Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

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U.S. Nuclear Regulatory Commission
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Watts Bar Nuclear Plant, Unit 2
Construction Permit No. CPPR-92
NRC Docket No. 50-391

Subject: **Watts Bar Nuclear Plant Unit 2 – Severe Accident Management Alternatives for Reactor Coolant Pump Seals (TAC No. MD8203)**

Reference: Letter from TVA to NRC, "Watts Bar Nuclear Plant (WBN) – Unit 2 – Response to Request for Additional Information Item Numbers 2, 3, 5 and 15 Regarding Severe Accident Management Design Alternative Review," dated May 25, 2011 (TAC No. MD8203) [ADAMS Accession No. ML11147A099]

The purpose of this letter is to provide the results of a re-evaluation of the benefits of Watts Bar Nuclear Plant (WBN) Unit 2 Severe Accident Management Alternatives (SAMAs) associated with Reactor Coolant Pump (RCP) seal integrity. Tennessee Valley Authority (TVA) has not installed the Westinghouse Shield Seal package on the WBN Unit 2 RCPs, necessitating this re-evaluation.

The enclosed report provides the revised cost and benefit for the seven RCP SAMAs. The results of the re-evaluation did not identify cost-beneficial hardware modifications. A proposed enhancement to the Loss of Component Cooling Water Abnormal Operating Instructions (AOI) was determined to be potentially cost beneficial. The enhancement suggested in this alternative has been incorporated into the applicable WBN AOIs.

This submittal fulfills the commitment TVA made in the reference to reassess the RCP SAMAs if the reliability of the Shield Seal had not been established. There are no new regulatory commitments associated with this submittal.

If you have any questions, please contact Gordon Arent at (423) 365-2004.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 10th day of April 2015.

Respectfully,

for JW Shea

J. W. Shea
Vice President, Nuclear Licensing

Enclosure: Re-evaluation of Seven Watts Bar Unit 2 Severe Accident Mitigation Alternatives for Reactor Coolant Pump Seals

cc (Enclosure):

NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Watts Bar Nuclear Plant, Unit 2
NRC Project Manager – Watts Bar Nuclear Plant, Unit 2

ABS Consulting

R-3087636-1931
Revision 1

Re-Evaluation of Seven Watts Bar Unit 2 Severe Accident Mitigation Alternatives

March 2014

Prepared for:

Tennessee Valley Authority

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Re-Evaluation of Seven Watts Bar Unit 2 Severe Accident Mitigation Alternatives

March 2014

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1. Introduction

On September 16, 2011, Tennessee Valley Authority (TVA) submitted to Nuclear Regulatory Commission (NRC) the revised Severe Accident Management Design Alternative Review (SAMDA) for Watts Bar Unit 2 (WBN 2). Among the set of Severe Accident Mitigation Alternatives (SAMA), SAMA 58 was the original SAMA dealing with the implementation of the improved reactor coolant pump (RCP) seals; i.e., the SHIELD Passive Thermal Shutdown Seal (SDS) design. In addition, a number of other SAMAs (e.g., SAMAs 50, 55, 56, 215, 226, and 242) are also applicable to the issue related to the RCP seal protection or mitigation of the RCP seal loss of coolant accident (LOCA). These other SAMAs are essentially mutually exclusive with SAMA 58 in that, if one is implemented, then the others are not required, because implementation of SAMA 58 would greatly reduce the benefits of these other SAMAs as well as SAMA 93.

Although SAMA 58 was qualitatively screened out in Phase I of the previously submitted Watts Bar Unit 2 SAMA analysis, TVA was interested in the new seal package technology being demonstrated by Westinghouse at Farley Nuclear Plant since this replacement RCP seal package was potentially more attractive than most of the hardware changes to reduce the likelihood of a seal LOCA.

However, for Watts Bar Unit 2, the high temperature RCP seals had already been received, so the current seal cartridges would have to be returned to the vendor and replaced with the new design if TVA chooses to implement it. Besides, there are still design, manufacturing, installation, operational, and technological uncertainties with long term use of the SHIELD seal design that had not been captured in the total estimated cost. Additional operation experience was considered necessary prior to the implementation of SAMA 58 at Watts Bar Unit 2. Nevertheless, TVA had committed to follow the progress and experience with this new seal package design, and, if proven reliable during operation, it would be installed at the earliest refueling outage following startup during normal seal package replacements.

Since the RCP seal integrity is important to the risk profile, if the Westinghouse SHIELD seal design is not proven reliable and will not be installed at Watts Bar Unit 2, the TVA committed to use the latest probabilistic risk assessment (PRA) model at the time to re-evaluate SAMAs 50, 55, 56, 215, 226, and 242 to determine if an alternative SAMA is cost beneficial for implementation to prevent RCP seal LOCA during a station blackout event, and implement the SAMA accordingly. In other words, TVA had further committed to re-evaluate the benefits of these other SAMAs for mitigation of RCP seal LOCA scenarios if SAMA 58 is not proven reliable. TVA also further committed to add SAMA 93 to this list for re-evaluation should SAMA 58 is not proven reliable.

Based on the description for Event Number 49217 in the NRC Event Notification Report for July 29, 2013, Westinghouse reported an identified inconsistency between the intended design functionality of the SDS and that observed during post-service testing; i.e., the removal of the seal revealed that it failed some of the performance objectives for the SDS. This raised some concerns and prompted forensic investigations and resulted in enhancements to improve reliability margins.

In response to the uncertain effectiveness of the Westinghouse SHIELD seal design, seven SAMAs, as listed below, are re-evaluated in this analysis for Watts Bar Unit 2. These seven specific SAMAs are Numbers 50, 55, 56, 93, 215, 226, and 242 from the SAMA analysis performed in 2010/2011. They are re-evaluated under the condition that SAMA 58 is not implemented.

- SAMA 50 – Enhance Loss of Component Cooling System (CCS) Procedure
- SAMA 55 – Independent Seal Injection System with Dedicated Diesel Generator (DG)
- SAMA 56 – Independent Seal Injection System without Dedicated Diesel Generator
- SAMA 93 – Hardened, Unfiltered Containment Vent
- SAMA 215 – Backup Thermal Barrier Cooling
- SAMA 226 – Self-Powered, Backup Charging Pump
- SAMA 242 – Permanent, Dedicated DG for the Charging Pump

This report documents the methods used in the evaluation of the alternatives for the SHIELD seal and the findings from the analysis, which was performed by the use of the latest CAFTA-based PRA model for Watts Bar Unit 2 to evaluate the cost-benefits of the seven SAMAs considered. This evaluation was performed independent of other design changes in response to Diverse and Flexible Coping Strategy (FLEX)/Fukushima regulations.

The results of this evaluation identify no potentially cost beneficial hardware changes and one potentially cost effective procedure change and its associated training activity (only in the conservative sensitivity case) that will be considered for implementation in the event the Westinghouse SHIELD seal design is not installed.

2. List of Acronyms

Acronyms	
AB	Auxiliary Building
AFW	Auxiliary Feedwater
AOI	Abnormal Operating Instruction
ATWS	Anticipated Transient Without Scram
CCI	Core Concrete Interaction
CCP	Centrifugal Charging Pump
CCS	Component Cooling Water System (WBN System Designation)
CCW	Condenser Circulating Water
CDF	Core Damage Frequency
CEF	Early Containment Failure
CET	Containment Event Tree
DC	Direct Current
DCH	Direct Containment Heating
DG	Diesel Generator
ERCW	Emergency Raw Cooling Water (WBN System Designation)
ESW	Emergency Service Water
F&O	Fact and Observation
FIVE	Fire Induced Vulnerability Evaluation
FLEX	Diverse and Flexible Coping Strategy
GEIS	Generic Environmental Impact Statement
HEP	Human Error Probability
HPI	High Pressure Injection
IPE	Individual Plant Examination
IPEEE	IPE for External Events
ISLOCA	Inter-System Loss of Coolant Accident
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power

Acronyms (Continued)	
MACR	Maximum Averted Cost Risk
MCR	Main Control Room
MOV	Motor-Operated Valve
MSR	Moisture Separator Reheater
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PDS	Plant Damage State
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RDR	Real Discount Rate
RWST	Refueling Water Storage Tank
SAMA	Severe Accident Mitigation Alternative
SAMDA	Severe Accident Management Design Alternative Review
SBO	Station Blackout
SDS	Shutdown Seal
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SW	Service Water
TB	Turbine Building
TD	Turbine Driven
TDAFW	Turbine Driven Auxiliary Feedwater
TVA	Tennessee Valley Authority
WBN 2	Watts Bar Unit 2
WOG	Westinghouse Owner's Group

3. Methodology

3.1 Technical Approach

The methodology selected for this re-evaluation of the seven SAMAs related to the postulated RCP seal LOCA event for Watts Bar Unit 2 (WBN 2) is again based on the Nuclear Energy Institute's (NEI) SAMA Analysis Guidance Document (NEI 2005) and involves determining whether or not the implementation of these candidates is beneficial on a cost-risk reduction basis. The metrics chosen to represent plant risk include core damage frequency (CDF), large early release frequency (LERF), the dose-risk, and the economic cost-risk. These values provide a measure of both the likelihood and consequences of a core damage event. The methodology used in the 2010/2011 assessment was repeated to determine the Maximum Averted Cost using the updated PRA model. This measure is necessary to complete the Phase II evaluation. The methodology for the determination of the Maximum Averted Cost is documented in Section 5.

The process of evaluating potential SAMAs follows an iterative, progressive screening approach. There are two phases in the evaluation process. All seven SAMAs received a Phase I assessment. SAMAs that survive the Phase I evaluation are retained for additional detailed consideration in Phase II. Phase I of the SAMA analysis provides the bases for screening some candidates that are not applicable to the WBN 2 design, are of low benefit in pressurized water reactors (PWR) such as WBN 2, have already been implemented at WBN 2 or whose benefits have been achieved at WBN 2 using other means, and those whose roughly-estimated cost exceeds the possible Maximum Averted Cost Risk.

In Phase II, the SAMAs that survive the Phase I screening are further evaluated by comparing the averted cost-risk associated with that particular SAMA to a more detailed cost analysis to identify the net cost-benefit. Those SAMAs that are not screened out in Phase I were evaluated in Phase II using the same methodology as in the 2010/2011 assessment. This is the baseline Phase II evaluation, which uses mean or nominal values in the assessment.

In addition, various sensitivity analyses are conducted as part of the SAMA evaluation. The analyses included investigation of the potential impact of varying the real discount rate (RDR), an evaluation of the impact of the uncertainties reflected in the PRA, an evaluation of the sensitivity of the results to assumed evacuation speed, and an evaluation of the multiplier used to account for the risks resulting from external events.

The master PRA fault tree model for Watts Bar Unit 2 used in this analysis consists of both the Level 1 model and the Level 2 model developed using the CAFTA suite of codes. The Level 1 portion of the WBN 2 model is directly linked to the Level 2 model by first mapping the core damage sequences to five different plant damage states (plant damage states [PDS], or bins), including:

1. Containment Not Bypassed, High Reactor Coolant System (RCS) Pressure, Wet Steam Generator (SG)

2. Containment Not Bypassed, High RCS Pressure, Dry SG

3. Containment Not Bypassed, Low RCS Pressure

4. Containment Bypassed – Inter-System Loss of Coolant Accident (ISLOCA) and Steam generator Tube Rupture (SGTR) with High RCS Pressure (including both large early and late releases)

5. Containment Bypassed – SGTR with High RCS Pressure, Wet SG and Small Early Releases

Via these five PDSs, the core damage sequences are then linked into the Level 2 Containment Event Tree (CET). There are two different Level 2 CETs: Non-Station Blackout (Non-SBO) and Station Blackout (SBO). In each of the SBO and non-SBO Level 2 event trees, there are 18 event tree questions. Most of the questions are applicable to both SBO and non-SBO events. The first five event tree questions in the CETs determine each of the five Level 1 PDS bins and the portion of the CETs linked to. These five questions are:

- Question 1: SBO or Non-SBO
- Question 2: Containment Bypassed
- Question 3: Containment Isolated
- Question 4: Break Size
- Question 5: Feedwater Available to SG

As such, all Level 1 core damage sequences are linked to the Level 2 CET in accordance with their PDS bin. The remaining 13 CET event tree questions determine the containment failure and CET end state assigned. They are:

- Question 6: Pressure Induced SG Tube Rupture
- Question 7: RCS Depressurization (early)
- Question 8: Thermally Induced SG Tube Rupture
- Question 9: RCS Depressurization (late)
- Question 10: Core Damage Stopped Prior to Vessel
- Question 11: Availability of Air Return Fan System
- Question 12: Igniters Available
- Question 13: Hydrogen Detonation
- Question 14: Direct Containment Heating (DCH)
- Question 15: Containment Failure (early)
- Question 16: Containment Heat Removal
- Question 17: Basemat Melt-Through
- Question 18: Large Early Release

The Level 2 CET end states are then mapped to four release categories for the SAMA analysis. The SAMA Model Release Categories are defined and listed below:

Release Category I Non-Bypass LERF – non-bypass large and early releases; containment failures due to severe accident phenomena at or near time of vessel failure, plus containment isolation failures. This release category includes the following contributors:

- LERF-HLERF and LERF-LLERF (early containment failure, CEF, end states) – large and early releases; containment failures due to severe accident phenomena at or near time of high pressure or low pressure vessel failure.

- LERF-ILERF (ISOLATION) – large pre-existing containment leaks or isolation failures.

- LERF-LSERF (SERF) – small and early releases wherein core damage arrest occurs prior to vessel breach (i.e., reactor vessel is not breached at time of core damage), but early containment failure occurs due to hydrogen detonation or other severe accident phenomena (conservatively categorized with LERF sequences for consequence/cost calculation because their consequences are larger than the SERF category consequences).

Release Category II BYPASS LERF – large and early releases from containment bypasses due to interfacing system LOCAs, early SGTRs, and pressure-induced and temperature-induced SGTR sequences

Release Category III LATE – containment fails late; either from basemat melt-through (assigned a probability of zero for WBN because the containment would overpressurize first in sequences without containment heat removal) or from containment overpressurization due to loss of containment heat removal. This release category includes the following contributors:

- Basemat Melt-Through Only, and with Auxiliary Feedwater (AFW) Available

- Basemat Melt-Through Only and without AFW Available

- Containment Overpressurization Following Loss of Containment Heat Removal but with AFW Successful

- Containment Overpressurization Following Loss of Containment Heat Removal but with AFW Not Available

Release Category IV SERF – small containment isolation failures and SGTR release scenarios but with some mitigation of release by phenomenological means. This release category includes the following contributors:

- SERF-ISERF (ISOLATION) – Small Preexisting Containment Leaks or Isolation Failures (i.e., leakage at roughly 25 times the design leak rate)

- SERF-SSERF (SGTR) – Late SGTR Bypass Release Sequences with SGs Wet (i.e., that are scrubbed)

Release Category V INTACT – containment remains intact even in long term, release via nominal leakage rate from containment (not used for SAMA analysis because of the minimal offsite consequences and as such negligible cost is assigned to offsite exposure and economic costs).

