stratification may be present, there is no significant potential for cycling as has been observed for the in-leakage case. The effect of TASCS will not be significant under these conditions and can be neglected.

➢ **Low flow TASCS**

In some situations, the transient startup of a system (e.g., shutdown cooling suction piping) creates the potential for fluid stratification as flow is established. In cases where no cold fluid source exists, the hot flowing fluid will fairly rapidly displace the cold fluid in stagnant lines, while fluid mixing will occur in the piping further removed from the hot source and stratified conditions will exist only briefly as the line fills with hot fluid. As such, since the situation is transient in nature, it can be assumed that the criteria for thermal transients (TT) will govern.

➢ **Valve leakage TASCS**

Sometimes a very small leakage flow of hot water can occur outward past a valve into a line that is relatively colder, creating a significant temperature difference. However, since this is generally a "steady-state" phenomenon with no potential for cyclic temperature changes, the effect of TASCS is not significant and can be neglected.

- **Convection Heating TASCS**

  Similarly, there sometimes exists the potential for heat transfer across a valve to an isolated section beyond the valve, resulting in fluid stratification due to natural convection. However, since there is no potential for cyclic temperature changes in this case, the effect of TASCS is not significant and can be neglected.

In summary, these additional considerations for determining the potential for thermal fatigue as a result of the effects of TASCS provide an allowance for considering cycle severity. The above criteria have previously been submitted by EPRI to the NRC for generic approval [letters dated February 28, 2001 and March 28, 2001, from P.J. O'Regan (EPRI) to Dr. B. Sheron (USNRC), *Extension of Risk-Informed Inservice Inspection Methodology*]. The methodology used in the W3 RIS_B application for assessing TASCS potential conforms to these updated criteria. Final materials reliability program (MRP) guidance on the subject of TASCS will be incorporated into the W3 RIS_B application, if warranted. It should be noted that the NRC has granted approval for RI-ISI relief requests incorporating these TASCS criteria at several facilities, including Comanche Peak (NRC letter dated September 28, 2001) and South Texas Project (NRC letter dated March 5, 2002).

3.3 Element and NDE Selection

Code Case N-716 and lessons learned from the Grand Gulf and DC Cook RIS_B applications provide criteria for identifying the number and location of required examinations. Ten percent of the HSS welds shall be selected for examination as follows:

(1) Examinations shall be prorated equally among systems to the extent practical, and each system shall individually meet the following requirements:

  (a) A minimum of 25% of the population identified as susceptible to each degradation mechanism and degradation mechanism combination shall be selected.

  (b) If the examinations selected above exceed 10% of the total number of HSS welds, the examinations may be reduced by prorating among each degradation mechanism and degradation mechanism combination, to the extent practical, such that at least 10% of the HSS population is inspected.

  (c) If the examinations selected above are not at least 10% of the HSS weld population, additional welds shall be selected so that the total number selected for examination is at least 10%.

(2) At least 10% of the RCPB welds shall be selected.

(3) For the RCPB, at least two-thirds of the examinations shall be located between the inside first isolation valve (IFIV) (i.e., isolation valve closest to the RPV) and the RPV.

(4) A minimum of 10% of the welds in that portion of the RCPB that lies outside containment (OC) (e.g., portions of the main feedwater system in BWRs) shall be selected.

(5) A minimum of 10% of the welds within the break exclusion region (BER) shall be selected.

Currently, there are one hundred eighty-five BER program welds at Waterford 3. One hundred twenty-three of these welds are on high energy lines and sixty-two are on moderate energy lines. The RI-BER program examines a total of 24 welds with fifteen welds on high energy piping and nine welds on moderate energy piping. This represents an inspection population that is 12% of the total High Energy BER population. This program was implemented via the W3 10 CFR 50.59 program. Per the requirements of N-716, a minimum of 10% of the High Energy BER population is to be inspected. For W3, this results in a total of 13 inspections. However, 14 welds were inspected to meet the 10% HSS requirement.

In contrast to a number of RI-ISI Program applications where the percentage of Class 1 piping locations selected for examination has fallen substantially below 10%, Code Case N-716 mandates that 10% be chosen. A brief summary is provided below, and the results of the selections are presented in Table 3.3. Section 4 of EPRI TR-112657 was used as guidance in determining the examination requirements for these locations.

Unit	Class 1 Welds(1)		Class 2 Welds(2)		NNS Welds(3)		All Piping Welds(4)	
	Total	Selected	Total	Selected	Total	Selected	Total	Selected
3	729	74	1674	25	4	0	2407	99

Notes

(1) Includes all Category B-F and B-J locations. All 729 Class 1 piping weld locations are HSS.
(2) Includes all Category C-F-1 and C-F-2 locations. Of the 1674 Class 2 piping weld locations, 221 are HSS and the remaining 1453 are LSS.
(3) All four of these non-nuclear safety (NNS) piping weld locations are HSS.
(4) Regardless of safety significance, Class 1 and 2 in-scope piping components will continue to be pressure tested as required by the ASME Section XI Program. VT-2 visual examinations are scheduled in accordance with the station's pressure test program that remains unaffected by the RIS_B Program.

Prior to developing the RIS_B Program, W3 had planned to inspect locations scheduled for examination in the traditional ASME Section XI inspection program. Examination activities during refueling outages are planned far in advance. In general, only designated plant areas and components are accessible for examination during a given refueling outage due to other ongoing plant maintenance and modification activities. Hence, any location previously scheduled for examination in the third period via the traditional program will remain scheduled for examination in the third period, if the location has also been selected for RIS_B Program purposes. To complete the sample size, additional locations will be selected, if necessary, to achieve equal representation of the degradation mechanisms. Other factors such as accessibility and scaffolding requirements will also be factored into the selection process.

3.3.1 Additional Examinations

If the flaw is original construction or otherwise acceptable, Code rules do not require any additional inspections. Any unacceptable flaw will be evaluated per the requirements of ASME Code Section XI, IWB-3500 and/or IWB-3600. As part of performing evaluation to IWB-3600, the degradation mechanism that is responsible for the flaw will be determined and accounted for in the evaluation. The process for ordinary flaws is to perform the evaluation using ASME Section XI. If the flaw meets the criteria, then it is noted and appropriate successive examinations scheduled. If the nature and type of the flaw is service-induced, then similar systems or trains will be examined. If the flaw is found unacceptable for continued operation, it will be repaired in accordance with IWA-4000 and/or applicable ASME Section XI Code Cases. The need for extensive root cause analysis beyond that required for IWB-3600 evaluation will be dependent on practical considerations (i.e., the practicality of performing additional NDE or removing the flaw for further evaluation during the outage). The NRC is involved in the process at several points. For preemptive weld overlays, a relief request in accordance with 10 CFR 50.55a(a)(3) is usually required for design and installation. Should a flaw be discovered during an examination, a notification in accordance with 10 CFR 50.72 or 10 CFR 50.73 may be required. IWB-3600 requires the evaluation to be submitted to the NRC. Finally, the documentation requirements will be documented in the corrective action program and the Owner submittals required by Section XI.

The evaluation will include whether other elements in the segment or additional segments are subject to the same root cause conditions. Additional examinations will be performed on those elements with the same root cause conditions or degradation mechanisms. The additional examinations will include HSS elements up to a number equivalent to the number of elements required to be inspected during the current outage. If unacceptable flaws or relevant conditions are again found similar to the initial problem, the remaining elements identified as susceptible will be examined during the current outage. No additional examinations need be performed if there are no additional elements identified as being susceptible to the same root cause conditions.

