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Subject:	SEVERE ACCIDENT MITIGATION ANALYSIS SITE AUDIT REGARDING WATERFORD STEAM ELECTRIC STATION, UNIT 3
Date:	Friday, October 21, 2016 1:52:00 PM
Attachments:	image001.png W 3 SAMA Audit Needs List.pdf W3 SAMA Audit Plan.pdf image003.png image004.emz image005.png

UNITED STATES

NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

Mr. Michael R. Chisum
Site Vice President
Entergy Operations, Inc.
Waterford 3

SUBJECT: SEVERE ACCIDENT MITIGATION ANALYSIS SITE AUDIT REGARDING WATERFORD STEAM ELECTRIC STATION, UNIT 3

Dear Mr. Chisum:

The U.S. Nuclear Regulatory Commission (NRC) is reviewing Entergy Operations, Inc. (Entergy) application for renewal of the operating license for Waterford Steam Electric Station, Unit 3 (WF3). As part of the environmental review, a site audit of the severe accident mitigation analysis (SAMA) evaluation will be conducted at WF3, by NRC staff, during the week of October 24, 2016. The SAMA audit will be conducted in accordance with the enclosed SAMA audit plan (enclosure 1).

To aid in developing the site specific supplemental environmental impact statement to NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," the NRC staff requests the information described in the enclosed SAMA audit needs list (enclosure 2) be made available, to the extent possible, during the site audit. The NRC staff transmitted this information to your staff (Leia Milster) via e-mail on September 9, 2016 and discussed in a conference call on September 22, 2016.

If you have any questions, please contact me by telephone at 301-415-8517 or by e-mail at Elaine.Keegan@nrc.gov.

Sincerely,

Elaine Keegan, Sr. Project Manager
Environmental Review and Project
Management Branch
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-382
Enclosures:
As Stated

cc: Listserv

ADAMS Accession no.: ML16294A511 ***via e-mail**

OFFICE	LA:DLR	PM:RERP:DLR	BC:RERP:DLR
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DATE	10/21/2016	10/20/2016	10/ /2016

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WATERFORD STEAM ELECTRIC STATION UNIT 3 (WF3)
LICENSE RENEWAL ENVIRONMENTAL SITE AUDIT NEEDS LIST
SEVERE ACCIDENT MITIGATION ALTERNATIVES

BACKGROUND

NRC staff will review the severe accident mitigation analysis (SAMA) evaluation and results as documented in Entergy's License Renewal Environmental Report (ER) and supporting documents. Staff will also review Level 1, 2 and 3 probabilistic safety analysis (PSA) results, as described in the ER, and supporting documents. The following are specific questions to be discussed and/or addressed at the audit. Note that additional questions may be developed and provided several days before the audit or during the audit.

DOCUMENTS REQUESTED TO BE AVAILABLE FOR REVIEW

1. The most recent PSA Peer Review and/or self-assessment reports for all hazards.

2. Documentation associated with more recent WF3 PSA reviews, particularly of the Level 2 and 3 PSAs.

3. Reference D.1-2, PSA-WF3-01-L2-01, “Waterford 3 Level 2 PRA”, Revision 0.

4. The comprehensive list of 201 candidate SAMAs considered cited in Section D.2.1 and the documentation of their disposition.

5. Documentation of the identification and disposition of SAMAs identified from the review of candidate SAMAs from five other PWR SAMA analyses.

6. Supporting documentation for the core inventory calculation.

7. Windows based Methods for Estimation of Leakages and Consequences of Releases (MELCOR) Accident Consequence Code System (WinMACCS) calculation documentation.

8. Reference D.1-33, W3-EP-14-00013, "Waterford 3 Level 3 PRA," Revision 0.

9. Reference D.1-34, WF3-EP-14-00012, Site-Specific MACCS2 Input Data for WF3, Revision 0.

INTERVIEWS

NRC staff needs to interview Entergy/contractor personnel knowledgeable of WF3 Level 1, 2 and 3 PSA development and results as well as SAMA identification and evaluation. Interviews will cover the attached questions and any other questions that may arise from the material reviewed during the audit.

