Entergy

Entergy Operations, Inc.
1340 Echelon Parkway
Jackson, Mississippi 39213-8298
Tel 601-368-5000

CNRO-2007-00022

May 23, 2007

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

SUBJECT: Request for Alternative GG-ISI-002
Responses to Requests for Additional Information

Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29

REFERENCE: Entergy Operations, Inc. letter CNRO-2006-00043 to the NRC, dated September 22, 2006

Dear Sir or Madam:

In the referenced letter, Entergy Operations, Inc. (Entergy) submitted Request for Alternative GG-ISI-002, which requests approval to implement a risk-informed inservice inspection (ISI) program at Grand Gulf Nuclear Station (GGNS). The program is to be based on ASME Code Case N-716, *Alternative Piping Classification and Examination Requirements, Section XI Division 1.* During their review of GG-ISI-002, the NRC staff provided, via e-mail, two sets of Requests for Additional Information (RAI). Entergy provided draft responses to the staff, which were reviewed and discussed at a meeting on May 7, 2007. Based on those discussions, Entergy is providing revised responses to the RAIs contained in Enclosures 1 and 2 of this letter. Changes are denoted by revision bars in the margins, where applicable. Should you have any questions regarding this submittal, please contact Guy Davant at (601) 368-5756.

As stated in the referenced letter, Entergy requests that the NRC approve GG-ISI-002 by September 22, 2007 in order to support the upcoming fall 2007 refueling outage at GGNS.

A047

This letter contains no new commitments.

Sincerely,

Steven A. Saunders
Manager, Engineering Projects

SAS/GHD/ghd

Enclosures:
1. Responses to Request for Additional Information Set #1
2. Responses to Request for Additional Information Set #2

cc:
Mr. J. S. Forbes (ECH)
Mr. O. Limpias (ECH)
Mr. W. R. Brian (GGNS)

Dr. Bruce S. Mallett
Regional Administrator, Region IV
U. S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

NRC Senior Resident Inspector
Grand Gulf Nuclear Station
Route 2, Box 399
Port Gibson, MS 39150

U.S. Nuclear Regulatory Commission
Attn: Mr. B. K. Vaidya
MS O-7 D1
Washington, DC 20555-0001

ENCLOSURE 1

CNRO-2007-00022

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION SET #1

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION SET #1 REGARDING REQUEST FOR ALTERNATIVE GG-ISI-002

1) Entergy, "requests authorization to implement a risk-informed inservice inspection (ISI) program based on American Society of Mechanical Engineers Boiler and Pressure Vessel Code Case N-716 (N-716)." There appears to be, however, some differences between the methodology in N-716 and the method applied by Entergy as described in the submittal.

   a) Table 3 in N-716 discusses high, medium, and low failure potential and pairs these potentials with degradation categories large brake, small leak, and none respectively. It does not appear that this table was used in the submittal. Was this table used in the submittal? If not, what was used in lieu of Table 3?

   Response

   The information contained in Table 3 of N-716 was used in the GGNS application and submittal. The information is identified in Table 3.4-1 and Table 5 of the submittal. The information is contained in the column identified as "Failure Potential." This column is further divided into two sub-columns (i.e., "DMs" and "Rank"). 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"). See response to Question 3b, below.

   b) 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 GGNS 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.

   c) 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).

d) Section 7 in N-716, "Program Updates," includes several steps that make up a program update. Page 14 of 28 in your submittal states that, "[u]pon approval of the RIS_B Program, procedures that comply with the guidelines described in Electric Power Research Institute (EPRI) TR-112657 (EPRI Topical) will be prepared to implement and monitor the program." Please identify the Sections in the EPRI Topical that describe the update program that Exelon intends to implement. Please describe and compare the update program that Exelon intends to implement against the characteristics of such a program as described in Section 7 of N-716.

Response

The wording in GG-ISI-002 is based on previous industry RI-ISI submittals. While the intent of both updating processes (EPRI TR-112657 and N-716) is the same, Entergy will meet the wording of N-716.

2) The relationship between N-716's guideline that, "any piping segment whose contribution to core damage frequency (CDF) is greater than 1E-6/year is a high safety significant (HSS) segment," and the EPRI Topical guidelines for safety significant categorization is unclear. For example, a low consequence segment in the EPRI Topical methodology has a CCDP less than 1E-6, an identical numerical value but a different metric than the 1E-6/year guideline in N-716. Page 3-8 in the EPRI Topical provides an explanation that the CCDP and conditional large early release probability (CLERP) ranges were selected, "to guarantee that all pipe locations ranked in the low consequence category do not have a potential CDF impact higher than 1E-8 per year or a potential large early release frequency (LERF) impact higher than 1E-9 per year." Inspection of Table 3.1 in your submittal also indicates that there are no entries in the "CDF > 1E-6" column indicating that no segments in the Grand Gulf flooding probabilistic risk assessment (PRA) exceeded this guideline.

a) The N-716 code case Section 2(5) does not include a LERF guideline analogous to the CDF guideline, and Table 3-1 in your submittal includes a column for CDF but not for LERF. Please explain why a LERF guideline is not included as a guideline in parallel with CDF.