3.3.2 Program Relief Requests

An attempt has been made to select RIS_B locations for examination such that a minimum of >90% coverage (i.e., Code Case N-460 criteria) is attainable. However, some limitations will not be known until the examination is performed since some locations may be examined for the first time by the specified techniques.

In instances where locations at the time of the examination fail to meet the >90% coverage requirement, the process outlined in 10 CFR 50.55a will be followed.

Consistent with previously approved RI-ISI submittals (e.g. ANO Unit 2 SER), Entergy will calculate coverage and use additional examinations or techniques in the same manner it has for traditional Section XI examinations. Experience has shown this process to be weld-specific (e.g., joint configuration). As such, the effect on risk, if any, will not be known until that time. Relief requests will be submitted per the guidance of 10 CFR 50.55a(g)(5)(iv) within one (1) year after the end of the interval.

Request for Alternative CEP-ISI-007 pertaining to the application of Code Case N-663 will be withdrawn for use at W3 upon NRC approval of the RIS_B Program submittal.

3.4 Risk Impact Assessment

The RIS_B Program has been conducted in accordance with Regulatory Guide 1.174 and the requirements of Code Case N-716, and the risk from implementation of this program is expected to remain neutral or decrease when compared to that estimated from current requirements.

This evaluation categorized segments as high safety significant or low safety significant in accordance with Code Case N-716, and then determined what inspection changes are proposed for each system. The changes include changing the number and location of inspections and in many cases improving the effectiveness of the inspection to account for the findings of the RIS_B degradation mechanism assessment. For example, examinations of locations subject to thermal fatigue will be conducted on an expanded volume and will be focused to enhance the probability of detection (POD) during the inspection process.

3.4.1 Quantitative Analysis

Code Case N-716 has adopted the EPRI TR-112657 process for risk impact analyses whereby limits are imposed to ensure that the change in risk of implementing the RIS_B Program meets the requirements of Regulatory Guides 1.174 and 1.178. The EPRI criterion requires that the cumulative change in CDF and LERF be less than 1E-07 and 1E-08 per year per system, respectively.

For LSS welds the CCDP/CLERP, values of 1E-4 / 1E-5 were conservatively used. The rationale for using these values is that the change-in-risk evaluation process of N-716 is similar to that of the EPRI RI-ISI methodology. As such, the goal is to determine CCDPs/CLERPs threshold values. For example, the threshold values between High and Medium consequence categories is 1E-4 (CCDP) / 1E-5 (CLERP) and between Medium and Low consequence categories are 1E-6 (CCDP) / 1E-7 (CLERP) from the EPRI RI-ISI Risk Matrix. Using these threshold values streamlines the change-in-risk evaluation as well as stabilizes the update process. For example, if a CCDP changes from 1E-5 to 3E-5 due to an update, it will remain below the 1E-4 threshold value; the change-in-risk evaluation would not require updating.

The above values were from the W3 internal flooding study which was performed to RA-Sb-2005 capabiity category II requirements. CCDP for in-scope LSS Class 2 piping previously being inspected is less than 1E-4 with no containment bypass breaks. Therefore, the 0.1 conditional LERF is also reasonable. The values are consistent with and conservatively above any CCDP value obtained for W3 in-scope Class 2 piping, and the CLERP value is appropriately scaled.

With respect to assigning failure potential for LSS piping, the criteria are defined by Table 3 of the Code Case. That is, those locations identified as susceptible to FAC (or another mechanism and also susceptible to water hammer) are assigned a high failure potential. Those locations susceptible to thermal fatigue, erosion-cavitation, corrosion or stress corrosion cracking are assigned to a medium failure potential and those locations that are identified as not susceptible to degradation are assigned a low failure potential.

In order to streamline the risk impact assessment, a review was conducted to verify that the LSS piping was not susceptible to FAC or water hammer. This review was conducted similar to that done for a traditional RI-ISI application. Thus, the High failure potential category is not applicable to LSS piping. In lieu of conducting a formal degradation mechanism evaluation for all LSS piping (e.g. to determine if thermal fatigue is applicable), these locations were conservatively assigned to the Medium failure potential ("Assume Medium" in Table 3.4-1) for use in the change-in-risk assessment. Experience with previous industry RI-ISI applications shows this to be conservative.

W3 has conducted a risk impact analysis per the requirements of Section 5 of Code Case N-716 that is consistent with the "Simplified Risk Quantification Method" described in Section 3.7 of EPRI TR-112657. The analysis estimates the net change in risk due to the positive and negative influences of adding and removing locations from the inspection program.

The conditional core damage probability (CCDP) and conditional large early release probability (CLERP) values used to assess risk impact were estimated based on pipe break location. Based on these estimated values, a corresponding consequence rank was assigned per the requirements of EPRI TR-112657 and upper bound threshold values were used as provided below. Consistent with the EPRI risk-informed methodology, the upper bound for all break locations that fall within the high consequence rank range was based on the highest CCDP value obtained (e.g., Large LOCA CCDP bounds the Intermediate LOCA (3.5E-03) and small LOCA (4.8E-04) CCDPS for W3).

CCDP and CLERP Values Based on Break Location

Break Location Designation	Estimated		Consequence Rank	Upper Bound	
	CCDP	CLERP		CCDP	CLERP
LOCA	4.20E-03	4.20E-04	**HIGH**	4.20E-03	4.20E-04
RCPB pipe breaks that result in a loss of coolant accident – The highest CCDP for Large LOCA was used (0.1 margin used for CLERP)					
ILOCA[(1)]	8.40E-06	8.40E-07	**MEDIUM**	1.00E-04	1.00E-05
RCPB pipe breaks that result in an isolable LOCA – Calculated based on Large LOCA CCDP of 4.2E-3 and valve fail to close probability of 2E-3 (0.1 margin used for CLERP)					
PLOCA[(1)]	4.20E-06	4.20E-07	**MEDIUM**	1.00E-04	1.00E-05
RCPB pipe breaks that result in a potential LOCA – Calculated based on Large LOCA CCDP of 4.2E-3 and valve rupture probability of <1E-3 (0.1 margin used for CLERP)					
PLOCA – SD[(1)]	1.00E-04	1.00E-05	**HIGH**	4.20E-03	4.20E-04
RCPB pipe breaks that occur in shutdown cooling piping and result in a potential LOCA – Values obtained from RI-BER analysis (0.1 margin used for CLERP)					
BER – SI1	1.00E-04	1.00E-05	**HIGH**	4.20E-03	4.20E-04
Class 2 pipe breaks that occur in shutdown cooling piping inside containment – Values obtained from RI-BER analysis (0.1 margin used for CLERP)					
BER – SI2[(2)]	1.00E-04	1.00E-04	**HIGH**	4.20E-03	4.20E-04
Class 2 pipe breaks that occur in shutdown cooling piping outside containment – Values obtained from RI-BER analysis (no margin used for CLERP)					
BER – FW1	7.00E-04	7.00E-05	**HIGH**	4.20E-03	4.20E-04
Class 2 pipe breaks that occur in main feedwater piping inside containment – Values obtained from RI-BER analysis (0.1 margin used for CLERP)					
BER – FW2	7.00E-04	7.00E-05	**HIGH**	4.20E-03	4.20E-04
Class 2 pipe breaks that occur in main feedwater and emergency feedwater piping outside containment – Values obtained from RI-BER analysis (0.1 margin used for CLERP)					
BER – MS	7.00E-04	7.00E-05	**HIGH**	4.20E-03	4.20E-04
Class 2 pipe breaks that occur in main steam piping inside and outside containment – Values obtained from RI-BER analysis (0.1 margin used for CLERP)					
Class 2 LSS	1.00E-04	1.00E-05	**MEDIUM**	1.00E-04	1.00E-05
Class 2 pipe breaks that occur in the remaining system piping designated as low safety significant – Estimated based on upper bound for Medium Consequence					

