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W3F1-2017-0030

April 21, 2017

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

SUBJECT: Responses to Request for Additional Information for the Environmental Review (SAMA Round 2) Regarding the License Renewal Application for Waterford Steam Electric Station, Unit 3 (Waterford 3)
Docket No. 50-382
License No. NPF-38

REFERENCES:

1. Entergy letter W3F1-2016-0012 "License Renewal Application, Waterford Steam Electric Station, Unit 3" dated March 23, 2016.
2. NRC letter to Entergy "Requests for Additional Information for the Environmental Review of Waterford Steam Electric Station, Unit 3" dated March 28, 2017.

Dear Sir or Madam:

By letter dated March 23, 2016, Entergy Operations, Inc. (Entergy) submitted a license renewal application (Reference 1).

In letter dated March 28, 2017 (Reference 2), the NRC staff made a Request for Additional Information (RAI) SAMA Round 2, needed to complete the environmental review. Enclosure 1 provides the responses to the SAMA Round 2 RAIs.

There are no new regulatory commitments contained in this submittal. If you require additional information, please contact the Regulatory Assurance Manager, John Jarrell, at 504-739-6685.

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 21, 2017.

Sincerely,

MRC/AJH

Enclosures: 1. SAMA Round 2 RAI Responses – Waterford 3 License Renewal Application

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Enclosure 1 to

W3F1-2017-0030

SAMA Round 2 RAI Responses

Waterford 3 License Renewal Application

RAI 1

Regulatory Basis for RAI 1

NEI 05-01 provides the following guidance on the Level 2 model information to be provided in the SAMA submittal:

> Provide a table or matrix describing the mapping of Level 1 accident sequences into Level 2 release categories and a description of the representative release sequences.

The ER did not provide a table or matrix describing the mapping of specific Level 1 accident sequences into the Level 2 release categories or a description of the representative release sequences. Entergy was therefore asked in RAI 2.e to provide a description of the sequences used to characterize the source terms for each of the significant release categories, the basis for this selection and its appropriateness for use in determining the benefit for the Phase II SAMAs evaluated. While the Entergy RAI response 2.e provided an expanded general discussion of the reviews made to select the representative sequence, no specific sequences were provided or a description of the representative release sequences provided.

Request

Provide a description of the specific Level 1 and Level 2 accident sequences used to characterize the significant release categories (H-E, H-I and M-I) and why the particular Level 1 and Level 2 accident sequences were chosen to be representative for those release categories used in determining the benefit of the Phase II SAMAs.

Waterford 3 Response

Release Category H-E

The H-E cutset results show a distribution of accident sequences as follows: transient (TQX_H) 81%, bypass (all induced SGTR) 11%, bypass (SGTR) 5.5%, reactor vessel rupture 1.7%, containment isolation failures < 1%; and ISLOCAs < 1%.

The MAAP scenario selected to represent the H-E release category is identified as TQX_H. Level 1 sequence TQX is a transient initiating event followed by successful reactor trip and RCS pressure control. The RCP seals develop a leak, due to loss of seal cooling, resulting in a small LOCA. HPSI is initially successful, but fails during recirculation after the RWSP inventory is exhausted. Level 2 sequence TQX_H is TQX with early failures of both containment sprays and containment fans. In TQX_H, containment failure occurs prior to core damage. Scenario TQX_H was selected to represent the H-E release category based on its dominant frequency within the release category as well as large release fractions of CsI. In addition, this accident sequence is unique in that several failure events occur in quick progression. High steaming rates and loss of containment safeguard systems lead to rapid containment over-pressurization occurring during the core damage phase but before reaching the core damage temperature of 2200 °F.

Other accident sequences included in the H-E release category included containment bypass, containment isolation failures, ISLOCA and reactor vessel ruptures.