Response

N-716 provides five criteria for determining the classification of welds. These criteria are specifically designed to address CDF considerations and LERF considerations [e.g., break exclusion region (BER)]. Additionally, Section 4 of the code case was also developed to specifically address CDF and LERF considerations. In particular, Section 4(d) (LOCA – outside containment) and Section 4(e) (BER) are important requirements of the Case, from a LERF perspective. As such, Sections 2 and 4 taken together are used to define the revised inspection program (i.e., the number and locations for inspection).

The CDF guideline for PRA internal flood segments was added to provide additional margin, as applicable, to the initial scope of HSS welds (i.e., a "belts and suspenders" approach). As discussed in the whitepaper, N-716 is based upon lessons learned from a large number of risk-informed applications (e.g., RI-ISI, RI-BER). With respect to defining the scope (e.g., HSS vs. LSS), these insights include both the impact on CDF *and LERF (e.g., RI-BER insights). In the whitepaper, eight plants (4 BWRs and 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(5) is not used.*

Additionally, as a final step, after Sections 2 and 4 have been completed, N-716 requires an assessment of the impact on plant risk, which includes both CDF and LERF. This change-in-risk assessment includes so-called "risk category 6 and 7" locations, which are not required to be included in the EPRI RI-ISI delta risk assessment. Risk acceptance criteria for these metrics, are consistent with other RI-ISI applications and meet Regulatory Guide (RG) 1.174 criteria.

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

Response

N-716 provides five criteria for determining the classification of welds. The CDF guideline was added to provide additional margin, as applicable, to the initial scope of high safety significant welds (i.e. a belts and suspenders approach). As discussed in the whitepaper, N-716 is based upon lessons learned from a large number of risk-informed applications (e.g., RI-ISI, RI-BER). In the whitepaper, eight plants (4 BWRs, 4 PWRs) were compared to the N-716 criteria and N-716 was shown to provide for more inspections than traditional RI-ISI approaches even when the criterion of section 2(5) is not used.

Section 2(5) of N-716 provides an additional criterion that can only potentially increase the scope of high safety significant locations (i.e. will only increase the number of inspections). Consistent with EPRI TR-112657 (Section 3.3.2), the value of 1E-6 (CDF) was chosen as a value that is suitably small and is consistent with the decision criteria for acceptable changes in CDF found in Reg. Guide 1.174. Further, the guideline value is consistent with the philosophy found in EPRI TR-105396 (PRA Applications Guide). Allocating resources (e.g., NDE) on components below this guideline value (e.g., section 4.2.2) will provide negligible risk benefit while expending unnecessary worker dose and radwaste. However, the purpose of this criterion is to act as a trigger value rather than a formal definition of risk significance.

From a practical perspective, this criterion has two potential impacts.

1. *It allows for reducing the inspection for some Class 2 piping. This is acceptable, because N-716 requires as a final check that the revised program per N-716 be compared against the previous ISI program and has requirements, on a system level, that changes in risk meet the acceptance criteria of NRC approved EPRI TR-112657. That is, for each individual system that witnesses a reduction in the number of inspections, an assessment of the impact of these reductions on both CDF and LERF is conducted. These acceptance criteria are consistent, albeit conservative, as compared to RG 1.174.*

2. *It could require inspection of some Class 3 and / non-nuclear safety (NNS) piping. This piping is not currently required to be inspected (i.e., NDE) by ASME or NRC. As such, use of this criterion, regardless of its value, can only result in a reduction in plant risk. It should be noted that all other NRC approved RI-ISI methodologies, allow for the use of limited scope applications (e.g., Class 1 only, Class 1 and 2 only). None of these approved methodologies require that Class 3 or NNS piping be included in the RI-ISI scope of application. Thus, N-716 and the use of the Section 2(5)(a) criterion is conservative as compared to existing RI-ISI methodologies in terms of scope definition.*

Finally, the assessment of the impact on plant risk (Section 5 of N-716) due to implementing N-716 provides an additional level of assurance that the overall impact on plant risk (CDF and LERF) will be acceptably low. Risk acceptance criteria for these metrics, are consistent with other RI-ISI applications and meet Reg. Guide 1.174 criteria. This change in risk assessment also includes so-called "risk category 6 and 7" locations, which are not required to be included in the EPRI RI-ISI delta risk assessment.

c) Please provide a list of the piping segments that were compared to the > 1E-6/year criterion along with the CDF and LERF estimates, the pipe failure frequency, and the CCDP and conditional large early release probability for each segment.