Notes

1. The W3 PRA does not explicitly model potential and isolable LOCA events, because such events are subsumed by the LOCA initiators in the PRA. That is, the frequency of a LOCA in this limited piping downstream of the first RCPB isolation valve times the probability that the valve fails is a small contributor to the total LOCA frequency. The N-716 methodology must evaluate these segments individually; thus, it is necessary to estimate their contribution. This is estimated by taking the LOCA CCDP and multiplying this by the valve failure probability.
2. The safety injection shutdown cooling function (injection to cold legs and suction from hot legs) has pipe segments outside containment and is in the N-716 scope because it is in the W3 BER program. This piping was assigned a relatively high CCDP=CLERP=1E-4 in the BER evaluation. This is dominated by considering pipe failure during shutdown; failure of two valves in series during power operation is less likely (see BER-SI2 in Section 3.4.1 of the submittal). The risk impact assessment (CDF and LERF) for applicable piping meets risk acceptance criteria for the N716 application with significant margin.

The likelihood of pressure boundary failure (PBF) is determined by the presence of different degradation mechanisms and the rank is based on the relative failure probability. The basic likelihood of PBF for a piping location with no degradation mechanism present is given as x_o and is expected to have a value less than 1E-08. Piping locations identified as medium failure potential have a likelihood of $20x_o$. These PBF likelihoods are consistent with References 9 and 14 of EPRI TR-112657. In addition, the analysis was performed both with and without taking credit for enhanced inspection effectiveness due to an increased POD from application of the RIS_B approach.

Table 3.4-1 presents a summary of the RIS_B Program versus 1992 ASME Section XI Code Edition program requirements on a "per system" basis. The presence of FAC was adjusted for in the quantitative analysis by excluding its impact on the failure potential rank. The exclusion of the impact of FAC on the failure potential rank and therefore in the determination of the change in risk is performed, because FAC is a damage mechanism managed by a separate, independent plant augmented inspection program. The RIS_B Program credits and relies upon this plant augmented inspection program to manage this damage mechanism. The plant FAC Program will continue to determine where and when examinations shall be performed. Hence, since the number of FAC examination locations remains the same "before" and "after" and no delta exist, there is no need to include the impact of FAC in the performance of the risk impact analysis.

As indicated in the following table, this evaluation has demonstrated that unacceptable risk impacts will not occur from implementation of the RIS_B Program, and satisfies the acceptance criteria of Regulatory Guide 1.174 and Code Case N-716.

W3 Risk Impact Results

System(1)	ΔR_{CDF} Results		ΔR_{LERF} Results	
	w/ POD	w/o POD	w/ POD	w/o POD
RC	-9.66E-10	6.09E-09	-9.66E-11	6.09E-10
CH	-8.36E-09	-4.66E-09	-8.36E-10	-4.66E-10
SI	-2.14E-08	-5.44E-09	-2.14E-09	-5.44E-10
EF	-1.76E-09	-4.20E-10	-1.76E-10	-4.20E-11
FW	-2.92E-09	-7.35E-10	-2.92E-10	-7.35E-11
MS	1.12E-10	1.12E-10	1.12E-11	1.12E-11
CS	1.50E-10	1.50E-10	1.50E-11	1.50E-11
TOTAL	-3.51E-08	-4.90E-09	-3.51E-09	-4.90E-10

Note

(1) Systems are described in Table 3.1.

3.4.2 Defense-in-Depth

The intent of the inspections mandated by ASME Section XI for piping welds is to identify conditions such as flaws or indications that may be precursors to leaks or ruptures in a system's pressure boundary. Currently, the process for picking inspection locations is based upon structural discontinuity and stress analysis results. As depicted in ASME White Paper 92-01-01 Rev. 1, *Evaluation of Inservice Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds*, this method has been ineffective in identifying leaks or failures. EPRI TR-112657 and Code Case N-716 provide a more robust selection process founded on actual service experience with nuclear plant piping failure data.

This process has two key independent ingredients; that is, a determination of each location's susceptibility to degradation and secondly, an independent assessment of the consequence of the piping failure. These two ingredients assure defense-in-depth is maintained. First, by evaluating a location's susceptibility to degradation, the likelihood of finding flaws or indications that may be precursors to leak or ruptures is increased. Secondly, a generic assessment of high-consequence sites has been determined by Code Case N-716 supplemented by plant-specific evaluations thereby requiring a minimum threshold of inspection for important piping whose failure would result in a LOCA or BER break. Finally, Code Case N-716 requires that any piping on a plant-specific basis that has a contribution to CDF of greater than 1E-06 (or 1E-07 for LERF) be included in the scope of the application. W3 did not identify any such piping.

All locations within the Class 1, 2, and 3 pressure boundaries will continue to be pressure tested in accordance with the Code, regardless of its safety significance.

4. IMPLEMENTATION AND MONITORING PROGRAM

Upon approval of the RIS_B Program, procedures that comply with the guidelines described in Code Case N-716 will be prepared to implement and monitor the program. The new program will be integrated into the second ISI interval. No changes to the Technical Specifications or Updated Final Safety Analysis Report are necessary for program implementation.

The applicable aspects of the ASME Code not affected by this change will be retained, such as inspection methods, acceptance guidelines, pressure testing, corrective measures, documentation requirements, and quality control requirements. Existing ASME Section XI program implementing procedures will be retained and modified to address the RIS_B process, as appropriate.

The monitoring and corrective action program will contain the following elements:

A. Identify

B. Characterize

C. (1) Evaluate, determine the cause and extent of the condition identified

(2) Evaluate, develop a corrective action plan or plans

D. Decide

E. Implement

F. Monitor

G. Trend

The RIS_B Program is a living program requiring feedback of new relevant information to ensure the appropriate identification of HSS piping locations. As a minimum, this review will be conducted on an ASME period basis. In addition, significant changes may require more frequent adjustment as directed by NRC Bulletin or Generic Letter requirements, or by industry and plant-specific feedback.

For preservice examinations, Entergy will follow the rules contained in Section 3.0 of N-716. Welds classified HSS require preservice inspection. The examination volumes, techniques, and procedures shall be in accordance with Table 1. Welds classified as LSS do not require preservice inspection.

5. PROPOSED ISI PROGRAM PLAN CHANGE

A comparison between the RIS_B Program and ASME Section XI 1992 Code Edition program requirements for in-scope piping is provided in Table 5.

W3 intends to start implementing the RIS_B Program during the plant's third period of the current (second) inspection interval. By the end of last refueling outage (RF-14), 65% of the piping weld examinations required by ASME Section XI have been completed thus far in the second ISI interval for Examination Categories B-F, B-J, C-F-1 and C-F-2. To ensure the performance of 100% of the required examinations during the current (second) ten-year ISI interval, 35% of the inspection locations selected for examination per the RIS_B process will be examined in the third period of the interval. The third ISI interval will implement 100% of the inspection locations selected for examination per the RIS_B Program. Examinations shall be performed such that the period percentage requirements of ASME Section XI are met.

