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W3F1-2008-0023

April 4, 2008

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

SUBJECT: Response to RAI Regarding PRA Flooding Analysis that Supports the Request for Alternative W3-ISI-005 Risk Based ISI program using Code Case N-716
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES
1. Entergy letter dated October 18, 2007, *Request for Alternative W3-ISI-005 Request to Use ASME Code Case N-716* (W3F1-2007-0046)
2. Entergy letter dated February 14, 2008, *Revision to Request for Alternative W3-ISI-005 Request to Use ASME Code Case N-716* (W3F1-2008-0013)
3. NRC letter to Entergy dated March 18, 2008, Request for Additional Information

Dear Sir or Madam:

Per Reference 1, Entergy requested NRC review and approval to implement a risk-informed Inservice Inspection (ISI) program based on ASME Code Case N-716 at Waterford 3 (W3). On February 14, 2008 Entergy provided the NRC a revised Request for Alternative W3-ISI-005 (Reference 2) that reflects the updated internal flooding portion of the W3 PRA, which conforms to ASME Standard RA-Sb-2005, and the incorporation of W3 specific responses to the GGNS RAI questions.

As part of the NRC Staff review of Reference 2, NRC Staff personnel conducted an audit of the updated internal flooding portion of the W3 PRA for conformance to the supporting requirements (SRs) identified in the ASME PRA standard RA-Sb-2005. A total of seven (7)

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NRR

RAI questions were identified by NRC Staff via letter dated March 18, 2008. Entergy's response to the RAI is included in Attachment 1.

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

Sincerely,

RJM/RLW

Attachment: Response to RAI Regarding PRA Internal Flooding Analysis that Supports the Request for Alternative W3-ISI-005 Risk Based ISI program using Code Case N-716

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Attachment 1

W3F1-2008-0023

Response to RAI Regarding PRA Internal Flooding Analysis that Supports the Request for Alternative W3-ISI-005 Risk Based ISI Program using Code Case N-716

Response to RAI Regarding PRA Flooding Analysis that Supports the Request for Alternative W3-ISI-005 Risk Based ISI Program using Code Case N-716

On March 6, 2008, Stephen Dinsmore from U.S. Nuclear Regulatory Commission (NRC) audited your recently completed flooding analysis. The flooding analysis was done to support the Waterford 3 February14, 2008, request to implement a risk-informed inservice inspection (RI-ISI) program.

The [NRC] audit compared your [W3] flooding analysis with the supporting requirements (SRs) identified in the American Society of Mechanical Engineers probabilistic risk assessment (ASME PRA) standard RA-Sb-2005. In general, consistency with the standard would satisfy the PRA quality requirements for supporting development of a RI-ISI program. However, since the PRA quality needs to be sufficient to support the proposed changes, if the appropriate capability category of the standard was not met, an evaluation was made as to whether the analysis was sufficient to support the proposed submittal. All SRs (except the documentation SRs) were evaluated during the audit. The NRC staff identified the following issues for resolution before it can reach a finding that the quality of the flooding analysis is sufficient to support the proposed application, i.e., that there is confidence that scenarios that exceed the quantitative guidelines are identified. For clarity, the issues are listed according to the SR requirements.

RAI 1 – *IF-C3 identifies the failure mechanisms that shall be evaluated to determine the susceptibility of each safety-related structure, system, and component (SSC) in a flood area to flood-induced failures. Capability category II identifies failure by submergence and spay as requiring detailed analysis. Capability category III includes jet impingement, pipe whip, and humidity, condensation, and temperature concerns. Waterford 3 reported that it relied on its high-energy line break (HELB) design analysis to conclude that jet impingement, pipe whip, humidity, condensation, and temperature concerns would not cause the failure of SSCs in a flood area. RI-ISI requires that all SSC failures induced by a pipe break be considered. Please demonstrate that all SCC failures that are induced by a pipe break are adequately addressed in your analysis.*

Entergy Response:

The internal flooding analysis has been revised to explicitly consider high-energy line breaks and their consequences although, in evaluating these consequences, we have relied extensively on earlier studies and, in particular, on those presented in the FSAR. There are four potential high-energy line breaks of concern at Waterford 3:

- Rupture of the steam generator blow down line in the wing area of the reactor auxiliary building, -4 ft elevation.
- Rupture of the letdown line on the -4 ft elevation of the reactor auxiliary building.
- Rupture of the main steam system.
- Rupture of the feed water system.