It must be noted that most sequences with early releases still do not lead to early health effects for the average release group fractions assigned but may, under adverse weather conditions or inadequate offsite emergency response, potentially lead to early health effects. LERF sequences are those which can potentially lead to early health effects involving the public offsite. The approach adopted for the SAMA analysis was to include all sequence categories involving substantial early containment failures into the "Non-Bypass LERF" category (i.e., Release Category I), but to exclude the bypass sequences. The bypass sequences, which also have the potential for causing early health effects, are instead evaluated in a separate release category (i.e., Release Category II) because of their greater release fractions. Nevertheless, release frequencies and associated release fractions of both Release Category I and Release Category II are considered in the SAMA evaluations.

Late containment failures result from core damage sequences in which there is an extended loss of containment heat removal. Such loss of containment heat removal sequences which also fail to have an early release are then assumed to result in late containment failure.

The SAMA analysis also considered Release Category IV. Sequences assigned to Release Category IV result from a reactor vessel breach with a small pre-existing hole in the containment. These releases occur early but are so limited that they are judged not to have the potential to cause early health effects and are therefore considered separately in the SAMA analysis.

The frequency of release category V was not used in the SAMA analysis for offsite exposure and economic costs. However, the core damage sequences with an intact containment are accounted for via the total core damage frequency in the evaluation of the onsite exposure and economic costs.

Thus, the CET event sequence end states and their associated sequences are mapped into Release Categories I through IV. Linked-fault tree Level 2 end state gates are defined for each of the major contributors to each of these four release categories. The release category frequencies are found by totaling the frequencies of the contributors.

The release characteristics and Level 3 dose and economic consequences were evaluated based on the source terms associated with the dominant core damage scenarios contributing to the plant damage states that contribute the most to the containment failure release scenarios for each of the release categories. The approach is to identify dominant scenarios for each release category specific to WBN 2, note their specific release characteristics in terms of accident progression, and then derive the release fractions for each radionuclide group in each release category (the mean release fractions for each release category can be calculated). The release fractions and other sequence attributes (e.g., for timing and release points) were used in MACCS2 to determine the consequences for each sequence sub-category.

Thus, the dominant Level 1 core damage sequences and Level 2 CET sequences were evaluated for characteristics that lead to containment failure and release. These scenario characteristics were used to derive the source terms and the release times, heights, and energies for each release category. The source terms for each set of accident characteristics were weighted in accordance with the percentage contributions for each release type. The release characteristics were then used as input to the Level 3 MACCS2 analysis. Each release characteristic type were calculated separately with MACCS2 and then weighted by frequency to determine the averaged dose and economic consequences for each release category. As such, the dose and economic consequence results were evaluated separately for each of the dominant core damage and containment failure release scenarios and then the results were weighted by frequency for the four release categories. The weighted Level 3 dose and economic consequence results were subsequently applied to the Phase II SAMAs for cost/benefit analysis and to assess the benefit/cost ratio.

In the determination of release times and durations, for Non-Bypass LERF, the sequence of events involves early containment failures due to DCH, hydrogen burns, rocket mode failures, or ex-vessel steam explosions, the earliest of which take place at or following reactor vessel breach. The dominant sequences for this release category, however, involve periods of AFW cooling (4 hours) or high pressure injection (HPI; until failure of high pressure recirculation) prior to beginning the RCS heatup. Therefore, reactor vessel breach is expected to be delayed by more than 4 hours. Instead of a vessel breach time of about 4 hours, a vessel breach time of at least 8 hours is expected. Therefore, the NUREG/CR-4551 time of release was adopted (i.e., 10 hours) as representative of Watts Bar specific sequences.

For BYPASS LERF, the dominant sequence, as determined in the analysis performed for the September 16, 2011 submittal, involves a flood-induced SBO during which the turbine driven (TD) AFW pump operates for at least 4 hours until the batteries deplete. There is some additional time of SG cooling after the level control valves lose air but was conservatively neglected. Once SG cooling is lost, this is followed by a gradual heatup of the RCS and hot gases reaching the boiled dry SGs; i.e., at least another 4 hours. This gives a minimum induced SG rupture time of about 4+4 = 8 hours. Warning time is measured from the first 4 hours or about 8-4 = 4 hours total.

For the LATE release category, the time of containment overpressure is a function of the specific accident sequences. Modular Accident Analysis Program cases were developed for Watts Bar in the Individual Plant Examination (IPE) Level 2 notebook (Pages 116 and 117) for the rise in pressure in the containment. A time of containment overpressure of 30 hours is consistent with these results which vary depending on the sequence and the assumed containment failure pressure. If a conditional containment failure probability due to overpressure of 0.1 is used, the times vary between 24 hours (Non-SBO) and 38 hours (SBO). Therefore, the 30 hour release time is used.

In summary, the consequences were calculated for the dominant release category types that make up each release category, with these results weighted by the contribution from each release category type to the release category itself. These weighted release category consequences for the four release categories were then used to determine the benefit of each SAMA.

In addition to the average weighted dose and economic consequences applied to the SAMA evaluations, as sensitivity, the worst accident sequence dose and consequences of the sub-categories from each release category were also applied to the SAMAs to identify if any of the Phase II SAMA results would then become cost beneficial. After applying the worst accident doses and consequences, the SAMAs were reviewed for cost benefit using the 95th percentile CDF sensitivity. This new sensitivity of cost and benefits was evaluated using the 2.70 multiplying factor (see Sections 3.3 and 9.2 for the basis for this factor) for the 95th percentile CDF. With the exception of one SAMA, the other six SAMAs remain well below the cost benefit threshold.

3.2 Evaluation Steps

The SAMA re-evaluation consisted of the following steps:

- The most current Watts Bar Nuclear Unit 2 PRA model of record (WBN_U1_U2_NEW dated February 7, 2014) was quantified and the results compared to those in the model quantification notebook. The comparison is considered as evidence that the model used was verified to yield results which agree with the TVA Model of Record. The PRA model WBN_U1_U2_NEW has already addressed the peer review facts and observations (F&O), it accounts for Unit 2 operation, and includes a Level 1 and Level 2 analyses of internal events, including internal floods.

- Modify the WBN 2 portion of the dual unit PRA model (WBN_U1_U2_NEW) to correct a few minor discrepancies identified during the SAMA re-evaluation process. The modified PRA model is named WBN_U1_U2_SAMA. For purposes of this SAMA re-evaluation, four major release categories are used to assess the consequences of an accident. These release categories are I – Non-Bypass Large Early Containment Failure (LERF); II – Containment Bypass; III – Late Containment Failure; and IV – Small Early Containment Failure. The Intact Containment Category results in negligible consequences and its frequency is not used in the SAMA re-evaluation. These release categories are summarized in Table 1. The WBN 2 PRA model changed in the preceding was then modified, as appropriate, to map sequences into the above four release categories. The results of the combined Level 1 and Level 2 analysis expressed as Release Category frequencies are input into the Level 3 analysis. The contributions of external events to the risk metrics selected are incorporated into this SAMA update as described in Section 4.3.

- Adopt the same Level 3 PRA analysis results previously completed in support of the prior SAMDA analysis submitted to NRC on September 16, 2011 (see Section 4.6).

- Calculate the monetary value of the unmitigated WBN Unit 2 severe accident risk using U.S. Nuclear Regulatory Commission regulatory analysis techniques (NRC 1997). The release category frequencies from the modified Watts Bar Unit 2 model are used in this re-evaluation. This monetary value becomes the maximum averted cost-risk (MACR) that is possible (Section 5).

- Perform a Phase I SAMA Analysis by screening out SAMA candidates related to the RCP seal LOCA that are of low benefit in PWRs such as WBN 2, candidates that have already been implemented at WBN 2 or whose benefits have been achieved at

WBN 2 using other means, and candidates whose roughly-estimated cost exceeds the possible MACR (the results of the Phase I assessment are documented in Section 7).

- The necessary changes were made to the model to allow for the seven SAMAs to be evaluated. These changes are documented in Sections 4.1, 4.2, and 8.

- Calculate the risk reduction attributable to each remaining SAMA candidate related to the postulated RCP seal LOCA event and perform a Phase II SAMA Analysis by comparing the averted cost-risk to a more detailed cost analysis to identify the net cost-benefit. PRA insights from the modified Watts Bar Unit 2 PRA model (i.e., WBN_U1_U2_SAMA) are also used to screen SAMA candidates in this phase (the results of the Phase II assessment are documented in Section 8). All seven potential SAMA candidates previously considered for the RCP seal LOCA event are re-evaluated using the modified Watts bar Unit 2 PRA model.

- Evaluate how changes in the SAMA analysis assumptions might affect the cost-benefit evaluation. As was done in the 2010/2011 SAMA analysis, the Phase II SAMAs were re-evaluated using an alternative Real Discount Rate, the 95th percentile CDF and release category frequencies, an alternative evacuation speed model, and a more conservative external events multiplying factor (the results of the sensitivity evaluation are documented in Section 9).

- Summarize results and identify conclusions (Section 10).

3.3 Assumptions

1. The results of the latest Level 3 analysis performed as documented in the September 16, 2011 response to the Requests for Additional Information (RAIs) from the NRC were adopted in this evaluation. This facilitates the comparison of the current results with the results of the previous analyses. Therefore, the mix of the core damage scenarios contributing to each of the significant plant damage states associated with each of the release categories is assumed to be the same.

2. Four major release categories were used to assess consequences. These release categories are: I – Large Early Containment Failure (Non-Bypass LERF); II – Containment Bypass (Bypass LERF); III – Late Containment Failure; and IV – Small Early Containment Failure. The Intact Containment Category results in negligible offsite consequences and its frequency is not used in the SAMA assessment, although the total core damage frequency, which includes the successful containment isolation cases, is used for the evaluation of onsite consequences.

3. Based on the uncertainty propagation performed for the most up-to-date WBN 2 PRA model (which includes Internal Events and Internal Flooding initiating events), the ratios of 95th percentile to mean are 1.52 and 1.78 for CDF and LERF, respectively. However, this uncertainty analysis only accounts for the propagation of parameter uncertainty. Modeling uncertainty is not included. Since no uncertainty analysis accounting for both parameter and modeling uncertainties is available for use in the SAMA CDF/LERF sensitivity analyses, the ratio of the 95th percentile to the mean (of

either CDF or LERF) was still assumed to be 2.7; i.e., the same value as used in previous WBN 2 SAMA analysis.

4. Also, because no updated results of the internal fire PRA, seismic PRA, and other external events PRA are available, a multiplier of 2 was applied to the evaluation results using only risk contribution from internal events to account for the risk contributions from the external events. This is the same as previously assumed in the original SAMA analysis.

4. Severe Accident Risk

4.1 WBN Unit 2 Level 1 SAMA Model

The modified Watts Bar Unit 2 PRA model used in this SAMA re-evaluation (i.e., WBN_U1_U2_SAMA, hereinafter referred to as the "SAMA Model") was developed from the most current model of record for Watts Bar Unit 2; i.e., WBN_U1_U2_NEW, hereinafter referred to as the "Record Model". There were no changes made to the Record Model to create the Level 1 portion (i.e., the core damage frequency, or CDF) of the SAMA Model.

4.2 WBN Unit 2 Level 2 SAMA Model

The containment event trees incorporated in the current Record Model for Watts Bar Unit 2 (i.e., WBN_U1_U2_NEW) were used as the starting point for this SAMA re-evaluation. The changes to the Record Model to create the Level 2 portion of the SAMA Model are listed below.

Level 2 Model Changes

The Record Model Level 2 analysis was changed to create the SAMA analysis base case model. The changes to the Level 2 Release Category Frequency model are described below:

1. Reorganization of Release Categories into newly defined Release Categories.

   The Record Model uses the following four end states and LERF subcategories as shown in Figure 1.

   LERF – Large Early Release Frequency (Gate U2_L2LERFTOP)

   a. High Pressure Release – LERF (Gate U2_L2HLERFP)
   b. Low Pressure Release – LERF (Gate U2_L2LLERFP)
   c. Isolation Failure – LERF (Gate U2_L2ILERFP)
   d. Bypass – LERF (Gate U2_L2BLERFP)

   LATE – Late Containment Failure (Gate U2_L2LATETOP)

   INTACT – Intact Containment (No failure, Gate U2_L2INTACTTOP)

   SERF – Small Early Release Frequency (Gate U2_L2SERFTOP)

   The Record Model release categories were reorganized for the SAMA Model. The SAMA model release categories are shown in Figure 2 and listed below:

   INTACT (Gate U2_L2INTACTTOP) – containment remains intact even in long term, release via nominal leakage rate from containment (not used for SAMA because of

the minimal offsite consequences and as such negligible cost is assigned to offsite exposure and economic costs).

Non-Bypass LERF (Gate U2_L2LERFSAMA2014) – Non-bypass large and early releases, including containment failures due to severe accident phenomena at or near time of vessel failure, and containment isolation failures. Containment bypasses (e.g., ISLOCA and early SGTRs) are excluded. This release category includes the following contributors:

a. LERF-HLERF (Gate U2_L2HLERFP) – Large and Early Releases; containment failures due to severe accident phenomena at or near time of vessel failure, assigned from high pressure vessel failure sequences.

b. LERF-LLERF (Gate U2_L2LLERFP) – Large and Early Releases; containment failures due to severe accident phenomena at or near time of vessel failure, assigned from low pressure vessel failure sequences.

c. LERF-ILERF (Gate U2_L2ILERFP) – Large preexisting containment leaks or isolation failures.

d. LERF-LSERF (Gate U2_L2LSERFP) – Small and Early Releases wherein core damage arrest occurs prior to vessel breach (i.e., reactor vessel is not breached at time of core damage), but early containment failure occurs due to hydrogen detonation or other severe accident phenomena (conservatively categorized with LERF sequences for consequence/cost calculation because their consequences are larger than the SERF category consequences). Includes sequences under the following gates in the Level 2 Record Model: U2_L2SERF003, 004, 005, 006, 007, 008, 009, 010, 011, 012, 013, 014, 015, 016, 017, 018, 019, 020, 021, 022, 023, 024, 025, 026, 027, 028, 029, 030, 031, 032, 033, 034, 035, 036, 037, and 038.

Bypass-LERF (Gate U2_L2BLERFP) – Large and Early Releases from containment bypasses due to interfacing system LOCAs, early SGTRs, and pressure-induced and temperature-induced SGTR sequences.

SERF (Gate U2_L2SERFSAMA2014) – Small containment isolation failures and SGTR release scenarios, but with some mitigation of release by phenomenological means. This Release Category includes the following contributors:

a. SERF-ISERF (ISOLATION) – Small preexisting containment leaks or isolation failures; i.e., leakage at roughly 25 times the design leak rate. Includes sequences under Gates U2_L2SERF001 and U2_L2SERF039 in the Level 2 Record Model.

b. SERF-SSERF (SGTR) – Late SGTR bypass release sequences with SGs wet; i.e., that are scrubbed. Includes sequences under Gates U2_L2SERF002 and U2_L2SERF040 in the Level 2 Record Model.

