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RBG-47803

December 5, 2017

Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: Response to License Renewal Application (LRA) NRC Request for Additional Information, Severe Accident Mitigation Alternatives
River Bend Station, Unit 1
Docket No. 50-458
License No. NPF-47

References: 1) Entergy Letter: License Renewal Application (RBG-47735 dated May 25, 2017)

2) NRC email: River Bend Station, Unit 1, Request for Additional Information, RBS Severe Accident Mitigation Alternatives - License Renewal Application – dated November 9, 2017. (ADAMS Accession No. ML17317A002).

Dear Sir or Madam:

In Reference 1, Entergy Operations, Inc (Entergy) submitted an application for renewal of the Operating License for River Bend Station (RBS) for an additional 20 years beyond the current expiration date. In an email dated November 9, 2017, (Reference 2) the NRC staff made a Request for Additional Information (RAI), needed to complete the License Renewal application review. Enclosure 1 provides the responses to the Severe Accident Mitigation Alternatives RAIs.

There are no regulatory commitments contained in this submittal. If you require additional information, please contact Mr. Tim Schenk at (225)-381-4177 or tschenk@entergy.com.

In accordance with 10 CFR 50.91(b)(1), Entergy is notifying the State of Louisiana and the State of Texas by transmitting a copy of this letter and attachment to the designated State Official.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 5, 2017.

Sincerely,

WFMaguire

WFM/RMC/alc

Enclosure 1: Severe Accident Mitigation Alternatives RAI Responses – River Bend Station

cc: (with Enclosure)

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RBF1-17-0147

Enclosure 1

Responses to Request for Additional Information

Severe Accident Mitigation Alternatives

SEVERE ACCIDENT MITIGATION ALTERNATIVES
LICENSE RENEWAL APPLICATION
RIVER BEND STATION, UNIT 1

Question 1:

Provide the following information regarding the Level 1 Probabilistic Risk Assessment (PRA) used for the River Bend Station (RBS) Severe Accident Mitigation Alternatives (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 RBS SAMA analysis, Nuclear Regulatory Commission (NRC) staff evaluates the applicant's treatment of internal events and calculation of core damage frequency in the Level 1 PRA model. The requested information is needed for the NRC staff to determine the sufficiency of the applicant's Level 1 PRA model for supporting the SAMA evaluation.

Question 1a:

Section D.1.1 states, "This model reflects the RBS as-built, as-operated configuration as of April 30, 2009 [D.1- 4]. No other planned major plant modifications, which could adversely impact the SAMA analysis results, have been identified." Clarify the intent of this statement relative to any changes made to plant operations, procedures and/or physical modifications in the eight years since the stated configuration date and any planned future changes and their potential impact on the SAMA analysis.

1.a Response:

SAMA analyses are based on PRA Revision 5A, which was an "Interim" PRA revision rather than a full periodic update. Data analysis is updated for full periodic revisions, thus Rev. 5A incorporates the plant specific data which was developed for PRA Revision 5. PRA Revision 6 was recently (Oct. 5, 2017) approved and is a full periodic update including an update to plant specific and industry generic data. Rev. 6 used updated plant specific data from 6/1/2009 to 5/31/2014; this reflects the latest data from when Rev. 6 work began in 2014. Significant plant changes since Rev. 5 was approved in 2011 are listed in the Rev. 6 Summary Report, and include:

- Incorporated Extended Loss of AC Power (ELAP) procedure (AOP-0065) and FLEX Support Guidelines (RBS-FSG-001 through -012) and FLEX equipment into the PSA model. Specifically, permanently installed plant equipment and strategies for extending DC power availability are credited in the baseline PSA model, as well as use of Upper Containment Pool suction for RCIC and alternate Containment Control (suppression pool cooling).
- Incorporated a model for containment venting based on deflating containment airlock seals based on EOP-0005 Enclosure 21 enhancements made in response to PRA Rev. 5 risk insights. This significantly reduced CDF contributions due to loss of all active containment decay heat removal.

These changes contributed to a reduction in the Full Power Internal Event CDF (without internal flooding) of 59%, from 2.60E-06/year for Rev. 5 to 1.07E-06/year for Rev. 6. Since the changes made to plant operations, procedures and physical modifications in the five years between the Rev. 5A configuration date and the Rev. 6 configuration date resulted in a CDF reduction, and did not increase the CDF of important sequences, they would not adversely affect the SAMA analysis.

The RBS design process includes screening modifications to identify PRA impacts. Procedure EN-DC-115, "Engineering Change Development," includes a required PSA impact screening in Attachment 9.3. Detailed criteria for determining if an impact screening is required are included in Attachment 9.4. Engineering Changes (EC) which could potentially impact the plant PRA are routed for review to the PSA group, which determines the actual impact on the PRA and determines whether the PRA model should be revised when the EC is implemented or as part of the next regular revision of the plant PRA.

Procedure EN-DC-151, "PSA Maintenance and Update," establishes the processes for maintaining Entergy's PRA models current with the as-built and as-operated plants. Specifically, this procedure governs the PRA Model Change Request (MCR) process which documents items to be assessed against the plant PRA to determine impacts, if any, to the model. Once initiated, MCRs are graded, including the determination of if the changes should be promptly incorporated in the PRA model, incorporated as part of the next regular model revision, or are considered enhancements or minor changes which can be deferred to the next periodic model update, or do not impact the PRA model.

The MCRs initiated in the three years since the Rev. 6 PRA configuration date do not include any existing or planned changes that would adversely affect the SAMA analysis.

In conclusion, if the plant changes in the time since the Rev. 5A configuration date and planned future changes were incorporated into the model used for the SAMA analyses, additional cost-beneficial SAMAs would not be identified.

Question 1.b:

Section D.1.4.8 indicates that plant specific data and initiating events frequencies through April and May 2009 were incorporated in the RBS Revision 5 (R5) PRA. Section D.1.4.9 did not cite any further data updates made to RBS R5A. Section D.1.4.11 states, "Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated regularly. EN-DC-151 suggests an update frequency of approximately every four years." Discuss the current status of the plant specific data and initiating event frequency updates and the potential impact on the SAMA analysis.

1.b Response:

SAMA analyses are based on PRA Revision 5A, which was an "Interim" PRA revision rather than a full periodic update. Data and Initiating Event frequencies are updated for full periodic revisions, thus Rev. 5A incorporates the data and Initiating Event frequencies from PRA Revision 5. PRA Revision 6 was recently (Oct .5, 2017) approved and is a full periodic update including updates to data and to plant specific (through May 2014) and industry generic initiator frequencies. This reflects the latest data from when Rev. 6 work began in 2014.

Use of the Rev. 5A model for SAMA analyses is expected to be conservative compared to the Rev. 6 model. The internal events only (without internal flooding) CDF was 2.79E-06/year for PRA Rev. 5A compared to 1.07E-06/year for Rev. 6. This reduction is attributed to the following changes in the PRA model for Rev. 6:

- Incorporated Extended Loss of AC Power (ELAP) procedure (AOP-0065) and FLEX Support Guidelines (RBS-FSG-001 through -012) and FLEX equipment into the PSA model. Specifically, permanently installed plant equipment and strategies for extending DC power availability are credited in the baseline PSA model, as well as use of Upper Containment Pool suction for RCIC and alternate Containment Control (suppression pool cooling).
- Incorporated a model for containment venting based on deflating containment airlock seals based on EOP-0005 Enclosure 21 enhancements made in response to PRA Rev. 5 risk insights. This significantly reduced CDF contributions due to loss of all active containment decay heat removal.
- Updated Control Building cooling logic reflecting new information developed on building heatup and equipment survivability which greatly decreased the risk contribution to CDF from the control building HVAC and control building chilled water systems.
- Improved RCIC heatup calculations performed in conjunction with the FLEX project which showed significantly less room heatup than previous calculations.
- Credited the Condensate Transfer systems as an injection source for the Reactor Pressure Vessel, per Emergency Operating Procedures (Enclosure 6).

Thus, because of the plant changes and PRA model improvements made in conjunction with PRA Rev. 6, it is conservative to use PRA Rev. 5A for SAMA analyses.

The CDF contribution from most of the important sequences decreased from Rev. 5A to Rev. 6. The CDF contribution from IE-T3B, Loss of the Feedwater / Condensate System Initiator, did increase from 6.8E-8/yr to 1.4E-7/yr. A change in initiator frequency from 4.79E-2/yr to 5.68E-2/yr influenced the CDF contribution increase. However, incorporating the revised IE-T3B frequency into the model used for the SAMA analyses would not result in additional cost-beneficial SAMAs.

<u>Question 1.c</u>:

Identify the RBS PRA revision that was reviewed in the 2011 peer review. If not RBS R5, provide additional information on the revision reviewed including core damage frequency (CDF), large early release frequency (LERF), and major changes to produce PRA RBS R5A.

<u>1.c Response</u>:

The 2011 BWROG PRA peer review was conducted against PRA Revision 5.

<u>Question 1.d</u>:

Confirm that no changes have been made to the RBS model used in the SAMA analysis since the peer review that would constitute an upgrade as defined by American society of mechanical engineers (ASME)/American nuclear society (ANS) RA-Sa-2009, as endorsed by RG 1.200, Revision 2.

<u>1.d Response</u>:

All changes made to the Rev. 5 PRA (peer reviewed revision) to create the Rev. 5A model (used for SAMA) were maintenance updates, not upgrades.

For the SAMA analysis, the full scope Level 2 model described in ER Section D.1.2 was created. The prior RBS LERF model (Rev. 5 and Rev. 5A) was a simplified NUREG/CR-6595 LERF model. Attachment I to PSA-RBS-01-L2-01 addresses compliance of the full scope Level 2 PRA with the ASME/ANS PRA Standard, ASME/ANS-RA-Sa-2009.

Changes to the LERF element of the Revision 6 PRA were based upon the Level 2 model developed for the SAMA analysis and were considered an upgrade. A BWROG focused scope peer review of the LERF and Internal Flooding elements of the Rev. 6 PRA was conducted the week of 11 September 2017. In this peer review, all LERF SRs were considered "met" although the following findings were noted.

1) SR LE-C6 back-references to SY SRs for modeling of the additional systems that may be added into the PSA for Level 2 accident progression modeling. SR SY-A4 CC-II requires plant walkdowns and interviews with plant personnel regarding the system modeling. The HCS and CIS system analyses (PSA-RBS-01-SY, Appendices 17 and 28, respectively) do not incorporate walkdowns or interviews with plant personnel as required by SR SY-A4.

This finding identifies a documentation issue, resolution of which would have no impact on the results of the SAMA analysis.

2) Section 4 of PSA-RBS-01-L2-01, Rev. 0, River Bend Level 2 PRA, states that accident sequences that are [depressurized] successfully in the Level 1 remain depressurized for the Level 2 analysis. This is reflected in the Level 2 PSA logic under the L2-LATE-INJ-FAIL gate where low pressure injection is allowed without questioning whether the SRVs remain functioning. The Level 2 analysis does not provide the justification for continued functioning of the SRVs during the extreme high temperatures during core damage severe accident progression.

Resolution of this finding would not impact the results of the SAMA analysis because the SRVs would remain functional under the temperature and pressure conditions they would experience during a severe accident. The RBS Level 2 PRA report has been updated to present the findings of a detailed analysis of the conditions the SRVs would experience during a severe accident. The analysis shows that temperature and pressure conditions (up to the time of vessel failure) would not cause failure of the SRVs.

3) The following QU SRs require additional effort to meet the requirements: QU-D1: A review of the significant accident sequences/cutsets was performed but there is no tabulation of the reviewed cutsets and specific insights (see Section 5.2.1 for non-significant cutsets). QU-E2: Section 6.7 of the Level 2 Report and Section 7.1 of LERF report provides sensitivities which address some key assumptions made in the development of the PRA model. It is not clear how these were selected or if the adequately address all relevant assumptions.