Response

The scope of piping reviewed against this criterion consisted of Class 2 piping not classified as HSS (e.g., BER, Class 3, and non-nuclear safety piping). The GGNS internal flooding study was used to conduct this comparison. The GGNS internal flooding study was performed in a step-by-step manner with an initial qualitative screening to identify the significant flood events and a quantitative analysis to determine the contribution to core damage for the most significant flood scenarios.

As opposed to a segment-by-segment evaluation, the GGNS internal flooding study was performed by defining flood zones, identification of their contents (e.g., important equipment), identification of potential flood sources, identification of flood propagation pathways, a qualitative screening analysis and a quantitative analysis of potentially important flood scenarios.

With respect to flood frequency, only the largest flood initiator per system in each flood zone was considered if the frequency and consequence of the larger flood initiator were approximately of the same magnitude as those of the smaller one. If the frequency of the smaller flood initiator was higher and its consequences similar to that of the larger initiator, the smaller flood initiator was considered the primary flood source for that particular system. For screening purposes, this is conservative from an internal flooding study perspective. It is also conservative from an N-716 perspective because some of these flooding sources (e.g., tanks) may not be within the N-716 scope (e.g., piping).

An example of the process is described as follows:

> *A flooding scenario in flood zone "A" revealed that a Flow Control Valve (FCV) in support of system "Z" would become submerged. Using the component failure matrix developed for the internal flooding study, this FCV is identified to fail when submerged. The fault tree for system "Z" is reviewed and the FCV failure is discovered to lead to the failure of the in-line pump, which results in system "Z" being unable to deliver flow to its loads. Therefore, the entire system "Z" fails due to submergence of the FCV. Subsequent to the analysis of the failure of system "Z", the dependency matrices were used to determine which other systems would fail [e.g., Instrument Air system failure would lead to failure of several mitigating systems, including Control Rod Drive (CRD), Containment Venting, Feedwater, Condensate, Component Cooling Water (CCW), Turbine Building Cooling Water (TBCW), and Plant Service Water (PSW)].*
>
> *A listing of failed mitigating systems for each flooding scenario, as well as available mitigative systems, was compiled for use in the qualitative analysis.*

Flood initiation frequency was on the order of 1E-3 to 1E-4 / year per zone. The failure of a single equipment train is on the order of 1E-2, except for some equipment [(e.g., Reactor Core Isolation Cooling (RCIC)] which can be higher (e.g., 1E-1). Therefore, the approximate likelihood of a flood plus two unrelated, random system failures is 1E-7 to 1E-8 / year. Due to the approximate nature of these estimated values, it is possible that a flood plus two random failures could occur with some significant probability. Thus, any flood scenario for which two or less random failures could produce core damage was analyzed in more detail. Similarly, any flood scenario for which three or more random system failures could produce core damage was screened out. Typically, this screening was done on a zone-by-zone basis. Thus, individual segments within the zone would have a likelihood of core damage less than that for the entire zone.

Based on the above, two flooding scenarios required detailed quantification. These scenarios involved the PSW system (CDF = 1.99E-7) and the standby service water system (CDF = 2.26E-8), which, after detailed quantitative assessment, fall below the criterion of section 2(5) of N-716.

d) Please provide any observations made during any independent reviews of the Grand Gulf flooding PRA or observations from the internal events review that are also applicable to the flooding analysis. Please describe how these observations have been resolved such that there is confidence that segments that have a CDF greater than the guideline value have been identified.

Response

As indicated in the initial submittal, the industry peer review of the GGNS PRA was conducted in August 1997. The facts and observations (F&O) from this review were characterized with regard to level of significance and given scores of A, B, C or D. An F&O with a level of significance of "A" is one that is extremely important and necessary to address to assure the technical adequacy of the PRA or the quality of the PRA. These should be addressed promptly. An F&O with a level of significance of "B" is one *that is important and necessary to address, but may be deferred until the next PRA update. "C" F&Os are of marginal importance, but are considered desirable to maintain maximum flexibility in PRA Applications and consistency in the industry. "D" F&Os are editorial or minor technical items left to the discretion of the utility. As such, the important F&Os to PRA technical adequacy and quality are those categorized as "A" or "B." Within the "A" & "B" F&Os, only two "B" F&Os are on the internal flooding analysis.*

The first "B" F&O stated that the dependency table in the internal flooding analysis did not list the Instrument Air system as a support system. No changes were necessary to address this comment since Instrument Air was clearly listed as a support system in various locations in the documentation, including the mitigating systems-versus-support systems dependency table and the support systems-versus-support systems table.