LATE – containment fails late either from basemat melt-through (assigned a probability of zero for WBN because the containment would overpressurize first in sequences without containment heat removal) or from containment

overpressurization due to loss of containment heat removal. This release category includes the following contributors:

a. Basemat Melt-Through Only, and with AFW Available

b. Basemat Melt-Through Only and without AFW Available

c. Containment Overpressurization Following Loss of Containment Heat Removal but with AFW Successful

d. Containment Overpressurization Following Loss of Containment Heat Removal but with AFW Not Available

The SAMA model fault tree gates in Figure 2 associated with the above release categories are summarized as follows:

Unit 2 Level 2 SAMA Fault Tree Gate	Description
U2_L2SAMA2014	Non-Bypass LERF, Bypass-LERF, LATE, INTACT and SERF combined Top Gate
U2_L2LERFSAMA2014	Non-Bypass Large Early Release Frequency (LERF) Top Gate
U2_L2HLERFP	Early Containment Failure, with High RCS Pressure at time of Vessel Failure, Large Early Release Frequency (HLERF) Sequences Top Gate
U2_L2LLERFP	Early Containment Failure, with Low RCS Pressure at time of Vessel Failure, Large Early Release Frequency (LLERF) Sequences Top Gate
U2_L2ILERFP	Isolation Failure Large Early Release Frequency (ILERF) Sequences Top Gate
U2_L2LSERFP	Detonation or Early Containment Failure LERF, Small Early Release Frequency (LSERF) Sequences Top Gate
U2_L2BLERFP	Bypass Large Early Release Frequency (BLERF) Sequences Top Gate
U2_L2LATETOP	Late Containment Failure Top Gate
U2_L2SERFSAMA2014	Small Early Release Frequency (SERF) Top Gate
U2_L2SLERF	SGTR LARGE EARLY RELEASE FREQUENCY (SLERF) SEQUENCES
U2_L2INTACTTOP	Containment Intact Sequences Top Gate

The LERF and SERF end state subcategories required modifications in order to properly assign the new release category definitions. The modifications are described below.

Non-Bypass LERF End State Modifications

LSERF are SERF sequences where there is no vessel breach at time of core melt but detonation or containment failure occurs. These sequences are changed from the SERF end state in the Record Model to the Non-Bypass LERF end state in the SAMA Model because their consequences are larger than the SERF category consequences.

SERF End State Modifications

As described for the Non-Bypass LERF end state, the LSERF sequences are moved from the SERF end state to the Non-Bypass LERF end state.

2. Remove SLOCA Sequences 004, 014, and 022 from PDS BIN 2.

   The SLOCA Event Tree Sequences 004 (Gate U2_SLOCA-004), 014 (Gate U2_SLOCA-014), and 022 (Gate U2_SLOCA-022) do not lead to core damage due to success of low pressure recirculation. These sequences are not included in the CDF model (Figure 6.4-3 of Accident Sequence Analysis Notebook, Calculation MDN-000-999-2008-0141, Rev. 2), but were inadvertently included in PDS BIN 2. These sequences were therefore removed from PDS BIN 2 in the SAMA model.

3. Addition of missing ISLOCA CDF sequences ISLM-004, 013, and 016 to the fault tree gate for BIN 4.

   ISLOCA sequences ISLM-004, 013, and 016 were identified for Bin 4 (Figure 6.4-10 of Accident Sequence Analysis Notebook, Rev. 2 of Calculation MDN-000-999-2008-0141, and Table 9-6 of Level 2 Analysis Notebook, Rev. 3 of Calculation MDN-000-999-2008-0148) but were included not in the fault tree model. There were added to Level 2 Gate U2_L2BIN-4 in the SAMA model.

4. Correct selected split fraction values for early containment failure events to remove contributions from rocket mode and ex-vessel steam explosions in sequences involving core damage arrest prior to vessel breach; i.e., no vessel breach occurs.

   For sequences in which the vessel is not breached at the time of core damage (i.e., core damage arrested prior to vessel breach due to offsite power recovery following a SBO event), the rocket mode and ex-vessel steam explosion failure modes cannot occur because there is no water in the reactor cavity. Therefore, the following split fraction values for the early containment failure events are corrected to reflect contribution from hydrogen burn only (to be consistent with the values listed in Table 6-35 of Level 2 Analysis Notebook, Calculation MDN-000-999-2008-0148, Rev. 3).

Branch	Split Fraction	SBO Value
CFE11	U2_L2_CFE11	0.00
No CFE11	U2_L2_NOTCFE11	1.00
CFE12	U2_L2_CFE12	0.07
No CFE12	U2_L2_NOTCFE12	0.93
CFE15	U2_L2_CFE15	0.00
No CFE15	U2_L2_NOTCFE15	1.00
CFE16	U2_L2_CFE16	0.04
No CFE16	U2_L2_NOTCFE16	0.96

No changes were made to the Level 2 event trees. The results of the above changes to the Level 2 Release Category frequencies are shown in Table 3.

4.3 Quantitative Strategy for External Events

The SAMA PRA model (i.e. WBN_U1_U2_SAMA) incorporates internal initiating events, including those originating from internal floods. External events were evaluated in the Individual Plant Evaluation of External Events (IPEEE) using seismic margins and the Electric Power Research Institute Fire Induced Vulnerability Evaluation (FIVE) methodologies. No vulnerabilities to external events were identified.

A multiplication factor of 2 is applied to the internal event results to account for the contribution to core damage from fire and other external events. The multiplication factor of two is based on the guidance from NEI 05-01 for developing the external events multiplier [for plants that used Fire-Induced Vulnerability Evaluation (FIVE) and Seismic Margin Assessment (SMA) for fire and seismic analyses, respectively) and based on a review of the SAMA submittals for a number of 4-loop Westinghouse plants including Wolf Creek (WCNOC 2006), Vogtle (SNC 2007), Catawba (DUKE 2001), McGuire [2001a], and D. C. Cook (AEP 2003). The first two were chosen because they represent recent applications while the latter three, while older applications, were chosen because they represent ice condenser plants.

Additionally, while the dominant core damage sequences will be different for seismic, fire and other external events, overall the contributions to release categories should be bounded by the internal events PRA sequences. For example, it is not expected that containment bypass sequences (SG tube ruptures and interfacing system LOCAs) will be dominant release sequences for fire and seismic initiators since these tend to result in loss of power to operate and control plant equipment. Also, RCP seal LOCAs are a significant contributor to fire risk and SAMAs directed at maintaining RCP seal cooling and are already considered for internal initiating events.

4.4 PRA Model Quality

The Watts Bar dual unit PRA model was peer reviewed by the Westinghouse Owner's Group (WOG) and all A and B level F&Os were resolved in the current model of record used as the starting point for this SAMA re-evaluation. The Level 2 model changes made for this analysis were also independently reviewed. Details of these changes are provided in Section 4.2.

4.5 WBN Unit 2 SAMA Base Model Quantification Results

The Unit 2 core damage frequency result for the modified SAMA Model is 1.43×10^{-5}, and the release category results are shown in Table 3. Table 4 identifies key scenarios for each of the release categories.

4.6 WBN Unit 2 Level 3 SAMA Model

The Level 3 portion of this updated SAMA analysis adopted the approach from the previous SAMA for Watts Bar Unit 2 in its entirety (in particular the SAMDA submitted to NRC on September 16, 2011). Briefly, the WinMACCS computer code, Version 3.6.0 (NRC 2007) was used to perform probabilistic analyses of radiological impacts. The WinMACCS code is the current version of the MACCS2 code. A detailed description of the MACCS model is provided in NUREG/CR-4691 (NRC 1990). The enhancements incorporated in MACCS2 are described in the MACCS2 User's Guide (NRC 1998).

Site-specific input parameters were used for population distribution, economy and agricultural data, radionuclide release, evacuation, and meteorology. The same representative scenarios for each release category were also adopted.

4.6.1 Analysis

Site-specific input parameters formed the basis for the analysis, including population distribution, economic parameters, and agricultural product. Plant-specific release data included nuclide release quantities, release timing and duration, release energy (thermal content), release frequency, and release category; i.e., early release, late release. The behavior of the population during a release (evacuation parameters) was based on declaration of a general emergency and the WBN Plant emergency planning zone evacuation time.

Generic input parameters given with the MACCS2 Sample Problem A, which includes the data used in NUREG 1150 (NRC 1989), supplemented the site-specific data.

This data, in combination with site-specific meteorology, were used to simulate the probability distribution of impact risks (exposure and economic cost) to the surrounding 80-kilometer (within 50 miles) population.

4.6.2 Population Distribution

The population surrounding the WBN Plant site was estimated for the year 2040. The distribution was given in terms of the population at 10 distances, ranging from 0 miles to 50 miles from the plant, in the direction of each of the 16 compass points (north,

north-northeast, northeast, etc.), a total of 160 segments. The population projections were determined using 2000 census population data. A map was prepared displaying county and census tract boundaries for all counties partly or totally within the 50-mile boundary. County population data for 2000 were allocated to the appropriate sectors, using census tracts to the extent feasible. For segments near the plant site, especially within 5 miles, aerial photos and TVA staff knowledge of the area were also used. The segments populations were projected for the year 2040 using growth rates from county population projections. The total projected population within 50 miles of the site was estimated to be 1,523,390 (see Table 5).

4.6.3 Economy and Agriculture Data

Agriculture production information was generated using SecPop2000 Version 3.13.1. SecPop2000 uses data from the 2000 Census and the 2002 Census of Agriculture to determine the population, land fraction, and a region index in each sector of the radial grid, centered on the plant, that is used to compute the results in WinMACCS. The region index is keyed to another section of the site file that provides economic data for each region. The dollar values were increased by a factor of 1.15254 to account for inflation from 2002 to 2007. This value was determined by using the United States Bureau of Labor Statistics CPI Inflation Calculator (BLS 2010). The population generated by SecPop2000 was replaced by the population information supplied by TVA, as documented in the previous section.

SecPop2000 provides only generic values for the number of watersheds, the watershed index, the watershed definition, and the crop season and share sections of the site file. The watershed index and watershed definition together define the drainage areas that are averaged to determine doses received from crops. Because open water does not contribute to this dose, a second watershed was added with no contribution to dose from crops, all sectors with a land fraction of 0.00 were changed from Watershed 1 to Watershed 2.

The crop share is the fraction of the area that is devoted to each of seven crop types: pasture, stored forage, grains, green leafy vegetables, legumes and seeds, roots and tubers, and others. Because most of the deposition (and related measures such as interdicted area) are within 10 miles of the plant, the crop share was computed for the closest 10 miles to the plant. This was estimated by visual examination to be 50% within Rhea County, 40% within Meigs County, and 10% within McMinn County. The United States Department of Agriculture 2007 Agricultural Census (USDA 2009) Tables 8, 25, and 30 were used to estimate the total area in each county in farms, and the area in each county used for each of the crops listed in these tables, if any. The crops were grouped into the seven crop categories used by the code, and each of these areas was divided by the total farm area in that county to determine the fraction of crop area in that county used for each food category. These fractions were weighted by the estimated area within ten mile of Watts Bar in each county, and used to replace the generic SecPop2000 values in the site file. The crop season for all crops was taken from an online source (CD 2010) as March 25 through November 5, or Day 84 through Day 309.

4.6.4 Radionuclide Release

Core damage sequences that lead to containment failure (failure mode defined as bypass, early, and late) and release of radioactive materials to the environment are

considered in this section. The core damage sequences from the Level 1 PRA are binned into plant damage states based on similar characteristics that control the accident progression following core damage and the timing and magnitude of fission product releases to the environment. The possible fission product releases are then binned into release categories that represent similar release magnitudes and timing. The Level 2 release categories are defined as conditional probabilities that, when combined with the plant damage state frequencies, yield release frequencies. The determination of the release characteristics for each release category is based on representative accident scenarios that reflect the post core damage behavior for the dominant sequence or sequences within a plant damage state. These core damage accident scenarios then become the major contributors to the release level categories associated with each of the containment failure modes.

The WBN 2 Level 2 model is represented by a large containment event tree that is based on the NUREG-1150 Level 2 assessment for Sequoyah. The event tree nodes and split fractions were reviewed to assure that the consequences, in terms of release frequencies, would be larger than expected with an updated Level 2 model. This will maximize the consequences, which in turn would maximize the economic benefits of the candidate SAMAs.

The release categories that are used in the SAMA assessment and examples of various accident scenarios leading to containment failure and/or bypass are presented below. These release categories represent a consolidation of release categories from the WBN 2 Level 2 PRA. The consolidation was performed to simplify the SAMA assessment by choosing the most severe release characteristics from the WBN 2 Level 2 PRA for each of the four SAMA release categories; i.e. excluding Release Category V for intact containment. This provides the largest potential benefit in terms of fission product release prevention or mitigation for the alternatives in the Phase I assessment.

- Release Category I results from a reactor vessel breach with a non-bypass large early containment failure.
- Release Category II results from a reactor vessel breach with containment bypass.
- Release Category III results from a reactor vessel breach with late containment failure.
- Release Category IV results from a reactor vessel breach with a small pre-existing hole in the containment.
- Release Category V, the remaining core damage sequences do not challenge the containment and result in an intact containment.

Table 4 provides a representation of the dominant accident scenarios that lead to each release category and the likelihood of their occurrence.

Table 6 shows the equilibrium reactor core radionuclide inventory at the time of a reactor trip. These are unchanged from the previous analysis.

Table 7 provides important information on time to core damage, containment failure, and release duration.

Table 8 shows the fission product release fractions associated with each of the release categories. The source terms calculated in NUREG/CR-4551 Volume 5 did not include any release categories analogous to Release Category IV, the small pre-existing hole. Because of this, the methodology described in that document, the accompanying data, and the assumption that the maximum release rate was 25 times the design leak rate were used to produce a source term for this release category in a manner similar to that used in NUREG/CR-4551. The assumed release rate that is 25 times the design release rate comes from a Westinghouse report (Westinghouse, 2004).

4.6.5 Evacuation

Evacuation data, including delay time before evacuation, area evacuated, average evacuation speed, and travel distance, was obtained from the *Tennessee Multi-Jurisdictional Radiological Emergency Response Plan for the Watts Bar Nuclear Plant*, Annex H (TVA 2006). For this analysis, the evacuation and sheltering models are the same as those in the previous SAMA analysis for Watts Bar Unit 2; i.e., see Table 9 for the evacuation times used.

4.6.6 Meteorology

Annual onsite meteorology data sets from 2001 through 2005 were used to prepare the sequential hourly data (8,760 hours) required for use in WinMACCS. This part of the analysis is unchanged from the previous SAMA analysis for Watts Bar Unit 2.

4.7 Severe Accident Risk Results

Table 10a summarizes the consequences for the four key release categories (without any SAMAs implemented), assuming mean meteorological conditions, within an 80-kilometer (50-mile) radius of the reactor site. The analysis assumes that a site emergency would have been declared early in the core damage accident sequence and that all nonessential site personnel would have evacuated the site in accordance with site emergency procedures before any radiological releases to the environment occurred. In addition, emergency action guidelines would be implemented to initiate evacuation of the public within 16.1 kilometers (10 miles) of the plant. The WinMACCS computer code models the evacuation sequence to estimate the dose to the general population within 80 kilometers (50 miles) of the accident.

The frequency of each key release category is given in Table 10b. Table 10b shows the population dose risks (accident consequence multiplied by the release category frequency) for each accident release category. These frequencies are based on the WBN_U1_U2_SAMA PRA model.