SAMA Audit Questions:

1. Level 1 PRA

   a. ER Section D.1.4 indicates that there is approximately a factor of 3 increase in core damage frequency (CDF) and a factor of 3 decrease in large early release frequency (LERF) from PSA 2009 R4 to 2015 PSA R5 used for the SAMA analysis. Discuss the major reasons for these changes.

   b. ER Section D.1.4.5 indicates that the 2009 peer review concluded that approximately 9% of the applicable PRA standard's supporting requirements (SRs) were met at Capability Category I (CCI) while 10% of the SRs were rated as not met. Provide the Entergy resolution of the Facts and Observations leading to these conclusions and confirm that all applicable SRs are now met at Capability Category I (CCII) or higher or that there is no impact on the SAMA analysis.

   c. Provide the "freeze date" or the date which corresponds to the WF3 design and operation incorporated into the WF3 PSA used for the SAMA analysis. Identify any design or operational (including fuel cycle) changes that have or, are planned, since this freeze date that might impact the SAMA analysis.

   d. Confirm that no changes have been made to the WF3 model used in the SAMA analysis since the peer review that would constitute an upgrade as defined by ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2.

   e. The revised Attachment W to the WF3 National Fire Protection Association (NFPA) 805 License Amendment Request (LAR) gives the internal events CDF and LERF as 6.5E-06 per reactor-year (rx-year) and 8.7E-08 per rx-year respectively. These values are approximately 60% of the results given for the 2015 (R5) PSA used for the SAMA analysis (internal events CDF and LERF as 1.05E-05 per rx-year and 1.36E-07 per rx-year respectively). Identify which of these values best represents WF3 for license renewal purposes, discuss the reasons for these differences and the impact on the SAMA analysis.

   f. Briefly describe the process and procedures in place to assure the technical adequacy of changes made to the WF3 PSA since the 2009 peer review.

   g. ER Section D.1.1 (p. D-26) states that the CDF uncertainty factor of 1.99 is based on the ratio of the 95th percentile CDF to the mean CDF. Since the PSA results used in the cost-benefit analysis are based on point estimates, the uncertainty factor should be based on the ratio of the 95th percentile CDF to the point estimate CDF. Describe the uncertainty analysis and provide the 95th, mean and point estimate results from this analysis. Discuss the impact of the revised uncertainty factor on the results of the SAMA analysis.

2. Level 2 PRA

    a. The table in ER Section D.1.4 gives LERF for the 2015 (R5) PSA as 1.36E-06 per rx-year, while Section D.1.2.1 (p. D-27) and Table D.1-12 gives 1.88E-06 per rx -year. Explain the difference.

    b. ER Section D.1.4.4 indicates that a full level 2 model was created for the 2015 (R5) PSA based on the 2015 internal events model. Describe the full level 2 model in comparison with the prior LERF only model reviewed in the 2009 peer review, the changes made to it to obtain the 2015 (R5) level 2 model and the steps taken to insure the technical adequacy of the full Level 2 model.

    c. ER Sections D.1.2.1 - Containment Performance Analysis (p. D-26 , D-27) and D.1.2.2.6 - Mapping of Level 1 Results into the Various Release Categories (p. D-54, D-55, D-56) both provide discussions regarding the transfer of Level 1 core damage results to the Level 2 fission product release analyses. The ER states:

       "For the WF3 Level 2 analysis, no grouping into PDS was performed to group accident sequences with similar safety features and containment failure responses. A more rigorous approach was taken where each Level 2 accident sequence was assessed individually based on the accident-specific containment response."

       "The WF3 Level 2 accident sequences were named using the two or three letter identification for the CD sequences from the Level 1 core damage event trees (i.e., AX, MU, SB, TQX, TKQ, and RB) and combined with a one-letter code to represent core melt sequences (core damage with containment safeguard systems)."