These scenarios are now all evaluated in the internal flood analysis:

- The rupture of the letdown line in the pipe chase and general area on the -4 ft elevation was addressed in the FSAR and the consequences evaluated. No pressure-related damage or pipe-whip damage within the chase was anticipated. Spray or jet impingement damage to valves is possible should the rupture occur in the valve gallery but this would not exacerbate the situation as none of the valves are safety-related. Two mitigating factors here are the low release rate (38 gpm) and the fact that any release would occur within a compartment and thus not cause sprays or jet impingement damage outside the compartment. Accordingly, this HELB would result in a %T1 event should the reactor be tripped.
- The rupture of the blowdown line in the wing area on the -4 ft elevation was addressed in the FSAR and the consequences evaluated. No environmental damage, pipe whip, jet impingement or spray damage to safety-related equipment was anticipated and, as the area is large, pressure build up would be small. That said, the continued release following the rupture of the blowdown system was addressed as this might result in submergence damage to safety-related equipment on the -35 ft elevation. This flooding scenario, comprising a %T1 event coupled with the loss of safety-related equipment, was therefore quantified.
- The ruptures of the main steam or the feed water systems do not have consequences beyond those already addressed in the internal events PRA: the mitigating factor here is that the main steam system and feedwater system lines are not routed through the reactor auxiliary building (RAB) but rather run above the roof. On the 46 ft elevation adjacent to the reactor building and outside the RAB structure, the MSIVs and other valves are located in a partially enclosed area, but no components except those related to the main steam and feedwater systems and a number of nitrogen accumulators are present in this structure. Accordingly, this event has been addressed adequately in the internal events risk model.

RAI 2 - *IF-C5 permits screening out of flood areas where flooding of the area does not cause an initiating event or a need for immediate plant shutdown and no mitigating equipment modeled in the PRA is failed. Waterford 3 reported that it screened out flood areas where flooding of the area does not cause an initiating event or did not require an immediate shutdown and only one train of mitigating equipment had failed. That is, if the flooding of an area failed only one train of mitigating equipment and also did not cause an initiating event or a need for immediate shutdown, the area was screened out and not further evaluated. As illustrated by the requirement across all capability categories for this SR in the Standard, even if the plant does not trip, a serious flooding event that fails important equipment is not acceptable. Please re-evaluate areas that were screened out because less than two trains of mitigating equipment had failed.*

Entergy Response:

The requested evaluations are now included in the internal flooding analysis. It is now assumed that, in addition to the occasions when the rupture and release demands a trip (e.g, a rupture of the component cooling water system), the reactor will be tripped whenever there is:

- a serious flooding event
  and
- either mitigating equipment/safety related train is lost or whenever a plant may need to be shut down quickly

RAI 3 - *IF-C6 permits screening out of flood areas based on, in part, the success of human actions to isolate and terminate the flood. Waterford 3 chose 20 or 30 minutes as the time when, if isolation was feasible, an operator would terminate the flood. Equipment damage was premised on the amount of water released during this time. This is a capability category I analysis. Crediting human actions is an integral but complex step in PRA. The endorsed RI-ISI methods require determination of the flood scenario with and without human intervention which corresponds to the capability category III, i.e. scenarios are not screened out based on human actions. Therefore a category III analysis would be acceptable and is preferable. Capability category II permits screening based on "high reliability" of the required actions. If capability category II is used, the human error probabilities should be quantified using methods consistent with the IF-E5 in the standard. Please re-evaluate the credit given to human actions to provide confidence that scenarios that might exceed the quantitative guideline are identified.*

Entergy Response:

Rather than simply assess the damage that would occur by a time at which operators might reasonably be expected to terminate a release, the analysis has been revised so that flooding scenarios are now allowed to persist until either the source of floodwater supplying them is exhausted, the basement area of the reactor auxiliary building is filled with water, or a considerable time (typically 24 hours) has elapsed. The flood damage that would occur at both the endpoint of the release and intermediate points is specified. In quantifying the contribution of the flooding scenario to the core damage frequency, most flood scenarios are now defined in terms of the greatest damage that would result should there be no early termination of the release. In a number of scenarios, however, human reliability analysis is used to quantify the likelihood that the release will be terminated after the release, but before a specific damage state is reached.