Overall, the dose risk results are small. Completion and operation of WBN Unit 2 would not significantly change the risks evaluated for WBN Unit 1 because the principal change to Unit 1 accident mitigation capabilities is the loss of the Unit 2 Component Cooling Water System (WBN System Designation) (CCS) 2A-A pump as a backup for Unit 1. The Unit 2 CCS 2A-A pump is currently only used for spent fuel pool cooling and so its assignment to the A train for Unit 2 does not affect the risk at Unit 1. CCS 1A-A

and 1B-B pumps continue to supply the A train header for Unit 1. CCS 2A-A and 2B-B pumps will supply the A train header for Unit 2. The CCS C-S pump will supply the common B header for Units 1 and 2. Additionally, the CCS 1B-B or 2B-B pumps may be aligned as a backup to supply water to the common B header. Therefore, the changes in CCS system alignment due to operation of Unit 2 have no risk significance for Unit 1. Changes to other systems, including shared systems, were found to have no significant impact on the Unit 1 risks. This is consistent with the conclusions of NRC's Generic Environmental Impact Statement for License Renewal of Nuclear Plants (GEIS) (NRC 1996). Accidents that could affect multiunit sites are typically initiated by external events.

Severe accidents initiated by external events such as tornadoes, floods, earthquakes, and fires traditionally have not been discussed in quantitative terms in final environmental statements and were not considered in the NRC's Generic Environmental Impact Statement for License Renewal of Nuclear Plants (NRC 1996). In the GEIS, however, NRC staff did evaluate existing impact assessments performed by NRC and the industry at 44 nuclear plants in the United States and concluded that the risk from beyond-design-basis earthquakes at existing nuclear power plants is small. Additionally, the staff concluded that the risks from other external events are adequately addressed by a generic consideration of internally initiated severe accidents. To account for the possible contribution of fires and other external events to the core damage frequency at Watts Bar Unit 2, the internal events core damage frequency was doubled. Thus, all candidate SAMAs are evaluated using the averted costs based on doubling the core damage frequency from the internal events PRA analysis.

5. Cost of Severe Accident Risk/Maximum Benefit

This section explains how to monetize the severe accident consequences based on the formulas in the Nuclear Energy Institute's SAMA Analysis Guidance Document (NEI 2005). This analysis is also used to establish the maximum benefit that could be achieved if all risk for reactor operation were eliminated; i.e., accident consequences without SAMA implementation.

5.1 Offsite Exposure Cost

The annual offsite exposure risk was converted to dollars using the conversion factor of $2,000 per person-rem, and discounted to present value using the following standard formula:

$$W_{pha} = C * Z_{pha} \tag{5-1}$$

Where:

W_{pha} = monetary value of public health risk after discounting ($)

C = $[1-\exp(-rt_f)]/r$ (years)

t_f = years remaining until end of facility life = 40 years

r = real discount rate (as fraction) = 0.07 per year

Z_{pha} = monetary value of public health (accident) risk per year before discounting ($ per year)

The Level 3 analysis showed a baseline annual total offsite population dose risk of about 14.8 person-rem. The calculated value for C using 40 years and a 7 percent discount rate is approximately 13.42 years. Calculating the discounted monetary equivalent of accident dose-risk involves multiplying the dose (person-rem per year) by $2,000 and by the C value (13.42). In this calculation, the delay until the initial time of operation is conservatively assumed to be zero. The calculated offsite exposure cost is estimated to be $398,309.

5.2 Offsite Economic Cost

The annual off-site economic risk was discounted to present value using the following standard formula:

$$W_{ea} = C * Z_{ea} \tag{5-2}$$

Where:

W_{ea} = monetary value of economic risk after discounting

C = $[1-\exp(-rt_f)]/r$ (years)

t_f = years remaining until end of facility life = 40 years

r = real discount rate (as fraction) = 0.07 per year

Z_{ea} = monetary value of economic (accident) risk per year before discounting ($ per year)

The Level 3 analysis showed a baseline annual offsite economic risk of $40,611. Calculated values for offsite economic costs caused by severe accidents must be discounted to present value. This is performed in the same manner as for public health risks and uses the same C value. The resulting value is $544,879.

5.3 Onsite Exposure Cost

The values for onsite (occupational) exposure consist of "immediate dose" and "long-term dose." The best estimate value provided in NUREG/BR-0184 (NRC 1997) for immediate occupational dose is 3,300 person-rem/event, and long-term occupational dose is 20,000 person-rem (over a 10-year clean-up period). The following equations are used to calculate monetary equivalents.

5.3.1 Immediate Dose

$$W_{IO} = R * F * D_{IO} * C \quad (5\text{-}3)$$

Where:

W_{IO} = monetary value of accident risk avoided due to immediate doses, after discounting

R = monetary equivalent of unit dose ($2,000 per person-rem)

F = accident frequency (i.e., CDF = 1.43 x 10-5 events per year)

D_{IO} = immediate occupational dose (3,300 person-rem per accident [NRC estimate])

C = $[1 - \exp(-rt_f)]/r$ (years)

r = real discount rate (0.07 per year)

t_f = years remaining until end of facility life (40 years).

The best estimate of the immediate dose cost for WBN Unit 2 is:

W_{IO}= 2,000 *1.43 x 10^{-5} * 3,300 * {[1 - exp(-0.07 * 40)]/0.07}

= $1,265

5.3.2 Long-Term Dose

$$W_{LTO} = R * F * D_{LTO} * C * \{[1 - \exp(-rm)]/rm\} \quad (5\text{-}4)$$

Where:

W_{LTO} = monetary value of accident risk for long-term onsite doses, after discounting, ($)

R = monetary equivalent of unit dose ($2,000 per person-rem)

F = accident frequency (i.e., CDF = 1.43 x 10^{-5} events per year)

D_{LTO} = long-term dose [20,000 person-rem per accident (NRC estimate)]

C = [1 - exp(-rtf)]/r (years)

r = real discount rate (0.07 per year)

t_f = years remaining until end of facility life (40 years).

m = years over which long-term doses accrue (as long as 10 years)

Using values defined for immediate dose, the best estimate of the long-term dose is:

W_{LTO} = 2,000 * 1.43 x 10^{-5} * 20,000 *{[1 - exp(-0.07 x 40)]/0.07} * {[1 -exp(-0.07 * 10)]/0.07 * 10}

= $5,514

5.3.3 Total Onsite Exposure

The total occupational exposure is then calculated by combining Equations 5-3 and 5-4 above. The total accident related onsite (occupational) exposure risk (W_O) is:

$W_O = W_{IO} + W_{LTO}$ = ($1,265 + $5,514) = $6,779

5.4 Onsite Economic Cost

Onsite economic cost includes cleanup and decontamination cost, and either replacement power cost or repair and refurbishment cost.

5.4.1 Onsite Cleanup and Decontamination Cost

The total undiscounted cost of a single event in constant year dollars (C_{CD}) that NRC provides for cleanup and decontamination is $1.5 billion (NRC 1997). The net present value of a single event is calculated as follows:

$$PV_{CD} = [C_{CD}/m] * \{[1-\exp(-rm)]/r\} \quad (5\text{-}5)$$

Where:

PV_{CD} = net present value of a single event ($)

C_{CD} = total undiscounted cost for a single accident in constant-year dollars

r = real discount rate (0.07)

m = years required to return site to a pre-accident state

The resulting net present value of a single event is:

$$PV_{CD} = [\$1.5 \times 10^9 / 10 \text{ years}] * \{[1-\exp(-0.07*10)]/0.07\}$$

$$= \$1.08 \times 10^9$$

The NEI 05-01 uses the following equation to integrate the net present value over the average number of remaining service years:

$$U_{CD} = PV_{CD} * C \quad (5\text{-}6)$$

Where:

U_{CD} = total cost of cleanup and decontamination over the analysis period ($-years)

PV_{CD} = net present value of a single event ($1.08 x 10^9)

C = $[1 - \exp(-rt_f)]/r$ (years)

r = real discount rate (0.07 per year)

t_f = years remaining until end of facility life (40 years)

The resulting net present value of cleanup integrated over the license term is

$$U_{CD} = \$1.08 \times 10^9 * \{[1-\exp(-0.07*40)]/0.07\}$$

$$= 1.45 \times 10^{10} \text{ \$-years (per single event)}$$

5.4.2 Replacement Power Cost

Long-term replacement power costs were determined following NRC methodology in NUREG/BR-0184 (NRC 1997). The net present value of replacement power for a single event, PV_{RP}, was determined using the following equation:

$$PV_{RP} = [B/r] * [1 - \exp(-rt_f)]^2 \quad (5\text{-}7)$$

Where:

PV_{RP} = net present value of replacement power for a single event, ($)

r = real discount rate (0.07)

t_f = 40 years (license period)

B = a constant representing a string of replacement power costs that occur over the lifetime of a reactor after an event (for a 910MWe "generic" reactor, NUREG/BR-0184 uses a value of $1.2E+8) ($/yr)

= $\$1.2 \times 10^8 * 1160/910 = \$1.53 \times 10^8$ for WBN power level of 1160 MWe

The resulting net present value of a single event is:

$$PV_{RP} = [\$1.53 \times 10^8/0.07] * [1 - \exp(-0.07*40)]^2$$

$$= \$1.93 \times 10^9$$

To attain a summation of the single-event costs over the entire license period, the following equation is used:

$$U_{RP} = [PV_{RP} /r] * [1 - \exp(-rt_f)]^2 \quad (5\text{-}8)$$

Where:

U_{RP} = net present value of replacement power over life of facility ($-year)

r = real discount rate (0.07)

t_f = 40 years (license period)

The resulting net present value of replacement power integrated over the license term is

$$U_{RP} = [\$1.93 \times 10^9/0.07] * [1-\exp(-0.07*40)]2$$

$$= 2.43 \times 10^{10} \text{ \$-years}$$

5.4.3 Total Onsite Economic Cost

The total onsite economic costs are calculated by summing cleanup/decontamination costs and replacement power costs, and multiplying this value by the internal events CDF.

Onsite economic cost = (1.45 x 10^{10} $-years + 2.43 x 10^{10} $-years) * 1.43 x 10^{-5}/year

= $553,766

5.5 Total Cost of Severe Accident Risk/Maximum Benefit

The sum of the baseline costs is as follows:

Offsite Exposure Cost = $398,309

Offsite Economic Cost = $544,879

Onsite Exposure Cost = $6,779

Onsite Economic Cost = $553,766

Total Cost = $1,503,733

The total cost risk represents the maximum averted cost risk if all risk were eliminated. The MACR ($1,503,733) is based on at-power internal events contributions.

The internal event MACR is doubled to account for external events contributions. The resulting modified MACR (MMACR) is $3,007,467 and was used in the Phase I screening process.

Compared to the MACR calculated in the RAI response submittal dated September 16, 2011, the total base cost of severe accident has decreased from $1,930,303 to $1,503,733 due to decreases in the core damage frequency and in the frequencies of all four release categories used in the SAMA analysis. The table below shows a comparison of the offsite and onsite exposure costs as well as the offsite and onsite economic costs.

Cost Category	September 16, 2011 SAMA Results in RAI Response Submittal	March 2014 SAMA Results
Off-Site Exposure Cost	$535,803	$398,309
Off-Site Economic Cost	$720,324	$544,879
On-Site Exposure Cost	$8,153	$6,779
On-Site Economic Cost	$666,023	$553,766
Total Base Severe Accident Cost	$1,930,303	$1,503,733
Base Cost with External Events Multiplier 2.0	**$3,860,606**	**$3,007,467**
Base Cost with External Events Multiplier 2.28	**$4,401,090**	**$3,428,512**

6. SAMAs Re-Evaluated

The following SAMAs were re-evaluated in this analysis:

- SAMA 50 – Enhance Loss of CCS Procedure
- SAMA 55 – Independent Seal Injection System with Dedicated Diesel Generator
- SAMA 56 – Independent Seal Injection System without Dedicated Diesel Generator
- SAMA 93 – Hardened, Unfiltered Containment Vent
- SAMA 215 – Backup Thermal Barrier Cooling
- SAMA 226 – Self-Powered, Backup Charging Pump
- SAMA 242 – Permanent, Dedicated DG for the Charging Pump

6.1 SAMA 50 – Enhance Loss of CCS Procedure

Enhance loss of CCS procedure to underscore RCS cooldown for small/medium LOCA response following a loss of RCP seal cooling event. Upon receipt of any RCP Seal No. 1 outlet temperature high alarm, AOI-15 and 24 require an RCS cooldown after isolation of the CCS path to the RCP thermal barrier and isolation of RCP seal injection. This order of actions is deemed appropriate for overall plant stabilization following a loss of CCS.

6.2 SAMA 55 – Independent Seal Injection System with Dedicated Diesel Generator

This alternative would add a 40 gpm, motor driven seal injection pump powered by a dedicated diesel generator and taking suction from the refueling water storage tank (RWST). This independent system would incorporate direct current (DC) motor-operated valves (MOV) and controls, also powered by the diesel generator. The system function would be limited to providing seal injection flow only and would not have the capacity to provide RCS charging / makeup flow. The seal return and letdown flows would be routed back to the RWST to mix with the cooler RWST inventory before recirculation. Controls would be provided in the Main Control Room (MCR) to allow system initiation within 13 minutes following a Station Blackout or loss of seal injection/seal cooling event. The seal injection pump could be located in the abandoned reciprocating charging pump room on Auxiliary Building (AB) Elev 692 and the supporting diesel generator could be located on the Turbine Building (TB) Elev 755 deck near the Moisture Separator Reheaters (MSR).

6.3 SAMA 56 – Independent Seal Injection System without Dedicated Diesel Generator

SAMA 56 is similar to SAMA 55 except that an independent turbine driven seal injection pump would be implemented, rather than the diesel generator and motor drive for SAMA 55. This alternative would require steam supply and exhaust lines for the turbine and DC MOVs, which would reduce the SBO battery capacity. The abandoned reciprocating charging pump room on Auxiliary Building Elev 692 would be the location

for this pump and turbine. Its proximity to the Turbine Driven Auxiliary Feedwater (TDAFW) pump room would allow steam to be supplied through a branch line from the TDAFW turbine steam supply. Likewise, the exhaust could be routed back to the TDAFW exhaust stack.

6.4 SAMA 93 – Hardened, Unfiltered Containment Vent

An extended SBO event with a consequential RCP Seal LOCA will ultimately uncover the core and cause fuel damage. This will increase the amount of hydrogen produced by the metal-water reaction and will directly raise the temperature and pressure inside the containment. SAMA 93 would install a hardened, unfiltered containment vent. This vent would be routed to an elevated release point, and would include pneumatically operated butterfly isolation valves inside and outside the containment boundary. These valves would be operated with nitrogen gas supplied by the backup compressed nitrogen storage system and controlled with DC solenoid valves. The nitrogen system would provide sufficient gas pressure for at least 24 hours of periodic venting. The venting would be initiated at or near containment failure pressure of 47 psig to preclude containment failure and to limit hydrogen concentration less than the deflagration limit.

6.5 SAMA 215 – Backup Thermal Barrier Cooling

The concept for SAMA 215 is to provide backup thermal barrier cooling via permanent crosstie to the discharge of the TDAFW pump. A new 3” diameter pipe would be connected to the 6” diameter TDAFW pump discharge line just upstream of the first steam generator supply tee. This pipe would be routed approximately 150 ft. to the Component Cooling Water System and would be connected to the common discharge of the thermal barrier booster pumps, upstream of the outboard containment isolation valve. DC MOVs and controls would be incorporated to operate the crosstie from the MCR within 13 minutes following a SBO or loss of seal injection/seal cooling event. The design would also incorporate a pressure breakdown orifice between the TDAFW connection and the first new DC MOV to reduce the TDAFW discharge pressure to less than the design pressure of the CCS piping.