This finding identifies a documentation issue, resolution of which would have no impact on the results of the SAMA analysis.

Question 1.e:

Discuss the various systems available to RBS that were not part of the original design basis or have been added as part of various industry programs to address beyond-design-basis events and the extent they are credited in the SAMA PRA.

1.e Response:

The following systems/components were not included in the RBS original design.

- Station Blackout Diesel (BYS-EG1)—Included in the PRA model (hardware, maintenance and human actions)

- Hydrogen Igniter Generator (HCS-ENG1)—Included in the Level 2 PRA model (hardware and human actions)

- Alternate Power to the SRV's (Battery Cart SRV-1)—Included in the PRA model (human action)

- Portable Diesel Driven Pump (FPW-P4)—Not included in the Rev 5A model

- FLEX Equipment (includes portable diesels for battery charging, portable diesel driven pumps, RCIC suction path for Upper Containment Pool, compressed air/nitrogen bottles for SRVs)—Not included in the Rev 5A model

Question 2:

Provide the following information relative to the Level 2 PRA 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 RBS SAMA analysis, NRC staff evaluates the applicant's treatment of accident propagation and radionuclide release in the Level 2 PRA model. The requested information is needed in order for the NRC staff to determine the adequacy of the applicant's Level 2 PRA model for supporting the SAMA evaluation.

Question 2.a

Source term categories (STCs) 5 and 6 are for large ruptures of containment while STCs 7 through 14 are for penetration failures. Briefly discuss the analysis of containment integrity that lead to these STC assignments and how this was modeled in the Level 2 PRA.

2.a Response:

A detailed, plant-specific evaluation of the RBS containment internal pressure fragility was performed in 1992. Median and 95% non-exceedance pressures were calculated for all relevant containment penetrations and for the regions of the containment structure itself. The analysis determined that as containment pressurizes, leakage failures of containment penetrations (e.g., the containment hatch, containment dome ventilation opening and drywell equipment hatch/personnel door) occur at much lower pressures than the large containment rupture failure mechanisms. For sequences with a gradual pressurization of containment, the predicted leakage failures were modeled in the MAAP code to predict a realistic release profile for RBS. The large containment rupture pressure was also input into the MAAP model, but the leakage area, which increases as pressure rises (and is modeled in MAAP with some conservatism), tends to prevent containment from reaching the median pressure of large containment failure. The observation that the leakage failures are expected to prevent long term pressure-induced rupture of containment would apply using any range of results from the containment fragility analysis. The median distribution was utilized in the MAAP analyses for best estimate, but the other range presented (95% confidence) demonstrates the same behavior – that leakage failures occur at much lower pressures than the predicted rupture pressure.

STCs 5 and 6 represent large ruptures of containment due to rapid pressure loading caused by hydrogen explosions. In the containment fragility analysis, the failure area of the large containment rupture was not calculated, but the RBS Level 2 PRA assumes it to be 25 square feet.

STCs 7 through 14 involve sequences with gradual pressurization of containment, and the MAAP analyses predict releases from the leakage failures of containment. The pressures do not reach the level of containment rupture.

Question 2.b:

Section D.1.2.3.2 states that "...that the frequency for STC 1, Intact Containment, has been increased to account for the difference between the Level 1 CDF and the total calculated Level 2 frequency." The difference being due to the impact of truncation on the Level 2 cutsets. Provide support for the assignment of this difference to STC 1, which has a low risk compared to other STCs, and discuss the impact of assigning these unaccounted for sequences to other STCs.

2.b Response:

The difference between the internal events CDF (2.79E-6/yr) and the total frequency of the internal events Level 2 STCs (2.58E-6/yr) is 2.09E-7/yr, or 8.1% of the CDF. The approach of adding the difference to STC-1 was intended to capture the total CDF for benefit calculations. The maximum benefit and the benefit calculation for each SAMA are comprised of onsite and offsite costs. Adding the CDF to STC-1 ensures that all onsite costs are captured. However, the extra frequency could be assigned to other STCs as evaluated below.

Excluding the excess CDF of 2.09E-7/yr, STC-1 comprises 14% of the total CDF. Therefore, approximately 86% could be allocated to other STCs.

From Table D.1-31 the baseline MACR contribution from onsite risk plus replacement power is $82,049; the offsite contribution is $173,623. Therefore, the offsite contribution is approximately 68% of the total MACR.

Combining these factors, the frequency that could be assigned to other STCs would be only 8.1% * 86% * 68% = 4.7% of the total CDF.

Since the same process was utilized for the baseline MACR calculation and for each SAMA benefit calculation, the underestimate of the SAMA benefit would be significantly offset by the underestimate of the MACR. In addition, as described in Section D.2.3, conservative, bounding modeling changes were used to estimate the benefit of each SAMA. These conservatisms would compensate for a change of this small magnitude. Therefore, redistribution of the truncation difference proportionally to other STCs would not change the conclusions of the SAMA analysis.

Question 2.c:

The discussion of the representative Modular Accident Analysis Program (MAAP) cases for STCs 9 and 10 (containment failed prior to vessel breach, late molten core concrete interactions (MCCI)) cite MAAP cases S2A-6, T-TB-1, T-TB-2, T-14 and T1-4. Similarly, for STCs 11 through 14 (containment intact at vessel breach, with and without MCCI) MAAP cases T-TB-3, RCIC-Inj, T-TB-6, T-TB-9, and T-51 and T-51-CV are cited. Provide a description of each of these cases including how late molten core concrete interaction (MCCI) is modeled, the frequency of the sequences that they represent, and discuss if there are any other sequences that could be important for evaluating the benefit of potential SAMAs.

2.c Response:

A description of these cases, the time MCCI is predicted to begin, and their frequencies are presented in Table 2.c-1. The sequences in Table 2.c-1 represent approximately 90% of the total CDF. Therefore, there is high confidence that the dominant sequences have been evaluated and that other sequences would not be important for their potential to affect the SAMA benefit calculations.

MAAP Case T-14 was used to represent STCs 7 through 10. As seen in Table 2.c-1, case S2A-6 experiences MCCI earlier than case T-14. A question was raised if the use of T-14 as being representative of STCs 7, 8, 9 and 10 could yield non-conservative SAMA benefit results, given that MCCI occurs at a later time than case S2A-6. It is concluded that the use of the T-14 results provides a more conservative benefit calculation for the following reasons:

- In case S2A-6, a General Emergency declaration occurs 10.4 hours prior to vessel failure (into failed containment). In case T-14, a General Emergency declaration occurs 6.3 hours prior to vessel failure (into failed containment). The shorter time available in case T-14 yields less time for evacuation.

- The releases from S2A-6 are significantly less than those of T-14. For example, the CsI release fraction (prior to MCCI) from S2A-6 is 7.1E-4, compared to 5.2E-3 for case T-14. The CsI release fraction (end of run in which MCCI occurs) is 1.1E-1, compared to 1.4E-1 for case T-14.

Therefore, the T-14 releases yield a more conservative SAMA benefit calculation.

Question 2.d:

The discussion of the representative MAAP cases for STCs 7 through 10 indicates that the same MAAP case (T-14) was used for both no MCCI categories, as well as those with MCCI. Similarly, for STCs 11 through 14, the same MAAP case (T-TB-3) was used for both no MCCI categories, as well as those with MCCI. However, the release fractions for otherwise similar categories (STC 7 and 9, STC 8 and 10, STC 11 and 13; as well as STC 12 and 14) are different, even though from the same MAAP case. From the discussion in the ER, it appears that the results prior to MCCI were used for the no MCCI categories while the results at the end of the run were used for the MCCI categories. Please clarify this and justify the use of MAAP results prior to MCCI occurring for the no MCCI categories rather than the end of run result for a MAAP case without MCCI.

2.d Response:

The discussion in the question is correct. For the STCs without MCCI, the releases predicted by MAAP were used up to the time of MCCI. For the STCs with MCCI, the cases were continued.

The RBS Level 2 PRA utilizes probabilities of successful or unsuccessful debris cooling from NUREG/CR-4551. Different conditional probabilities are based on the spread of the debris, which depends on the reactor pressure at the time of vessel failure and the degree of debris entrainment. These conditional probabilities are not MAAP inputs, but are taken from NUREG/CR-4551.

The MAAP code contains input parameters and assumptions that make MCCI more or less likely to occur. For any given sequence with water covering the debris spread in containment, the MAAP parameters can be set to nearly guarantee that MCCI occurs, or nearly guarantee that debris is quenched. Due to MAAP settings in the RBS Level 2 MAAP model, MCCI occurred in nearly all cases. Therefore, to estimate the releases for cases that should not have MCCI (based on the NUREG/CR-4551 probabilities), the release fractions were taken up to the time of MCCI prediction.

In some of the cases evaluated, this time was long after containment failure. For example, the case used to represent STCs 7-10 did not predict MCCI until more than 30 hours after core damage and containment failure had both occurred. Therefore, the releases had stabilized, and continuing the run beyond 63.4 hours would have had a negligible impact on the releases.

In other cases, MCCI occurred before containment failure. For example, the case used to represent STCs 11-14 predicted MCCI at 17.6 hours, while containment failure did not occur until 22.9 hours. These cases would more appropriately be represented by MAAP analyses with MCCI suppressed, but with the case continued beyond containment failure. The selection of MAAP case T-TB-3 up to the time of the onset of MCCI has noticeably under-predicted the noble gas release fraction. The effect is due to the fact that the T-TB-3 case predicted MCCI early in the event, prior to containment failure. The release categorization should have been continued past containment failure in order to observe the release fraction plateaus, post-containment failure.

In order to evaluate the impact, a new MAAP analysis has been performed for T-TB-3, in which the MAAP inputs are identical except for modification of some internal parameters to suppress MCCI. The new analysis was run to 20 hours post-containment failure. The release fractions for this case are presented in Table 2.e-1 for STCs 11 and 12.

Most of the release fractions in Table 2.e-1 are approximately a factor of 5 higher than those previously calculated. However, other than the noble gas release fraction, the others are all of very low magnitude. In addition, STCs 11 and 12 each have a frequency of 2.99E-10/year. The combination of low frequency and low consequence yields the conclusion that these STCs have a negligible impact on the SAMA analysis.

Question 2.e:

Table D.1-7 gives a noble gas release fraction of 0.018 for STCs 11 and 12 (containment intact at vessel breach, no MCCI) as obtained from MAAP case T-TB-3. This result is much less than that for STC 1, intact containment. Discuss the modeling of no MCCI scenarios and explain this low noble gas release fraction result.

2.e Response:

In the RBS Level 2 PRA, ex-vessel debris quenching has the potential for success in sequences in which the debris remains covered with water. The probability of success depends on several factors, including the spreading of the debris (which is related to reactor pressure at vessel failure) and the fraction of the core debris that is retained in the reactor. Where debris quenching is successful and MCCI is prevented, subsequent releases from containment are not significant. Therefore, the RBS Level 2 evaluated the releases of these sequences up to the time of the start of MCCI.

The representative MAAP case for STCs 11 and 12 was selected by reviewing the MAAP results (up to the time of MCCI) for the dominant sequences that contributed to these STCs. This included review of MAAP cases T-TB-3, RCIC-Inj, T-TB-6, T-TB-9, T-51 and T-51-CV. Of these, T-TB-3 had the highest release fractions, and was therefore conservatively selected to be representative.

However, as noted in the question, the selection of T-TB-3 up to the time of the onset of MCCI has noticeably under-predicted the noble gas release fraction. The effect is due to the fact that the T-TB-3 case predicted MCCI early in the event, prior to containment failure. The release categorization should have been continued past containment failure in order to observe the release fraction plateaus, post-containment failure.