       Provide additional information on this process including a listing of the Level 2 accident sequences evaluated, the Level 1 and Level 2 sequence naming nomenclature and the assignment of the Level 2 sequences to the Level 2 release categories.

    d. ER Section D.1.2.1 states that 4 containment event trees (CETs) were used to model the core melt progression and radioactive releases. Four trees, Trees B, D, F and H, representing four combinations of containment heat removal, are subsequently discussed. Confirm that these are the four CETs used and describe the use of the four trees considering that the two containment heat removal systems are explicitly represented by CET nodes.

    e. ER Section D.1.2.2.7 indicates that for: Containment Bypass Sequences, Containment Isolation Sequences, Reactor Vessel Rupture Events and Interfacing System Loss of Coolant Accident (LOCA) Events; there was no consideration of fission product (FP) scrubbing, retention, or deposition and all were assigned to the High-Early release category (RC). Clarify this statement since with no scrubbing, retention or deposition, 100 percent release of volatile FPs would be expected.

f. ER Section D.1.5.2.9 states:

> "The representative accident sequences selected for each release category represented both the dominant accident class based on the Level 2 results and the maximum release of fission products from the MAAP analyses."

Provide a further discussion of this process including a description of the Level 2 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. Note that using the dominate sequence in each RC to characterize the releases for that category may not necessarily lead to the correct benefit for the individual SAMA cost-benefit analyses.

g. The start of release times given in ER Table D.1-10 are not consistent with the RC definitions in Table D.1-8 for a number of release categories. For example: for RC H-E (start of release less than 4 hours after general emergency declaration), the time of the start of release (plume 1) is 13.4 hours while the time of declaration of a general emergency is 15 minutes; and for RC H-I (start of release is greater than 4 hours after general emergency declaration), the time of release is 2.0 hours. Provide a discussion of the reasons for these differences and the impact on the results of the SAMA analysis.

h. ER Section D.1.2.2.6 indicates that level 2 accident sequences were evaluated deterministically using the Modular Accident Analysis Program (MAAP) 4.0.6 code and a 36-hour accident time period, and that this time period was selected to ensure that sufficient time was allotted to allow for late failures and to capture the peak steady-state fission product (FP) release concentrations. Provide support that the 36 hour accident time period yields the peak FP release over the 48 hour time period beginning at the time of declaration of a general emergency. If the peak FP release does not occur using the 36 hour accident time period, discuss the impact on the SAMA analysis if the analysis is extended to 48 hours after the declaration of a general emergency.

i. ER Table D.1-9 states that the frequency of the "intact" RC is obtained from the difference between the base CDF and the total of the other release categories. Provide the results for the "intact" RC from the sum of the no containment failure containment event tree end states. Discuss the impact of cut set truncation on the CDF and RC frequencies and the validity of the approach taken to determining the RC frequencies.

3. External events

   a. In response to NRC requests following the accident at the Fukushima Daiichi Nuclear Power Plant, new seismic hazard curves have been developed for each nuclear power plant site. Based on this information, EPRI has produced updates to the GI-199 seismic CDFs (see NEI Letter March 12, 2014) (ADAMS Package ML14083A596).

      i. Provide the WF3 seismic CDF from this analysis and discuss the impact and appropriateness of using this seismic CDF on the WF3 SAMA analysis.

      ii. In response to NRC staff RAIs on the WF3 NFPA 805 transition LAR, Entergy provided an assessment of the seismic CDF that is different from that given in the ILRT interval extension LAR used in the SAMA analysis. Update this

NFPA 805 LAR value to be based on the new post Fukushima hazard curves and discuss the impact of using this seismic CDF on the WF3 SAMA analysis.

b. As stated above, the revised Attachment W to the WF3 NFPA 805 LAR gives the internal events CDF and LERF as 6.5E-06 per rx-year and 8.7E-08 per rx-year respectively. These values are approximately 60% of the results given for the 2015 (R5) PSA used for the SAMA analysis. If the 2015 (R5) value is the most appropriate for use in the license renewal application (LRA), provide an assessment of the impact of this more recent internal events model on the results of the fire PSA used in the SAMA and the resulting impact on the SAMA analysis.

c. ER Section D.1.3.4 indicates that internal flooding is not included in the 2015 internal events PSA used for the SAMA analysis. It is also stated that changes were made to internal flooding analysis that allowed the internal flooding analysis to satisfy the requirements in the ASME Standard and RG 1.200. Provide further information on this analysis including consistency with the system modeling in the 2015 (R5) PSA, the impact of any differences on the internal flood CDF and the SAMA analysis, and the process used to ensure the technical adequacy of the internal flooding analysis.

d. As discussed in the NRC staff's "Interim Staff Response to Reevaluated Flood Hazards" at Waterford dated April 12, 2016, there are a number of reevaluated flood hazards that exceed the current design-basis. Provide a discussion of the current status of the WF3 Mitigation Strategy Assessment (MSA) and integrated assessment or focused evaluation, and a discussion of the impact of flood hazards on the WF3 risk. Provide support for the ER's conclusion that flood hazards are negligible and need not be included in the external events multiplier.