RAI 4 - *IF-C8 permits screening out flood sources based, in part, on human actions analogous to IF-C6 above which dealt with flood areas. The comment provided for IF-C6 also applies to IF-C8.*

Entergy Response:

We have revised the analysis and have not screened out any flood sources or truncated any consequences based on reliance on human actions.

RAI 5 - *IF-D5a addresses the development of flood initiating (pipe rupture) frequencies for use during the scenario development. Capability category I permits the use of generic operating experience. The combined category II/III requires collection and incorporation of plant specific experience. Waterford 3 reported that they collected plant specific pipe failure experience, but this emphasized actual, and unlikely, failures. Pipes tend to fail where degradation mechanisms exist and RI-ISI is premised on inspecting locations with the highest risk, driven mostly by failure frequency. The plant specific information collected and used should include experience related to degradation mechanisms that could indicate increased likelihood of pipe failure at particular locations. Waterford should re-assess its plant specific operating experience to identify experience related to degradation mechanism and incorporate any relevant experience in the development of pipe failure frequencies.*

Entergy Response:

To address the plant specific information on plant design, operating practices and conditions that may impact flood likelihood, the service history of the plant was reviewed. This review consisted of the following:

1. Performance of various queries of the plant corrective action program, which was started in February 1993, to identify plant conditions associated with piping/component ruptures, flooding events, and plant piping/component thermal stress and fatigue conditions;
2. Results of evaluations from NRC Bulletin 88-08 on thermal stresses in piping connected to the reactor coolant system (RCS);
3. Continuing evaluations being performed under the Materials Reliability Program (MRP-146) on management of thermal fatigue in stagnant non-isolable RCS branch lines;
4. Programs that monitor known degradation mechanisms in susceptible piping systems, such as the Microbiologically Influenced Corrosion (MIC) Program and the Flow Accelerated Corrosion (FAC) Program.

<u>Corrective Action Program Queries</u>

The various queries of the plant corrective action program, using keywords rupture, flood, thermal stress, and thermal fatigue, produced over 170 condition reports. These reports were further reviewed for specific details on the nature of the condition and whether the condition qualified as a condition that may impact flood likelihood. From the list of condition reports, Condition report CR-WF3-1995-1017 identified a Fire Protection (FP) system pipe rupture in the Shutdown Cooling Heat Exchanger B Room. The failed area of the pipe constituted a hairline crack of approximately 15 inches. This FP failure was reminiscent of a similar pipe failure in September 1992. This 1992 pipe failure (line 7FP2-125) was the subject of an INPO SOR 92-014. There were no other Condition reports that met the criteria for inclusion in the list of conditions that impact flood likelihood.

NRC Bulletin 88-08

The evaluations performed as a result of NRC Bulletin 88-08 determined that Pressurizer Spray and Auxiliary Spray piping met the criteria for unisolable piping that could be subjected to temperature distributions or stratification which could result in unacceptable thermal stresses. As a result, thirty-one (31) ultrasonic examinations were performed during the first period of the Inservice inspection on the Pressurizer Spray and Auxiliary Spray piping using special ultrasonic techniques designed for the detection of cracking in austenitic stainless steel piping. No pipe cracking has been identified to date. These weld inspections continue to be an augmented part of the ISI program.

MRP-146 Evaluations

EPRI Materials Reliability Program issued MRP-146 in June 2005 to present required screening, evaluation and inspection recommendations for assessing potential thermal fatigue cracking due to swirl penetration and /or valve in-leakage that may occur in normally stagnant non-isolable piping systems attached to pressurized water reactor main RCS piping. An assessment of the Waterford 3 RCS normally stagnant lines for the effects of thermal fatigue is documented in Condition Report LO-HQNLO-2007-00068 CA#3. The assessment identified three RCS crossover drain lines (1RC2-46RL1A, 47RL1B, and 48RL2A) that require monitoring, refined stress analysis, or repair/replacement/mitigation. Further actions associated with the inspections of these lines have been scheduled for prior to the end of refueling outage 16 (November 2009), as specified in Report LO-HQNLO-2007-00068 CA#6. However, these inspections are based on the need for supplemental guidance from EPRI on refined evaluation methods for the known conservatisms in the calculation methodology used. Further actions will be taken following any revision to the calculation.