In order to evaluate the impact, a new MAAP analysis has been performed for T-TB-3, in which the MAAP inputs are identical except for modification of some internal parameters to suppress MCCI. The new analysis was run to 20 hours post-containment failure. The release fractions for this case are presented in Table 2.e-1 for STCs 11 and 12.

Most of the release fractions in Table 2.e-1 are approximately a factor of 5 higher than those previously calculated. However, other than the noble gas release fraction, the others are all of very low magnitude. In addition, STCs 11 and 12 each have a frequency of 2.99E-10/year. The combination of low frequency and low consequence yields the conclusion that these STCs have a negligible impact on the SAMA analysis.

Question 2.f:

Section D.1.2.3.1, with regard to the MAAP analysis of fission product release, states "In general, cases were run to a minimum of 140 hours to ensure that any late MCCI effects are understood. Provide the MAAP run times for each STC, as well as the time of declaration of a general emergency, the time of core damage, the time of containment failure and the time of the start of release. Also, clarify the meaning/definitions for the plume durations and plume delays given in Table D.1-21. If any of the run times are less than 48 hours after the time of declaration of general emergency justify the duration of the run time.

2.f Response:

The requested information about each STC MAAP evaluation is presented in Table 2.f-1.

In Table 2.f-1, only the run times for the T-14 and T-TB-3 runs before MCCI occurs are less than 48 hours after the time of declaration of general emergency. As described in the response to 2.d, ex-vessel debris quenching has the potential for success in sequences in which the debris remains covered with water. The probability of success depends on several factors, including the spreading of the debris (which is related to reactor pressure at vessel failure) and the fraction of the core debris that is retained in the reactor. Where debris quenching is successful and MCCI is prevented, subsequent releases from containment are not significant. Therefore, the Level 2 model evaluated the releases for STC-7, STC-8, STC-11, and STC-12 up to the time of the start of MCCI. Thus, for these STCs, the run time is the time of the start of MCCI.

To evaluate the impact of this assumption, a sensitivity analysis was performed. The representative MAAP case (T-14) for STC-7 and STC-8 was re-evaluated, with MAAP input parameters changed to suppress MCCI to at least 48 hours after the declaration of a General Emergency, extending the T-14 case to 77.2 hours. The STC-11 and STC-12 MAAP case (T-TB-3) was not included in this sensitivity analysis because T-TB-3 was extended to 20 hours post-containment failure with MCCI suppressed, as described in the response to 2.e, and the STC-11 and STC-12 frequencies are negligible (each is more than two orders of magnitude lower than STCs 7 and 8, and approximately 0.01% of the total RBS CDF).

The STC-7 and STC-8 sensitivity MAAP case for T-14 yielded higher releases than in the baseline analysis, but the releases were still low and classified as "Small" releases per the criterion defined in the RBS Level 2 PRA (Cesium, Iodine and Tellurium radionuclide releases are <2.5% of the core inventory). The sensitivity release fractions were input into the MACCS code to calculate the impact on the MACR. The result for a 7% discount rate was that the MACR increased from $255,672 to $255,946 (a 0.1% increase). This negligible increase demonstrates that the RBS SAMA analysis is not sensitive to the assumptions about calculation of releases for STCs with no MCCI.

The plume delay times given in Table D.1-21 are the time from the start of the event to the start of the plume release. The plume duration in Table D.1-21 is the duration of the plume release. During each plume release, the fraction of each isotope group in Table D.1-20 is modeled as a linear release during the duration of the plume. For example, for STC-1, two plumes are modeled. Plume 1 begins at 1800 sec and lasts for 28,800 sec. During this plume, a linear release of the isotope fractions in the top line of Table D.1-20 occurs. Plume 2 begins at 90,000 sec and lasts for 36,000 sec. During this plume, a linear release of the isotope fractions in the 2nd line of Table D.1-20 occurs.

Question 2.g:

Section D.1.4.10 states that the LERF model was peer reviewed in July 2011. Section D.1.2.1 states, with regard to the Level 2 PRA, that "It was prepared and reviewed by qualified personnel in accordance with existing industry standards" and further...a team of RBS experts representing various site organizations (e.g. Operations, System Engineering, Mechanical/Safety Analysis, PRA License Renewal) performed a review of the results to confirm that the model is representative of the plant and the results are reasonable." Discuss the results of any self-assessment of the LERF portion of the Level 2 PRA model against the LERF requirements of the ASME/ANS PRA standard.

<u>2.g Response</u>:

The 2011 LERF model was updated in 2016 to develop a full Level 2 PRA to support the license renewal SAMA analysis. The self-assessment of the 2016 Level 2 PRA found that all of the ASME/ANS PRA Standard LE supporting requirements were met to at least Category II except for the following:

- LE-C13 – Met to Category I because of conservative treatment of containment bypass. This is also conservative to the SAMA analysis, as it has the potential to overestimate the benefit of some SAMAs. Note this supporting requirement was considered Met by the subsequent Peer Review, with no F&O's assigned to this SR.

- LE-D5 – Not applicable to RBS

- LE-D6 – Not applicable to RBS

<u>Question 2.h:</u>

Section D.1.2.2 states that, "Each Level 1 CDF accident sequence was grouped into one of six groups for use in the Level 2 analysis." Discuss how Level 1 station blackout (SBO) sequences were treated in the Level 2 analysis.

<u>2.h Response</u>:

The Level 1 sequences that dominate the SBO contribution to CDF are T-TB-1, 2, 3, 5, 6 and 9. Together, these comprise over 97% of the SBO CDF. These sequences were evaluated with the RBS MAAP model. To ensure the accident progression was fully understood for RBS, the MAAP analyses included base case analyses of the SBO sequences, along with variations that included with and without depressurization, with and without successful hydrogen control, and with and without containment venting.

There is a single Containment Event Tree (CET) in the RBS Level 2 PRA; the CET is supported by two Decomposition Event Trees (DETs). The CET evaluates all the Level 1 core damage sequences (SBO and non-SBO). The Level 1 SBO sequences were grouped in a manner similar to the non-SBO sequences, considering factors such as status of injection, status of reactor pressure, and whether or not containment integrity is intact at the time of vessel failure. Evaluation of the accident progression of all sequences includes consideration of hydrogen effects, including the mass of hydrogen generated, status of the Hydrogen Control System (HCS), and the potential for hydrogen-induced containment failure.

As with the CET modeling, SBO and non-SBO sequences are both evaluated in the DET modeling. The DET "BYPOOL" evaluates the potential for pedestal failure to lead to a suppression pool bypass event. The DET "DEBCOOL" evaluates the coolability of ex-vessel core debris, after vessel failure.

Question 3:

Provide the following information with regard to the treatment and inclusion of external events 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 RBS SAMA analysis, NRC staff evaluates the applicant's treatment of external events in the PRA models. The requested information is needed in order for the NRC staff to determine the sufficiency of the applicant's PRA models for supporting the SAMA evaluation.

Question 3.a:

In the ER, Entergy reported a RBS seismic CDF (SCDF) of 2.5E-06/Rx-yr using more realistic plant specific fragility values instead of the more conservative values used by the NRC in the GI-199 safety/risk assessment. Provide more information on, and support for, these more realistic fragility values.

3.a Response:

In GSI-199, the NRC conservatively used the SSE PGA (Peak Ground Acceleration) as the HCLPF in calculating C_{50} and the spectral shape of the SSE to determine the spectral factors. This was necessary because RBS was a reduced scope plant and was not required to develop plant specific HCLPFs for the IPEEE. It is very conservative to use the SSE as the HCLPF. The NRC did this because it did not have the information needed to make a better, less conservative, estimate of the HCLPF and because it did not need a good estimate of the SCDF to evaluate whether the plant might be impacted by new seismic hazard curves (it was looking at the difference in SCDF between the new and old curves and only needed a conservative fragility). There are two general methods for calculating fragility, the hybrid method and the separation of variables method (full fragility analysis). River Bend used both methods to develop more realistic plant level fragilities (Engineering Report RBS-SA-11-00001). The separation of variables method resulted in the more conservative fragilities which were used to calculate seismic CDF for the plant using the methods NRC used in GSI-199. The separation of variables methodology estimates various factors of conservatism and uncertainty for the factors contributing to the fragility. The factors include strength, inelastic energy absorption, spectral shape, damping, wave incoherence, modeling, directional components and modal combination. Each of these factors was reviewed and conservative values were applied to them for the fragility calculation.

EPRI used the GSI-199 RBS fragility values for RBS in its seismic evaluation documented in "Fleet Seismic Core Damage Frequency Estimates for Central and Eastern U.S. Nuclear Power Plants Using New Site-Specific Seismic Hazard Estimates" (ML14083A586). The more realistic fragility analysis performed by RBS is deemed to be more appropriate for use in calculating the SCDF.

<u>Question 3.b</u>:

Section D.1.3.4 indicates that the internal flood analysis was revised and updated in 2012. Identify the internal events model used in this updated flood analysis, characterize it with respect to the internal events model used in the SAMA analysis (RBS 5A), and assess the impact of any difference between the two models.

<u>3.b Response</u>:

The 2012 internal flooding fault tree model is based on the RBS Revision 5 PRA model. The major differences between the Revision 5 model and the Revision 5A model are described in section D.1.4.9 of the ER SAMA attachment and Table D.1-12 provides the Internal Events CDF for the two revisions. The Revision 5A model CDF is ~7% higher that the Revision 5 CDF. The Revision 5A model had a significant decrease in transient CDF contribution due to improvements in modeling of the loss of service water initiators. There was also an increase in small break LOCA and stuck open relief valve CDF because of the application of more rigorous and higher non-recovery probability for long-term decay heat removal.

The differences between the Revision 5 and 5A models are not expected to have significant impact on the internal flooding overall CDF. The internal flooding initiators are mapped to transient initiators and associated logic in the Rev 5 fault tree in order to create flooding sequences. The transient accident logic in the Revision 5A model is essentially the same as the transient logic in Revision 5. Therefore, no significant differences between results of internal flooding PRAs based on Revision 5 and Revision 5A fault trees are expected. Any difference would be similar to that of the full power internal event Revision 5 and Revision 5A CDFs. Therefore, the internal flooding CDF used in the SAMA analysis is acceptable.

In addition, Revision 6 of the RBS PRA was approved on October 5, 2017, with an overall CDF of 3.03E-06/yr. The internal flooding contribution to this overall CDF is 1.96E-06/yr, which is approximately 40% of the Revision 5 internal flooding CDF. This result confirms that the internal flooding CDF used in the SAMA analysis is conservative.

<u>Question 3.c</u>:

Section D.1.4.10 indicates that the majority of the supporting requirements assessed as "not met" in the 2011 peer review were related to internal flooding. Further Section D.1.4.11 states "... following each periodic PRA model update, Entergy performs a self-assessment to assure that the PRA quality and expectations for all current applications are met." Discuss the results of the self-assessment of the 2012 internal flood model as well as the status of meeting those requirements "not met" in the 2011 peer review. For any internal flood requirements "not met" discuss the impact on the SAMA analysis.

<u>3.c Response</u>:

During the 2011 peer review, 18 of the 59 Findings were related to the Internal Flooding PRA element. 14 of these 18 Findings were considered closed by the update to the Internal Flooding analysis approved in 2012, leaving four Findings open to be resolved as part of PRA Revision 6. A self-assessment of the 2012 internal flood model was not documented since the R5A model which was an "Interim" PRA revision rather than a full periodic update. Table D.1-13 lists the remaining open peer review findings from the 2011 peer review, which were not closed in the internal events model used for the SAMA analysis. The "not met" supporting requirements are identified in this table. Resolution of these items is not expected to significantly change the total internal events CDF for RBS. Thus, the remaining open peer review findings have no impact on the conclusions of the SAMA analysis.