4. Please provide the following information regarding the Level 3 PRA used in the SAMA analysis. The basis for this request is as follows: Applicants for license renewal are required by 10 CFR 51.53(c)(3)(ii)(L) to consider SAMAs, if not previously considered, in an environmental impact assessment, related supplement, or environmental assessment for the plant. As part of its review of the Waterford SAMA analyses, NRC staff evaluates the applicant's analysis of accident consequences in the Level 3 PRA. The requested information is needed in order for the NRC staff to reach a conclusion on the sufficiency of the applicant's Level 3 PRA model for supporting the SAMA evaluations.

a. ER Table D.1-10 includes a time to declaration of general emergency (GE) and a warning time that is said to include a 15 minute GE declaration. The GE declaration time is 15 minutes for all release categories while the warning time ranges from 15 minutes to 9 hours. Discuss the use of these times in the consequence analysis and how they were determined. The GE declaration time would be expected to be sequence specific and based on site procedures.

b. It is noted in ER Table D.1-12 that the population dose for the Low-Intermediate (L-I) RC is greater than that for the High-Early (H-E) while the cesium and iodine release fractions given in Table D.1-10 are about one 25th of those for H-E. Similarly, the population dose for the Moderate-Intermediate (M-I) RC is higher than that for the H-E RC, while the cesium and iodine release fractions are about one third and one eighth of those for the H-E. Explain the reason for this unexpected result and the impact on the SAMA cost-benefit analysis. Note that if the L-I result is incorrect, or conservatively high, then the benefit evaluation of SAMA 61 in Case 33 – Reduce Consequences of Steam Generator Tube Ruptures – would be non-conservative.

c. ER Table D.1-11 provides the estimated core inventory input to the Level 3 analysis; however, there is no description regarding how this input was developed. Clarify that the core inventory estimates applied in support of the Level 3 analysis are specific to WF3. Additionally, clarify whether additional adjustments of the core inventory values are necessary to account for differences between fuel cycles expected during the period of extended operation and the fuel cycle upon which the Level 3 analysis is based (e.g., to account for any changes in future fuel management practices or fuel design).

d. Regarding ER Section D.1.5.3, the NRC staff notes that the consequence analysis assumed site-specific meteorological data from year 2010, given that it generated the highest population dose and offsite economic cost. However, Section D.1.5.2.6 indicates that certain meteorological data, including that for year 2010, was not available and was addressed, at least in part, by using "data from approved data substitution methods as needed". Quantify the amount of missing meteorological data, which were estimated using data substitution, and clarify the methods used.

e. ER Section D.1.5.2.1 discusses population data. Explain why the population distribution used in the analysis is appropriate, and justify the method used for population extrapolation. In doing so, describe how those parishes with declining population projections were addressed (if applicable). Additionally, clarify whether transient and special facility populations were included, and if not, justify their exclusion.

f. ER Section D.1.5.2 describes the assumptions used for many of the parameters applied in support of the Level 3 analysis, but gaps exist in the information provided. Specifically, the guidance in Section 3.4.2 of NEI 05-01 identifies several economic parameters utilized in the WinMACCS model that are not discussed (e.g., cost of evacuation, cost of temporary relocation, cost of land decontamination, etc.). Describe how each of these cost parameters were developed, and provide the values and technical basis for any inflation/escalation factors utilized.

g. NUREG-1530, Revision 1, Reassessment of NRC's Dollar per Person-Rem Conversion Factor Policy (DRAFT) is publicly available in ADAMs at ML15049A114. It is anticipated that this NUREG will be completed by the end of this year and submitted to the Commission for approval. Waterford used the old value of $2,000 per person-rem in the current SAMA analysis. Since the approval of the NUREG is expected by the middle of 2017, this would be new information that would need to be evaluated before the Waterford License Renewal is issued. In anticipation of this change, please provide a sensitivity analysis using $5,200 per person-rem.