The remaining "B" F&O documented issues associated with a single flooding sequence. The first issue questioned whether there was a thermal hydraulic calculation which supports the use of a single CRD pump for success following a manual emergency depressurization. This issue was addressed by developing a calculation for CRD success criteria. As a result of the calculation, CRD is now credited only after another system (such as RCIC or HPCS) has provided core injection for approximately 5 hours and two CRD pumps operate. This modeling is incorporated into the modeling used to develop CCDPs for the N-716 analysis. The second issue pointed out that the text description of a sequence indicated that it resulted in core damage while the event tree indicated that the core was OK. The text description was in error and, since the event tree was the input into the development of the overall model fault tree, there was no related impact on the PRA results. The remaining issue stated that containment venting is not asked in the sequence; therefore, containment heat removal capability is unknown. That was basically a true statement as it was not necessary to vent the containment in order to determine the outcome of this sequence. Containment failure does not directly lead to failure of operating injection pumps since the most likely failure location is high in the containment and any steam released into the Auxiliary Building is not expected to impact these pumps which are low in the Auxiliary Building.

3) Section 5(c) in N-716 does not clearly specify what population of welds should be included in the change of risk estimates and what welds may be excluded. The description of the parameters in the equations in Section 5(c) indicates that any weld that was inspected under Section XI or that will be inspected under the RI-ISI program will be included in the change in risk estimate.

   a) 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 GG-ISI-002 lists the surface examination requirements prior to GGNS 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.

b) If all welds that were or will be inspected are included in the change-in-risk estimates in Table 4.4-1 in your submittal, how are the CCDP, CLERP, and the failure frequency estimated for LSS welds?

Response

For 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 derived from the GGNS internal flooding study. The 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 GGNS 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 application, 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.

4) Page 11 of 28 describes how the CCDP and CLERP of different categories of pipe breaks are estimated in support of the change-in-risk estimates. For example, bounding values for pipe breaks that result in isolable loss-of-coolant accidents (LOCAs) are derived as the product of the CCDP from unisolable LOCAs and the probability of a motor operated valve failing to close on demand. This type of an evaluation can be very analyst specific and essentially bypasses the PRA peer review process upon which the NRC relies to minimize the staff review of the plant specific PRA for each risk-informed submittal.

a) The submittal states that it used bounding CCDP and CLERP values for pipe breaks that result in a LOCA. What are the current CCDP and CLERP values for the different LOCA sizes in the current Grand Gulf PRA? Was one LOCA size selected for all LOCAs and, if so, why is one size sufficient?

Response

The GGNS PRA models a variety of LOCA sizes. LOCA CCDPs were re-calculated to support the previously completed RI-BER application. These values are provided below. As can be seen, the intermediate LOCA is the bounding event. Also, a CCDP/CLERP value of 0.1 was conservatively assigned to develop a corresponding/bounding CLERP. These values (CCDP = 5.4E-4 and CLERP = 5.4E-5) were used in the N-716 change-in-risk assessment for locations that would result in a LOCA.

Initiator	***Description***	***CCDP***
%A	*Large LOCA*	*5.19E-04*
%S1	*Intermediate LOCA*	*5.40E-04*
%S2	*Small LOCA*	*5.31E-06*

b) Please identify events modeled in the Grand Gulf PRA that are similar to the isolable LOCA and potential LOCA events quantified on page 11 of your submittal or further clarify why the Grand Gulf PRA can not be used to develop the required estimates. If applicable events in the PRA can be identified, please provide a description of these events and the bounding CCDP and CLERP values for these types of breaks derived from the PRA.

Response

The GGNS 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.

c) Please describe how the CCDP and CLERP values for "non reactor coolant pressure boundary pipe breaks that occur in standby system piping" were developed from the Grand Gulf flooding PRA. What is the relationship between this analysis, and the analysis used to implement the N-716 guideline that any segment with a CDF > 1E-6/year should be categorized high safety significant?

Response

Please see the responses to Questions 2(c) and 3(b), above.

d) In the "Break Location" column in Table 3.4-1 in your submittal, there are some entries labeled "Class 2". What characteristics results in a "Class 2" designation and how are the CCDPs and CLERPs of these welds developed?

