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W3F1-2007-0061

November 30, 2007

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

SUBJECT: Supplement 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. NRC letter dated September 21, 2007, Regarding Authorization of Proposed Alternative GG-ISI-002.

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. In a telephone call held on November 29, 2007, Entergy discussed with the staff how the Waterford 3 ASME Code Case N-716 submittal addressed the NRC Safety Evaluation and NRC Requests for Additional Information (RAI) that were the basis for NRC approval of the Grand Gulf ASME Code Case N-716 Application (Reference 2). In order to facilitate NRC review of the Waterford 3 ASME Code Case N-716 application, the attached supplemental information is docketed to supplement the Waterford 3 request.

A047

URR

This letter contains no commitments. Should you have any questions regarding this submittal, please contact Ron Williams at (504) 739-6255.

Sincerely,

A handwritten signature in black ink, appearing to read "Ron Williams", with a long, sweeping horizontal stroke at the end.

RJM/RLW

Enclosure: 1. Supplement to Request for Alternative W3-ISI-005

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ENCLOSURE 1

W3F1-2007-0061

**SUPPLEMENT to REQUEST FOR ALTERNATIVE
W3-ISI-005**

Introduction

In preparing the Waterford-3 (W3) Request to Use ASME Code Case N-716 submittal and the template for the other Entergy plants, Entergy considered much of the information provided via the Request for Information (RAI) to support the Grand Gulf Nuclear Station (GGNS) submittal as general information about the code case and the methods of applying the code case to any facility. This information was not repeated for the Waterford 3 application because it was believed to be generic information that had been previously provided through the GGNS RAI response. Also, it was believed that some of the information in the GGNS RAI response was information only that did not appear to have been used as basis for NRC approval based on review of the GGNS Safety Evaluation. Therefore, this information was omitted from the W3 submittal to streamline the submittal. However, upon additional review in response to your concerns, we did identify a small amount of W3 specific information that should have been included in the original submittal to correspond with information provided for GGNS. This information is minimal, but believed to be useful to the Staff. Entergy accepts the Staff's desire to have all information docketed for each code case N-716 application to assist in the Staff's review.

We have prepared additional information to assist the Staff in quickly evaluating the W3 submittal. Attached are the 37 Grand Gulf RAI questions (GGNS is still in the question) that supported the GGNS submittal and NRC approval. For each question, we have left it specific to GGNS; however, we are providing a response applicable to W3. To assist the Staff's review, Entergy has identified the differences in two ways. First, information that was included in the original W3 submittal (W3-ISI-005) is underlined with its location in W3-ISI-005 in parentheses below the information. Second, the information not contained in W3-ISI-005, Reference 1, or that is different from what was provided for GGNS, is shaded

RESPONSES TO GGNS REQUEST FOR ADDITIONAL INFORMATION SET #1

- 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 break, 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 **W3** 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 susceptibility 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 **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.*

- 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 W3-ISI-005 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

Entergy agrees that most PRA applications with a CDF guideline include a LERF guideline, as well. Therefore, Entergy has added a LERF guideline of 1E-07/year to the requirements of Section 2(a)(5) of Code Case N-716. W3 has reviewed low safety significant (LSS) piping [e.g., non HSS Class 2, Class 3, and non-nuclear safety (NNS) piping] against the new LERF requirement. As a result of this review, Entergy has confirmed that, in addition to having a CDF contribution of less than 1E-06/year, this piping also has a LERF contribution of less than 1E-07/year.

(W3-ISI-005 Page 7 of 25 Paragraph (5) and page 18 of 25 Table 3.1)

- 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

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. Class 3 / NNS Piping

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).

- 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 as well as BER, Class 3, and NNS piping. The W3 internal flooding study was used to conduct this comparison. The W3 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 W3 internal flooding study included the following general tasks:

- *Preliminary flood scenario development*
- *Plant walkdown*
- *Flood scenario importance screening*
- *Refinement of analysis bases and assumptions*
- *Detailed quantification of important flood scenarios*

The internal flooding analysis team developed preliminary flood scenario tables for the plant. These tables were used as a basis for flood scenario definition and analysis in subsequent tasks and included the following information:

- *Component - location cross reference development*
- *Hazard source - location cross reference development*
- *Mitigating/Isolation features - location cross reference development*
- *Preliminary flood propagation path analysis*

The internal flooding analysis team refined the preliminary internal flood scenario tables based on the results of plant walkdowns and developed quantitative screening criteria for flood scenario importance to total core damage frequency. A quantitative screening or cutoff frequency of $1 \text{ E-}06$ (CDF) of each potential flood scenario was utilized. The team then proceeded to develop and record flood scenario initiation frequencies for each flood scenario documented in the internal flood scenario tables. The team then performed importance screening based on assuming a maximum impact on equipment in all zones in the propagation path.

In order to screen internal flood scenarios at W3, it was first necessary to estimate an annual frequency of flooding in each flood zone defined in the analysis. The internal flooding analysis team found no significant plant specific flooding data at W3. Therefore, industry data was used to develop flood zone flood initiation screening frequencies in this analysis. This was accomplished by taking the total internal flood frequency for a specific plant building and "spreading" it throughout the flood zones defined within that building. The following physical factors were used in developing a weighting factor to accomplish this frequency spread in the W3 internal flood analysis:

- room volume
- flood source density

Room volumes were taken from existing plant specific calculations or were calculated using architectural drawings and associated civil engineering information for W3 buildings. The floor area was estimated from this information (in ft^2) and then multiplied by the estimated room height (in ft) to calculate the room volume.