h. On May 4, 2016, the Commission issued a decision (CLI-16-07) in the Indian Point license renewal proceeding, in which it directed the Staff to supplement the Indian Point SAMA analysis with sensitivity analyses. Specifically, the Commission held that documentation was lacking for two inputs (TIMDEC and CDNFRM) used in the MACCS computer analyses, and that uncertainties in those input values could potentially affect the SAMA analysis cost-benefit conclusions. The Commission therefore directed the Staff to perform additional sensitivity analyses.

   The two inputs (TIMDEC and CDNFRM) are commonly used in the SAMA analyses performed for LRAs. These two input values were generally based on the values provided in NUREG 1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" and NUREG/CR 3673, "Economic Risks of Nuclear Power Reactor Accidents." The TIMDEC input value defines the time required for

completing decontamination to a specified degree. The CDNFRM input parameter defines the cost (on a per person basis) of decontaminating non-farmland by a specified decontamination factor. The CDNFRM values used in NUREG 1150 ($3,000/person for decontamination factor of 3 and $8,000/person for decontamination factor of 15) stem from decontamination cost estimates provided in NUREG/CR 3673, the same 1984 economic risk study referenced in support of the decontamination time inputs. These decontamination cost inputs are commonly escalated to account for inflation.

The NRC staff believes the Commission's decision in CLI-16-07 may be applicable to the SAMA analysis performed for WF3, inasmuch as that analysis may have also relied upon the NUREG 1150 values for TIMDEC and CDNFRM. We therefore request that Entergy either justify why CLI-16-07 does not apply to the SAMA analysis performed for WF3 or supplement the SAMA analysis with sensitivity analyses for the CDNFRM and TIMDEC values. Entergy is requested to review the input values specified in CLI-16-07 for the Indian Point LRA, and (1) to apply the maximum values specified by the Commission (one year (365 days) for TIMDEC and $100,000 for the CDNFRM values for the decontamination factor of 15) or, in the alternative, (2) to explain, with sufficient justification, its rationale for choosing any other value(s) for its sensitivity analyses. In any event, Entergy should execute sensitivity analyses for the release categories modeled that exceed 10^{15} Becquerels of Cs-137 released. Entergy is requested to evaluate how these sensitivity analyses may affect its identification of potentially cost-beneficial SAMAs. Finally, upon completing its sensitivity analysis, Entergy is requested to submit the spreadsheet (or equivalent table if another method is used) that conveys the population dose and off-site economic cost for each release category and integrates the results into a Population Dose Risk and an Offsite Economic Cost Risk for WF3.

5. SAMA identification and Screening

   a. Review of importance analysis in ER Tables D.1-2

      i. For events %T1 and %T3, reactor trip and turbine trip respectively, the identified SAMAs are improvements in emergency core cooling system (ECCS) and component cooling systems. Consider a reliability improvement program for normal secondary cooling.

      ii. The risk reduction worth (RRW) for event %TAC3 - Loss of 4.16Kv Bus 3A3-S (1.0914) is considerably less than that for %TAC4 - Loss of 4.16Kv Bus 3B3-S (1.318). Explain the reasons for this difference and consider a potential SAMA that addresses the cause of this difference.

      iii. Event EHFALNAB_P - Failure to energize bus 3AB3-S from bus opposite initial supply--recovery flag, is failure of a human action flag and is addressed by several hardware related SAMAs. Discuss the potential for SAMAs relating to improvements in procedures and training to reduce the impact of this human error and other human error events (e.g. Events HHFALNAB_P, HHFISOMINP, OHFRCPTRIP, SHFABCCWRP, OHFSGTRCDP, GHFFANM, and QHFCSPEMPP)

      iv. There are numerous events involving failure of motor driven exhaust fans (EMFEXFANAA, etc.). Discuss the potential for a SAMA involving adding a redundant (and perhaps diverse) exhaust fan. Note that while SAMA 35 - Provide a redundant train or means of ventilation, was determined to not be

cost-beneficial, the scope of this SAMA is not clear. Also, the reliability of cost-beneficial SAMA 36 - Implement procedures for temporary heating ventilation and air conditioning (HVAC) may be less than for redundant/diverse fans.