6.6 SAMA 226 – Self-Powered, Backup Charging Pump

This alternative is similar to SAMA 55, but it would add a turbine driven charging pump with the full capacity to provide RCS makeup plus seal injection flow. The pump and turbine would be located in the abandoned reciprocating charging pump room on Auxiliary Building Elev 692 and the supporting diesel generator could be located on the Turbine Building Elev 755 deck near the MSRs. Steam would be supplied to the turbine from a branch line off the TDAFW pump Main Steam supply line. The turbine exhaust would be routed back to the TDAFW turbine exhaust stack. Cooling water for RCS letdown and seal return flow would be provided by a small diesel generator and motor driven pump taking suction from the Condenser Cooling Water inlet to the condenser and discharging to the 16” CCS miscellaneous equipment header. After passing through the Letdown and Seal Water heat exchangers, the cooling water would be discharged to the lake through the CCS to Emergency Raw Cooling Water (WBN System Designation) (ERCW) crosstie.

6.7 SAMA 242 – Permanent, Dedicated DG for the Charging Pump

The existing centrifugal charging pumps (CCP) are rated at 150 gpm and 5800 ft. of head. This equates to 720 shaft horsepower or 537 kW. SAMA 242 would provide a permanent, dedicated diesel generator system with cabling, switchgear and controls to allow the system to be placed in service from the MCR within 13 minutes following a SBO or loss of seal injection/seal cooling event. The switchgear would be designed to allow either CCP to be selected and powered from the new dedicated diesel. The diesel would be located at grade level, outside the Turbine Building in a weather proof enclosure. The skid mounted DG would include a self-contained fuel tank with capacity to support 24 hours of full load service.

7. Phase I SAMA Analysis

The purpose of the Phase I analysis is to use high-level knowledge of the plant and SAMAs to preclude the need to perform detailed cost-benefit analyses on them. As in the earlier SAMA analysis, the following screening criteria were again used:

- Not Applicable: If a proposed SAMA does not apply to the WBN design, it is not retained.
- Already Implemented: If the SAMA or equivalent was previously implemented and is accounted for in the PRA model, it is not retained.
- Combined with Another SAMA: If a SAMA is similar in nature and can be combined with another SAMA to develop a more comprehensive or plant specific SAMA, only the combined SAMA is further evaluated.
- Excessive Implementation Cost: If the estimated cost of implementation is greater than the modified Maximum Averted Cost-Risk, the SAMA cannot be cost beneficial and is screened from further analysis.
- Very Low Benefit: If the SAMA is related to a non-risk significant system which is known to have negligible impact on the risk profile, it is not retained

Table 11 also provides a description of how each SAMA was or can be dispositioned in Phase I for this study. The SAMAs were compared to the new MACR of $3.01 million to determine if they can be dispositioned as “Excessive Implementation Cost”. Those SAMAs that required a more detailed cost-benefit analysis are retained for Phase II and evaluated in Section 8.

8. Phase II SAMA Analysis

The purpose of the Phase II analysis is to perform a cost-benefit analysis on the SAMAs that were not screened out in Phase I. However, although some of the SAMAs can potentially be screened out in Phase I, they are still retained for further detailed analysis in Phase II. The risk benefit for each of the seven SAMAs was analyzed using the PRA model described in Section 4. The cost of implementation of six of the SAMAs was estimated in "Watts Bar Nuclear Plant Unit 2 SAMA Detailed Design Cost Study for RCP Seal LOCA Event" to identify those SAMAs that are potentially cost beneficial. The cost of implementation for SAMA 50 was estimated by TVA separately. The results of the Phase II analysis are shown in Table 12 and each SAMA is described below.

SAMA 50: Enhance Loss of CCS Procedure

Description: This is to enhance the loss of CCS procedure to underscore RCS cooldown for small/medium LOCA response following a loss of RCP seal cooling event. Upon receipt of any RCP seal No. 1 outlet temperature high alarm, AOI-15 and 24 already require an RCS cooldown after isolation of the CCS path to the RCP thermal barrier and isolation of RCP seal injection. This order of actions is deemed appropriate for overall plant stabilization following a loss of CCS.

Risk Benefit: In the absence of RCP seal injection and thermal barrier cooling, the RCP seals will fail in approximately 13 minutes and a seal LOCA cannot be prevented because of the rapid progression of the seal leak. Nevertheless, an improved loss of CCS procedure could enhance the mitigation efforts (cooldown/depressurization) in response to small and medium LOCA scenarios following a loss of RCP seal cooling event. It is, however, possible that the enhanced procedure may not reduce the risk significantly; i.e., may only have minimal risk improvement. Therefore, the intent of this SAMA is to determine if the actual benefit is minimal. In this evaluation, the risk benefit was bounded by calculating the change with all three operator failure probabilities for AFWOP1 (depressurize/cooldown to low pressure injection following MLOCA), AFWOP2 (depressurize/cooldown to LPI following small break LOCA with failure of HPI), and AFWOP3 (depressurize/cooldown to low pressure injection following small LOCA with failure of high pressure makeup) reduced to 1.0E-03. The resulting CDF is 1.41E-5. Calculating the averted risk cost relative to the base case using the method described in Section 5 results in a net benefit of $16,789. It must be noted that this estimated net benefit is very conservative because the basic HEP values for these operator actions were reduced, but the correction factors applied to the associated core damage and release category cutsets were not increased. These correction factors were applied to ensure that the joint failure probabilities of more than one operator failure action for the corresponding minimum cutsets are not below a reasonable value. As the basic HEP values decrease, the values of these correction factors tend to increase. Without revising the values of these correction factors, the net benefit estimated would be greater than the realistic value.

Cost: The TVA estimated cost of this SAMA is $31,675.

SAMA 55: Install an Independent Seal Injection System with Dedicated Diesel Generator

Description: This alternative would add a 40 gpm, motor driven seal injection pump powered by a dedicated diesel generator and taking suction from the RWST. A diesel generator for the pump was added to allow its use when electrical power was not available. The system would be limited to providing seal injection flow only. The seal return and letdown flows would be routed back to the RWST to mix with the cooler RWST inventory before recirculation. Since the dedicated pump would need to operate from the control room to mitigate seal damage within the required time frame, controls would be provided in the MCR to allow system initiation within 13 minutes following an SBO or loss of seal injection/seal cooling event. The seal injection pump could be located in the abandoned reciprocating charging pump room in the AB and the dedicated diesel generator could be located in the TB.

Risk Benefit: The risk benefit was bounded by calculating the change due to the addition of an independent motor-driven seal injection system with a dedicated diesel generator as a backup to the CCPs. Failure of the independent seal injection system is represented by failures of a motor-operated seal injection pump, a diesel generator, four motor-operated valves (which must operate to change position), and an operator action to align and initiate this independent seal injection system from the MCR within 13 minutes. The risk model was revised by adding a fault tree gate (which models failures of the additional equipment and operator action) ANDed with the two CCPs. A very conservative Human Error Probability (HEP) of 1.0E-03 was used considering the very short time window within which this operator action must be completed (to prevent degradation of the RCP seals) among all of the other immediate, post-trip actions following a SBO event. Normal conditions are applied otherwise. The resulting CDF is 9.36E-6. Calculating the averted risk cost relative to the base case using the method described in Section 5 results in a net benefit of $1,060,052.

Cost: The TVA estimated cost of this SAMA is $5,438,754.

SAMA 56: Install an Independent Seal Injection System, without Dedicated Diesel Generator

Description: This alternative would add an independent turbine-driven seal injection pump, along with steam supply and exhaust lines for the turbine and DC MOVs. This pump and turbine would be installed in the abandoned reciprocating charging pump room in AB. The required steam would be supplied through a branch line from the TDAFW turbine steam supply. The steam exhaust would be routed back to the TDAFW exhaust stack. Suction piping, discharge piping, and power are available in the PD pump room. The current PD pump would be dismantled.

Risk Benefit: The risk benefit was bounded by calculating the change due to the addition of an independent, turbine-driven seal injection system as a backup to the CCPs. Failure of the independent, turbine-driven seal injection system is represented by failures of a turbine driven seal injection pump, five motor-operated valves (which must operate to change position), and an operator action to align and initiate this independent seal injection system from the MCR within 13 minutes. The risk model was revised by adding a fault tree gate (which models failures of the additional equipment and operator action) ANDed with the two CCPs. A very conservative HEP of 1.0E-03 was used

considering the very short time window within which this operator action must be completed (to prevent degradation of the RCP seals) among all of the other immediate, post-trip actions following a SBO event. Normal conditions are applied otherwise. The resulting CDF is 9.32E-6. Calculating the averted risk cost relative to the base case using the method described in Section 5 results in a net benefit of $1,067,736.

Cost: The TVA estimated cost of this SAMA is $5,213,021.

SAMA 93: Install an Unfiltered, Hardened Containment Vent to Eliminate the Containment Overpressure Failure

Description: The potential enhancement of this SAMA is to increase decay heat removal capability for non-Anticipated Transient Without Scram (ATWS) events, without scrubbing released fission products. This alternative would allow the operators to reduce pressure in the containment during a severe accident in order to prevent late containment overpressure; i.e., it is for the prevention of containment failure in late release sequences.

Risk Benefit: The WBN Unit 2 CAFTA model adopts an approach similar to NUREG/CR-4550, Volume 5, Rev. 1, Part 1, which evaluates the frequency of core damage independent of containment heat removal. Then, the subsequent loss of containment heat removal only affects the assigned release category, but not the frequency of core damage. Due to the very low frequencies of sequences involving successful core cooling initially but with failure of containment heat removal leading to eventual core damage following containment failure, the additional benefit provided by installing containment venting for the loss of containment heat removal scenarios to potentially reduce the core damage frequency is very small.

Therefore, if implemented, this alternative would reduce the Level 2 release consequences of the LATE Release Category but would not affect the core damage frequency. The mix of sequences contributing to the late release category would also not be changed by implementing this alternative.

All containment failure modes contain an early part for RCS release phase and a late part for the core-concrete interaction (CCI) phase. The effects of containment unfiltered venting are accounted for by the containment failure mode of Vent Release. For the Vent Release scenario, the RCS release was assumed identical to an intact containment, which was modeled as the design leak rate. And, the CCI release phase was modeled as half the release for the late rupture. This SAMA assumes that the containment is vented to the environment directly just prior to containment overpressure (during late time frame). Thus, the LATE release characteristics associated with the LATE scenarios were modified to allow the containment to be vented unfiltered at a time just prior to containment overpressure. The risk benefit was bounded by adjusting the total dose and offsite economic costs for the LATE Release Category frequency to better represent the releases that would result if the vent were to be actuated just prior to containment overpressure. The dose for this approximate scenario was computed to be 4.24E+05 man-rem, and the offsite economic cost was determined to be $1.23E+09.

This approach is conservative because it assumes that the operators would only operate the vent for those scenarios in which a late overpressure of the containment is expected; i.e., there is no downside risk assumed. This approach makes no changes to the CDF

value; i.e., assumes no impact on CDF value. Calculating the averted risk cost relative to the base case using the method described in Section 5 results in a net benefit of $860,709.

Cost: The TVA estimated cost of this SAMA is $4,757,266.

SAMA 215: Provide Backup Thermal Barrier Cooling to Ensure RCP Seal Cooling So That RCP Seal LOCAs Are Precluded for SBO Events

Description: The option considered is to provide a backup cooling water supply to the RCP Thermal Barrier Heat Exchangers within 13 minutes after a loss of CCS or SBO event. Install a cross-tie from the TDAFW pump discharge line to the discharge of the Thermal Barrier Booster Pumps in the CCS. The return flow from the thermal barriers would be routed to the ERCW return header. DC powered MOVs and controls are used to operate the crosstie from the MCR. Such a strategy would also benefit loss of emergency service water (ESW; i.e., ERCW at WBN 2) and loss of CCS events.

Risk Benefit: The risk benefit was bounded by calculating the change due to the addition of a crosstie from the TDAFW pump discharge to the inlet of the thermal barrier heat exchangers. Failure of the cross-tying capability is represented by failures of the TDAFW pump to deliver flow, two motor-operated MOVs which must operate to change position, and an operator action to align and initiate this backup thermal barrier cooling from the MCR within 13 minutes. The risk model was revised by adding a fault tree gate (which models failures of the additional equipment and operator action) ANDed with the CCS supply to the thermal barrier cooling. A very conservative HEP of 1.0E-03 was used considering the very short time window within which this operator action must be completed (to prevent degradation of the RCP seals) among all of the other immediate, post-trip actions following a SBO event. Normal conditions are applied otherwise. This SAMA will result in a reduction of risk from seal LOCAs during SBO events. The resulting CDF is 1.12E-5. Calculating the averted risk cost relative to the base case using the method described in Section 5 results in a net benefit of $631,382.

Cost: The TVA estimated cost of this SAMA is approximately $2,181,540.

SAMA 226: Provide Turbine-Driven, High Pressure Backup Charging Pump

Description: This SAMA provides a means of reducing the likelihood and limiting the size of a seal LOCA. It would provide a turbine-driven charging pump (with the full capacity to provide RCS makeup plus seal injection flow) that can be rapidly aligned to the RCP seals from the MCR. Steam would be supplied to the turbine from a branch line off the TDAFW pump Main Steam supply line. The turbine exhaust would be routed back to the TDAFW turbine exhaust stack. Cooling water for RCS letdown and seal return flow would be provided by a small diesel generator and a diesel-powered motor-driven pump. The pump and turbine would be located in the abandoned reciprocating charging pump room in AB and the supporting diesel generator could be located in the TB. Long term secondary side cooling can be provided through the operation of the turbine driven AFW pump using existing procedures. This arrangement would make it possible to provide adequate core cooling in extended SBO evolutions.

Risk Benefit: The risk benefit was bounded by calculating the change due to the installation of a turbine-driven backup charging pump and its supporting RCS letdown

and seal return flow cooling water pump and the associated diesel generator. Failure of the backup charging pump and its supporting equipment is represented by failures of a turbine-driven pump, a motor-driven pump, a diesel generator, six motor-operated valves (which must operate to change position), and an operator action to align and initiate this new backup charging pump and the supporting equipment from the MCR within 13 minutes. The risk model was revised by adding a fault tree gate (which models failures of the additional equipment and operator action) ANDed with the existing CCPs. A very conservative HEP of 1.0E-03 was used considering the very short time window within which this operator action must be completed (to prevent degradation of the RCP seals) among all of the other immediate, post-trip actions following a SBO event. Normal conditions are applied otherwise. The resulting CDF is 9.36E-6. Calculating the averted risk cost relative to the base case using the method described in Section 5 results in a net benefit of $1,059,788.

Cost: The TVA estimated cost of this SAMA is $7,727,232.