Due to changes in approach for the Internal Flooding analysis performed for Revision 6 and due to the number of original findings from the 2011 peer review, Entergy determined the Revision 6 Internal Flooding PRA should be considered an Upgrade. It was the subject of a focused scope BWROG PRA peer review conducted the week of 11 September 2017. PRA Revision 6 was approved 5 October 2017. The focused scope peer review report has not been issued. Preliminary results were one "Not Met" related to documentation of sources of model uncertainty, and eight Findings related to Internal Flooding. None of these are expected to appreciably impact the results of the Internal Flooding PRA. Since the Rev. 6 internal flooding model is a complete upgrade, findings from the focused-scope peer review are unrelated to the internal flooding model used in the SAMA analysis.

The CDF associated with Internal Flooding for Rev. 6 is 1.96E-06/year. This is 39% of the Rev. 5 Internal Flood CDF of 4.97E-06/year. Thus, use of the 2012 Internal Flooding analysis for SAMA analyses is conservative compared to use of the updated Rev. 6 analysis.

The dominant contributors to internal flood risk in the Rev. 6 model were examined for possible mitigation, but no additional SAMAs were postulated. None of the 348 individual internal flooding scenarios exceeded 1.0E-06/yr for CDF or 1.0E-08/yr for LERF.

<u>Question 3.d</u>:

Following the accident at the Fukushima Dai-ichi nuclear power plant, Entergy responded to an NRC 10 CFR 50.54(f) request for information. This response included a reevaluation of the external flood hazards, the development of mitigating strategies for external floods and a focused evaluation of the external flooding mechanisms for which the re-evaluated flooding hazards is not bounded by the current design basis. Entergy's evaluations concluded that permanent passive protection is in place for the Probable Maximum Flood (PMF) on West Creek and PMF on the Mississippi River and the Local Intense Precipitation (LIP) flood-causing mechanisms (Entergy, June 28, 2017, ADAMS Accession No. ML17207A105). This focused evaluation was a deterministic (that is, not a probabilistic) evaluation. Provide a discussion of these external flood hazards and the associated impact on RBS to support the conclusion that they would not contribute to the external events multiplier nor lead to any cost-beneficial SAMAs.

<u>3.d Response</u>:

The RBS external flooding focused evaluation (ML17207A105) demonstrated that there was adequate physical margin for the LIP hazard and the PMF hazards on West Creek and the Mississippi River. The evaluation did not utilize PRA methodologies and, therefore, no core damage frequencies were generated. The analyses used several conservative inputs, assumptions, and/or methods in the reevaluation of the hazards. The LIP conservative items included:

1. Small openings in each Vehicle Barrier System (VBS) block were conservatively assumed to be blocked (i.e., the VBS is impervious).

2. Conservative hydrometeorological report data, used to determine the greatest rainfall rates theoretically possible for the United States east of the 105th meridian, were used for the probable maximum precipitation (PMP) input. A site-specific study would have reduced LIP results substantially and likely below the protected elevation of 98.0 ft MSL.

3. The site drainage network was assumed to be non-functional. Culverts were considered to be blocked, and storm sewers were not considered. The culverts that convey flow below South Plant Road were considered non-functional.

The PMF conservative items included:

1. Culverts on West Creek under the access road were assumed to be completely blocked by debris. Bridges and culverts upstream of RBS were conservatively ignored.

2. A conservative antecedent rainfall condition curve, which describes runoff potential of the watershed, was used for the PMF simulation.

3. The Louisiana State Highway 10 Bridge over Grants Bayou was conservatively assumed to be 50% blocked by debris. All other bridges downstream of RBS on Grants Bayou were conservatively assumed to be completely blocked by debris. The Louisiana State Highway 10 Bridge is over 1,000 feet long and has an opening that is more than 50 feet high at the centerline of the channel. Significant debris blockage of the bridge is unlikely due to the large size of the bridge opening. Bridges and culverts upstream of RBS were conservatively ignored.

4. Conservative hydrometeorological report data, used to determine the greatest rainfall rates theoretically possible for the United States east of the 105th meridian, were used for the PMP input. A site-specific study would have reduced LIP results substantially and likely below grade elevation of 95 ft MSL.

Since permanent, passive protection is in place for these conservatively analyzed floods, the conclusion remains that external flood hazards do not contribute to the external events multiplier or result in cost-beneficial SAMAs.

<u>Question 4</u>:

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 RBS 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.

<u>Question 4.a:</u>

ER Table D.1-32 provides the results of several sensitivity analyses. The change in population dose risk is reported to be unchanged from the base case (1.21 person-rem/year) for all of the sensitivity analyses. No explanation is provided for this non-intuitive result, especially for those sensitivity cases that would have been expected to have some impact on the population dose risk (i.e., evacuation speed, fraction of public evacuating, and time to declaration of a general emergency). For each of the sensitivity cases, explain why there is no impact on the population dose risk. Also, provide the decontamination time (TIMDEC) value assumed in the baseline SAMA analysis.

4.a Response:

The population dose risk results are insensitive to the variables regarding evacuation speed, fraction of public evacuating, and time to declaration of general emergency due to the population distribution around RBS and the timing of the early declaration of the general emergency with respect to the releases as discussed below.

Of the estimated 2045 population around RBS as shown in Table D.1-14 there is an estimated total of 28,970 people within the 10 mile EPZ and 1,475,914 people within the 50 mile radius. Therefore, the fraction of individual evacuated is ~2% of the total population. The sensitivity cases with ± 5% of the evacuating fraction would change the results for approximately 1,450 people. Since a relatively small number of people are affected by evacuation, the sensitivity analyses show that the changes in evacuation speed and fraction of public evacuating do not noticeably change the population dose risk results.

Table D.1-21 presents the delay time from the start of the event and the duration of the various plumes modeled in the MACCS analysis. This table also presents the earliest time an alarm would be declared resulting in the evacuation of the 10 mile EPZ. For STCs 2, through 6, 13 and 14 the alarm occurs after the initial plume but prior to any subsequent plumes. For STCs 7 through 10 the alarm occurs prior to the initial plume. Only in STCs 11 and 12 is the evacuation started after all plume releases. Procedurally, the assessment, classification, and declaration of an emergency condition is expected to be completed within 15 minutes after the availability of indications (i.e. plant instrumentation, plant alarms, computer displays, or incoming verbal reports) to plant operators that an emergency action level (EAL) has been exceeded. The 15 minute criterion is not to be construed as a grace period to restore plant conditions to avoid declaring the event. Thus, the emergency declaration should be made promptly without waiting for the 15 minute period to elapse once the EAL is recognized as being exceeded. Nevertheless, the sensitivity on the time of declaration of a general emergency (alarm time) added a 15 minute (900 sec) delay to assess the impact of this potential delay. For each of the STCs, adding 900 sec to the alarm time did not change the relation of the alarm to the plume release times. For STCs 2, through 6, 13 and 14 the alarm still occurs after the initial plume but prior to any subsequent plumes. For STCs 7 through 10 the alarm still occurs prior to the initial plume. And for STCs 11 and 12, the alarm still occurs after all plume releases. In addition, as noted previously, evacuation only impacts ~2% of the total population. Thus, the sensitivity analysis shows that the potential 15 minute delay does not noticeably change the population dose risk results.

Small variations in the population dose are seen in the results for each individual sensitivity in the MACCS analysis, however these minor changes are insufficient to result in a change in the total population dose risk.

The baseline value of the variable TIMDEC for a 3% decontamination factor was 5.184E+6 seconds (60 days). The baseline value for a 15% decontamination factor was 1.0368E+7 seconds (120 days).

For the decontamination costs sensitivity case, TIMDEC for the 15% decontamination factor along with the non-farmland decontamination costs were adjusted to the maximum possible values of 3.15E+7 sec (~365 days) and $100,000/person, respectively. Both of these variables impact the offsite economic cost risk without any changes to the population dose risk.

Question 4.b:

Table D.1-16 provides the estimated core inventory input to the Level 3 analysis. Clarify whether 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).

4.b Response:

The core inventory source term used in the Level 3 analysis was extracted from the ORIGEN evaluation to support the current 24-month cycle operations at RBS with GNF-2 fuel. Currently, RBS is in the initial scoping efforts to incorporate a change to GNF-3 fuel for future operations, and the overall impact is too early to determine at this time. However, the preliminary GNF-3 equilibrium study identifies that the total core mass and end-of-cycle exposure may increase by ~2%. Individual isotope activities can be expected to have a similar order of magnitude increase. Changes of this magnitude are bounded by the SAMA 95th percentile uncertainty sensitivity case.

Question 4.c:

Section D.1.5.2.6 indicates that meteorological data for the year 2013, the most conservative data set for the years 2008 through 2014, was used in the consequence analysis. Discuss the basis for the conclusion that the year 2013 data is the most conservative, the extent to which there was missing data, and how missing data was accounted for in the SAMA analysis.

4.c Response:

The 14 release categories were run using all available meteorological data sets (2008, 2009, 2010min, 2010max, 2011min, 2011max, 2012min, 2012max, 2013min, 2013max, 2014min, 2014max) to determine which meteorological data set resulted in the highest population dose risk and offsite economic risk. The 2013 data using the minimum mixing height averages from previous years resulted in the highest population dose risk and offsite economic risk.

As described in ER SAMA Section D.1.5.2.6, the required data was obtained from the RBS meteorological monitoring system and from National Climatic Data Center (NCDC) data.

Temperatures at the 30ft and 150ft level, which are used to determine the stability factor, were obtained from the RBS meteorological monitoring system. Over the entire seven year period, 406 hours (0.7%) of temperature data were missing. The 2013 site meteorological dataset had 22 hours of missing data. The largest gap of sequential missing data was three hours. The hours of missing data were filled by interpolation.

Wind speed, wind direction, precipitation, and mixing height data were obtained from the NCDC. Over the entire seven year period, 2,588 hours of wind direction (4.2%), 268 hours of wind speed (0.4%), and 1,121 hours of precipitation (1.8%) were missing. The 2013 NCDC data contained 36 hours of missing wind data and 58 hours of missing precipitation data. The largest gap of sequential missing wind data was eight hours and the largest gap of sequential missing precipitation data was 23 hours. Missing wind data was filled using interpolation (see example below) except for 8 sequential missing hours which were filled with data from 2014 for the same hours. Missing precipitation data were filled with zero.

Wind data interpolation example:

Wind speed and direction data were missing for hour 17:00 on February 12, 2013. At 16:00 on 2/12/2013, the wind speed was 6.2 m/s with a direction of 80 deg from North and at 18:00, the wind speed was 5.1 m/s with a direction of 100 deg from North. The wind speed and direction at 17:00 were set halfway between these values; 5.65 m/s and 90 deg from North. Care was taken to ensure that interpolation between wind direction values on either side of 0 deg / 360 deg was performed correctly. For example, a missing wind direction for an hour after a 0 deg hour and before a 300 deg hour was set to 330 deg. (A lookup table was used to assign a downwind sector to each "degrees from" value for use in the MACCS2 model.)

The wind direction data from NCDC contained a direction value of "variable" in the dataset which is not compatible with the MACCS2 model. A variable wind direction occurs when the wind direction fluctuates and the wind speed is less than 6 knots (6.9 mph) or when the wind speed is greater than 6 knots and the wind direction fluctuates by more than 60 degree or more during a two minute period. Although a variable wind direction is not missing data, a data point for each hour must be provided for the MACCS2 model. In these cases interpolation (see example above) was used to provide a MACCS2 model wind direction. There were 402 hours of wind direction data in 2013 that had a variable wind direction.