b. ER Section D.1.3.4 indicates that, while the internal flooding analysis is not integrated with the internal events analysis, changes were made to the internal flooding analysis that allowed the internal flooding analysis to satisfy the requirements in the ASME Standard and RG 1.200. Two SAMAs, SAMA 67 – "Improve internal flooding response procedures and training to improve the response to internal flooding events," and SAMA 68 - "Install flood doors to prevent water propagation in the electric board room" were included in the in the Phase II evaluation. Provide a discussion of the identification of additional candidate SAMAs for mitigating internal flooding risk based on review of important contributors to the internal flooding CDF consistent with the importance analysis review for internal events risk.

c. The ER indicates that the WF3 fire PRA was utilized to identify potential SAMAs. Three fire related SAMAs (74, 75 and 76) are included in the SAMA analysis based on their being commitments in the Waterford 3 NFPA 805 LAR. The WF3 fire PRA model gives a CDF for internal fires that is 1.7 times higher than the internal events CDF after crediting these commitments. Provide a discussion of the identification of other candidate SAMAs for mitigating internal fire risk based on review of important contributors to the internal fire CDF consistent with the importance analysis review of internal events risk.

d. The disposition of Individual Plant Examination (IPE) and Individual Plant Examination of External Events (IPEEE) insights is given in ER Table D.2-1.

   i. Phase I SAMA 184 - Install a portable generator to charge the AB battery is screened out as "already installed". The stated disposition indicates that the intent of this SAMA is met by the ability to manually control the turbine-driven emergency feedwater pump after loss of direct current (DC). Provide the importance of this human action and the potential for a SAMA involving the use of a portable generator.

   ii. Phase I SAMA 185 - Add guidance for aligning the low pressure safety injection (LPSI) pump for containment spray is screened out because it is "already installed". The procedure implemented is stated to address use of LPSI pumps for containment spray only for Large LOCAs. Discuss the benefit of this SAMA for other LOCAs or transients.

e. Identify the number of Phase I SAMA candidates identified from the various sources (i.e., NEI 05-01 Generic List, other industry documents of PWR SAMAs, the WF3 IPE and IPEEE, plant specific internal events importance analysis and other sources). If the total number of Phase 1 SAMA candidates is different than the 201 identified in Section D.2.1 of the ER, then provide an explanation for this difference.

f. Section D.1.2.1 states that Table D.1-5 provides the correlation between all level 2 release states RRW risk significant events down to 1.005 identified from the WF3 PRA Level 2 model and the SAMAs evaluated in Section D.2. Clarify specifically which release categories are included in the importance analysis: all release categories, all except the intact RC, or all except intact and high-early release categories?

6. Cost-benefit Analysis

   a. The benefit of SAMA 31 - Install a digital feedwater upgrade is addressed by Case 2 - Improve Feedwater Reliability. Case 2 was evaluated by eliminating the loss of feedwater initiating event. Discuss the added benefit that might occur if the upgrade would increase the availability of feedwater subsequent to other initiating events.

   b. The assumptions for Case 7 - Reduced Frequency of Loss of Auxiliary Component Cooling Water (ACCW) given in ER Table D.2-2 is the elimination of failure of ACCW. Section D.2.3 indicates that model was changed by adding the ability to cross-tie the ACCW. Provide further information on the modeling to clarify this apparent difference.

   c. SAMA 19 - Add redundant DC control power for SW pumps is evaluated in Case 12 by eliminating the DC control power gates to the ACCW pumps. While this SAMA is from the generic PWR list in NEI 05-01 and does not necessarily represent an important failure mode at WF3, discuss the benefit associated with eliminating DC control power failures for the component cooling water (CCW) pumps, in addition to the ACCW pumps.

   d. SAMA 33 - Add a motor-driven feedwater pump was evaluated by Case 17. WF3 has an existing motor driven auxiliary feedwater (AFW) pump. Discuss if this pump is credited in the WF3 PSA and the potential impact of the existence of the AFW on the cost of SAMA 33.