SAMA 242: Install a Permanent, Dedicated Diesel Generator for One CCP, with Local Operation of TDAFW after 125V Battery Depletion

Description: This alternative provides a means of reducing the likelihood and limiting the size of a seal LOCA and providing primary side makeup through the installation of a diesel generator that can be rapidly aligned to one CCP from the MCR. Providing dedicated power to CCP requires a dedicated DG with remote manual initiation capability to meet the 13-minute criteria to prevent seal LOCA. The self-contained DG will provide RCS makeup and RCP seal injection. It will also power a battery charger to maintain control power for MCR indication and necessary controls. Additionally, cooling for the letdown and seal water heat exchangers will be provided by a new DG powered backup CCS pump located in the TB.

Long term secondary side cooling can be provided through the operation of the turbine driven AFW pump using existing Wolf Creek procedures. This arrangement would make it possible to provide adequate core cooling in extended SBO evolutions.

Local operation of the TDAFWP is currently proceduralized and already accounted in the PRA offsite power recovery analysis during a SBO event.

Risk Benefit: This SAMA provides additional reliability to mitigate SBO events. The risk benefit was bounded by calculating the change due to the addition of a permanent, dedicated diesel generator supplying backup power to one CCP. Failure of the backup power input to this CCP is represented by failures of a diesel generator, four motor-operated valves (must operate to change position), a motor-driven pump, and an operator action to align and initiate this new diesel generator power supply to one CCP from the MCR within 13 minutes. The risk model was revised by adding a fault tree gate (which models failures of the additional equipment and operator action) ANDed with the normal power supply to one of the CCPs (Gate ABL). It is conservatively assumed that the new diesel generator can also provide backup power to the CCP A room cooler (normally supplied by 480VAC Control & Auxiliary Building Ventilation Board 2-MCC-214-A001-A, 2A1-A). A very conservative HEP of 1.0E-03 was used considering the very short time window within which this operator action must be completed (to prevent degradation of the RCP seals) among all of the other immediate, post-trip actions following a SBO event. Normal conditions are applied otherwise. The

resulting CDF is 1.41E-5. Calculating the averted risk cost relative to the base case using the method described in Section 5 results in a net benefit of $65,727.

Cost: The TVA estimated cost of this SAMA is approximately $8,043,882. The cost of the proposed change for this alternative is dominated by the implementation of the new hardware, control system capability to ensure remote actuation of the alternate power supply (timing required to prevent RCP seal LOCA), and the design work to incorporate a new diesel generator to the plant design.

With the exception of SAMA 50, the costs shown above for all of the other SAMAs are generally dominated by the plant physical change itself, rather than the life cycle cost of procedure development, training, surveillance test, and maintenance. This makes the design, safety analysis, change to the licensing basis and installation of the new hardware prohibitive relative to the potential risk reduction.

9. Uncertainty Analysis

Sensitivity cases were run for the following conditions to assess their impact on the overall SAMA evaluation:

- Use a real discount rate of 3 percent, instead of the 7 percent value used in the base case analysis.
- Use the 95th percentile for the core damage and release category frequencies in place of the mean or point estimate frequencies.
- Use alternate MACCS2 input variables for selected cases.
- Use more conservative external events multiplication factor.

9.1 Real Discount Rate

A sensitivity study has been performed in order to identify how the conclusions of the SAMA analysis might change based on the value assigned to the real discount rate. The original RDR of 7 percent has been changed to 3 percent, which could be viewed as conservative, and the MACR was recalculated using the methodology outlined previously.

Implementation of the 3 percent RDR increased the MMACR by 77 percent compared with the case where a 7 percent RDR was used. This corresponds to an increase in the MACR from $3,007,467 to $5,322,887.

The Phase I SAMAs were reviewed to determine if such an increase in the MACR would impact the disposition of any SAMAs. It was determined that some additional SAMAs could have been screened in the Phase I if an RDR of 3 percent were used in place of the 7 percent value. However, all seven SAMAs were retained for detailed analysis to ensure that the final decisions are made based on a more accurate analysis.

The Phase II SAMAs are dispositioned based on detailed analysis. As shown in Table 13, the determination of cost effectiveness was only changed for SAMA 50 when the 3 percent RDR was used in lieu of 7 percent. However, the margin by which the SAMA becomes “cost beneficial” is small and it does not mean that this SAMA would be selected for consideration if a 3 percent real discount rate were applied in the SAMA analysis as other factors influence the decision making process, such as the 95th percentile sensitivity analysis.

9.2 95th Percentile PRA Results

The results of the SAMA analysis can be impacted by implementing conservative values from the PRA’s uncertainty distribution. If the best estimate failure probability values were consistently lower than the “actual” failure probabilities, the PRA model would underestimate plant risk and yield lower than “actual” averted cost-risk values for

potential SAMAs. Re-assessing the cost benefit calculations using the high end of the failure probability distributions is a means of identifying the impact of having consistently underestimated failure probabilities for plant equipment and operator actions included in the PRA model. This sensitivity uses the 95th percentile results to examine the impact of uncertainty in the PRA model.

For WBN 2, propagation of the uncertainty distributions associated with parameters used in the current CAFTA PRA model (WBN_U1_U2_NEW) shows that the ratios of the 95th percentile to the mean value are 1.52 and 1.78 for CDF and LERF, respectively. However, this representation of the uncertainty only reflects parameter uncertainty and does not account for modeling uncertainty. The results of the previous RISKMAN version of the Level 1 internal events model and Level 2 model uncertainty analysis was used as estimates of the uncertainty associated with the 2010/2011 SAMA model. The ratio of the 95th percentile CDF to the mean CDF from the RISKMAN model was 2.7. The ratio of the 95th percentile LERF to the mean LERF from the RISKMAN model was 2.6. Based on the same CDF ratio of 2.7, the CDF results and the Level 2 release category results were assessed in this SAMA re-evaluation against a factor of 2.7 higher (i.e., the point estimate mean CDF and Release Category frequencies were increased by a factor of 2.7) to provide an estimate of the impact of using the 95th percentile results.

As shown in Table 14, the determination of cost effectiveness changed for only one Phase II SAMA (i.e., SAMA 50) when the 95th percentile parameter uncertainty was used in lieu of the mean values. However, the margin by which the SAMA becomes “cost beneficial” is still limited (when an external events multiplying factor of 2.0 is used) and it does not mean that this SAMA would be selected for consideration if a 95th percentile CDF/LERF were applied in the SAMA analysis as other factors influence the decision making process.

9.3 WinMACCS Input Variations

The MACCS2 model was developed using the best information available for the WBN site; however, reasonable changes to modeling assumptions can lead to variations in the Level 3 results. In order to determine how certain assumptions could impact the SAMA results, sensitivity assessments were performed on a group of parameters that has previously been shown to impact the Level 3 results. These parameters include:

- Meteorological Data
- Population Estimates
- Evacuation Effectiveness
- Radionuclide Release Height

Meteorological data and radionuclide release height have been studied extensively (e.g., the Vogtle and Wolf Creek SAMA Uncertainty analyses) and have been shown to result in relatively small changes in overall risk.

On the other hand, population density and evacuation speed have been shown to have the greatest effect on risk. Population density increases have been accounted for in the WBN 2 SAMA assessments by using the projected 2040 population densities in the

50-mile radius of the WBN site. Smaller population increases would serve to reduce the cost effectiveness of various SAMA alternatives.

The impact of evacuation speed was investigated by performing a sensitivity analysis with MACCS2 where the evacuation speed was reduced from 2.2 mph (1 meter/sec) to 1.6 mph and another where the evacuation speed was increased to 3.4 mph. The results of the variations in evacuation speed on population dose and economic costs are shown in Table 15. The results, in terms of impact on the baseline SAMA cost benefit are provided in Table 16 for the seven SAMAs evaluated in Phase II. As shown in Table 16, the cost effectiveness of all SAMAs does not change with changes in evacuation speed. This is due to the relatively low contribution of offsite exposure cost to the overall cost as shown in Section 4.5.

9.4 External Events Multiplication Factor

The use of an external events multiplier of 2 is consistent with the Watts Bar Unit 1 licensing bases, consistent with recent and ice condenser plant specific external events multipliers, consistent with guidance for the development of an external events multiplier per NEI-05-01 and confirmed through the closure of GI-194. However, an external events multiplier of 2.28 was also used as a sensitivity evaluation. This increase in impact for external events is judged to adequately bound the potential impact on results for external events, compared to the internal events model used. The impact of a higher external events multiplier of 2.28 on the benefit/cost analysis is shown in Tables 12, 13, 14, and 16.

10. Conclusions

The benefits of revising the operational strategies in place at Watts Bar and/or implementing hardware modifications can be evaluated without the insight from a risk-based analysis. However, use of the PRA in conjunction with cost-benefit analysis methodologies provides an enhanced understanding of the effects of the proposed changes relative to the cost of implementation and projected impact on offsite dose and economic impacts.

In previous SAMA analyses, SAMA 58 (Westinghouse SHIELD seal design) was not considered further because insufficient operational experience had been accumulated. However, TVA committed to follow the progress and experience in this new RCP seal design. Following failures in the post-service testing of the SHIELD seals used at Farley and Beaver Valley Unit 2, TVA has decided to re-evaluate in this analysis the seven SAMAs related to the RCP seal LOCA event as alternates to SAMA 58 due to the uncertain effectiveness in the implementation of the SHIELD seal design.

Based on the results of the detailed SAMA Phase II baseline analysis, all seven SAMAs are not cost beneficial. However, the results of the uncertainty analysis for this study indicate that, while not cost-beneficial in the baseline evaluation, SAMA 50 would be potentially cost beneficial at the 95th percentile, or the 3 percent RDR based on very conservative HRA treatment. SAMA 50 involves the enhancement of the CCS procedure to underscore RCS cooldown for small/medium LOCA response following a loss of RCP seal cooling event. Although the realistic benefit may not be significant due to the possible rapid degradation of the RCP seals under these conditions, it is possible that TVA may still consider implementing this SAMA.

Table 1. Definition and Causes of Containment Failure Mode Classes

Failure Mode	Definition and Causes
I Non-Bypass Early Containment Failure (LERF)	Involves structure failure of the containment before, during, or slightly after (within a few hours of) reactor vessel failure. A variety of mechanisms can cause structure failure, including direct contact of core debris with containment, rapid pressure and temperature loads, hydrogen combustion, and fuel coolant interaction (ex-vessel steam explosion). Failure to isolate containment or to provide early venting of containment after core damage also is classified as early containment failures.
II Containment Bypass (BYPASS)	Involves failure of the pressure boundary between the high-pressure reactor coolant and low-pressure auxiliary system. For pressurized water reactors, steam generator tube rupture, either as an initiating event or as a result of severe accident conditions, will lead to containment bypass. In this scenario, if core damage occurs, a direct path to the environment can exist.
III Late Containment Failure (LATE)	Involves structural failure of the containment several hours after reactor vessel failure. A variety of mechanisms can cause late structure failure, including gradual pressure and temperature increase, and hydrogen combustion. Venting containment late in the accident also is classified as a late containment failure.
IV Small pre-existing hole (SERF)	Involves no structural failure or bypass of the containment except for a small pre-existing hole that cannot be isolated. If core damage occurs, fission products are released from the containment but at a small rate limited to approximately 25 times the design basis leak rate.
V Intact Containment	Involves no structural failure or bypass of the containment. If core damage occurs, fission products are retained in the containment and there is no release to the environment.

Table 2. Unit 2 Core Damage Frequency Results

Record Model CDF	SAMA Model CDF
1.43E-05	1.43E-05

Table 3. Unit 2 Release Category Results

Release Category	Record Model Frequency	SAMA Model Frequency
I. Non-Bypass Large Early Release Frequency (NON-BYPASS-LERF), Note 1	**1.16E-06**	**5.63E-07**
II. Bypass Release Frequency (BYPASS-LERF)	**Note 1**	**6.07E-07**
III. Late Release Frequency (LATE)	**1.03E-05**	**1.03E-05**
IV. Small, pre-existing hole (Early) Release Frequency (SERF)	**1.68E-06**	**1.67E-06**

Note 1: For the record model, the total LERF is 1.16E-06 which includes the frequencies of both non-bypass early containment failure and bypass sceanrios; i.e., bypass frequency is part of LERF.

Table 4. Release Category Dominant Scenarios

Release Category	Base Case Frequency	Example Scenario
I – NON-BYPASS LERF	5.63E-07	The major accident contributors to this release event are initiated by flooding scenarios causing a SBO and core damage followed by early containment failure. Other significant contributors include small LOCA event followed by failure of AFW cooldown, sump recirculation or RWST makeup failure, core damage, and early containment failure; and loss of CCS or ERCW followed by failure of CCP alternate cooling, seal LOCA, core damage, and early containment failure.
II – BYPASS LERF	6.07E-07	The major accident contributors to this release event are initiated by flooding scenarios causing a SBO followed by a temperature-induced or pressure-induced SGTR, and core damage. Other significant contributors include an ISLOCA in conjunction with operator errors leading to failure of the coolant system and makeup to the RCS; and SBO initiated by LOSP followed by operator failure to locally control AFW, a temperature-induced or pressure-induced SGTR, and core damage.
III – LATE	1.03E-05	The major accident contributors to this release event are initiated by flooding scenarios causing core damage and loss of containment heat removal. Other significant contributors include loss of CCS or ERCW followed by failure of CCP alternate cooling, seal LOCA, core damage, and early containment failure.
IV – SERF	1.67E-06	This category involves core damage sequences with an intact containment except for a small pre-existing hole that cannot be isolated. Containment heat removal is available so that neither late overpressure nor basemat melt-thru occurs.

Table 5. Projected 2040 Population Distribution within 80 Kilometers (50 miles)

Direction	Miles										
	0-1	**1-2**	**2-3**	**3-4**	**4-5**	**5-10**	**10-20**	**20-30**	**30-40**	**40-50**	**0-50**
N	0	18	0	0	135	2,465	1,885	2,778	4,768	6,172	18,222
NNE	0	0	18	411	185	1,536	11,762	18,766	14,502	2,547	49,727
NE	0	0	18	308	287	827	3,783	16,734	29,838	78,334	130,130
ENE	0	0	18	308	287	497	3,553	29,539	63,798	25,383 1	351,832
E	0	8	431	308	616	552	11,352	18,647	30,063	44,013	105,990
ESE	0	0	0	27	41	68	6,230	20,120	5,068	3,280	34,833
SE	8	0	0	29	39	135	19,852	15,185	3,950	4,822	44,020
SSE	21	0	0	246	413	103	8,951	12,907	2,918	48,593	74,151
S	16	0	0	0	1,983	3,824	4,586	42,883	56,430	17985	127,707
SSW	0	0	21	0	0	546	5,725	42,517	46,281	106,39 2	201,482
SW	0	0	0	0	0	1,051	12,978	14,499	62,307	111,79 5	202,630
WSW	0	6	36	59	126	711	12,791	2,837	2,840	3,372	22,778
W	0	14	22	101	90	710	3,406	5,555	2,944	5,474	18,316
WNW	0	0	22	126	79	490	2,091	4,372	5,654	20,511	33,345
NW	0	108	332	376	526	2,655	2,889	18,634	10,462	15,956	51,940
NNW	0	0	0	173	123	3,116	1,536	33,843	11,609	5,890	56,290
Total	45	154	918	2,4 72	4,930	19,286	11,3370	299,816	353,432	728,96 7	1,523,39 0

Source (SAIC 2007); same as used in the September 2011 RAI response submittal.
Note: To convert from mile to kilometer multiply the value by 1.609.