To date, Entergy has examined the following number of Class 1 and 2 piping welds in the third period:

B-F = 0

B-J = 0

C-F-1 = 18

C-F-2 = 3

6. REFERENCES/DOCUMENTATION

USNRC Safety Evaluation on the use of ASME Code Case N-663, dated August 26, 2003 (letter CNRI-2003-00010)

EPRI Report 1006937, *Extension of EPRI Risk Informed ISI Methodology to Break Exclusion Region Programs*

EPRI TR-112657, *Revised Risk-Informed Inservice Inspection Evaluation Procedure*, Rev. B-A

ASME Code Case N-716, *Alternative Piping Classification and Examination Requirements, Section XI Division 1*

Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis*

Regulatory Guide 1.178, *An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping*

USNRC Safety Evaluation for Grand Gulf Nuclear Station Unit 1, Request for Alternative GG-ISI-002-Implement Risk-Informed ISI based on ASME Code Case N-716, dated September 21, 2007

USNRC Safety Evaluation for DC Cook Nuclear Plant, Units 1 and 2, Risk-Informed Safety-Based ISI program for Class 1 and 2 Piping Welds, dated September 28, 2007

Supporting Onsite Documentation

ENTP-19Q-301, *Degradation Mechanism Evaluation for Waterford*, Revision 0

ENTP-19Q-302, *RI-BER Evaluation for Waterford Unit 3*, Revision 0

ENTP-19Q-303, *N-716 Evaluation of Waterford 3*, Revision 0

PLG-0624, *Internal Flood Frequencies during Shutdown and Operation for Nuclear Power Plants*, N. O. Siu, et al., prepared for Public Service of New Hampshire, Pickard, Lowe and Garrick, Inc., dated May 1988

Table 3.1 N-716 Safety Significance Determination								
System Description	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	CDF > 1E-6 LERF > 1E-7	High	Low
RC – Reactor Coolant	282	✓					✓	
CH – Chemical and Volume Control	129	✓					✓	
SI – Safety Injection	277	✓	✓				✓	
	41	✓					✓	
	28		✓		✓		✓	
	12		✓				✓	
	34				✓		✓	
	1209							✓
EF – Emergency Feedwater	22			✓			✓	
FW – Main Feedwater	44			✓	✓		✓	
	6			✓			✓	
	2				✓		✓	
MS – Main Steam	77				✓		✓	
	43							✓
CS – Containment Spray	201							✓
SUMMARY RESULTS FOR ALL SYSTEMS	277	✓	✓				✓	
	452	✓					✓	
	28		✓		✓		✓	
	12		✓				✓	
	44			✓	✓		✓	
	28			✓			✓	
	113				✓		✓	
	1465							✓
TOTALS	**2407**						**954**	**1453**

Table 3.2 Failure Potential Assessment Summary											
System[1]	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive	
	TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC
RC	✓	✓				✓					
CH		✓									
SI[2]	✓	✓	✓			✓					
EF		✓									✓
FW	✓										✓
MS[2]											
CS[2]											

Notes

1. Systems are described in Table 3.1.
2. A degradation mechanism assessment was not performed on low safety significant piping segments. This includes the CS system in its entirety, as well as portions of the SI and MS systems.

Table 3.3 N-716 Element Selections									
System[1]	Weld Count		N-716 Selection Considerations					Selections
	HSS	LSS	DMs	RCPB	RCPB IF/IV	RCPB OC	BER	
RC	1		TASCS, TT, PWSCC	✓	✓			1
RC	19		TASCS, TT	✓	✓			6
RC	2		TT, PWSCC	✓	✓			1
RC	16		TASCS	✓	✓			4
RC	3		TT	✓	✓			0
RC	14		PWSCC	✓	✓			4
RC	213		None	✓	✓			13
RC	14		None	✓				0
CH	36		TT	✓	✓			11
CH	5		TT	✓				0
CH	76		None	✓	✓			2
CH	12		None	✓				0
SI	8		TT, IGSCC	✓				2
SI	14		TASCS	✓	✓			4
SI	52		TT	✓	✓			17
SI	106		TT	✓				8
SI	36		TT				✓	8
SI	4		TT					0
SI	2		PWSCC	✓	✓			1
SI	40		None	✓	✓			0
SI	96		None	✓				0
SI	26		None				✓	0
SI	8		None					0
SI		1209						
EF	2		TT, (FAC)					0
EF	18		TT					3
EF	2		None (FAC)					0
FW	6		TASCS, (FAC)				✓	2
FW	3		TASCS				✓	3
FW	2		TASCS					0
FW	8		None (FAC)				✓	0
FW	29		None				✓	1
FW	4		None					0
MS	77		None				✓	8
MS		43						
CS		201						

Table 3.3 (Cont'd) N-716 Element Selections								
System[1]	Weld Count		N-716 Selection Considerations					Selections
	HSS	LSS	DMs	RCPB	RCPB$^{IF/IV}$	RCPB^{OC}	BER	
SUMMARY RESULTS FOR ALL SYSTEMS	1		TASCS, TT, PWSCC	✓	✓			1
	19		TASCS, TT	✓	✓			6
	6		TASCS, (FAC)				✓	2
	8		TT, IGSCC	✓				2
	2		TT, PWSCC	✓	✓			1
	2		TT, (FAC)					0
	30		TASCS	✓	✓			8
	3		TASCS				✓	3
	2		TASCS					0
	91		TT	✓	✓			28
	111		TT	✓				8
	36		TT				✓	8
	22		TT					3
	16		PWSCC	✓	✓			5
	8		None (FAC)				✓	0
	2		None (FAC)					0
	329		None	✓	✓			15
	122		None	✓				0
	132		None				✓	9
	12		None					0
		1453						
TOTALS	954	1453						99