As described in the response to RAI 2.d[1], these scenarios are classified as early releases because the initiating event failure leads to an immediate release pathway from the containment structure. In addition, these scenarios are classified as high severity because the containment release pathway precludes the mitigation or retention of fission products due to scrubbing, retention, or deposition mechanisms that occur within the containment structure. MAAP analyses were not performed for these H-E sequences, due to the high uncertainties associated with evaluation of these types of sequences in the MAAP code, so the actual CsI release fractions were not calculated. More rigorous assessment of these scenarios using MAAP would result in the distribution of these classes of accidents across release categories of lower severity including M-E, L-E and LL-E. Individually, these accident sequences (I-SGTR, SGTR, V, CI, and ISLOCA) contribute no more than 11% to the H-E release category and combined contribute < 20% to the H-E release category. It is judged that the existing conservatisms in the individual SAMA case analyses more than compensate for the potential for higher CsI release fractions from these sequences. For example, if the objective of the SAMA was to reduce the likelihood of a certain failure mode, the failure mode was completely removed from the model to estimate the benefit, even though the SAMA would not be expected to be 100% effective in eliminating the failure. In addition, as shown in the revised Table D.2-2 provided in the earlier RAI response, Phase II SAMAs 61 and 71, which were evaluated to reduce the risk from SGTR, are potentially cost-beneficial. However, the other SAMAs related to these accident sequences are far from being potentially cost-beneficial in the revised Table D.2-2. [Phase II SAMAs 54, 56, 57, 58, 59, 60 were evaluated to reduce the risk from I-SGTR and SGTR, and Phase II SAMA 55 was evaluated to reduce the risk from containment isolation failure]. Therefore, scenario TQX_H was selected as representative of the H-E release category.

A summary of the dominant H-E cutset results is provided below.

Dominant H-E Release Category Accident Sequences			
Sequence ID	Contribution	CsI %	Description
TQX_H	~81%	35%	Transient followed by successful reactor trip and RCS pressure control. The RCP seals develop a leak due to loss of seal cooling resulting in a small LOCA. HPSI is initially successful, but fails during recirculation after the RWSP inventory is exhausted. Containment fans and sprays fail early; containment fails due to over-pressurization during core uncovery, prior to core damage (>2200 °F).

[1] Entergy Letter, W3F1-2017-0001 (ADAMS Accession Number ML17038A436)

Dominant H-E Release Category Accident Sequences			
Sequence ID	Contribution	CsI %	Description
I-SGTR	~11%	not calculated	Pressure and thermally induced Steam Generator Tube Ruptures
SGTR	~5.5 %	not calculated	Steam Generator Tube Ruptures
			RB, <1% This sequence represents a SGTR followed by reactor trip and loss of RCS and core heat removal. This results in RCS pressurization above the HPSI shutoff head and loss of RCS inventory without adequate makeup from HPSI or the charging system. The result is early core damage at high pressure.
			RX, ~5% This sequence represents the case when a SGTR occurs followed by reactor trip and successful primary to secondary heat removal. However, since the RCS is not depressurized, it remains at high pressure and inventory is lost through the steam generator and out the MSSVs. RCS inventory control will be lost when the RWSP is depleted. This event results in late core damage at medium pressure.
			RU, <1% This sequence represents a SGTR followed by reactor trip with successful RCS heat removal and depressurization. However, RCS inventory control is lost due to the failure of the HPSI and charging system to make-up inventory lost out of the ruptured, unisolated steam generator. This results in early core damage at medium pressure.
V	~1.7 %	not calculated	Reactor Vessel Rupture
CI	<1 %	not calculated	Containment Isolation Failures
ISLOCA	<1%	not calculated	Isolating System LOCA

Release Category H-I

The H-I cutset results show a distribution of accident sequences as follows: station blackout (SBO_E) 66%, TB_H 28%, TQU_H 5%.

The MAAP scenario selected to represent the H-I release category is identified as SBO_E. Level 1 sequence SBO is a loss of offsite power followed by failure of both emergency diesel generators and failure of the turbine-driven EFW pump to start or run. This results in early core damage at high pressure. Level 2 sequence SBO_E is the Level 1 SBO sequence with the early loss of both containment sprays and containment fans due to loss of power. The SBO_E scenario was selected to represent the H-I release category based on its dominant frequency in the release category as well as the largest release fractions of CsI. This sequence results in containment failure prior to vessel breach.

Other accident sequences included in the H-I release category included TB_H and TQU_H.

Level 1 sequence TB is a transient initiating event followed by successful reactor trip, RCS pressure control, and RCS pressure boundary integrity. Decay heat is not removed from the steam generators by means of the MFW, EFW, or any backup systems. Sequences that result in lifting of the pressurizer SRV (RCS pressure control fails), but successful closure of the SRV are included in this sequence. Failure to recover primary to secondary heat removal results in RCS inventory loss through the pressurizer safety relief valves and early core damage at high pressure. Level 2 sequence TB_H is TB with early failures of both containment sprays and containment fans. This sequence results in a late containment failure (> 4 hours) after vessel breach.

Level 1 sequence TQU is a transient initiating event followed by successful reactor trip and RCS pressure control. However, the RCP seals develop a leak due to the loss of seal cooling resulting in a small LOCA. High pressure safety injection fails to inject sufficient water from the RWSP. This event leads to early core damage at medium pressure. Level 2 sequence TQU_H is TQU with early failures of both containment sprays and containment fans. This sequence results in late containment failure (> 4 hours) after vessel breach.