   e. SAMA 18 - Create a reactor coolant depressurization system was evaluated by Case 20 by eliminating small LOCA events. Discuss the added benefit that might occur for mitigating medium LOCAs, steam generator tube ruptures (SGTRs) and transients.

   f. Provide more details on the WF3 specific cost estimate for SAMA 35 - Provide a redundant train or means of ventilation. It is not clear if the scope of the cost estimate is consistent with the assumed elimination of failure of emergency diesel generator (EDG) room 3A cooling for Case 23 used to assess the benefit of SAMA 35.

   g. Clarify that the scope of SAMA 36, Implement procedures for temporary HVAC, is applicable to rooms other than EDG room 3A. Analysis of this SAMA only assumed elimination of failure of EDG room 3A cooling (Case 23). Based on the benefit results for Case 23, it appears likely that the implementation of temporary HVAC for other rooms may also be potentially cost-beneficial.

   h. Case 24 - Debris Coolability and Core Concrete Interaction was evaluated by eliminating failure of debris coolability and core concrete interaction to determine the benefit associated with the relatively low cost SAMAs; 38, 47, 72 and 73. These low cost SAMAs provide water to the cavity or otherwise improve core coolability or reduce core concrete interaction. Case 28 - Increase Cooling and Containment of Molten Core Debris was evaluated by eliminating containment core melt propagation and was used to determine the benefit associated with relatively high cost SAMAs 44, 45, and 46. The benefit associated with Case 28 is approximately $6.9M compared to that for Case 24 of $61K. It would appear that the SAMAs evaluated by Case 24 would achieve much of the benefit associated with SAMA 28. Discuss the reasons for this significant difference and the potential for SAMAs 38, 47, 72 and 73, or some combination of them, to be cost-beneficial.

i. Case 34 - Reduce ATWS Frequency was evaluated by eliminating ATWS contribution. The result is a 1.4% reduction in CDF while only a 0.3 percent and 0.2 percent reduction in person-rem risk and offsite economic cost risk (OECR), respectively. Based on other submittals, it is expected that the impact of anticipated transient without scram (ATWS) on risk would be similar to or greater than that on the CDF. Discuss the Level 2 modeling that leads to the WF3 result.

j. Case 39 - Eliminate Thermally Induced Tube Ruptures Following Core Damage was evaluated by eliminating thermal induced SGTR. This case was used to estimate the benefit for SAMA 54 - Modify procedures such that the water loop seals in the reactor cooling system (RCS) cold legs are not cleared following core damage. The reduction in person-rem risk and OECR was found to be 0.2 percent and 0.3 percent, respectively. Other assessments have found significantly larger risk reductions and were deemed cost-beneficial. Describe the modeling of the water loop seals in the WF3 Level 2 analysis that leads to this result.

k. Case 43 - Gagging Device to Close a Stuck Open Safety Valve is evaluated by eliminating failure events for stuck open relief valves and was used to estimate the benefit of SAMA 71 - Manufacture a Gagging Device for a Steam Generator Safety Valve and developing a procedure or work order for closing a stuck-open valve. Provide a further description for the failure events listed and their relevance to limiting release following a SGTR event.

l. Case 41 - Improve Internal Flooding Response Procedures and Training and Case 42 - Water Tight Doors for the Largest Contributor to Internal Flooding were evaluated by assuming that the reduction in risk was proportional to the reduction in internal flooding CDF. SAMAs evaluated by these cases were SAMA 67 - Improve internal flooding response procedures and training to improve the response to internal flooding events and SAMA 68 - Install flood doors to prevent water propagation in the electric board room. An examination of the reductions in risk given in ER Table D.2-2 for other cases indicates that this assumption may be non-conservative depending on the failures resulting from the specific flooding events mitigated. Describe the system failures involved in the internal flood events mitigated by these SAMA and select evaluation cases that would be more representative for these specific internal flooding SAMAs.

m. The cost for SAMA 68 - Install flood doors to prevent water propagation in the electric board room is given as $4,695,000 and stated to be from the Sequoyah cost estimate. The Sequoyah LRA ER indicates that this is the cost for both Sequoyah units. Further, the cost of such a modification would appear to be strongly dependent on a specific plant layout. Provide a cost that is valid for the WF3 plant configuration. Also discuss if something less than a full flood door, such as a flood barrier, might achieve the same risk reduction benefit.

n. The cost for SAMA 8 – Use fire water system as a backup source for diesel cooling, is given as $2,000,000 and stated to be from the Seabrook cost estimate. The Seabrook cost estimate is based on plant-specific changes necessary to use fire water as an EDG cooling source. A similar SAMA for the Grand Gulf plant (SAMA 10) is only $100,000. Provide a WF3-specific justification for the cost estimate for SAMA 8.