Response

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). With respect to CCDPs/CLERPs, please see the response to Question 3(b), above.

e) How does GGNS evaluate interfacing system LOCAs as part of GG-ISI-002?

Response

The CCDP estimates for LOCA outside containment is based on the following:

- *For piping in the BER scope - a plant-specific BER evaluation is used to estimate CCDP.*
- *For Class 1 piping not in BER scope - the CCDP estimate is based on passive failure of 1st pressure boundary valve (~ 1E-3) and mitigation failure from GGNS ISLOCA analysis (~ 1E-3).*
- *For Class 2 piping not in BER scope - the 1E-4 CCDP used for LSS Class 2 piping bounds the probability of multiple valves failures that are required to initiate a possible LOCA outside containment.*

The risk impact assessment (CDF and LERF) for applicable piping meets risk acceptance criteria for the N716 application with significant margin.

5) The fourth bullet on page 11 of 28 in your submittal states that CCDP and CLERP values were determined based on the risk informed break exclusion region (RI-BER) evaluation performed for Grand Gulf. How many welds were being inspected in the RI-BER program and how many will be inspected in the proposed RIS_B program? Please summarize the reasons for any change in the number of welds to be inspected in the BER.

Response

Currently, there are 24 inspections included within the RI-BER program. This represents an inspection population that is 7% of the total BER population. This program was implemented via the GGNS 10 CFR 50.59 program. Per the requirements of N-716, a minimum of 10% of the BER population is to be inspected. For GGNS, this results in a total of 35 inspections. However, N-716 contains an additional requirement that pertains to the BER scope at GGNS. That is, the number of inspections is also weighted towards those locations that are potentially susceptible to degradation versus those locations that do not have a degradation mechanism identified. This requirement increases the number of inspections in the BER portion of this N-716 application to 45 inspections (please see Table 3.3 of GG-ISI-002 for a breakdown of these locations).

6) Note 2 in Table 5 of your submittal explains that the column "other" in the table was not filled in. Please update Table 5 by filling in the "other" column. Notes 3 and 4 will provide the needed differentiation between "other" inspections credited versus not credited in the RIS_B program.

Response

Please see attached the updated table.

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			Vol/Sur	Sur Only	RIS_B	Other[(2)]
RPV	✓		LOCA	TASCS, TT, (IGSCC)	Medium (Medium)	B-F	6	6	0	4[(3)]	4
RPV	✓		LOCA	TT, (IGSCC)	Medium (Medium)	B-F	1	1	0	0	–
						B-J	1	1	0	0	–
RPV	✓		LOCA	None (IGSCC)	Low (Medium)	B-F	20	20	0	0	–
						B-J	6	6	0	0	–
RPV	✓		LOCA	None	Low	B-F	1	1	0	0	–
						B-J	5	5	0	0	–
FW	✓		LOCA	TASCS, TT	Medium	B-J	60	18	0	9	–
FW	✓		ILOCA	TASCS, TT	Medium	B-J	10	8	2	7	–
FW	✓		BER	TASCS, TT	Medium	C-F-2	10	1	0	3	–
FW	✓		ILOCA	TASCS	Medium	B-J	4	0	4	4	–
FW	✓		LOCA	TT	Medium	B-J	3	2	0	3	–
FW	✓		ILOCA	None	Low	B-J	4	0	1	0	–
FW	✓		BER	None	Low	C-F-2	17	1	0	0	–
MS	✓		LOCA	None	Low	B-J	107	9	4	4	–
MS	✓		ILOCA	None	Low	B-J	64	8	34	0	–
MS	✓		PLOCA	None	Low	B-J	2	0	2	0	–
MS	✓		BER	None	Low	C-F-2	20	2	0	0	–
SD	✓		LOCA	None	Low	B-J	37	0	4	4	–

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			Vol/Sur	Sur Only	RIS_B	Other[2]
SD	✓		ILOCA	None	Low	B-J	4	0	0	0	–
SP	✓		LOCA	None	Low	B-J	5	0	0	1	–
RCR	✓		LOCA	None (IGSCC)	Low (Medium)	B-J	25	6	0	8[4]	8
RCR	✓		LOCA	None	Low	B-J	161	38	4	12	–
RCR	✓		PLOCA	None	Low	B-J	8	0	4	0	–
CRD		✓	Class 2	N/A	Assume Medium	C-F-2	63	5	0	0	–
SLC	✓		LOCA	None	Low	B-J	5	0	0	4	–
SLC	✓		PLOCA	None	Low	B-J	37	0	4	1	–
RHR	✓		BER	TT, CC	Medium	C-F-2	4	0	0	1	–
RHR	✓		BER	TT	Medium	C-F-2	13	4	0	4	–
RHR	✓		LOCA	None	Low	B-J	24	8	0	7	–
RHR	✓		PLOCA	None	Low	B-J	55	10	0	1	–
RHR	✓		BER	None	Low	C-F-2	18	3	0	0	–
RHR		✓	Class 2	N/A	Assume Medium	C-F-2	500	32	2	0	–
LPCS	✓		LOCA	None	Low	B-J	7	4	0	3	–
LPCS	✓		PLOCA	None	Low	B-J	25	4	0	1	–
LPCS		✓	Class 2	N/A	Assume Medium	C-F-2	64	5	0	0	–
HPCS	✓		LOCA	TT	Medium	B-J	4	3	0	2	–
HPCS	✓		LOCA	None	Low	B-J	8	3	1	2	–
HPCS	✓		PLOCA	None	Low	B-J	30	2	0	1	–