Table 6. Watts Bar Core Inventory

Nuclide	Isotope	Group [a]	Curies
Cobalt	Co-58	6	1.11E+06
	Co-60	6	8.67E+05
Krypton	Kr-83m	1	1.15E+07
	Kr-85m	1	2.39E+07
	Kr-85	1	1.03E+06
	Kr-87	1	4.81E+07
	Kr-88	1	6.66E+07
Xenon	Xe-131m	1	1.05E+06
	Xe-133m	1	6.16E+06
	Xe-133	1	1.91E+08
	Xe-135m	1	4.05E+07
	Xe-135	1	6.43E+07
	Xe-138	1	1.67E+08
Iodine	I-130	2	1.93E+06
	I-131	2	9.46E+07
	I-132	2	1.39E+08
	I-133	2	1.95E+08
	I-134	2	2.16E+08
	I-135	2	1.86E+08
Bromine	Br-83	2	1.15E+07
	Br-84	2	2.14E+07
Cesium	Cs-134	3	1.66E+07
	Cs-135	3	0.00E+00
	Cs-136	3	5.89E+06
	Cs-137	3	1.17E+07
	Cs-138	3	1.81E+08
Rubidium	Rb-86	3	1.87E+05
	Rb-88	3	6.83E+07
	Rb-89	3	8.92E+07

Table 6. Watts Bar Core Inventory (Continued)

Nuclide	Isotope	Group [a]	Curies
Strontium	Sr-89	4	9.34E+07
	Sr-90	5	8.94E+06
	Sr-91	5	1.16E+08
	Sr-92	5	1.24E+08
Yttrium	Y-90	7	9.48E+06
	Y-91m	7	6.76E+07
	Y-91	7	1.21E+08
	Y-92	7	1.25E+08
	Y-93	7	9.48E+07
	Y-94	7	1.51E+08
	Y-95	7	1.57E+08
Zirconium	Zr-95	7	1.67E+08
	Zr-97	7	1.61E+08
Niobium	Nb-95	7	1.69E+08
	Nb-97m	7	1.53E+08
	Nb-97	7	1.62E+08
Molybdenum	Mo-99	6	1.78E+08
Technetium	Tc-99m	6	1.57E+08
	Tc-99	6	0.00E+00
	Tc-101	6	1.61E+08
Ruthenium	Ru-103	6	1.48E+08
	Ru-105	6	1.00E+08
	Ru-106	6	5.00E+07
Rhodium	Rh-103m	6	1.48E+08
	Rh-105	6	9.55E+07
	Rh-106	6	5.33E+07
	Rh-107	6	5.77E+07
Antimony	Sb-127	4	8.05E+06
	Sb-129	4	3.03E+07
	Sb-130	4	1.00E+07
Tellurium	Te-125m	4	1.93E+04
	Te-127m	4	1.33E+06
	Te-127	4	7.93E+06

Table 6. Watts Bar Core Inventory (Continued)

Nuclide	Isotope	Group [a]	Curies
	Te-129m	4	5.81E+06
	Te-129	4	2.88E+07
	Te-131m	4	1.86E+07
	Te-131	4	7.99E+07
	Te-132	4	1.36E+08
	Te-133	4	1.06E+08
	Te-134	4	1.73E+08
Barium	Ba-137m	5	1.11E+07
	Ba-139	5	1.73E+08
	Ba-140	5	1.73E+08
	Ba-141	5	1.56E+08
	Ba-142	5	1.49E+08
Lanthanum	La-140	7	1.79E+08
	La-141	7	1.58E+08
	La-142	7	1.54E+08
	La-143	7	1.46E+08
Cerium	Ce-141	8	1.59E+08
	Ce-143	8	1.48E+08
	Ce-144	8	1.29E+08
Praseodymium	Pr-143	7	1.44E+08
	Pr-144	7	1.30E+08
	Pr-145	7	1.01E+08
Neodymium	Nd-147	7	6.39E+07
Neptunium	Np-239	8	1.87E+09
Plutonium	Pu-238	8	3.15E+05
	Pu-239	8	3.48E+04
	Pu-240	8	4.38E+04
	Pu-241	8	1.49E+07
	Pu-243	8	2.86E+07
Americium	Am-241	7	9.80E+03
	Am-242	7	7.93E+06

Table 6. Watts Bar Core Inventory (Continued)

Nuclide	Isotope	Group [a]	Curies
Curium	Cm-242	7	3.98E+06
	Cm-244	7	1.61E+05

[a] The grouping is based on NUREG-1465.

Source (SAIC 2007); same as used in the September 2011 RAI response submittal.

Table 7. Release Times, Heights, and Energies for Release Categories

Release Category	Release Height (meters)	Warning Time (hours)	Release Time (hours)	Release Duration (hours)	Release Energy [a] (megawatts)
I – Non-Bypass LERF	10.00	8	10	2	28
II – BYPASS LERF	10.00	4	8	4	1
III – LATE	10.00	20	30	10	3.5
IV – SERF	10.00	8	10	2	3.5

[a] These values were taken from similar accident scenarios given in NUREG/CR-4551.

Source (SAIC 2007); same as used in the September 2011 RAI response submittal.

Release characteristics are the same for the separate dominant cases in each release category.

Table 8. Weighted Fission Product Source Terms for Four Release Categories

Release Category	NG	I	Cs	Te	Ba	Sr	Ru	La	Ce
I – Non-Bypass LERF	8.5E-01	8.4E-02	4.7E-02	4.0E-02	1.4E-02	1.5E-02	9.9E-03	1.0E-02	1.5E-02
II – BYPASS LERF	8.5E-01	9.0E-02	4.9E-02	3.6E-02	1.1E-02	1.1E-02	7.2E-03	7.5E-03	1.2E-02
III – LATE	8.5E-01	1.3E-02	7.2E-03	7.9E-03	3.0E-03	4.7E-03	2.5E-03	2.4E-03	3.1E-03
IV – SERF	8.5E-03	7.3E-04	6.3E-04	6.0E-04	5.6E-04	5.8E-04	5.5E-04	5.5E-04	5.6E-04

NG = Noble gases.

Mo is not a separate entry and is included within the Ru group.

The fission product source terms for Release Categories Non-Bypass LERF, LATE, and SERF are weighted by the relative frequencies of their respective dominant cases contributing to each release category (4 cases for Non-Bypass LERF and LATE; and 2 cases for SERF).

Same as shown in the September 2011 RAI response submittal.

Table 9. Evacuation Times 0-to-16-Kilometer (0-to-10-mile) Area

Evacuation Paths	Permanent Population, Adverse Condition (hrs-min)	Special Population, Adverse Condition (hrs-min)	General Population, Adverse Condition (hrs-min)
1	6 - 40	3 - 40	5 - 12
2	4 - 23	2 - 41	3 - 47
3	4 - 21	2 - 43	5 - 0
4	4 - 10	2 - 36	3 - 41
5	4 - 37	2 - 53	4 - 05
6	4 - 25	2 - 45	3 - 54
7	4 - 21	2 - 43	3 - 51
8	4 - 25	2 - 45	3 - 54
9	3 - 26	2 - 15	3 - 30
10	3 - 26	2 - 15	3 - 30
11	3 - 26	2 - 30	3 - 50
12	3 - 26	2 - 30	3 - 54
13	3 - 26	2 - 0	3 - 30
14	3 - 26	1 - 35	3 - 30
15	3 - 20	1 - 30	3 - 25
Total	61 - 20	37 - 21	58 - 33
Average Hours	4 - 5	2 - 29	3 - 54
Average Speed over 10 Miles (miles per hour)	2.45	4.02	2.56
(meters per second)	1.1	1.8	1.15

Source (SAIC 2007); same as used in the September 2011 RAI response submittal.

Table 10a Severe Reactor Accident Conditional Risks

Release Category	Offsite Population Dose within 80 Kilometers (50 miles) (person-rem)	Offsite Economic Cost within 80 Kilometers (50 miles) (dollars)
I – Non-Bypass LERF	2.96×10^6	6.35×10^9
II – BYPASS LERF	2.39×10^6	5.31×10^9
III – LATE	1.09×10^6	3.19×10^9
IV – SERF	3.22×10^5	5.85×10^8

Table 10b. Annual 80-Kilometer (50-mile) Population Dose and Economic Cost Risk

Release Category	Population Dose Risk (person-rem/year)	Economic Cost Risk (dollars/year)
I – Non-Bypass LERF	1.67	3.58×10^3
II – BYPASS LERF	1.45	3.22×10^3
III – LATE	11.2	3.28×10^4
IV – SERF	0.537	9.76×10^2

Table 11. Phase I SAMA Candidates

SAMA Number	SAMA Title	SAMA Discussion	Source	Phase I Comments	Disposition
50	Enhance loss of CCS procedure to underscore RCS cooldown for small/medium LOCA response following a loss of RCP seal cooling event	Reduced impacts of RCP seal failure.	NEI 05-01 (Rev A)	Basis for Screening: Upon receipt of any RCP seal No. 1 outlet temperature high alarm, AOI-15 & 24 already require an RCS cooldown after isolation of the CCS path to the RCP thermal barrier and isolation of RCP seal injection. This order of actions is deemed appropriate for overall plant stabilization following a loss of CCS. If both RCP thermal barrier cooling and seal injection are lost, RCP seal LOCA would occur within about 13 minutes. Improved loss of CCS procedure cannot prevent a seal LOCA to occur under this condition because of the rapid progression of the seal leak, although it can potentially reduce the impact of a RCP seal LOCA. The extent of risk reduction by an enhanced procedure is not known (may not reduce the risk significantly; i.e., may only have minimal risk improvement). Total cost of implementation of this procedure improvement including procedure change and training is estimated to be $31,675.This SAMA is retained for further evaluation.	Retain For Phase II Analysis

Table 11. Phase I SAMA Candidates (Continued)

SAMA Number	SAMA Title	SAMA Discussion	Source	Phase I Comments	Disposition
55	Install an independent motor-driven RCP seal injection system, with dedicated diesel.	Reduced frequency of core damage from loss of component cooling water, service water, or station blackout.	NEI 05-01 (Rev A)	For a plant with significant construction already completed, the estimated cost of implementation is likely to exceed the bounding benefit. This evaluation considers installing an independent, motor-operated RCP seal injection system with dedicated diesel in the location of the abandoned reciprocating charging pump in the AB. It can be initiated from the MCR within 13 minutes after a loss of CCS or SBO event. Total cost including hardware/materials, engineering, modification implementation, etc. is estimated to be in excess of $5.4M. This SAMA is retained for further detailed evaluation.	Excessive Implementation Cost, but Retain For Phase II Analysis

Table 11. Phase I SAMA Candidates (Continued)

SAMA Number	SAMA Title	SAMA Discussion	Source	Phase I Comments	Disposition
56	Install an independent turbine-driven RCP seal injection system, without dedicated diesel.	Reduced frequency of core damage from loss of component cooling water or service water, or station blackout.	NEI 05-01 (Rev A)	Install a turbine driven seal injection pump in the location of the abandoned reciprocating charging pump in AB. Use the old suction and discharge piping connections with new DC MOVs to provide the capability of MCR initiation within 13 minutes after a loss of CCS or SBO event. The turbine steam supply will be provided by a branch line from the Main Steam supply line to the TDAFW pump. Reroute the turbine exhaust line back to the TDAFW pump exhaust stack and share that discharge to atmosphere. Will provide a normal seal injection flow rate of approximately 40 gpm, but not the normal RCS makeup. No cooling for the letdown and seal return flows which will be rerouted to the RWST to mix with the cooler RWST inventory. Total cost including hardware/materials, engineering, modification implementation, etc. is estimated to be in excess of $5.2M. This SAMA is retained for further evaluation.	Excessive Implementation Cost, but Retain For Phase II Analysis

Table 11. Phase I SAMA Candidates (Continued)

SAMA Number	SAMA Title	SAMA Discussion	Source	Phase I Comments	Disposition
93	Install an unfiltered, hardened containment vent.	Increased decay heat removal capability for non-ATWS events, without scrubbing released fission products.	NEI 05-01 (Rev A)	For a plant with significant construction already completed, the estimated cost of implementation could exceed the bounding benefit. Provide an isolatable pathway using a spare penetration to an elevated discharge point, which would preserve the primary containment boundary following an RCP seal LOCA and subsequent core damage event. The air operated containment isolation valves would be provided inside and outside the containment penetration and would be operated from a MCR panel with a new DC battery subsystem including a battery charger. The HCVS would be actuated at less than or equal to containment failure pressure to discharge post-accident combustible gases and/or steam, thus precluding breach of the containment boundary. The implementation cost is estimated to be in excess of $4.7M. This SAMA is retained for further evaluation.	Excessive Implementation Cost, but Retain For Phase II Analysis

Table 11. Phase I SAMA Candidates (Continued)

SAMA Number	SAMA Title	SAMA Discussion	Source	Phase I Comments	Disposition
215	Provide backup thermal barrier cooling to ensure RCP seal cooling so that RCP seal LOCAs are precluded for SBO events.	Option considered is to use a cross-tie connection from the TDAFW pump discharge to provide backup cooling to RCP thermal barriers. Such a strategy would also benefit loss of ESW/ERCW and loss of CCS events.	Cook	The option considered may not be cost beneficial. For a plant with significant construction already completed, the estimated cost of implementation is likely to exceed the bounding benefit. The option evaluated in this analysis is to provide a backup cooling water supply to the RCP Thermal Barrier Heat Exchangers within 13 minutes after a loss of CCS or SBO event. Install a cross-tie from the TDAFW pump discharge line to the discharge of the Thermal Barrier Booster Pumps in the CCS. The return flow from the thermal barriers would be routed to the ERCW return header. Use DC powered MOVs. The estimated cost is in excess of $2.1M. This SAMA is retained for further evaluation.	Retain For Phase II Analysis

Table 11. Phase I SAMA Candidates (Continued)

SAMA Number	SAMA Title	SAMA Discussion	Source	Phase I Comments	Disposition
226	Install a turbine-driven charging pump to backup CCPs.	This SAMA provides a means of reducing the likelihood and limiting the size of a seal LOCA. This SAMA would provide a turbine-driven, high pressure charging pump that can be rapidly aligned to the RCP seals from the MCR. Long term secondary side cooling can be provided through the operation of the turbine driven AFW pump using existing Vogtle procedures. This arrangement would make it possible to provide adequate core cooling in extended SBO evolutions.	Vogtle	This alternative would provide a backup, turbine-driven charging pump that can be initiated from the MCR within 13 minutes after a loss of CCS or SBO event. The turbine driven, high pressure charging pump would be installed in the abandoned location of the reciprocating charging pump in the AB. The steam supply for the turbine will be provided by a branch line off the Main Steam supply to the TDAFW pump. The turbine exhaust will be routed back to the TDAFW pump exhaust stack. Cooling for letdown and seal water heat exchangers will be provided by a new diesel generator and a diesel powered backup CCS pump. The cost of this enhancement has been estimated to be more than $7.7M based on a conceptual design of the backup pump which exceeds the bounding benefit. This SAMA is retained for further evaluation.	Excessive Implementation Cost, but still Retain For Phase II Analysis

Table 11. Phase I SAMA Candidates (Continued)