Daily mixing height values (for morning and afternoon) for the vicinity of RBS for the period 2000 through 2009 were obtained from the NCDC. Mixing Height data for years beyond 2009 is no longer available. The morning values were calculated by NCDC using the lowest surface temperature for each day. The afternoon values were calculated in a similar way, except the maximum surface temperature was used. Using these NCDC data, seasonal mixing height averages were calculated for 2000 through 2009. For the years 2008 and 2009 the average seasonal values were used as calculated. Because data was not available for 2010 through 2014, the minimum and maximum average seasonal values for the years 2000 through 2009 were used.

<u>Question 5</u>:

Provide the following information with regard to the selection and screening of Phase I SAMA candidates. 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 RBS SAMA analysis, NRC staff evaluates the applicant's basis for the selection and screening Phase I SAMA candidates. The requested information is needed for the NRC staff to determine the adequacy of the applicant's Phase I SAMA selection and screening process for the SAMA evaluation.

Question 5.a:

The ER indicates that, based on the best available information, the CDF for each of the external events (seismic, internal fire and internal floods) are approximately equal to, or greater than, the internal events CDF. Discuss the steps taken to identify potential SAMAs that would mitigate the RBS specific risks due to these hazards.

5.a Response:

For seismic, internal fire, and internal flooding hazards, SAMAs were considered to mitigate vulnerabilities or dominant contributors.

Seismic –

Phase I SAMA 170, to increase seismic ruggedness of plant components, SAMA 171, to provide additional restraints for CO2 tanks to increase availability of fire protection given a seismic event, SAMA 172 to modify safety related condensate storage tanks to improve availability and reduce flooding potential following a seismic event, SAMA 173, to replace diesel generator anchor bolts, and SAMA 174, to reinforce block walls, were considered, but were either not applicable or were not judged to be necessary to mitigate seismic risk.

No additional seismic SAMAs were postulated since a seismic PRA does not exist, the IPEEE did not identify seismic vulnerabilities, and RBS is in a region of low seismicity. Also, the only recommendation from the updated fragility analyses was to re-examine one 480V motor control center and one containment outboard penetration termination cabinet for seismic interaction concerns, since the walkdown information was not well documented. Subsequently, seismic walkdowns were conducted to gather and report information from the plant related to degraded, non-conforming, or unanalyzed conditions for resolution of Fukushima Near-Term Task Force Recommendation 2.3. These walkdowns did not identify adverse seismic interaction concerns for 480V MCCs or for containment outboard penetration termination cabinets. Thus, Entergy believes this is a documentation issue with no impact on seismic risk and an associated SAMA is not postulated.

Internal Fire –

Phase I SAMA 175, to replace mercury switches in fire protection system, SAMA 176, to upgrade fire compartment barrier, SAMA 178, to enhance procedures to use alternate shutdown methods if the control room becomes uninhabitable, SAMA 179, to enhance fire brigade awareness, SAMA 180, to enhance control of combustibles and ignition sources, SAMA 182, to improve alternate shutdown training and equipment, SAMA 184 and SAMA 188, to proceduralize the use of a fire pumper truck to pressurize the fire service water system, SAMA 186, to add a bus cross-tie to reduce the impact of fires in a switchgear room, SAMA 187, to relocate relief valve cables, circuitry, and components, as well as other modifications, to ensure one train of core spray remains unaffected by fire, were considered, but were either not applicable or were not judged to be necessary to mitigate fire risk.

SAMA 177, to install additional transfer and isolation switches, SAMA 181, to improve alternate shutdown panel, and SAMA 185, to upgrade the ASDS panel to include additional system controls for opposite division, were combined into SAMA 185 and retained for evaluation.

SAMA 183, to add automatic fire suppression system was retained for evaluation.

The dominant contributors to fire risk were examined for possible mitigation, but no additional SAMAs were postulated.

Internal Flooding –

Phase I SAMA 165, to seal penetrations between turbine building basement and switchgear rooms, SAMA 166, to improve inspection of rubber expansion joints on main condenser, SAMA 167, to modify swing direction of doors separating turbine building basement from areas containing safeguards equipment, SAMA 168, to install an interlock to open the door to hot machine shop and change swing direction of door to plant administration building, were considered, but were not judged to be necessary to mitigate internal flood risk.

SAMA 169, to improve internal flooding procedures was retained for evaluation.

The dominant contributors to internal flood risk in the model used for the SAMA analysis were examined for possible mitigation, but no additional SAMAs were postulated.

External Flooding -

The response to question 3.d discusses evaluation of the dominant contributors to external 5lood risk in the Rev. 6 model for possible mitigation alternatives.

<u>Question 5.b:</u>

Address the following with respect to the review of the importance analysis in ER Table D.1-2

<u>Question 5.b.i</u>

Event E12-MDP-MA-C002A "[Residual heat removal] RHR pump A is unavailable due to maintenance" is addressed by a number of SAMAs that are either not applicable to this event (SAMAs 79 and 198) or involve costly new systems (SAMAs 110, 115 and 120). Consider other alternatives to mitigate this event such as eliminating or reducing on-line maintenance of the RHR pump.

<u>5.b.i Response</u>:

Entergy agrees that SAMAs 79 and 198 are not applicable to event E12-MDP-MA-C002A and that they should be removed from the disposition.

Reducing on-line maintenance would lower risk but there are substantial costs associated with moving the required maintenance to an outage, including extended outage time and cost of replacement power. Also, the risk reduction worth of event E12-MDP-MA-C002A in the recently completed Rev 6 PRA is significantly lower (1.0009) than its importance in the Rev 5A model (1.143). The difference in importance is primarily due to addition of a venting path (personnel air lock) for removal of decay heat from the containment in the R6 PRA model, as well as credit for FLEX equipment and procedures for Suppression Pool Cooling under ELAP conditions. Thus, any potential benefit associated with SAMAs for event E12-MDP-MA-C002A has been reduced by recent improvements in plant procedures in combination with FLEX strategies.

Question 5.b.ii:

Event FPW-XHE-LO-T2SBO "operator fails to follow attachment 2 for [station black out] (SBO)" is addressed by several hardware modifications. This event is given a failure probability of 0.5 and has a fairly high RRW of 1.117. Discuss the potential for a SAMA to improve the procedure or training. [Note: Phase I SAMA 72 to improve training on alternate injection via the fire water system was screened out as already installed. Also, it would appear that fire water injection into the reactor vessel would be important for increasing the likelihood of preventing molten core concrete interaction.]

5.b.ii Response:

The 0.5 probability for event FPW-XHE-LO-T2SBO noted in the question is a screening value that is applied during initial quantification. After quantification, recovery rules apply events that represent the failure probability from the human reliability analysis to individual HFEs and dependent HFEs in the cutsets. During this process, the screening HFEs are set to 1.0 and a new HFE representing the human failure is added to the cutset. For example, if FPW-XHE-LO-T2SBO is the only HFE in a cutset, it is set to 1.0 and event ZHE-FO-FPWSTARTSB (probability 0.1) is added to the cutset. If there are multiple screening HFEs in a cutset, all are set to 1.0 and a single dependent HFE is added to the cutset. Thus, the actual failure probability for this event, when alone in a cutset, is 0.1.

A new SAMA to improve the procedures and training on injection with the fire water system was performed. This SAMA assumes that the failure to align firewater for injection event (ZHE-FO-FPWSTARTSB) is reduced by 50%. The 50% is also applied to dependent HFEs that include event FPW-XHE-LO-T2SBO. The results are summarized below.

Internal & External Benefit	3% Discount Rate	95th Percentile Uncertainty
$73,317	$111,845	$293,268

The estimated implementation cost for a procedural change and training ranges from $50,000 to $200,000. Since this SAMA requires both procedure and training improvements, the cost would be at least $100,000. Given the above results, the SAMA is potentially cost-beneficial in the 3% discount rate and 95th percentile uncertainty sensitivity analyses.

Although the above SAMA candidate does not relate to adequately managing the effects of aging during the period of extended operation, it has been entered into the action tracking process to be evaluated for implementation.

Question 5.b.iii:

For Event IE-T3C "Initiator, Inadvertent opening of SRV," is addressed by SAMA 108 - Improve [safety relief valve] SRV and [main steam isolation valve] MSIV pneumatic components and SAMA 160 - Increase SRV reseat reliability. It is not clear that either of these SAMAs address this event. Discuss the potential for other SAMAs that address or mitigate this event.

5.b.iii Response:

SAMA 160 is applicable to IE-T3C. The initiating event, although named "Inadvertent opening of SRV," also includes the fact that the SRV subsequently sticks open (does not reseat). This SAMA was addressed by setting events associated with failure of the SRVs to close (i.e. IE-T3C, P1 and P2) to false in the model.

The disposition of initiator IE-T3C incorrectly referenced SAMA 108. SAMA 108 is to improve SRV and MSIV pneumatic components. Since the RBS SRVs utilize pneumatic components only to open the SRVs and mechanical springs to close the SRVs, this SAMA would not mitigate this initiating event.

No additional SAMAs have been identified to mitigate this initiating event.

Question 5.b.iv:

It is noted that the table includes a number of standby service water (SSW) pumps B and D failure events but not any events for failure of SSW pumps A or C. Explain the reasons for this difference and discuss if the reasons suggest any potential SAMAs.

5.b.iv Response:

The standby service water (SSW) system is divided into two trains of two pumps each. Train A SSW contains pumps A and C while Train B SSW contains pumps B and D. The SSW System is normally aligned so that Train B is providing cooling water to the component cooling water heat exchangers. Due to differences in flow requirements for the different trains because of the additional Train B flow required for the CCP system, failure of either pump B or D causes failure of Train B SSW whereas failures of both pumps A and C are needed to fail Train A SSW. Train B SSW also provides a flow path for low pressure injection of SSW and Fire Protection Water to the reactor vessel, through valves E12-MOVF094 and E12-MOVF096 which comprise a cross-connect piping path from Train B SSW to Train B of the RHR system.

SAMA 80, to add a SSW pump, addresses the SSW train asymmetry. To remove the asymmetry, while ensuring redundancy and allowing for maintenance, both Train B pumps would have to be replaced with larger pumps capable of providing sufficient flow. That modification would cost more than SAMA 80 and would also not be cost-beneficial. Other SAMAs, such as SAMA 21 and SAMA 22 which provide backup diesel cooling, and SAMA 17 which provides backup flow to the RHR heat exchanger, also mitigate failures of the SSW pumps. No additional SAMAs addressing the asymmetry between Train A and Train B SSW have been identified.

Question 5.b.v:

Event SWP-MOV-CC-F055A "motor operated valve [MOV] 1SWP*MOV55A fails to open on demand" is said to be addressed by SAMAs 75 and 80, both of which pertain to the service water pumps. Discuss the potential for a SAMA for the operator to manually open the valve.

5.b.v Response:

The RBS model includes an operator action (SWP-XHE-FO-F055A) to manually open valve

SWP-MOV55A if it fails due to loss of power. This action is needed for SBO accident sequences because valve SWP-AOV599 which provides the initial HPCS diesel service water return path to the cooling towers has a 4 hour air supply. The time window for completing the action (opening SWP-MOV55A) is 4 hours to ensure continued operation of the HPCS diesel. The Station Blackout abnormal operating procedure provides directions for performing this action. The procedure was reviewed to identify enhancements, but none were identified.

Question 5.b.vi:

For ADS-XHE-FO-INDIV "operator fails to start ADS by opening individual ADS valves or SRVs," consider improvements in procedures and training.