The flood source density rating was based on the physical density of piping, pipe joints, pipe flanges, tanks, or other liquid sources in the room being evaluated for internal flood impact. Flood source density was used as a parameter to evaluate the contribution to internal flood due to material failure. If liquid sources (piping, tanks, valves, flanges, etc.) occupied greater than one third of the total room volume, a flood source density rating of "high" was assigned. If liquid sources occupied more than one-tenth, but no more than one third of the total room volume, a flood source density of "medium" was assigned. If liquid sources existed in the room but occupied no more than one-tenth of the total room volume, then a flood source density rating of "low" was assigned. If no liquid sources existed in the room, then a density rating of "none" was assigned.

Other factors such as liquid source system volume, and the density of connection points such as valves, joints, and flanges pressure were also subjectively considered in the assignment of flood source density ratings. Reference 5.2 (see below) was the source for annual flood frequencies used in the W3 internal flooding analysis.

Based upon the above, the W3 IPE internal flood screening study successfully evaluated potential flood scenarios for all areas of the WSES-3 plant. Only one flood scenario was still greater than the screening frequency of 1.00E-06/year after a series of successively less conservative quantifications of flood scenarios. This remaining scenario involves a flood originating in the turbine building zone designated TGB. The area is located at elevation 46 feet, essentially plant grade. A detailed cutset review was performed for this scenario, resulting in an estimated core damage frequency of 1.12E-06/year. This value was identified as still containing conservatism related to flood initiator frequency and flood-induced failure assumptions and after removal of these conservatisms, falls below the criterion of Section 2(a)(5) of N-716.

Note: Reference 5.2 = "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., PLG-0624, May 1988.

- 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

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) was 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 11 & 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.

As part of the Revision 3 update of the PSA, most of the important observations resulting from the peer review were also addressed. Following Revision 3 of 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 early 2008. W3 will also attempt to close many of the 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.

Request for Alternative W3-ISI-005 is based on the W3 PSA Revision 3 model and the W3 LERF model. The base case 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.

(W3-ISI-005, page 2 & 3 of 25, Section 1.2)

- 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 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.

- 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 W3 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 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 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 W3 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 Large 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 = 4.2E-3 and CLERP = 4.2E-4) were used in the N-716 change-in-risk assessment for locations that would result in a LOCA.

Initiator	Description	CCDP
%A	Large LOCA	4.2E-03
%S1	Intermediate LOCA	3.5E-03
%S2	Small LOCA	4.8E-04

- 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 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.

- 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 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.

- 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 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.

- 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

Note 2 was changed in the W3 submittal to explain that 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.

(W3-ISI-005, Page 25 of 25, Table 5)

RESPONSES TO GGNS REQUEST FOR ADDITIONAL INFORMATION SET #2

- 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 susceptibility 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 1(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 W3 internal flooding study, in particular. Another reason the failure frequencies used in the W3 internal flooding study are larger than the N-716 values is because the W3 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 ≥ 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].

- 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., ≤ 2 NPS) susceptible to thermal fatigue shall be a volumetric exam of the piping base metal within $\frac{1}{2}$ 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.

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

Response

Entergy has selected a 10.2% sample of all Class 1 butt welds regardless of NPS.

(W3-ISI-005, Page 10 of 25, Table in paragraph (5))

- 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 Waterford total change in risk in the table on page 14 of 25 is negative for both CDF and LERF when credit is not taken for improved detection and therefore this question is not applicable. However, even if it were positive the N-716 application will use inspection techniques that are expected to increase the inspection effectiveness as compared to current ASME Section XI requirements. Thus, 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.

(W3-ISI-005, Page 14 of 25)

- 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 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^{IFV}"). 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 25, 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. At least 10% of the RCPB piping weld population must be selected for examination per system.
4. At 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 that lie 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.

- 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:

$$\boxed{B-F = 0}$$

$$\boxed{B-J = 0}$$

$$\boxed{C-F-1 = 18}$$

$$\boxed{C-F-2 = 3}$$

- 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

To date Entergy has examined 65% of Class 1 and 2 piping welds. Therefore, it is Entergy's intentions to examine 35% 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.

(W3-ISI-005, Page 16 of 25, Section 5)

- 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, 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.

- 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. No socket welds will be inspected.

- 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 WV3 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.

August 20, 2007 Telephone Call

1. Specify which method or technique was use to estimate piping failure frequency in the GGNS flooding analysis, which supports the request

Response

The failure frequencies used in the W3 flooding study were not based on W3 plant specific data as there had not been significant flooding experience at W3. As such, failure frequencies were obtained from PLG-0624 (see Reference list). This report provides flooding frequencies based on plant areas and are derived from industry experience with flooding events due to failures in piping, piping connections, tanks and other sources. This data reflects the various causes of components failures (e.g. degradation mechanism). These building level failure frequencies are then spread across W3 flood zones to provide scenario level flood frequencies. This spreading is accomplished by developing weighting factors based upon room volume and flood source density (i.e. physical density of piping, piping components, tanks and other flood sources).

(W3-ISI-005, Page 7 of 25, Section 3.2)