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			Vol/Sur	Sur Only	RIS_B	Other[2]
HPCS		✓	Class 2	N/A	Assume Medium	C-F-2	82	6	0	0	–
MSLC	✓		ILOCA	None	Low	B-J	31	0	1	4	–
FWLC	✓		PLOCA	None	Low	B-J	11	0	0	2	–
RCIC	✓		LOCA	None	Low	B-J	7	0	0	2	–
RCIC	✓		PLOCA	None	Low	B-J	5	0	0	1	–
RCIC	✓		BER	None	Low	C-F-2	12	5	0	0	–
RCIC		✓	Class 2	N/A	Assume Medium	C-F-2	107	4	0	0	–
CGC		✓	Class 2	N/A	Assume Medium	C-F-1	3	3	0	0	–
						C-F-2	5	0	0	0	–
RWCU	✓		LOCA	None	Low	B-J	65	11	1	10	–
RWCU	✓		ILOCA	None	Low	B-J	25	8	0	2	–
RWCU	✓		BER	None	Low	B-J	4	0	0	0	–
						C-F-2	22	2	0	0	–
						Class 3	11	0	0	2	–
						Other	1	0	0	0	–
RWCU	✓		Class 2	None	Low	B-J [5]	3	0	0	0	–
RWCU		✓	Class 2	N/A	Assume Medium	C-F-2	2	0	0	0	–

Notes for Table 5

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 N-716. N-716 allows the existing plant augmented inspection program for IGSCC (Categories B through G) to be credited toward the 10% requirement. GGNS selected a 10% sampling without relying on IGSCC Program locations beyond those selected for RIS_B purposes either due to the presence of other damage mechanisms, or where no other damage mechanism is present.
3. These four piping welds have been selected for examination per the plant augmented inspection program for IGSCC (Category C) and for RIS_B purposes due to the presence of other damage mechanisms.
4. These eight piping welds have been selected for examination per the plant augmented inspection program for IGSCC (Category B) and are being credited for RIS_B purposes.
5. Although this piping classifies as Class 2 piping, GGNS conservatively treats it (i.e., NDE) as examination category B-J for inspection purposes.

ENCLOSURE 2

CNRO-2007-00022

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION SET #2

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION SET #2 REGARDING REQUEST FOR ALTERNATIVE GG-ISI-002

1) Regulatory Guide (RG) 1.178 describes one acceptable process for developing a RI-ISI program. Please explain:

a) 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.

b) How the process used to assess piping failure potential for the plant-specific probabilistic risk assessment (PRA) of pressure boundary failures compares to the process outlines in Section 2.1.5 of RG 1.178;

Response

For application, failure potential is used in two ways:

1. *Confirm on a plant-specific basis that there is no other piping that should be considered as HSS per Section 2(a) of N-716. [Please see the response to Question 3(c) below and the response to Question 2(c) from the first set of RAIs.]*
2. *Once the HSS population has been determined for the plant, the failure potential evaluation is identical to that in EPRI TR-112657 as applied to a number of NRC-approved RI-ISI applications. That is, the degradation mechanisms assessed, the evaluation criteria (e.g., attributes such as operating temperatures, allowable delta Ts, susceptible materials, flow velocities, etc.), and the failure potential ranking are the same.*

c) How the quantitative results of the pipe failure frequency that resulted from the failure potential assessment compares to the weld failure frequencies proposed in Section 5(a) of N-716 that are eventually used in your change in risk estimates;

Response

Because the failure frequencies in Section 5(a) of N-716 are at the weld level, they are substantially smaller than what is used in conducting an internal flooding study in general, and the GGNS internal flooding study, in particular. Another reason the failure frequencies used in the GGNS internal flooding study are larger than the N-716 values is because the GGNS internal flooding study includes the impact of flood sources beyond piping (e.g., tanks, pumps, heat exchangers, etc.). For screening purposes, this is conservative from an internal flooding study perspective. It is also conservative from a N-716 perspective because some of these flooding sources and, therefore, their contribution to failure frequency (e.g., tanks) are not within the N-716 scope of application (i.e., piping).

d) How the consequence evaluation performed as part of the plant-specific PRA of pressure boundary failures compares with the process outlined under Section 2.1.6 of RG 1.178.