Note

1. Systems are described in Table 3.1.

Table 3.4-1 Risk Impact Analysis Results												
System[1]	Safety Significance	Break Location	Failure Potential		Inspections			CDF Impact		LERF Impact		
			DMs	Rank[3]	SXI[2]	RIS_B	Delta	w/ POD	w/o POD	w/ POD	w/o POD	
RC	High	LOCA	TASCS, TT, PWSCC	Medium	1	1	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
RC	High	LOCA	TASCS, TT	Medium	19	6	-13	2.52E-10	5.46E-09	2.52E-11	5.46E-10	
RC	High	LOCA	TT, PWSCC	Medium	2	1	-1	4.20E-10	4.20E-10	4.20E-11	4.20E-11	
RC	High	LOCA	TASCS	Medium	0	4	4	-3.02E-09	-1.68E-09	-3.02E-10	-1.68E-10	
RC	High	LOCA	TT	Medium	3	0	-3	7.56E-10	1.26E-09	7.56E-11	1.26E-10	
RC	High	LOCA	PWSCC	Medium	3	4	1	-4.20E-10	-4.20E-10	-4.20E-11	-4.20E-11	
RC	High	LOCA	None	Low	63	13	-50	1.05E-09	1.05E-09	1.05E-10	1.05E-10	
RC	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
RC TOTAL								**-9.66E-10**	**6.09E-09**	**-9.66E-11**	**6.09E-10**	
CH	High	LOCA	TT	Medium	0	11	11	-8.32E-09	-4.62E-09	-8.32E-10	-4.62E-10	
CH	High	ILOCA	TT	Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
CH	High	LOCA	None	Low	0	2	2	-4.20E-11	-4.20E-11	-4.20E-12	-4.20E-12	
CH	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
CH TOTAL								**-8.36E-09**	**-4.66E-09**	**-8.36E-10**	**-4.66E-10**	
SI	High	PLOCA – SD	TT, IGSCC	Medium	2	2	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
SI	High	LOCA	TASCS	Medium	3	4	1	-2.27E-09	-4.20E-10	-2.27E-10	-4.20E-11	
SI	High	LOCA	TT	Medium	8	17	9	-1.08E-08	-3.78E-09	-1.08E-09	-3.78E-10	
SI	High	PLOCA – SD	TT	Medium	4	8	4	-5.04E-09	-1.68E-09	-5.04E-10	-1.68E-10	
SI	High	BER – SI1	TT	Medium	2	0	-2	5.04E-10	8.40E-10	5.04E-11	8.40E-11	
SI	High	BER – SI2	TT	Medium	4	8	4	-5.04E-09	-1.68E-09	-5.04E-10	-1.68E-10	
SI	High	LOCA	PWSCC	Medium	2	1	-1	4.20E-10	4.20E-10	4.20E-11	4.20E-11	
SI	High	LOCA	None	Low	6	0	-6	1.26E-10	1.26E-10	1.26E-11	1.26E-11	
SI	High	PLOCA – SD	None	Low	1	0	-1	2.10E-11	2.10E-11	2.10E-12	2.10E-12	
SI	High	PLOCA	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
SI	High	BER – SI1	None	Low	4	0	-4	8.40E-11	8.40E-11	8.40E-12	8.40E-12	
SI	High	BER – SI2	None	Low	2	0	-2	4.20E-11	4.20E-11	4.20E-12	4.20E-12	
SI	Low	Class 2 LSS[4]	N/A	Assume Medium	59	0	-59	5.90E-10	5.90E-10	5.90E-11	5.90E-11	
SI TOTAL								**-2.14E-08**	**-5.44E-09**	**-2.14E-09**	**-5.44E-10**	

Table 3.4-1 (Cont'd) Risk Impact Analysis Results											
System[1]	Safety Significance	Break Location	Failure Potential		Inspections			CDF Impact		LERF Impact	
			DMs	Rank[3]	SXI[2]	RIS_B	Delta	w/ POD	w/o POD	w/ POD	w/o POD
EF	High	BER – FW2	TT, (FAC)	Medium (High)	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
EF	High	BER – FW2	TT	Medium	2	3	1	-1.76E-09	-4.20E-10	-1.76E-10	-4.20E-11
EF	High	BER – FW2	None (FAC)	Low (High)	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
EF TOTAL								**-1.76E-09**	**-4.20E-10**	**-1.76E-10**	**-4.20E-11**
FW	High	BER – FW1	TASCS, (FAC)	Medium (High)	0	2	2	-1.51E-09	-8.40E-10	-1.51E-10	-8.40E-11
FW	High	BER – FW1	TASCS	Medium	3	3	0	-1.51E-09	0.00E+00	-1.51E-10	0.00E+00
FW	High	BER – FW1	None (FAC)	Low (High)	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
FW	High	BER – FW2	None (FAC)	Low (High)	1	0	-1	2.10E-11	2.10E-11	2.10E-12	2.10E-12
FW	High	BER – FW1	None	Low	2	1	-1	2.10E-11	2.10E-11	2.10E-12	2.10E-12
FW	High	BER – FW2	None	Low	3	0	-3	6.30E-11	6.30E-11	6.30E-12	6.30E-12
FW TOTAL								**-2.92E-09**	**-7.35E-10**	**-2.92E-10**	**-7.35E-11**
MS	High	BER – MS	None	Low	10	8	-2	4.20E-11	4.20E-11	4.20E-12	4.20E-12
MS	Low	Class 2 LSS[4]	N/A	Assume Medium	7	0	-7	7.00E-11	7.00E-11	7.00E-12	7.00E-12
MS TOTAL								**1.12E-10**	**1.12E-10**	**1.12E-11**	**1.12E-11**
CS	Low	Class 2 LSS[4]	N/A	Assume Medium	15	0	-15	1.50E-10	1.50E-10	1.50E-11	1.50E-11
CS TOTAL								**1.50E-10**	**1.50E-10**	**1.50E-11**	**1.50E-11**
GRAND TOTAL								**-3.51E-08**	**-4.90E-09**	**-3.51E-09**	**-4.90E-10**

Notes

1. Systems are described in Table 3.1.
2. Only those ASME Section XI Code inspection locations that received a volumetric examination in addition to a surface examination are included in the count. Inspection locations previously subjected to a surface examination only were not considered in accordance with Section 3.7.1 of EPRI TR-112657.
3. The failure potential rank for high safety significant (HSS) locations is then assigned as "High", "Medium", or "Low" depending upon potential susceptibly to the various types of degradation. [Note: Low safety significant (LSS) locations were conservatively assumed to be a rank of Medium (i.e., "Assume Medium").
4. The "Class 2" designation in Table 3.4-1 is used to identify those Code Class 2 locations that are not HSS because they do not meet any of the five HSS criteria of Section 2(a) of N-716 (e.g., not part of the BER scope).

Table 5 Inspection Location Selection Comparison Between ASME Section XI Code and Code Case N-716											
System(1)	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N-716	
	High	Low		DMs	Rank(3)			Vol/Sur	Sur Only	RIS_B	Other(2)
RC	✓		LOCA	TASCS, TT, PWSCC	Medium	B-J^{DMW}	1	1	0	1	–
RC	✓		LOCA	TASCS, TT	Medium	B-J	19	19	0	6	–
RC	✓		LOCA	TT, PWSCC	Medium	B-F	2	2	0	1	–
RC	✓		LOCA	TASCS	Medium	B-J	16	0	0	4	–
RC	✓		LOCA	TT	Medium	B-J	3	3	0	0	–
RC	✓		LOCA	PWSCC	Medium	B-F	3	3	0	1	–
						B-J^{DMW}	11	0	1	3	–
RC	✓		LOCA	None	Low	B-J^{DMW}	13	8	5	0	–
						B-J	200	55	36	13	–
RC	✓		PLOCA	None	Low	B-J	14	0	1	0	–
CH	✓		LOCA	TT	Medium	B-J^{DMW}	2	0	2	0	–
						B-J	34	0	8	11	–
CH	✓		ILOCA	TT	Medium	B-J	5	0	3	0	–
CH	✓		LOCA	None	Low	B-J^{DMW}	1	0	1	0	–
						B-J	75	0	1	2	–
CH	✓		PLOCA	None	Low	B-J	12	0	1	0	–
SI	✓		PLOCA – SD	TT, IGSCC	Medium	B-J	8	2	0	2	–
SI	✓		LOCA	TASCS	Medium	B-J	14	3	0	4	–
SI	✓		LOCA	TT	Medium	B-J^{DMW}	4	4	0	0	–
						B-J	48	4	0	17	–
SI	✓		PLOCA – SD	TT	Medium	B-J	106	4	1	8	–
SI	✓		BER – SI1	TT	Medium	C-F-1	8	2	0	0	–
SI	✓		BER – SI2	TT	Medium	C-F-1	32	4	0	8	–
SI	✓		LOCA	PWSCC	Medium	B-J^{DMW}	2	2	0	1	–
SI	✓		LOCA	None	Low	B-J	40	6	7	0	–
SI	✓		PLOCA – SD	None	Low	B-J	55	1	3	0	–
SI	✓		PLOCA	None	Low	B-J	41	0	7	0	–
SI	✓		BER – SI1	None	Low	C-F-1	26	4	0	0	–
SI	✓		BER – SI2	None	Low	C-F-1	8	2	0	0	–
SI		✓	Class 2 LSS	N/A	Assume Medium	C-F-1	1209	59	25	0	–