5.b.vi Response:

A new SAMA case was performed to evaluate the potential benefit from improvements in procedures and training for this event. The case assumes the improvements reduce the failure probability of ADS-XHE-FO-INDIV by a conservative percentage of 50%. The 50% reduction was also applied to dependent HFEs which include ADS-XHE-FO-INDIV unless the dependent event probability was at the minimum allowed value of 1E-06.

Internal & External Benefit	$16,077
3% Discount Rate	$24,747
95th Percentile Uncertainty	$64,308

The estimated implementation cost for a procedural change and training ranges from $50,000 to $200,000. Since this SAMA requires both procedure and training improvements, the cost would be at least $100,000. Thus, this SAMA would not be cost-beneficial.

Question 5.c:

The disposition of a number of late large release Level 2 risk significant terms in Table D.1-5 states, "This item is a split fraction. No SAMAs need to be correlated." While these events are in some cases related to deterministic phenomenological analysis or assumptions and not hardware or other failures, they do indicate the importance of a number of these events and the associated assumptions. In addition, as indicated by the base case risk results, source term categories (STCs) 9 and 10 dominate the risk and involve penetration failures with and without scrubbing in the auxiliary building. It would therefore appear that steps that could be taken to reduce the impact of these STCs should be considered. For example, consideration of means to increase the likelihood or effectiveness of scrubbing in the auxiliary building in the area of the penetrations would appear worthwhile. Discuss this particular example and the more general question addressing potential SAMAs suggested by review of the Level 2 split fractions.

5.c Response:

Split fractions are used to represent the likelihood of various phenomenological events from the deterministic analysis of the physical processes for the spectrum of severe accident progressions. In general, SAMAs do not need to be correlated for split fractions because split fractions add up to 1.0 and result in cutsets which are duplicates except for the split fraction events themselves. Reducing one part of the fraction necessitates increasing the other part of the fraction. Therefore, to mitigate cutsets containing split fractions, other event(s) in the cutsets should be mitigated.

Consider the following example:

> Table D.1-5 lists event L2-PROB-INJ-PIPE-INT, "No piping failure to disrupt injection," with a probability of 0.667. It also lists event L2-PROB-INJ-PIPE-FAIL, "Piping failure disrupts injection," with a probability of 0.333. (Events renamed INT and FAIL for short in this example.)
>
> If a SAMA was postulated to decrease the probability of INT to 0.333, it would also increase the probability of FAIL to 0.667. So, the probability of all cutsets containing INT would be decreased and the probability of all cutsets containing FAIL would be increased and the total plant risk would not be substantially changed.
>
> However, these split fractions are in cutsets with hardware failures and human action failures and the dominant hardware failures and human action failures also show up in the RRW tables. So, if a SAMA decreases the probability of a hardware or human action failure, the probabilities of cutsets containing INT would be decreased and the probabilities of cutsets containing FAIL would also be decreased and the total plant risk would be substantially reduced.

Since SAMAs have been evaluated for all of the important hardware and human action failure events, there is no need to evaluate SAMAs for the split fraction events. In addition, a SAMA to reduce the probability of injection pipe failure was not considered feasible.

The case of split fractions L2-ABSCRUB-FAIL and L2-ABSCRUB-SUCCESS is different from the other split fractions because these split fractions were used to alter the radioactive releases to simulate attenuation of fission products from auxiliary building scrubbing.

Since RBS is a Mark III containment design, the auxiliary building encloses the containment. Some particulate releases from containment would deposit on equipment and structures within the auxiliary building, rather than transit directly to the environment. This reduction in particulate release is a scrubbing benefit that cannot be precisely quantified as there is no MAAP model of the auxiliary building. In addition, the actual location of failures into the auxiliary building is unknown. Failures into the auxiliary building are expected to occur in the ducting that services the annulus, since that ducting has a low design pressure, but the specifics are unknown. The elevation of the failure, the downstream distance of the failure in the ducting, and whether or not a distributed failure occurs over a sustained length of the ducting are unknown variables that add to the uncertainty in modelling scrubbing of releases into the auxiliary building. However, there is a clear benefit that a best-estimate Level 2 analysis should credit. Therefore, for non-LERF releases, a probability of 0.5 was estimated for successful reduction in fission product releases to the environment due to auxiliary building attenuation. This probability was applied to the sequences with penetration failures of containment, with successful attenuation assumed to reduce the fission product release fractions by 50% (except for noble gases, for which no attenuation was credited).

Split fractions L2-ABSCRUB-FAIL and L2-ABSCRUB-SUCCESS were used to make two copies of every non-LERF cutset that resulted in a radioactive release through the auxiliary building due to penetration failure. This created four pairs of source term categories (STCs), which have the same probability. These are STC-7 and STC-8, STC-9 and STC-10, STC-11 and STC-12, STC-13 and STC-14. For each of the pairs of STCs, the fission product release fractions from the assigned MAAP case were applied to the larger numbered STC and half of the release fractions were applied to the smaller numbered STC to reflect the 50% attenuation (except noble gases, for which the entire release was applied in both STCs).

It is judged that this scrubbing estimate is likely conservative. In fact, it is believed that the reduction in releases to the environment would be greater, and also would likely reduce some of the Large, Early Releases to levels that would not require them to be considered large and early. However, there is a great deal of uncertainty in this assumption, so a sensitivity analysis was run to determine the impact on the release profile. This sensitivity analysis and the resulting impact on the release profile are presented in the following table.

Impact of Auxiliary Building Scrubbing Credit on LERF and LLRF

Case Description	LERF (/yr)	SERF (/yr)	LLRF (/yr)	SLRF (/yr)
a) Baseline (no credit for LERF, and the credit for late releases is not sufficient to reduce the releases from large to small)	2.32E-8	0.0	2.08E-6	1.29E-7
b) Aux Building scrubbing is sufficient to reduce LERF by 50% and Large, Late Release Frequency (LLRF) by 50%	1.16E-8	1.16E-8	1.04E-6	1.17E-6
c) Aux Building scrubbing is sufficient to reduce LERF by 90% and Large, Late Release Frequency (LLRF) by 90%	2.32E-9	2.09E-8	2.08E-7	2.00E-6

Notes:
LERF = Large, Early Release Frequency
SERF = Small, Early Release Frequency
LLRF = Large, Late Release Frequency
SLRF = Small, Late Release Frequency

As noted in the above table, taking additional credit for auxiliary building scrubbing has the potential to significantly reduce the large releases in the Level 2 model. However, without a defensible basis for the reduction in fission product releases, the modeling of a 50% chance of a 50% reduction in late particulate releases was utilized. Developing a defensible basis would require either modeling in MAAP (or other fission product transport code), or some credible tests, generic or plant specific.

The actual location of releases into the auxiliary building is unknown and there would be negative consequences, both spraying safety-related components at power and flood risk, from using a spray system in the auxiliary building to scrub the releases. Thus, no potential SAMAs were evaluated.

Question 5.d:

From the information in Tables D.1-1 and D.1-2 the frequency of initiating event IE-TNSW, Failure of the Normal Service Water (NSW)/Service Water Cooling (SWC) System, is an input into the internal events PRA as a value rather than as a fault tree model. The basic events that contribute to this frequency will therefore not appear in the list of risk significant terms in Table D.1-2. Describe NSW and SWC systems, their operation, and modeling in the PRA, particularly with respect to operation in hot weather and discuss the identification of candidate SAMAs, other than SAMA 197 (Generation Risk Assessment implementation into plant activities), that would mitigate the risk of this initiator.

5.d Response:

The service water system is comprised of two interconnected subsystems: the Normal Service Water (NSW) sub-system and the Standby Service Water (SSW) sub-system.

One portion of the service water system is non-safety-related; the other is safety-related. Under normal conditions, the NSW pumps provide cooling to both safety and non-safety related loads via a closed loop which is cooled by the Service Water Cooling (SWC) system. Under certain emergency conditions such as a loss of NSW or a low component cooling water pressure initiation of SSW, the two portions of the system will be isolated from each other. The NSW pumps will serve the non-safety-related loads while the SSW pumps will serve the safety-related loads using the standby cooling tower as a heat sink.

The operation of two pumps for NSW (or one NSW pump with SSW initiation) and one pump for SWC, six heat exchangers and a number of SWC fans dependent on ambient temperature are required to prevent a scram initiator. However, only one NSW pump is needed to cool the non-safety related loads following an initiator.

The SWC system is modeled simply as the failure of two of three SWC pumps, the heat exchangers, and the SWC fans. Failure of the SWC system is included in the TNSW initiator frequency since failure of SWC is expected to result in a loss of NSW. Six of eight heat exchangers are required to support normal service water operation. SWC fan success criteria were conservatively developed based on assumed ambient temperature conditions as follows.

Assumed Temperatures	Normal Operating Conditions	Post-event Operating Conditions
Summer (≥85°F maximum daily temperature)	3 of 5 fans	4 of 5 fans
Spring / Fall (60-85°F maximum daily temperature)	2 of 5 fans	3 of 5 fans
Winter (<60F maximum daily temperature)	1 of 5 fans	2 of 5 fans

The event with the highest RRW in the IE-TNSW cutsets is SWC-PHN-DN-SCHOT, "Summer Temperatures require four of five fans to run to maintain SWC temps." This is followed by mechanical failure events and maintenance events for SWC fans and NSW pumps. Thus, only large cost SAMAs, such as adding fans or pumps, would have a significant impact on this initiator. Therefore, no additional SAMAs are proposed to mitigate this initiator.

Question 5.e:

Section D.2.1 indicates that the initial list of SAMA candidates was developed from the review of a list of industry documents. Provide additional information on how this review was performed and how the decision was made to include individual items in the Phase I list. Specifically address:

Question 5.e.i:

Which potential plant improvements from the 13 other SAMA evaluations were considered.

5.e.i Response:

The NUREG 1437 Plant-Specific Supplements for the plants listed below were reviewed. SAMAs identified as potentially cost-beneficial in the supplements were included in the initial list of SAMA candidates.

- FitzPatrick Nuclear Power Plant SAMA Analysis
- ~~Columbia Generating Station SAMA Analysis~~
- Cooper Nuclear Station SAMA Analysis
- Oyster Creek Nuclear Generating Station SAMA Analysis
- Monticello Nuclear Generating Plant SAMA Analysis
- Brunswick Steam Electric Plant SAMA Analysis
- Pilgrim Nuclear Power Station SAMA Analysis
- Susquehanna Steam Electric Station SAMA Analysis
- Vermont Yankee Nuclear Power Station SAMA Analysis
- Duane Arnold Energy Center SAMA Analysis
- LaSalle County Station SAMA Analysis
- Grand Gulf Nuclear Station SAMA Analysis
- Fermi 2 Nuclear Power Plant SAMA Analysis

In total, 70 unique SAMA candidates from 12 other SAMA evaluations were considered. [Eleven of the SAMA candidates from other SAMA evaluations are also generic SAMA candidates listed in NEI 05-01A.]

During development of this response, it was noted that the Columbia Generating Station SAMA Analysis was included on the list above in ER Section D.1.1 although potentially cost-beneficial SAMAs from NUREG-1437, Supplement 47 were not included in the initial list of SAMA candidates. Since so many other SAMA candidates have already been considered, it is expected that addition of Columbia's potentially cost-beneficial SAMAs would not result in new retained SAMAs for RBS. Thus, Columbia is being removed from the list.

Question 5.e.ii:

The inclusion of items from the RBS Individual Plant Examination (IPE) or Individual Plant Examination of External Events (IPEEE) or NUREG-1742.

5.e.ii Response:

The IPE identified three enhancements that were implemented before the IPE was submitted and three modifications due to IPE insights to be implemented at a later date. The three pending modifications were to add a diesel generator as a permanent backup power source to the 125 VDC battery chargers, to remove the internals from check valves in fire water to standby service water cross-ties, and to revise the station blackout procedure for fire water injection to the reactor to eliminate the need for containment entry. These modifications have been implemented and, therefore, were not included in the list of SAMA candidates.