SAMA Number	SAMA Title	SAMA Discussion	Source	Phase I Comments	Disposition
242	Install a permanent, dedicated diesel generator for one CCP with local operation of TD AFW after 125V battery depletion.	This SAMA provides a means of reducing the likelihood and limiting the size of a seal LOCA and providing primary side makeup through the installation of a diesel generator that can be rapidly aligned to one CCP from the MCR. Long term secondary side cooling can be provided through the operation of the turbine driven AFW pump using existing Wolf Creek procedures. This arrangement would make it possible to provide adequate core cooling in extended SBO evolutions.	Wolf Creek	Local operation of the TDAFWP is currently proceduralized and already accounted in the PRA offsite power recovery analysis during a SBO event. Providing dedicated power to CCP requires a dedicated DG with remote manual initiation capability to meet the 13 minute criteria to prevent seal LOCA. The self-contained DG will provide RCS makeup and RCP seal injection. Further, it will also power a battery charger to maintain control power for MCR indication and necessary controls. Additionally, cooling for the letdown and seal water heat exchangers will be provided by a new DG powered backup CCS pump located in the Turbine Building. The cost for installing a dedicated DG and a new CCS pump is estimated to be in excess of $8.0M. This SAMA is retained for further evaluation.	Excessive Implementation Cost, but still Retain For Phase II Analysis

Table 12. Phase II Analysis Results

SAMA No.	SAMA Title	External Events Multiplier = 2.0				External Events Multiplier = 2.28			
		Estimated Benefit	Estimated Cost	Benefit/ Cost Ratio	Change in Conclusion	Estimated Benefit	Estimated Cost	Benefit/ Cost Ratio	Change in Conclusion
50	Enhance loss of CCS procedure to underscore RCS cooldown for small/medium LOCA response following a loss of RCP seal cooling event	$16,789	$31,675	0.53	Not cost beneficial	$19,140	$31,675	0.60	Not cost beneficial
55	Install an independent motor-driven RCP seal injection system with dedicated diesel generator	$1,060,052	$5,438,754	0.19	Not cost beneficial	$1,208,459	$5,438,754	0.22	Not cost beneficial
56	Install an independent turbine-driven RCP seal injection system, without dedicated diesel.	$1,067,736	$5,213,021	0.20	Not cost beneficial	$1,217,219	$5,213,021	0.23	Not cost beneficial
93	Install an unfiltered, hardened containment vent.	$860,709	$4,757,266	0.18	Not cost beneficial	$981,208	$4,757,266	0.21	Not cost beneficial
215	Provide a backup thermal barrier cooling to ensure RCP seal cooling so that RCP seal LOCAs are precluded for SBO events.	$631,382	$2,181,540	0.29	Not cost beneficial	$719,775	$2,181,540	0.33	Not cost beneficial
226	Provide a turbine-driven charging pump to back up CCPs.	$1,059,788	$7,727,232	0.14	Not cost beneficial	$1,208,158	$7,727,232	0.16	Not cost beneficial
242	Install a permanent, dedicated diesel generator for one CCP with local operation of TDAFW after 125V battery depletion.	$65,727	$8,043,882	0.01	Not cost beneficial	$74,929	$8,043,882	0.01	Not cost beneficial

Table 13. RDR Sensitivity Results

SAMA No.	SAMA Title	External Events Multiplier = 2.0			External Events Multiplier = 2.28		
		Benefit/ Cost Ratio 7% RDR	Benefit/ Cost Ratio 3% RDR	Change in Conclusion	Benefit/ Cost Ratio 7% RDR	Benefit/ Cost Ratio 3% RDR	Change in Conclusion
50	Enhance loss of CCS procedure to underscore RCS cooldown for small/medium LOCA response following a loss of RCP seal cooling event	0.53	0.95	NO	0.60	1.08	YES; based on very conservative HRA treatment
55	Install an independent motor-driven RCP seal injection system with dedicated diesel generator	0.19	0.36	NO	0.22	0.41	NO
56	Install an independent turbine-driven RCP seal injection system, without dedicated diesel.	0.20	0.38	NO	0.23	0.43	NO
93	Install an unfiltered, hardened containment vent.	0.18	0.32	NO	0.21	0.37	NO
215	Provide a backup thermal barrier cooling to ensure RCP seal cooling so that RCP seal LOCAs are precluded for SBO events.	0.29	0.54	NO	0.33	0.61	NO
226	Provide a turbine-driven charging pump to back up CCPs.	0.14	0.25	NO	0.16	0.29	NO
242	Install a permanent, dedicated diesel generator for one CCP with local operation of TDAFW after 125V battery depletion.	0.01	0.01	NO	0.01	0.02	NO

Table 14. CDF/LERF Sensitivity Results

SAMA No.	SAMA Title	External Events Multiplier = 2.0			External Events Multiplier = 2.28		
		Benefit/ Cost Ratio Mean CDF (Base Case)	Benefit/ Cost Ratio 95th%CDF	Change in Conclusion	Benefit/ Cost Ratio Mean CDF (Base Case)	Benefit/ Cost Ratio 95th%CDF	Change in Conclusion
50	Enhance loss of CCS procedure to underscore RCS cooldown for small/medium LOCA response following a loss of RCP seal cooling event	0.53	1.43	YES; based on very conservative HRA treatment	0.60	1.63	YES; based on very conservative HRA treatment
55	Install an independent motor-driven RCP seal injection system with dedicated diesel generator	0.19	0.53	NO	0.22	0.60	NO
56	Install an independent turbine-driven RCP seal injection system, without dedicated diesel.	0.20	0.55	NO	0.23	0.63	NO
93	Install an unfiltered, hardened containment vent.	0.18	0.49	NO	0.21	0.56	NO
215	Provide a backup thermal barrier cooling to ensure RCP seal cooling so that RCP seal LOCAs are precluded for SBO events.	0.29	0.78	NO	0.33	0.89	NO
226	Provide a turbine-driven charging pump to back up CCPs.	0.14	0.37	NO	0.16	0.42	NO
242	Install a permanent, dedicated diesel generator for one CCP with local operation of TDAFW after 125V battery depletion.	0.01	0.02	NO	0.01	0.03	NO

Table 15. Evacuation Speed Sensitivity Dose and Economic Cost Results

Release Category/Case	Base Case (2.2m/s)		Faster Evacuation (3.4m/s)		Slower Evacuation (1.6m/s)	
	Total Person-Rem	Economic Cost ($)	Total Person-Rem	Economic Cost ($)	Total Person-Rem	Economic Cost ($)
I. NON-BYPASS EARLY	2.96E+06	6.35E+09	2.83E+06	6.35E+09	3.19E+06	6.35E+09
II. BYPASS	2.39E+06	5.31E+09	2.35E+06	5.31E+09	2.46E+06	5.31E+09
III. LATE	1.09E+06	3.19E+09	1.09E+06	3.19E+09	1.09E+06	3.19E+09
IV. SERF	3.22E+05	5.85E+08	3.16E+05	5.85E+08	3.34E+05	5.85E+08

Table 16. Evacuation Speed Sensitivity SAMA Case Results

SAMA No.	SAMA Title	External Events Multiplier = 2.0				External Events Multiplier = 2.28			
		Benefit/Cost Ratio 2.2 mph	Benefit/Cost Ratio 3.4 mph	Benefit/Cost Ratio 1.6 mph	Change in Conclusion	Benefit/Cost Ratio 2.2 mph	Benefit/Cost Ratio 3.4 mph	Benefit/Cost Ratio 1.6 mph	Change in Conclusion
50	Enhance loss of CCS procedure to underscore RCS cooldown for small/medium LOCA response following a loss of RCP seal cooling event	0.53	0.53 0.53		NO	0.60	0.60 0.61		NO
55	Install an independent motor-driven RCP seal injection system with dedicated diesel generator	0.19	0.43 0.20		NO	0.22	0.49 0.23		NO
56	Install an independent turbine-driven RCP seal injection system, without dedicated diesel.	0.20	0.21 0.21		NO	0.23	0.24 0.24		NO
93	Install an unfiltered, hardened containment vent.	0.18	0.28 0.18		NO	0.21	0.32 0.21		NO
215	Provide a backup thermal barrier cooling to ensure RCP seal cooling so that RCP seal LOCAs are precluded for SBO events.	0.29	0.30 0.30		NO	0.33	0.35 0.35		NO
226	Provide a turbine-driven charging pump to back up CCPs.	0.14	0.14 0.14		NO	0.16	0.16 0.16		NO
242	Install a permanent, dedicated diesel generator for one CCP with local operation of TDAFW after 125V battery depletion.	0.01	0.01 0.01		NO	0.01	0.01 0.01		NO

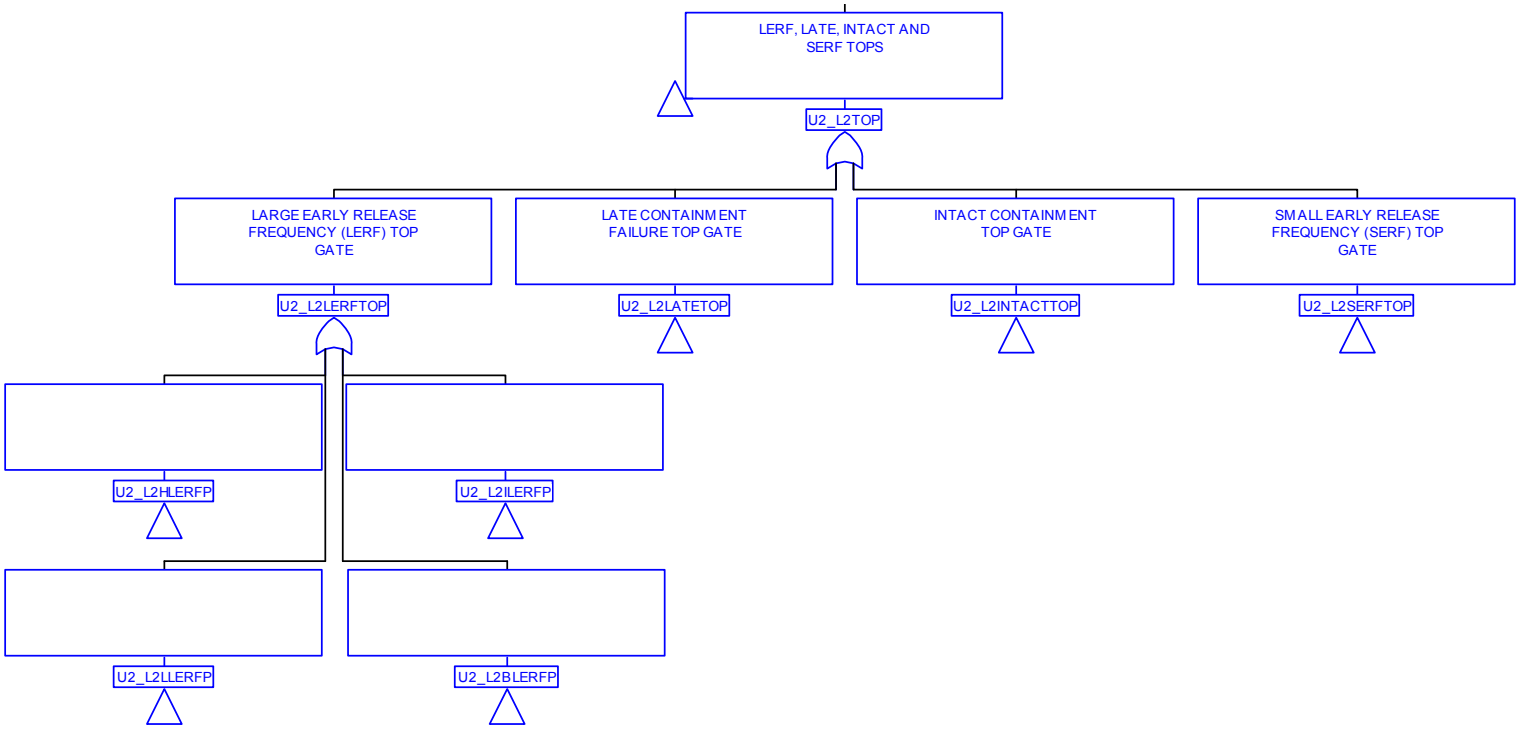

Figure 1. Record Model Level 2 Release Category Fault Tree

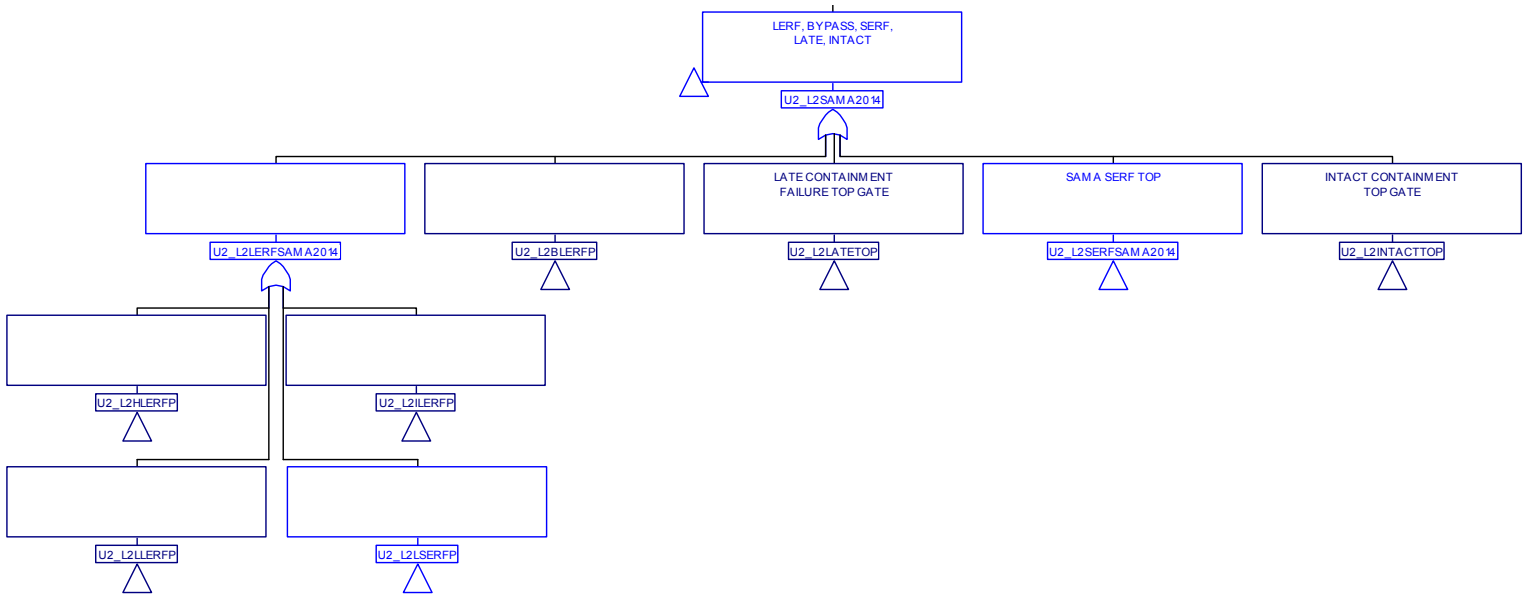

Figure 2. SAMA Model Level 2 Release Category Fault Tree

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