Table 5 (Cont'd) Inspection Location Selection Comparison Between ASME Section XI Code and Code Case N-716											
System[1]	Safety Significance		Break Location	Failure Potential		Code Category	Weld Count	Section XI		Code Case N-716	
	High	Low		DMs	Rank[3]			Vol/Sur	Sur Only	RIS_B	Other[2]
EF	✓		BER – FW2	TT, (FAC)	Medium (High)	C-F-2	2	0	0	0	–
EF	✓		BER – FW2	TT	Medium	C-F-2	18	2	0	3	–
EF	✓		BER – FW2	None (FAC)	Low (High)	C-F-2	2	0	1	0	–
FW	✓		BER – FW1	TASCS, (FAC)	Medium (High)	C-F-2	6	0	0	2	–
FW	✓		BER – FW1	TASCS	Medium	C-F-2	5	3	0	3	–
FW	✓		BER – FW1	None (FAC)	Low (High)	C-F-2	4	0	0	0	–
FW	✓		BER – FW2	None (FAC)	Low (High)	C-F-2	4	1	0	0	–
FW	✓		BER – FW1	None	Low	C-F-2	17	2	0	1	–
FW	✓		BER – FW2	None	Low	C-F-2	14	3	0	0	–
						NNS	2	0	0	0	–
MS	✓		BER – MS	None	Low	C-F-2	75	10	0	8	–
						NNS	2	0	0	0	–
MS		✓	Class 2 LSS	N/A	Assume Medium	C-F-2	43	7	0	0	–
CS		✓	Class 2 LSS	N/A	Assume Medium	C-F-1	201	15	0	0	–

Notes

1. Systems are described in Table 3.1.
2. The column labeled "Other" is generally used to identify plant augmented inspection program locations credited per Section 4 of Code Case N-716. Code Case N-716 allows the existing plant augmented inspection program for IGSCC (Categories B through G) in a BWR to be credited toward the 10% requirement. This option is not applicable for the W3 RIS_B application. The "Other" column has been retained in this table solely for uniformity purposes with other RIS_B application template submittals.
3. The failure potential rank for high safety significant (HSS) locations is then assigned as "High", "Medium", or "Low" depending upon potential susceptibly to the various types of degradation. [Note: Low safety significant (LSS) locations were conservatively assumed to be a rank of Medium (i.e., "Assume Medium").

ENCLOSURE 2

W3F1-2008-0013

Waterford 3 Specific Responses to GGNS RAI Questions
Not Incorporated into the Revised Request for Alternative W3-ISI-005

Waterford 3 Specific Responses to GGNS RAI Questions
Not Incorporated into the Revised Request for Alternative W3-ISI-005

Methodology Clarifications

1. Section 5(c) in N–716 does not appear to provide a "with probability of detection (POD)" and "without POD" option in the calculation but the submittal includes one set of estimates for "with POD" and another "w/o POD" in Table 3.4-1. Please clarify how the "with POD" and "w/o POD" columns in Table 3.4-1 are consistent with Section 5(c) in N-716.

   Response

   It is true that N-716 does not discuss the two options presented above. The W3 submittal contained both options in order to be consistent with previous RI-ISI submittals which contained both options. These two sets of analyses are typically conducted to provide a sensitivity of the delta risk evaluation with respect to assumptions on POD.

2. The estimates in the "w/o POD" column in Table 3.4-1 seem to include a standard POD of 0.5. Is this correct? If not, please provide some examples using the conditional core damage probability (CCDP) values from page 11 of 28 to produce the entries in Table 3.4-1.

   Response

   That is correct; the "w/o POD" column applies a POD of 0.5 for both the Section XI program and the N-716 program. Thus, there is no extra credit assumed for an N-716 inspection as compared to Section XI inspection as to inspection effectiveness (e.g., due to larger inspection volumes in the N-716 program).

3. Please provide a discussion justifying the guideline value for CDF selected in Section 2(5) in N-716 (i.e., 1E-6/year).

   Response

   As discussed in the response to RAI 2a), Entergy has added a criterion for LERF of 1E-07/year.

   From a practical perspective, the criterion used in Section 2(a)(5) of N-716 has two potential impacts. Each is discussed below.

   1. Class 2 Piping

      Any piping that has inspections added or removed per this code case, regardless of the value of this criterion, is required to be assessed as to its impact on risk. This risk impact analysis is conducted on an individual system basis, which includes the cumulative effect of LSS Class 2 piping currently being inspected. The change-in-risk acceptance criteria on a system basis are defined as 1E-07/year (CDF) and 1E-08/year (LERF). These criteria are derived from Regulatory Guide (RG) 1.174 and were approved by the NRC in EPRI TR-112657. If the change-in-risk acceptance criteria are not met, additional inspections are to be defined until these criteria are met [N-716 Section 5(d)]. Therefore, regardless of the number of segments (or inspections) that fall below these criteria, unacceptable risk changes will not occur and the safety objectives of risk-informed regulation will be met.

The change-in-risk analysis could be conducted without the benefit of these criteria [i.e., Section 2(a)(5) of N-716 and LERF per RAI 2a)] and shown to have acceptable changes in plant risk. In fact, this was demonstrated in the N-716 whitepaper where eight plants (4 BWRs, 4 PWRs) were compared to the N-716 criteria. N-716 was shown to provide for more inspections than traditional RI-ISI approaches even when the criterion of Section 2(a)(5) was not used. And, as expected, the change-in-risk acceptance criteria of 1E-07/year (CDF) and 1E-08/year (LERF) were met for these eight plants. However, implementation of this ancillary criteria [Section 2(a)(5) of N-716 and LERF per RAI 2a)] provides increased confidence that the change-in-risk acceptance criteria will be met without the need for additional inspections as would be required by Section 5(d) of N-716. Thus, any risk outliers, if they exist in Class 2 piping [(e.g., piping that exceeds the Section 2(a)(5) criterion and LERF per RAI 2a)], would require that, on a plant-specific basis, piping be added to the scope of HSS piping and subjected to inspection.

2. <u>Class 3 / NNS Piping</u>

   Currently, there are no Section XI NDE requirements for this piping. As such, use of this ancillary criteria [Section 2(a)(5) of N-716 and LERF per RAI 2a)], regardless of its value, can only result in a reduction in plant risk further supporting the safety objectives of risk-informed regulation. These additional inspections would be imposed on piping identified by the criterion of Section 2(a)(5) of N-716 and LERF per RAI 2 a) and cannot be used to reduce inspections in other HSS piping [see N-716 Section 4(b)].

From a more global perspective, the ancillary criteria of Section 2(a)(5) of N-716 and of LERF per RAI 2a) provide additional criteria that can only potentially increase the scope of HSS locations (i.e., will only increase the number of inspections). Although, the criteria of Sections 2(a)(1) through 2(a)(4) of N-716 were created based on the large number of risk-informed applications performed to date, Section 2(a)(5) of N-716 and LERF per RAI 2a) were added as a defense-in-depth measure to N-716 to provide a method of ensuring that any plant-specific locations that are important to safety are identified.