The IPEEE did not identify any vulnerabilities requiring enhancements to be included in the initial list of SAMA candidates. Also, NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program", did not identify any enhancements for RBS.

The fire areas identified as top contributors in the IPEEE were evaluated for potential SAMA candidates. Two SAMA candidates that were potentially cost beneficial at other plants were retained for further evaluation because they were applicable to fire areas identified as top contributors in the IPEEE. Specifically, SAMA 183, "Add automatic fire suppression system," was retained because the division 1 standby switchgear room does not have automatic suppression and was one of the top contributors to fire CDF in the IPEEE. Also, SAMA 185, "Upgrade the ASDS panel to include additional system controls for opposite division," was retained because the control room was one of the top contributors to fire CDF in the IPEEE.

No SAMAs were identified that would improve the other fire areas identified as top contributors in the IPEEE.

<u>Question 5.e.iii:</u>

How the RBS updated PRA list of significant contributors (the RRW correlation tables) were examined to identify RBS plant specific SAMA candidates as opposed to generic or other plant SAMAs.

<u>5.e.iii Response</u>:

The initial list of SAMA candidates was developed from the review of several industry documents. The generic SAMA candidates for BWRs from NEI 05-01A were added first. Then, the SAMAs identified as potentially cost-beneficial in 12 other SAMA evaluations were added, as described in the response to 5.e.i. Also, the IPE and IPEEE were checked for potential candidates, as described in the response to 5.e.ii.

This initial list of SAMA candidates was screened as described in Section D.2.1 to determine which SAMA candidates should be retained for further evaluation.

Finally, the RRW correlation tables were used to identify additional SAMA candidates. The events in the RRW correlation tables were examined, along with fault tree and cutset files, to identify retained SAMAs that would mitigate each dominant event (except split fractions as discussed in the response to 5.c). In this manner, the massive amount of industry SAMA experience was leveraged to identify mitigation alternatives for the dominant risk contributors. While each dominant contributor was being evaluated and industry SAMAs were being correlated to them, if a less costly alternative could be postulated to mitigate the event, one was added to the list of retained SAMAs. During this process, the following three SAMAs were identified for further evaluation.

- SAMA 204, Add capability to cross-tie fuel oil supply to emergency diesel generators
- SAMA 205, Revise FLEX procedures (FSG's) to allow use of FLEX equipment in non-ELAP conditions
- SAMA 206, Improve flow capacity of Service Water Cooling fans for summer conditions

SAMA 205 was proposed during the RRW correlation process, but was not actually listed for individual events in the RRW tables. This change would ensure availability of a large cool source of suction water and DC power for RCIC to operate for extended time periods and provide an alternate method of suppression pool cooling using the Suppression Pool Cleanup (SPC) system and a portable generator. Thus, it would mitigate multiple dominant contributors.

Question 5.f:

The RBS IPEEE did not identify any fire related vulnerabilities or improvements, however, five "enhancements" were identified. Discuss the status of the implementation of these "enhancements."

5.f Response:

A suggested procedure enhancement to address lack of condensate storage tank level indication due to a fire has been implemented in AOP-0052, *Fire Outside the Main Control Room in Areas Containing Safety Related Equipment*.

Also, a suggested procedure enhancement to warn that long-term operation of the SRVs may be compromised due to fire-related failures of compressed air has been implemented in AOP-0052, *Fire Outside the Main Control Room in Areas Containing Safety Related Equipment*.

The remaining identified enhancements were suggested changes to take credit in the safe shutdown analysis for additional, existing components that are already modeled in the PRA. Therefore, the suggested enhancements would not provide any risk reduction and were not considered as severe accident mitigation alternatives.

Question 5.g:

During the audit the NRC staff reviewed the process for identification and disposition of Phase I SAMA candidates. Address the following with regard to this process:

Question 5.g.i:

SAMA 93 regarding heating ventilation and air conditioning (HVAC) discusses a recent analysis of control building shows reduced HVAC importance and that this SAMA does not consider the control building loss of HVAC. This is also stated for SAMA 94 involving enhanced loss of HVAC procedures. Clarify the results of the recent analysis that supports this disposition.

5.g.i Response:

The recent control building HVAC analyses included revised GOTHIC control building heat-up calculations as well as revised equipment survivability assessments. The survivability studies demonstrated that the electrical equipment would survive and continue to operate at temperatures higher than their conservative design temperature limits. The heat-up calculations demonstrated that the most limiting equipment in a specific room would function for the room temperature profiles resulting from a loss of room cooling. An interim PRA model was used to evaluate the changes in the control building HVAC requirements. This model demonstrated significantly lower risk with a single division of chillers or control building HVAC out of service due to realistic treatment of equipment survivability and the limited actions required to recover switchgear room cooling. Incorporation of the above changes into the SAMA PRA model would result in a much lower control building HVAC contribution to risk such that, any related SAMAs would not be cost beneficial.

Question 5.g.ii:

SAMA 120, to install a hardened containment vent, cites a containment vent study. Describe this study, its scope and results. In addition, Section D.1.4.8 mentions the deletion of credit for a 3-inch containment vent. Describe and discuss the current status of this vent path.

5.g.ii Response:

The purpose of calculation G13.18.12.4-030 was to assess potential containment venting strategies in the event of loss of all containment decay heat removal. Three basic venting paths were analyzed: 1) Hard vent path (3-inch) using containment purge line discharging to the shield building annulus; 2) Venting through Auxiliary Building (AB) personnel airlock deflated inner door seals; and, 3) Venting through Fuel Building (FB) personnel airlock deflated inner door seals. Cases for the first path included venting to the annulus with and without standby gas system operating. Cases for the AB path included no venting of the AB, use of standby gas to vent AB, and venting AB to the environment through open doors. Cases for the FB path included no venting of the FB, use of HVAC system to vent FB, and venting FB to the environment through open doors. The study resulted in the following conclusions.

- Venting to the shield building annulus with the 3-inch vent will result in containment failure irrespective of operation of the standby gas treatment system.

- Containment failure is avoided when containment is vented through one of the containment airlocks provided a pathway to the environment through an open door is established.

- Venting of containment atmosphere into the AB or FB without a pathway through an open door to the environment will result in pressures well in excess of those used in structural design calculations for the AB and FB.

The 3-inch vent path is included in emergency operation and severe accident procedures. The instructions for implementing the vent path are contained in Enclosure 21 of procedure EOP-0005. This enclosure also provides instructions for venting with the AB personnel airlock.

Question 5.g.iii:

SAMA 170 to increase seismic ruggedness of plant components is stated to be already installed based on improving RBS components whose seismic ruggedness could be improved and was identified in the IPEEE. The RBS IPEEE does not identify any such components. Discuss the potential for reducing seismic risk by improving seismic ruggedness of selected RBS components.

5.g.iii Response:

The IPEEE concluded that RBS is seismically rugged based on reviewing design documents and performance of a seismic walkdown. The RBS Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident was provided in letter RBG-47453 on March 26, 2014 (ML14091A426) and concluded the following:

> "In accordance with the 50.54(f) request for information (U.S. NRC, 2012a), a seismic hazard and screening evaluation was performed for RBS. A GMRS was developed solely for the purpose of screening for additional evaluations in accordance with the SPID (EPRI, 2013a)."
>
> "In the 1 to 10 Hz part of the response spectrum, the SSE exceeds the GMRS. Therefore, a risk evaluation will not be performed."

Therefore, due to low seismicity at RBS, further seismic risk evaluation for RBS is was not required.

As described in Section D.1.3.5, an external event multiplier of 7 (rounded up from the calculated value of 6.9) was used in the SAMA analyses to account for the risk contribution from external events in the SAMA evaluations

Given the above discussion, improving the seismic ruggedness of selected components would not significantly reduce seismic risk nor change the SAMA evaluation results.

Question 6.:

Provide the following information with regard to the Phase II cost-benefit evaluations. The basis for this request is as follows: Applicants for license renewal are required by 10 CFR 51.53(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 RBS SAMA analysis, NRC staff evaluates the applicant's cost-benefit analysis of Phase II SAMAs. The requested information is needed in order for the NRC staff to determine the acceptability of the applicant's cost estimations for individual SAMAs and cost-benefit evaluation

Question 6a:

Describe what changes/modifications are associated with the implementation of SAMA 38, Protect service transformers from failure evaluated by Case 9. Address whether the scope of this modification can be reduced to obtain the same or similar benefit.

6.a Response:

The modification for SAMA 38 is construction of structures that would provide protection (from hurricane or tornado flying debris and from explosion of a nearby transformer) for transformers and associated cabling and buses running from the transformers into nearby buildings. The protection methods are to include walls, metal caging, or other suitable barriers. Protection is provided for three normal transformers and four preferred transformers. Power feed lines from offsite to the transformers are not protected by this SAMA.

The most likely failures of a transformer from external causes are from missiles generated from hurricanes, tornados or explosions from a nearby transformer. The conceptual design to address this SAMA includes design and construction of two buildings that provide protection from high wind missiles and explosions from nearby transformers. Failure of any of the seven service transformers has the potential to trip the plant, so all are protected. One building (~9600 sq ft) houses five transformers which are relatively close and located in the same transformer yard. The other building (2600 sq ft) houses two transformers that are near each other and located in a different yard. The buildings would be constructed with reinforced concrete designed for tornado missile protection. The size of the buildings and the need for tornado missile protection resulted in significant estimated costs for this SAMA. No other lower cost alternative designs which satisfied the SAMA's objectives were identified.

Question 6.b:

Case 19 evaluates the benefit of SAMA 87, Install digital feedwater upgrade, by setting the loss of feedwater system initiating event (IE-T3B) to false in the base model Level 1 and Level 2 cutsets. Discuss the potential for additional benefit of the upgrade resulting from the reduction in the potential for loss of feedwater following other initiators.

6.b Response:

Case 19 was re-evaluated assuming that feedwater also does not fail following other initiators. The SAMA was evaluated by setting gates PCS040, PCS009A, PCS030, PCS065, PCS017, and PCS033 to false in addition to IE-T3B. The potential benefits and cost are summarized below.

Internal & External Benefit	3% Discount Rate	95^{th} Percentile Uncertainty	Cost Estimate
\$57,422	\$88,416	\$229,689	\$900,000

Revised SAMA 87 is not cost-beneficial.

Question 6.c:

SAMA 183, add automatic fire suppression (specifically, addition of incipient detection and suppression division 1 switchgear (Div 1 Swgr) Room), and SAMA 185, upgrade the alternate shut down system (ASDS) panel to include additional system controls for opposite division, were evaluated in Cases 36 and 37 by eliminating the contribution to CDF from the respective fire zones. Address the following with respect to these two SAMA and cases:

Question 6.c.i

Case 37, Reduce Risk from Fires That Require Control Room Evacuation, was used to evaluate the benefit for SAMA 185 by assuming this SAMA eliminated Control Room fires from the RBS fire CDF. It is noted that fires other than in the control room may require control room evacuation and could benefit from the upgrade of the ASDS panel. Identify the other control room abandonment areas and discuss the impact on the cost-benefit analysis of SAMA 185 from crediting the risk reduction benefit of this SAMA for the identified abandonment areas.

6.c.i Response:

As noted, Case 37 only addressed Control Room fires. Upon further review, fires in one other area in the control building have the potential to result in control room evacuation. Smoke generated from fires in area C-17, Control Room Ventilation Room could be transferred to the control room and ultimately require evacuation because of low visibility. The Control Room accounts for approximately 22% of RBS fire CDF and the Control Room Ventilation Room accounts for 20%, for a total of 42%. A bounding analysis was performed by eliminating the CDF associated with these two rooms. A revised benefit is calculated using the same method as that described for Case 37 in Attachment D of the Environmental Report.