A summary of the dominant H-I cutset results is provided below.

Dominant H-I Release Category Accident Sequences			
Sequence ID	Contribution	CsI %	Description
SBO_E	~66 %	32%	Loss of offsite power followed by failure of both emergency diesel generators and failure of the turbine-driven EFW pump to start/run. Containment fans and sprays not available due to loss of power. Containment failure occurs prior to vessel breach.
TB_H	~28 %	25%	Transient with successful RCS pressure control and boundary integrity with loss of decay heat removal and failure to recover RCS inventory; early failure of containment fans and sprays; vessel breach with late (> 4 hours) containment failure.
TQU_H	~5 %	23%	Transient with seal LOCA and successful RCS pressure control but failure of inventory control; early failure of containment fans and sprays; vessel breach with late (> 4 hours) containment failure.

Release Category M-I

The M-I cutset results show a distribution of accident sequences as follows: SU_H 77%, TB_F 12%, TB_B 10%.

The MAAP scenario selected to represent the M-I release category is identified as TB_B. Level 1 sequence TB is a transient initiating event followed by successful reactor trip, RCS pressure control, and RCS pressure boundary integrity. Decay heat is not removed from the steam generators by means of the MFW, EFW, or any backup systems. Sequences that result in lifting of the pressurizer SRV (RCS pressure control fails), but successful closure of the SRV, are included in this sequence. Failure to recover primary to secondary heat removal results in RCS inventory loss through the pressurizer safety relief valves and early core damage at high pressure. Level 2 sequence TB_B is TB with both containment sprays and containment fans available for operation. In this sequence, the vessel remains intact with containment failure due to hydrogen burn occurring at about 14 hours. Scenario TB_B was selected to represent the M-I release category based on its applicability to the SAMAs being evaluated and earlier containment failure time.

Other accident sequences included in the M-I release category included small LOCA (SU_H) and TB_F.

Level 1 sequence SU is a small break LOCA with reactor trip, but failure of the HPSI to inject sufficient water from the RWSP. This event leads to early core damage at medium pressure. Level 2 sequence SU_H is SU with early failures of both containment sprays and containment fans. This sequence results in containment failure due to over-pressurization. SU_H is the dominant contributor to the M-I release category with a CsI fraction of 7.7%. As described in the prior response to RAI 4.b[2], SU_H was identified as an outlier based on the ratio of barium to iodide in comparison to the other accident scenarios. It is acceptable to exclude SU_H because the M-I release category is < 2% of the total level 2 release frequency and the CsI release fraction is similar to the selected scenario. Also, as shown in the revised Table D.2-2 provided in the earlier RAI response[2], Phase II SAMAs 13 and 18, which were evaluated to reduce the frequency of core melt from a small LOCA, are far from being potentially cost-beneficial.

As described above for TB_B, Level 1 sequence TB is a transient initiating event followed by successful reactor trip, RCS pressure control, and RCS pressure boundary integrity. Decay heat is not removed from the steam generators by means of the MFW, EFW, or any backup systems. Sequences that result in lifting of the pressurizer SRV (RCS pressure control fails), but successful closure of the SRV, are included in this sequence. Failure to recover primary to secondary heat removal results in RCS inventory loss through the pressurizer safety relief valves and early core damage at high pressure. Level 2 sequence TB_F is TB with containment sprays available, but containment fan coolers failed. The reactor vessel remains intact, but the containment fails late (19 hours) due to containment over-pressurization.

A summary of the dominant M-I cutset results is provided below.

Dominant M-I Release Category Accident Sequences			
Sequence ID	Contribution	CsI %	Description
TB_B	~10 %	6.3 %	Transient with successful RCS pressure control and boundary integrity with loss of decay heat removal and failure to recover RCS inventory. The vessel remains intact with containment failure occurring at 14 hours due to hydrogen burn. Both containment fans and sprays are available.
SU_H	~77 %	7.7%	Small LOCA with containment failure due to over-pressurization; failure of containment fans and sprays.

[2] Entergy Letter, W3F1-2017-0001 (ADAMS Accession Number ML17038A436)

Dominant M-I Release Category Accident Sequences			
Sequence ID	Contribution	CsI %	Description
TB_F	~12 %	7.9 %	Transient with successful RCS pressure control and boundary integrity with loss of decay heat removal and failure to recover RCS inventory. The vessel remains intact with containment failure due to over-pressurization occurring at 19 hours. Containment spray is available, but containment fans are failed.