Adopting RI-ISI programs permits a reduction in inspection by focusing inspections on the more important locations while, at the same time, maintaining or improving public health and safety. Use of this ancillary guideline and a technically adequate, plant-specific flooding evaluation to identify relatively important locations (e.g., Class 2, 3, or NNS piping) provides additional confidence that inspections will be focused on the more important locations.

According to the guidelines in RG 1.174, plant changes (permitting the reallocation of resources) that increase risk less than 1E-06/year (CDF) / 1E-07/year (LERF) would normally be considered very small and acceptable as long as the other principles are satisfied. This is considered to be a reasonable metric for identifying significant pipe segments since the potential reduction in CDF (LERF) from inclusion of such segments in the ISI program would also be very small. Additionally, use of the guideline value of 1E-06/year for CDF (1E-07/year for LERF) taken together with the system level change-in-risk limits of 1E-07/year for CDF (1E-08/year for LERF) provides additional assurance that plant-specific application of N-716 will meet the acceptance criteria of Region III in Figures 3 and 4 of RG 1.174. Thus, assuring any increase would be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

Finally, traditional RI-ISI approaches can be applied on a partial scope basis. That is, many plants have applied RI-ISI to Class 1 piping only. Thus, these plants have not witnessed the additional safety benefit of identifying and inspecting Class 2, 3, or NNS piping per criterion Section 2(a)(5) of N-716 and LERF per RAI 2a)

4. Is the population of welds that should be included in the N-716 change in risk estimate all welds that were inspected under Section XI and that will be inspected under the RI-ISI program? If not, where in code Case N-716 is the guidance that reduces the population of welds that should be included in the change-in-risk estimate.

   Response

   The population of welds to be included in the change-in-risk assessment includes all welds receiving NDE except for those that receive only a surface examination and are not susceptible to outside diameter attack [e.g., external chloride stress corrosion cracking (ECSCC)]. This population includes so-called "risk category 6 and 7" locations, which are not required to be included in the RI-ISI delta risk assessment. (Note: Table 5 of W3-ISI-005 lists the surface examination requirements prior to W3 implementation of ASME Code Case N-663.)

   It is the intent of the Code Case authors to update N-716 to reflect this requirement (i.e. exclusion of surface-only examinations without outside diameter attack) as well as any other relevant feedback from the pilot plant process.

5. How the approach used to analyze piping system failures for the plant specific PRA of pressure boundary failures compares to the approach described in Section 2.1.4 of RG 1.178.

   Response

   The purpose of segments and segment definitions are identical between the ASME Code Case N-716 (N-716) approach and that of the EPRI RI-ISI methodology. In both methodologies, segments are used only as an accounting/tracking tool. That is, whether the weld is tracked individually or as part of a segment, the results of the risk ranking and element selection part of the methodology will not change. In both approaches, whether the segment is small (e.g., a single weld) or large (e.g., many welds), all of the welds will be ranked and then subject to a fixed sampling percentage for determining the size of the inspection population.

   As an example, if the population of high safety significant (HSS) welds is 100, whether they are tracked as ten (10) segments (e.g., ten welds per segment) or two (2) segments (50 welds per segment), all 100 welds would be subject to the element selection process. For example, 25% of HSS welds with susceptibly to a degradation mechanism would be selected for applications and 25% of welds identified as Risk Category 2 would be selected for EPRI RI-ISI applications.

6. Is the guideline to examine a minimum 10% of all HSS welds, 10% of all HSS butt welds, 10% of all HSS butt welds $\geq$ 4 NPS, or something else?

   Response

   Yes, the guideline is to examine a minimum of 10% of HSS welds. For W3, this population includes butt welds that are both less than, equal to, and greater than 4 NPS. W3 has no HSS socket welds.

   Additionally, a lessons learned from the GGNS application was that the wording of N-716 could be clearer in its intent to require inspection of at least 10% of the reactor coolant pressure boundary (RCPB). While the W3 application meets this intent, it is also the author's intent to revise N-716 to make this requirement clearer, as well as other lessons learned from its application [see the response to Question 3(a) from the first set of RAIs].

7. What type of inspections can be counted as part of the required population? For example, can visual examinations or wall thickness exams be counted in the 10%?

   Response

   Per N-716, wall thickness exams as part of the FAC and localized corrosion (excluding crevice corrosion) programs cannot be counted as part of the 10% required population. Because of the nature of the degradation, wall thinning examination for locations potentially susceptible to erosion-cavitation will be conducted.

   Per N-716, the requirements for examination of socket welds and smaller bore branch connections (i.e., $\leq$ 2 NPS) susceptible to thermal fatigue shall be a volumetric exam of the piping base metal within ½ inch of the toe of the weld and a visual of the fitting itself.

   Thus, HSS inspections required by N-716 shall be volumetric exams as part of the W3 application.

8. At the top of page 9 of 10, GG-ISI-002 identifies four (4) primary guidelines on selecting inspection locations, or six (6) guidelines if each sub-bullet in (1) is counted as a guideline. Please describe briefly how each of these six guidelines was applied (e.g., how many inspections were influenced by the guideline and if application of the guideline resulted in changes to the original locations) when you were selecting inspection locations at Grand Gulf. Also, discuss whether there were any inspections added due to change in risk considerations.

   Response

   The process of defining the inspection population of an N-716 application is an iterative process. The first step is to define the scope of HSS welds on a "per system" basis. As a starting point, N-716 requires that 10% of the HSS welds, on a "per system" basis, be selected for inspection (see Table below, column entitled "HSS"). The next step is to assure that 10% of Class 1 welds are selected (see Table below, column entitled "RCPB"). It should be noted that a lesson learned from the GGNS application is that this requirement could be more clearly stated in N-716 and it is the author's intent to revise the code case to reflect this and other lessons learned, as applicable. The next step is to assure that 25% of locations identified as potentially susceptible to some type of degradation mechanism be selected (see Table below column entitled "DMs"). The next step is to confirm that two thirds of the identified inspections for the RCPB are within the first isolation valve or move inspections from between the two isolation valves to within the first isolation valve to compensate, if necessary (see Table below, column entitled "RCPB^{IFIV}"). The next step is to confirm, or select if necessary, so that 10% of the RCPB that lies outside containment is inspected (see Table below, column entitled "RCPB^{OC}"). Finally, inspections are chosen so that 10% of the break exclusion region (BER) populations are chosen (see Table below, column entitled "BER"). Again, this may have already been accomplished by the preceding criteria, but needs to be confirmed or adjusted accordingly.

Depending upon how the element selection process is ordered, it may be necessary to iterate once or twice to assure the criteria are met. Because of rounding up, the selection being done on a system-by-systems basis, and the multiple criteria, it is expected that a greater than a 10% inspection population will be attained (e.g., W3 witnessed 10.41%).

With respect to change-in-risk considerations, no changes to the number or locations of inspections were required.