Response

The plant-specific PRA of pressure boundary failures is consistent with that discussed in Section 2.1.6 of RG 1.178 in that plant walkdowns were conducted to identify flood initiators and the locations of critical components. Additionally, for each flood zone and/or scenario, the impact of both direct and indirect effects was considered. Direct effects included loss of a train or system (e.g., loss or diversion of flow), an initiating event, or both. Indirect effects included spatial effects, such as spray, pipe whip, etc., as well as loss of inventory effects (e.g., loss of a common tank).

2) Please fully define the population of welds to which the 10% guideline is applied. Please explain the following:

a) 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 GGNS, this population includes welds that are both less than, equal to, and greater than 4 NPS. It also includes butt welds and sockets 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 GGNS 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].

b) 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 GGNS application.

c) What percentage of Class 1 butt welds (regardless of NPS) will be inspected in the proposed risk-informed program?

Response

Entergy has selected an 11.5% sample of all Class 1 butt welds regardless of NPS.

3) Under Section 3.4 on Page 10 of 28 of GG-ISI-002, your submittal states, "the risk of implementing this program is expected to remain neutral or decrease when compared to that estimated from current requirements." However, the total change in risk in the table on page 13 of 28 is positive for both CDF and LERF when credit is not taken for improved detection. Please explain why additional inspections were not provided to bring the estimated risk increase to a risk neutral or risk decrease as proposed under Section 3.4.

Response

The GGNS N-716 application will use inspection techniques that are expected to increase the inspection effectiveness as compared to current ASME Section XI requirements. Thus, as shown on page 13 of 28, the expected impact on risk is a risk reduction. The "w/o POD" case provided in the submittal is a sensitivity study and not the true representation of the expected impact on risk of the application. Even so, the "w/o POD" sensitivity study shows that even when not crediting the improved inspection effectiveness, only a small increase in risk would be witnessed.

4) 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

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 attached Table 4-1, column entitled "HSS"). The next step is to assure that 10% of Class 1 welds are selected (see attached Table 4-1, column entitled "Class 1"). 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 attached Table 4-1, 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 attached Table 4-1, 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 attached Table 4-1, column entitled "$RCPB^{OC}$"). Finally, inspections are chosen so that 10% of the break exclusion region (BER) populations are chosen (see attached Table 4-1, 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., GGNS witnessed slightly less than 11%).

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

TABLE 4-1 [(1)]

ID	System	Selections	HSS	Class 1	DMs	RCPB^IFIV	RCPB^OC	BER
B13	RPV	Required	4 of 40	4 of 40	2 of 8	n/a	n/a	n/a
		Actual	4 of 40	4 of 40	4 of 8	n/a	n/a	n/a
B21	FW, MS, SD and SP	Required	35 of 347	30 of 300	22 of 87	20	8 of 72	16 of 155
		Actual	35 of 347	32 of 300	26 of 87	21	10 of 72	20 of 155
B33	RCR	Required	20 of 194	20 of 194	n/a	14	n/a	n/a
		Actual	20 of 194	20 of 194	n/a	20	n/a	n/a
C41	SLC	Required	5 of 42	5 of 42	n/a	4	n/a	n/a
		Actual	5 of 42	5 of 42	n/a	4	n/a	n/a
E12	RHR	Required	12 of 114	8 of 79	5 of 17	6	1 of 4	4 of 35
		Actual	13 of 114	8 of 79	5 of 17	7	1 of 4	5 of 35
E21	LPCS	Required	4 of 32	4 of 32	n/a	3	1 of 2	n/a
		Actual	4 of 32	4 of 32	n/a	3	1 of 2	n/a
E22	HPCS	Required	5 of 42	5 of 42	1 of 4	4	1 of 7	n/a
		Actual	5 of 42	5 of 42	2 of 4	4	1 of 7	n/a
E32	MSLC	Required	4 of 31	4 of 31	n/a	n/a	4 of 31	4 of 31
		Actual	4 of 31	4 of 31	n/a	n/a	4 of 31	4 of 31
E38	FWLC	Required	2 of 11	2 of 11	n/a	n/a	2 of 11	n/a
		Actual	2 of 11	2 of 11	n/a	n/a	2 of 11	n/a
E51	RCIC	Required	3 of 24	2 of 12	n/a	2	1 of 2	2 of 18
		Actual	3 of 24	3 of 12	n/a	2	1 of 2	2 of 18
G33	RWCU	Required	14 of 131	10 of 97	n/a	7	1 of 9	9 of 89
		Actual	14 of 131	12 of 97	n/a	10	2 of 9	14 of 89