RAI 2

Regulatory Basis for RAI 2

NEI 05-01 provides the following guidance for the identification of SAMAs:

> SAMAs may be hardware changes, procedure changes, or enhancements to programs, including training and surveillance programs. Hardware changes should not be limited to permanent changes involving addition of new, safety-grade equipment, but should also include lower cost alternatives, such as temporary connections using commercial grade equipment (e.g., portable generators and temporary cross-ties).

NEI 05-01 further states:

> If a SAMA was not evaluated for a dominant risk contributor, justify why SAMAs to further reduce the contributor would not be cost-beneficial.

The dominant contributor to internal flooding risk is water propagation in the electric board room and was evaluated in SAMA 68. In SAMA 68, Entergy proposed to install permanent flood doors to prevent water propagation to the electric board room. The WF3 plant specific cost estimate for the flood doors was determined to be approximately 1.3 million dollar. In comparison, the benefit for removing this internal flooding risk is approximately three hundred thousand dollars. Therefore, Entergy found SAMA 68 to not be cost-beneficial.

The staff requested in RAI 6.i for Entergy to consider a lower cost alternative than a permanent flood door, such as a flood barrier.

Request

Consider a lower cost alternative to the flood doors, such as a flood barrier, for SAMA 68 or explain why a lower cost alternative was not considered or necessary. Provide a cost-benefit comparison or justify why SAMAs to further reduce the water propagation in the electric board room would not be cost-beneficial.

Waterford 3 Response

Summary of Internal Flooding Analysis for the B Switchgear / Electrical Board Room [Flood Zone RAB21-212/225B]

An 8” fire protection line runs vertically through flood zone RAB21-212/225B. The guillotine rupture of this line will result in the release ~2700 gpm of water. It is assumed that 15 minutes will elapse before the release is terminated. In this time water will accumulate to a 7” depth in switchgear area “B” and the electrical penetration room. This will cause submergence damage to the reactor trip switchgear, multiplexer RA-2104, 4.6-kV switchgear 3B-3S, several B train 480-V switchgear and motor control centers, B train battery charger 3B2-S, and other B train electrical components. As a

result of submergence damage to multiplexer RA-2104, the reactor would trip. Simultaneously, damage to 4.6 kV switchgear 3B would result in the loss of the “B” trains of HPSI, LPSI, CS and EFW, the “B” EDG and the “B” component cooling and “B” ACCW pumps.

Floodwater released into this flood zone will tend to accumulate as there is no rapid or direct drainage to lower elevations and the only door leading outside the reactor auxiliary building, door D51 that opens into the cooling tower area, is tight. Floodwater will flow under or through doors into several rooms, most importantly into the motor-generator set room, switchgear room A/B, and switchgear room A.

Should the release following the guillotine rupture of the firewater line be terminated in 15 minutes, the level of floodwater will rise to only ~1” in switchgear room A/B and will not enter switchgear room “A” (a 4” curb lies inside the door). This is flood scenario RAB21-212-225B-15MIN.

Should the release last for more than 15 minutes, damage to the A/B switchgear might occur. This is flood scenario RAB21-212-225B-15-45MIN.

Should the release last for more than 45 minutes, damage to switchgear in switchgear room A might also occur. This is flood scenario RAB21-212-225B-45MIN.

If the release is still not terminated, once flood levels reach ~ 3’, doors will burst open and water will flood down to the -4’ and -35’ elevations of the reactor auxiliary building. Submergence damage at these lower levels will occur, resulting in submergence damage to the EFW pumps. Assuming that all water released descends to the -35’ elevation, sufficient water will be discharged in 80 minutes after a guillotine rupture to cause such damage. Should the flood persist for 2 to 3 hours, sufficient water would enter the safeguard rooms through the drains if the drain sump pumps failed to cause submergence damage to the HPSI, LPSI and containment spray pumps in these flood zones. This is flood scenario RAB21-212-225B-80MIN.

Flood Scenario	**CDF (/rx-yr)**
RAB21-212-225B-80MIN	7.19E-07
RAB21-212-225B-45MIN	3.47E-10
RAB21-212-225B-15MIN	8.40E-11
RAB21-212-225B-15-45MIN	2.76E-11
Total	7.19E-07

Analysis Case 42 for SAMA 68, "Water Tight Doors for the Largest Contributor to Internal Flooding"

This analysis case was used to evaluate the change in plant risk from installing flood doors to prevent water propagation in the electric board room. The electrical equipment rooms at WF3 do not have water tight flood doors. Specifically this SAMA will evaluate water tight doors for the largest contributor to internal flooding, which is flood zone RAB21-212/225B. This analysis case was used to model the benefit of phase II SAMA 68.