W3-ISI-005, Page 21 of 26, Table 3.3)

System	Selections	HSS(1)	DMs(2)	RCPB(3)	RCPB IFIV(4)	RCPB OC(5)	BER(6)
RC	Required	29 of 282	14 of 55	29 of 282	20	n/a	n/a
	Made	29	16	29	29	n/a	n/a
CH	Required	13 of 129	11 of 41	13 of 129	9	n/a	n/a
	Made	13	11	13	13	n/a	n/a
SI	Required	40 of 392	40(56) of 222	32 of 318	22	n/a	7 of 62
	Made	40	40	32	22	n/a	8
EF	Required	3 of 22	3(5) of 20	n/a	n/a	n/a	n/a
	Made	3	3	n/a	n/a	n/a	n/a
FW	Required	6 of 52	3 of 11	n/a	n/a	n/a	5 of 46
	Made	6	5	n/a	n/a	n/a	6
MS	Required	8 of 77	n/a	n/a	n/a	n/a	8 of 77
	Made	8	n/a	n/a	n/a	n/a	8
CS	Required	n/a	n/a	n/a	n/a	n/a	n/a
	Made	n/a	n/a	n/a	n/a	n/a	n/a
TOTAL	**Made**	**99**	**75**	**74**	**64**	**n/a**	**22**

Notes

1. Ten percent of the HSS piping welds shall be selected for examination per system.
2. A minimum of 25% of the piping weld population identified as susceptible to each degradation mechanism and degradation mechanism combination shall be selected for examination per system. For the safety injection and emergency feedwater systems, no more than 10% of the HSS piping welds are required to be selected for examination.
3. Al least 10% of the RCPB piping weld population must be selected for examination per system.
4. Al least 2/3 of the RCPB piping welds selected for examination must be located between the first isolation valve and the reactor pressure vessel per system.
5. A minimum of 10% of the RCPB piping welds outside containment must be selected for examination per system [W3 does not have RCPB piping welds outside containment, therefore, this requirement does not apply].
6. A minimum of 10% of the BER piping welds must be selected for examination per system.

9. Please describe how volumetric examinations will be performed. Will, at a minimum, volumetric examinations include the volume required for ASME Section XI examinations? Will ASME Section XI, Appendix VIII qualified examiners and procedures be used for all volumetric exams? Will the examination volume be scanned for both axial and transverse indications for all exams? Please describe and justify your answers.

   <u>Response</u>

   Volumetric examinations will be performed as required by Table 1 of N-716. The table requires an examination volume as defined in the ASME Section XI IWB figures. This would require examination of at least the ASME Section XI volume. (More volume may be required based on the notes on Table 1.) N-716 does not take any exceptions to the paragraphs of the Code that govern volumetric examinations and the request for alternative does not take exception to any 10 CFR limitations. Therefore, Entergy will examine these welds using the same personnel and procedure requirements as a traditional Section XI piping volumetric examination.

10. Section 3.3.2 also states that an attempt was made to select locations for examination such that a minimum >90% coverage is attained. Discuss how this attempt was conducted. If less than 90% examination is completed, discuss whether additional weld(s) will be examined to compensate for the limited examination coverage.

    <u>Response</u>

    As discussed in EPRI TR-112657, accessibility is an important consideration in the element selection process of a RI-ISI application. As such, for the W3 N-716 application, locations have generally been selected for examination where the desired coverage is achievable. This is typically accomplished by utilizing previous inspection history, plant access considerations, and knowledgeable plant personnel. However, some limitations will not be known until the examination is performed since some locations will be examined for the first time.

    In addition, other considerations may take precedence and dictate the selection of locations where greater than 90% examination coverage is physically impossible. This is especially true for element selections where a degradation mechanism may be operative (e.g., risk categories 1, 2, 3 and 5 of EPRI TR-112657). For these locations, elements are generally selected for examination on the basis of predicted degradation severity. For example, in the emergency core cooling system (ECCS) injection lines of PWRs, the piping section immediately upstream of the first isolation check valve is considered susceptible to intergranular stress corrosion cracking (IGSCC), assuming a sufficiently high temperature and oxygenated water supply. The piping element (pipe-to-valve weld) located nearest the heat source will be subjected to the highest temperature (conduction heating). As such, this location will generally be selected for examination since it is considered more susceptible than locations further removed from the heat source, even though a pipe-to-valve weld is inherently more difficult to examine and obtain full coverage than most other configurations (e.g., pipe-to-elbow weld). In this example, less than 90% coverage of this location will yield far more valuable information than 100% coverage of a less susceptible location.

For locations with no identified degradation mechanisms (i.e., similar to risk category 4 of EPRI TR-112657), a greater degree of flexibility exists in choosing inspection locations. As such, if at the time of examination an N-716 element selection is found to be obstructed, a more suitable location may be substituted instead.

Therefore, Entergy will review each instance of limited coverage and take the appropriate steps (e.g., relief requests) consistent with its impact on the basis of the N-716 application.

ENCLOSURE 3

W3F1-2008-0013

GGNS RAI Question Cross Reference List to Location of W3 Specific Responses within Enclosures 1 and 2

GGNS RAI Set 1	
Question	**W3 Specific Response**
1a)	Enclosure 1 – Table 3.4-1 note 3 and Table 5 note 3
1b)	Enclosure 2 - Methodology Clarifications Q1
1c)	Enclosure 2 - Methodology Clarifications Q2
1d)	Enclosure 1 - Section 4 Para 1
2a)	Enclosure 1 –Previously included in section 3.1(5) and Table 3.1
2b)	Enclosure 2 - Methodology Clarifications Q3
2c)	Subsumed by new flooding study
2d)	Enclosure 1 - Previously included in Section 1.2
3a)	Enclosure 2 - Methodology Clarifications Q4
3b)	Enclosure 1 - Section 3.4.1 Para 2-5
4a)	Enclosure 1 - Section 3.4.1, Para 7
4b)	Enclosure 1 - Page 14, note 1 to CCDP and CLERP Values Based on Break Location Table
4c)	Same as 2c) and 3b) above
4d)	Enclosure 1 - Table 3.4-1 note 4
4e)	Enclosure 1 - Page 14, note 2 to CCDP and CLERP Values Based on Break Location Table
5)	Enclosure 1 - Section 3.3 (5)
6)	Enclosure 1 - Previously included Table 5

GGNS RAI Set 2	
Question	**W3 Specific Response**
1a)	Enclosure 2 - Methodology Clarifications Q5
1b)	Subsumed by new flooding study
1c)	Enclosure 1 – Section 1.2
1d)	Subsumed by new flooding study
2a)	Enclosure 2 - Methodology Clarifications Q6
2b)	Enclosure 2 - Methodology Clarifications Q7
2c)	Enclosure 1 - Previously included in Table in Section 3.3(5)
3)	Question not applicable to W3 since change in risk is negative in all cases.
4)	Enclosure 2 - Methodology Clarifications Q8
5a)	Enclosure 1 - Section 5
5b)	Enclosure 1 - Previously included in Section 5
5c)	Enclosure 1 - Section 3.3
6)	Enclosure 2 - Methodology Clarifications Q9
7)	Enclosure 1 - Section 4
8)	Enclosure 1 - Section 3.3.1
8a)	Enclosure 1 - Section 3.3.1
8b)	Enclosure 1 - Section 3.3.1
8c)	Enclosure 1 - Section 3.3.1
9)	Enclosure 1 - Section 3.3.2
10)	Enclosure 2 - Methodology Clarifications Q10

W3 Specific Response to GGNS 8/20/2007 Telephone call	
1	Subsumed by new flooding study

ENCLOSURE 4

W3F1-2008-0013

LICENSEE-IDENTIFIED COMMITMENTS

LICENSEE-IDENTIFIED COMMITMENTS

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
W3 is in the process of evaluating MRP-146, *Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines*, and these results will be incorporated into the RIS_B Program, if warranted.	✓		June 30, 2011
Request for Alternative CEP-ISI-007 pertaining to the application of Code Case N-663 will be withdrawn for use at W3 upon NRC approval of the RIS_B Program submittal.	✓		Upon NRC approval of W3-ISI-005