Plant Degradation Mechanism Monitoring Programs

Microbiologically Influenced Corrosion Program

The plant has on-going piping and component monitoring programs such as MIC and FAC to detect and monitor various degradation mechanisms. The MIC program is a comprehensive condition monitoring process that uses various examination methods to detect the effects of MIC, corrosion caused or promoted by the presence of microorganisms, in instrument tubing, small and large-bore piping and components of susceptible plant systems. The program also includes chemical treatment of susceptible systems (except fire protection) with inhibitors and biocides to minimize corrosion, deposits and scale along with bacterial growth and biological fouling. The primary susceptible systems include Auxiliary Component Cooling Water, Component Cooling Water, and Fire Protection. Other susceptible systems also being monitored are Chilled Water, Emergency Diesel Generator Cooling water, Stator Cooling Water, and Turbine Building Cooling Water.

The MIC program has been successful in minimizing the growth of MIC within the system, thus minimizing the corrosion failure rates of piping and components. Periodic inspections, performed each refueling outage, during regular maintenance activities and as requested from Operation, Chemistry, System Engineering, and Maintenance, has provided a barrier to identify areas of degradation prior to significant leakage and catastrophic failure.

Flow Accelerated Corrosion (FAC) Program

The FAC Program is intended to prevent or mitigate the effects of material degradation that results in internal wall thinning of carbon steel piping and fittings under certain flow and chemistry conditions. The FAC program utilizes a computer model that uses plant operating data and non-destructive examination (NDE) ultrasonic measurements to trend (predict) pipe wall thinning in susceptible piping systems. The program includes methodology for selecting inspection locations, component gridding, inspection activities, requirements for evaluating inspection data, disposition of results, sample expansion criteria and repair/replace guidance. Examination methods consist of visual observation and nondestructive testing, such as radiography, ultrasonic, eddy current, liquid penetrant and magnetic particle.

This program provides an early warning of FAC-induced wall thinning that may cause a pipe to leak or rupture, potentially affecting plant safety through injury to plant personnel and/or damage to plant equipment.

Conclusion

Based on the plant specific operating history, the 1992 and 1995 failures of red brass piping within the fire protection system were used to Bayesian Update the generic flooding frequency from the EPRI report "Pipe Rupture Frequencies for Internal Flooding PRAs, Revision 1 (EPRI Report 1013141, Final Report, March 2006) for the applicable scenarios in the WF3 Internal Flooding Analysis. The remaining plant specific operating experience appears to illustrate that there are no other material degradation issues at Waterford 3. This operating experience provides confidence that W3 experience is at least as good as the generic plant database used to calculate the flood initiating event frequencies for the various internal flooding scenario groups.

RAI 6 - IF-E5a addresses human actions developed in the internal PRA that are carried over into the flooding analysis by using the internal events sequences to quantify the flooding sequences. Waterford 3 stated that it had not reviewed these human actions from the internal event PRA to ensure that they are still feasible given the flooding event that caused the flooding sequence. Please re-evaluate the credit given to these human actions to provide confidence that scenarios that might exceed the quantitative guideline are identified.

Entergy Response:

Event sequences have been reviewed to identify all operator actions that might be hampered by the consequences of internal flooding. If an operator action is on the same elevation as the flood, no credit is taken for that operator action in the quantification of risk.

RAI 7 - *IF-E7 requires a review of the large early release frequency (LERF) sequences used in the internal event PRA to ensure that they remain applicable when carried over into the analysis of the flooding sequences. Waterford stated that it did not perform a LERF analysis. The RI-ISI guidelines for identifying high-safety significant sequences used by Waterford require that LERF be considered and any piping that contributes to LERF sequences greater than 1E-7/year be placed in the high-safety significant category. Please re-evaluate your results to identify as high-safety significant any sequences that result in an estimated LERF greater than 1E-7/year.*

Entergy Response:

A LERF analysis has now been performed. Results have been re-evaluated and none of the internal flooding scenarios result in a LERF sequence greater than 1E-7/year; therefore, no piping needs to be added to the high-safety significant category as a result of LERF considerations.