7. Potential Lower Cost or More Effective Alternative SAMAs

    a. SAMA 27 - Install an additional component cooling water pump is evaluated as a means to increase cooling water availability. Consider a potentially lower cost modification of replacing one of the pumps with a diverse design that would lower the common cause pump failure.

    b. Also regarding SAMA 27 – Install an additional component cooling water pump. Table D.1-2 indicates a portion of this benefit is due to eliminating the operator failure to align CCW train AB to replace lost Train A or B. Provide an assessment of a potentially lower cost SAMA candidate to provide diverse backup auto-start signals for the standby CCW trains on loss of the running train.

LICENSE RENEWAL SEVERE ACCIDENT MITIGATION ANALYSIS AUDIT PLAN
WATERFORD 3

1. Background

On March 30, 2016, the U.S. Nuclear Regulatory Commission (NRC) received the Entergy Operations, Inc. (Entergy) application for renewal of operating license for Waterford Steam Electric Station, Unit 3 (WF3). In support of the application and in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 51.53(c) and Part 54, Entergy also submitted an environmental report (ER) for WF3.

As part of the license renewal and environmental review, the NRC staff will conduct a severe accident mitigation analysis (SAMA) audit at WF3. This audit is conducted with the intent to gain understanding, to verify, and to identify information that will require docketing to support the basis of the licensing or regulatory decision. Specifically, the NRC staff will identify pertinent environmental data, and obtain clarifications regarding information provided in the applicant's ER. Per NRC guidance, the NRC staff prepares a regulatory audit plan that provides a clear overview of audit activities and scope, and team assignments.

2. Environmental Audit Bases

License renewal requirements are specified in 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." Licensees are required by 10 CFR 54.23 to submit an ER that complies with the requirements in 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," as part of the LRA. Review guidance for the NRC staff is provided in NUREG-1555, Revision 1 "Standard Review Plans for Environmental Reviews for Nuclear Power Plants: Supplement 1 – Operating License Renewal."

NRC staff is required to prepare a site-specific supplement to NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants." According to NRC regulation 10 CFR 51.70(b), "The NRC staff will independently evaluate and be responsible for the reliability of all information used in the draft environmental impact statement." In addition, NRC staff is required by 10 CFR 51.29(a) to define the proposed action, identify significant issues which must be studied in depth, and to identify those issues that can be eliminated from further study.

3. SAMA Audit Scope

The scope of this SAMA audit is to discuss NRC staff's specific questions regarding the SAMA evaluation and results documented in the ER. Audit team members will focus on reviewing the documents and requested information listed in the Waterford 3 SAMA Audit Needs List (Enclosure 2) and discussing the information with the applicant's subject matter experts. Additional questions may develop during the audit.

4. Information and Other Material Necessary for the SAMA Audit

As described in the SAMA Audit Needs List (Enclosure 2).

5. Tentative Team Assignments by Discipline

The SAMA audit team members and their specific discipline assignments are shown in the following table:

SAMA Review Team Members	
Discipline	**Team Members**
Environmental Project Manager	Elaine Keegan, NRC
SAMA	Jerry Dozier, NRC William (Bill) Ivans, PNNL E. R. (Bob) Schmidt, Contractor

6. Logistics

The SAMA audit will be conducted at WF 3 starting on Tuesday, October 25, 2016. An entrance meeting will not be held at the beginning of the audit. An exit meeting may be held at the end of the audit.

7. Special Requests

The NRC staff requests the applicant make available the information identified on the SAMA Audit Needs List. Plant staff who are subject matter experts in the disciplines listed on the SAMA Audit Needs List should be available for interviews.

8. Deliverables

A report should be issued by the NRC staff to the applicant within 90 days from the end of the SAMA audit.