Given,
Maximum internal events (IE) benefit = $255,681
Total Fire CDF = 9.0E-06/rx-yr
Internal events (IE) CDF = 2.79E-06/rx-yr

Maximum Fire benefit = Maximum IE benefit X Total Fire CDF/Total IE CDF
Maximum Fire benefit = $255,681 X (9.0E-06/2.79E-06) = $824,776
Revised Case 37 benefit = 42% X (Maximum Fire benefit) = 0.42 X $824,776
Revised Case 37 benefit = $346,406

The benefits and costs are summarized below:

Internal & External Benefit	3% Discount Rate	95th Percentile Uncertainty	Cost Estimate
$346,406	$536,929	$1,385,624	$790,000

Therefore, revised SAMA 185 would only be cost-beneficial for the 95th Percentile Uncertainty sensitivity case. Although not related to adequately managing the effects of aging during the period of extended operation, the above, potentially cost-beneficial SAMA will be entered into the action tracking process to be evaluated for implementation.

Question 6.c.ii:

The benefit for these two fire related cases was determined based on the assumption that the percentage reduction in person-rem risk and offsite economic risk (OECR) is the same as the percentage reduction in CDF. An examination of the results for the risk reductions given in Table D.2-1 indicates that this assumption is not necessarily conservative. Depending on the case, the reduction in OECR is often a factor of 1.1 to 1.4 times the CDF reduction. For Case 17 the OECR reduction is 2.4 times the CDF reduction. Discuss the impact of this assumption on the cost-benefit analyses of SAMAs 183 and 185.

6.c.ii Response:

As indicated in the previous response, re-evaluated SAMA 185 is being considered potentially cost-beneficial and has been entered into the action tracking process to be evaluated for implementation.

Examination of the risk reductions in Table D.2-1 indicates that the PDR reduction ranges from 0.6 to 3.8 times the CDF reduction, with an average of 1.4. Also, the OECR reduction ranges from 0.8 to 3.7 times the CDF reduction, with an average of 1.3.

If the benefit values for SAMA 183 are increased by a factor of 1.4, the following results are obtained.

Internal & External Benefit	3% Discount Rate	95th Percentile Uncertainty	Cost Estimate
$242,484	$375,851	$969,937	$1,100,000

Therefore, SAMA 183 remains not cost-beneficial.

Question 7.:

For certain SAMAs considered in the RBS Environmental Report, there may be lower cost or more effective alternatives that could achieve much of the risk reduction. In this regard, provide an evaluation of the following SAMAs. 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 RBS SAMA analysis, NRC staff considers additional SAMAs that may be more effective or have lower implementation costs than the other SAMAs evaluated by the applicant. The requested information is needed in order for the NRC staff to determine the adequacy of the applicant's determination of cost-beneficial SAMAs.

Question 7a:

SAMA 34, Provide alternate feeds to essential loads directly from an alternate emergency bus, is evaluated using a plant-specific cost estimate of $2.3M. Case 4, evaluating this SAMA, is stated in ER Section D.2.3 to assume an added independent AC [alternating current] power source to each safety-related 4160v bus. Clarify the scope of the design used for the cost estimate and, if it includes independent supplies to all safety-related buses, consider the cost-benefit of a design that included a supply to only a single safety-related 4160v bus.

7.a Response:

The SAMA 34 implementation cost estimate was for a power supply from an alternate bus to a single safety-related 4160VAC bus.

The wording of the Case 4 description could have been more precise. The following statement in the description should be revised as noted.

> "A bounding analysis was performed by modifying the fault tree model by adding an

independent power source ***capable of supplying each of*** ~~to~~ the safety related 4160VAC buses."

Question 7.b:

SAMAs 80, 110, 115, and 120 involve major new systems to mitigate loss of cooling events and are cited for a large number of significant basic events in Table D.1-2 including, for example: SWP-XHE-FO-RETRN, Operator fails to open SWP manual isolation valve before containment over-pressurization failure, SWP-XHE-RE-F055A, operator fails to restore XOV downstream of F055A, and E12-MOV- OO-F048A, water diverted from RHR A HXS because bypass valve MOV F048A fails to close. Describe these events and other similar events regarding the possibility of lower cost alternatives such as: simpler hardware changes, changes in system operation in order that fewer changes in valve position are necessary, and/or procedure and training improvements.

7.b Response:

Pre-initiator event SWP-XHE-RE-F055A is the failure to restore manual valve SWP-V3302 to the open position following maintenance on Train A of the SSW system. This manual valve is the last valve in the SSW return line to the cooling tower. Event SWP-XHE-FO-RETRN is a post-initiator operator action to open the manual valve if SWP-XHE-RE-F055A fails. SWP-XHE-FO-RETRN is also applied to decay heat removal related RHR and SSW valve failure events. The event is only relevant for long term sequences associated with loss of decay heat removal. Event E12-MOV-OO-F048A is the failure of the RHR heat exchanger bypass valve to close. The valve is normally open to ensure full injection flow to the vessel following a LOCA signal. The valve must be closed to initiate heat removal by the RHR heat exchanger. Similar valves to those discussed are included in Division 2 SSW and RHR train.

These valves are important for loss of decay heat scenarios in the Rev. 5A model. However, the core damage contribution for the valves is significantly reduced in the Rev. 6 model (by as much as two orders of magnitude). The difference is primarily due to addition of a venting path (personnel air lock) for removal of decay heat from containment in the Rev 6 model, as well as credit for FLEX equipment and procedures for Suppression Pool Cooling under ELAP conditions. Therefore, lower cost alternatives such as those suggested are not viable mitigation alternatives. Changes in system operation such that fewer changes in valve position are necessary are not applicable since the manual valves are typically operated only for required testing and maintenance. The RHR valve is typically operated for required testing and maintenance and when the heat exchanger is required for heat removal. Procedure and training improvements would provide very limited benefit since manual operation of valves is simple and within the skill of the craft. Hardware modifications that would reduce the risk associated with these valves include addition of motor operated bypass valves around the SWP valves to create an alternate flow path and addition of a valve in line with E12-F048A to isolate the heat exchanger bypass line. These modifications are relatively complex since they would also require electrical power and control circuits in addition to the piping modifications and, therefore, would not be cost-beneficial. Less complicated hardware changes were not identified.

Table 2.c-1		
MAAP Case	**Description**	**Frequency**
S2A-6	Small LOCA; success of HPCS; RHR and containment fan failure leads to containment failure prior to core damage. MCCI is predicted to begin at 35.8 hours.	9.05E-07
T-TB-1	SBO; no SORV; HPCS successful; no offsite power recovery; long term core damage after containment failure. MCCI is predicted to begin at 63.9 hours.	1.40E-07
T-TB-2	SBO; no SORV; HPCS failure; RCIC successful while battery available; reactor depressurized; Fire Protection water injection successful; no offsite power recovery; long term core damage after containment failure. MCCI is predicted to begin at 66.7 hours.	6.61E-07
T-14	Transient with failure of feedwater; success of HPCS; failure of Power Conversion System; failure of RHR and containment fans; core damage after containment failure. MCCI is predicted to begin at 63.4 hours.	1.51E-07
T1-4	Loss of offsite power; success of HPCS; failure of RHR and containment fans; core damage after containment failure. MCCI is predicted to begin at 64.0 hours.	3.76E-07
T-TB-3	SBO; no SORV; HPCS failure; RCIC successful while battery available; reactor depressurized; Fire Protection water injection fails; no offsite power recovery; long term core damage after containment failure. MCCI is predicted to begin at 17.5 hours.	2.48E-07
RCIC-Inj	Transient with success of RCIC; no containment heat removal; depressurization successful but no low pressure injection; core damage occurs prior to containment failure; representative of various sequences with RCIC success but no low pressure injection or containment heat removal. Sequence frequency is combination of T1-16 and T-33. MCCI is predicted to begin at 16.1 hours.	9.34E-10
T-TB-6	SBO, no SORV, HPCS/RCIC failed; depressurization successful; failure of fire water. MCCI is predicted to begin at 6.7 hours.	5.94E-08
T-TB-9	SBO; two SRVs stuck open; HPCS failure. MCCI is predicted to begin at 6.6 hours.	3.93E-08
T-51	Transient; failure of all injection; failure to depressurize. MCCI is predicted to begin at 4.2 hours.	3.28E-08
T-51-CV	Same as T-51, but with containment venting	Same as T-51

Table 2.e-1

STC	Description	FREL(1) Nobles	FREL (2) CsI	FREL (3) TeO2	FREL (4) SrO	FREL(5) MoO2	FREL(6) CsOH	FREL(7) BaO	FREL(8) La2O3	FREL(9) CeO2	FREL(10) Sb	FREL(11) Te2	FREL(12) UO2
STC 11	No CFBVB, no MCCI, AB scrub (note 2)	1.0	1.5E-04	2.1E-5	2.9E-6	1.8E-5	2.9E-04	9.6E-06	1.6E-07	1.0E-06	4.4E-04	2.5E-07	4.2E-09
STC 12	No CFBVB, no MCCI, no AB scrub	1.0	3.1E-04	4.2E-05	5.9E-06	3.6E-05	5.8E-04	1.9E-05	3.1E-07	2.0E-06	8.8E-04	5.0E-07	8.3E-09

Table 2.f-1

MAAP Case #	STC	Time of CD	Time of Start of Release	Time of Containment Failure	Run Time	Time Gen Emerg Declaration
T-51-NF	1	36.9 min	0.5 hr	N/A	140 hr	2.97 hr
T-51-SPB	2	36.9 min	0.5 hr	2.97 hr	140 hr	2.97 hr
T-51-CIF	3	36.9 min	0.5 hr	39 psig not reached, because of containment isolation failure	140 hr	2.96 hr
VRUP-CF	4	3.9 sec	0 hr	0.0 sec (containment failure forced at vessel rupture)	140 hr	3.91 sec
T-51-SPB	5	36.9 min	0.5 hr	2.97 hr	140 hr	2.97 hr
T-TB-3-D-R	6	8.4 hr	8.2 hr	Leakage starts at 22.87 hr. Containment rupture set at 45 hours to evaluate late rupture of containment	160 hr	16.9 hr
T-14	7 (before MCCI occurs) with AB scrub	30.3 hr	30 hr	23.38 hr	63.4 hr (time of MCCI start)	29.24 hr
T-14	8 (before MCCI occurs) no AB scrub	30.3 hr	30 hr	23.38 hr	63.4 hr (time of MCCI start)	29.24 hr
T-14	9 (MCCI occurs, continue to end of run) with AB scrub	30.3 hr	30 hr	23.38 hr	140 hr	29.24 hr
T-14	10 (MCCI occurs, continue to end of run) no AB scrub	30.3 hr	30 hr	23.38 hr	140 hr	29.24 hr
T-TB-3	11 (before MCCI occurs) with AB scrub	8.4 hr	8.3 hr	22.9 hr	17.6 hr (time of MCCI start)	17.2 hr

MAAP Case #	STC	Time of CD	Time of Start of Release	Time of Containment Failure	Run Time	Time Gen Emerg Declaration
T-TB-3	12 (before MCCI occurs) no AB scrub	8.4 hr	8.3 hr	22.9 hr	17.6 hr (time of MCCI start)	17.2 hr
T-TB-3	13 (MCCI occurs, continue to end of run) with AB scrub	8.4 hr	8.3 hr	22.9 hr	160 hr	17.2 hr
T-TB-3	14 (MCCI occurs, continue to end of run) no AB scrub	8.4 hr	8.3 hr	22.9 hr	160 hr	17.2 hr