Since the internal event risk analysis does not include internal flooding, this internal flooding SAMA would not mitigate internal event risk. A bounding analysis was performed by assuming the SAMA would eliminate the contribution to internal flooding CDF from flood zone RAB21-212/225B. The SAMA case 42 benefit with uncertainty was estimated to be $332,233.

Analysis of Lower Cost Alternatives for SAMA 68

As stated above, *"Once flood levels reach ~ 3', doors will burst open and water will flood down to the -4' and -35' elevations of the reactor auxiliary building. Submergence damage at these lower levels will occur resulting in submergence damage to the EFW pumps. Assuming that all water released descends to the -35' elevation, sufficient water will be discharged in 80 minutes after a guillotine rupture to cause such damage."* This flood is designated RAB21-212-225B-80MIN and is the risk significant scenario. Since this flood scenario assumes that no actions are taken and the flood levels reach 3', the scenario can only be mitigated by a SAMA that would prevent the doors from opening when the flood reaches the 3' level, or by extensive room drain modifications that would prevent water accumulation. It would not be mitigated by lower cost alternatives like a curb or flood barrier. Thus, water tight doors were postulated to mitigate this flood.

Lower cost alternatives like a curb or flood barrier could, however, mitigate other flood scenarios in this room. As described above, the floods that last more than 15 minutes, but less than 80 minutes could damage the A/B switchgear and the A switchgear. Therefore, a lower cost alternative would eliminate the risk from floods RAB21-212-225B-15-45MIN (2.76E-11/rx-yr) and RAB21-212-225B-45MIN (3.47E-10/rx-yr), for a total of 3.75E-10/rx-yr. This is approximately 0.02% of the total internal flooding CDF (2.48E-06/rx-yr).

The benefit of eliminating this flood risk is calculated below, following the Case 42 method of calculating the benefit.

Given,
Maximum internal benefit is $2,355,400
Total internal flooding CDF = 2.48E-06/rx-yr
Internal events CDF = 1.05E-05/rx-yr

Maximum internal flooding benefit = Maximum internal benefit x Total internal flooding CDF/Internal events CDF

Maximum internal flooding benefit = $2,355,400 x (2.48E-06/1.05E-05) = $556,323

SAMA benefit = (averted internal flood CDF/Internal events CDF) x (Maximum internal flooding benefit) = (3.75E-10/2.48E-06) x $556,323

SAMA benefit = $84

Applying the uncertainty factor of 2.06,

SAMA benefit with uncertainty = $84 x 2.06 = $173

Since the minimum implementation estimate for a hardware modification is $100,000, the postulated lower cost alternatives would not be potentially cost-beneficial.

In response to RAI 6.h[3], Entergy indicated that even if one were to multiply the benefit produced by Cases 41 and 42 by three, SAMAs 67 and 68 would remain not cost-beneficial. Similarly, if one were to multiply the new benefit with uncertainty of $173 by three, the SAMA would remain not cost-beneficial.

[3] Entergy Letter, W3F1-2017-0001 (ADAMS Accession Number ML17038A436)

RAI 3

Regulatory Basis for RAI 3

NEI 05-01 provides the following regarding potentially cost beneficial SAMAs:

> This analysis may not estimate all of the benefits or all of the costs of a SAMA. For instance, it may not consider increases or decreases in maintenance or operation costs following SAMA implementation. Also, it may not consider the possible adverse consequences of procedure changes, such as additional personnel dose. Since the SAMA analysis is not a complete engineering project cost-benefit analysis, the SAMAs that are cost-beneficial after the Phase II analysis and sensitivity analyses are only potentially cost-beneficial.

Thus, in the ER, Entergy stated:

> Although the above SAMA candidates do not relate to adequately managing the effects of aging during the period of extended operation, they have been submitted for detailed engineering project cost-benefit analysis to further evaluate implementation of these potentially cost beneficial SAMAs.

In the response to several RAIs, Entergy revised the calculated benefits for several of the Phase II SAMAs, performed additional sensitivity analyses, and evaluated potentially lower cost alternatives. As a result of these analyses, 5 new potentially cost-beneficial SAMA candidates were identified.

RAI 3

Will the potential cost beneficial SAMAs added as a result of the RAIs dated 11/22/2016 and 03/28/2017 be submitted for detailed engineering project cost-benefit analysis to further evaluate implementation of these potentially cost beneficial SAMAs?

Waterford 3 Response

Engineering change requests have been initiated for detailed engineering project cost-benefit analysis in accordance with the engineering change process for the potentially cost-beneficial SAMA candidates identified as a result of RAIs.