(1) For columns entitled "HSS", "Class 1", "DMs", "RCPM^OC" and "BER", the information provided is in the format of number of inspections per population of welds (e.g., a 10% requirement for a population of forty (40) welds would be "4 of 40"). For the column entitled RCPB^IFIV, this criterion is that 2/3 of the Class 1 inspections inside the first isolation valve. Thus, this column identifies, on a "per system" basis, how many inspections were required per this criterion (row entitled "Required") and how many were actually selected to meet this criterion (row entitled "Actual").

5) In Section 5 of the licensee's submittal, the licensee states that it will implement the RIS_B program during the plant's third period of the current (second) inspection interval by performing 29% of the inspection locations selected for examination per the RIS_B process since 71% of the piping weld examinations required by ASME Section XI have been completed.

a) Please discuss what B-F, B-J, C-F-1, and C-F-2 weld examinations have been completed during the third ISI period of the second interval.

Response

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

B-F = 6

B-J = 21

C-F-1 = 0 (GGNS only has 3 C-F-1 welds)

C-F-2 = 24

b) Table IWB-2412-1, allows credit for up to 67% of examinations completed by the end of seven (7) years (second period) in the inspection interval. Please confirm that the plant will perform a minimum of 33% of the RIS_B selected examinations during the third period of the current inspection interval.

Response

Based on the NRC SER dated 20 February 2007, approving Entergy's Request for Relief CEP-GGNS-003, Entergy has suspended piping examinations while developing a Risk Informed ISI program. Entergy has completed examinations in one outage of the third period as well as on-line examinations. To date Entergy has examined 71% of Class 1 and 2 piping welds. Therefore, it is Entergy's intentions to examine 29% of the RIS_B selected examinations during the remainder of the third period of the current (second) inspection interval. Furthermore, Entergy believes that only allowing credit for 67% of the examinations would not be in line with current NRC guidance as demonstrated in the NRC approval of Code Case N-598 in RG 1.147. This Code case allows up to 75% of examinations to be credited during the second inspection period.

c) Describe how the licensee will determine which examinations to perform during the remainder of the second 10-year ISI interval.

Response

Prior to developing the RIS_B Program, GGNS 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.*

6) 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.

Response

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. Socket welds will be inspected as discussed in the response to Question 2b, above.

7) Please describe how preservice examinations will be performed for repair/replacement activities.

Response

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.

8) On Page 10 of 28 the licensee discusses additional examinations. Please describe what will be used to perform the engineering evaluation to determine the cause of any unacceptable flaw or relevant condition. Recent industry practice has been to perform corrective actions (i.e., overlays, replacement, etc.) prior to a root cause being determined (e.g., use of a qualified procedure and personnel).

Response

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. If the flaw is found unacceptable for continued *operation, it will be repaired in accordance with IWB-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).*

a) In some cases no materials are removed for metallurgical analysis. Please discuss the process used for this engineering evaluation, how will it be documented, and will the NRC be involved in the process?

Response

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.

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 Owner submits NIS-1 and NIS-2 forms, which summarize the inspections and repairs performed during the outage.

b) Discuss what process will be used to perform fracture mechanics evaluations

Response

ASME Section XI, IWB-3600 contains the rules for flaw evaluation and fracture mechanics, which include a requirement to submit the results of the evaluation to the NRC.

c) Discuss under what conditions would there be no additional examinations. Discuss how the licensee will document their justification.

Response

If the flaw is original construction or otherwise acceptable, Code rules do not require any additional inspections. If the nature and type of the flaw is service-induced, then similar systems or trains will be examined. The documentation requirements will be documented in the corrective action program and a summary will be submitted in the NIS-1 package.

9) On page 10 of 28 of GG-ISI-002, the licensee provides guidance in Section 3.3.2, "Program Relief Requests." For program relief requests, the licensee refers to the process outlined for 10 CFR 50.55a that will be used. Please describe the process for assessing limited examination coverage. Discuss whether additional examinations will be performed and whether additional techniques will be used to improve examination coverage. Discuss how the effect on risk of the incomplete examination coverage will be assessed. In what time frame will relief requests be submitted?

Response

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.

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.

Response

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 GGNS 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.